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5 CONTAINMENT SYSTEM

The containment for this plant consists of two systems which are:

a. The Reactor Building to provide biological and missile shielding and to contain the energy and material that could be released by an accident.

b. The engineered safeguards systems which limit the maximum value of the energy released by an accident.

5.1 REACTOR BUILDING

The Reactor Building will be a reinforced concrete Class I structure composed of cylindrical walls with a flat foundation mat and a shallow dome roof. The foundation slab will be reinforced with conventional steel reinforcing. The cylindrical walls will be prestressed with a post-tensioning system in the vertical and horizontal directions. The dome roof will be prestressed utilizing a three-way post tensioning system. The inside surface of the Reactor Building will be lined with a carbon steel liner to insure a high degree of tightness during operating and accident conditions. Liner plate thickness will be 3/8 in. for the cylinder and dome and 1/4 in. for the base.

The foundation mat will be founded on competent bearing material and will be approximately 12.5 ft thick with a 2 ft thick concrete slab above the bottom liner plate. The cylinder portion will have an inside diameter of 130 ft, a wall thickness of 3 ft 6 in., and a height of 157 ft from the top of the foundation slab to the spring line. The shallow dome roof will have a large radius of 110 ft, a transition radius of 20 ft 6 in., and a thickness of 3 ft. The Reactor Building is shown in Figure 5-1. 7

A definition of the design criteria plus an explanation of the design methods are included in Appendices 5A, "Structural Design Bases," 5B, "Design Program for Reactor Building," and 5C, "Design Criteria For Reactor Building." Delineation of quality control responsibilities and extracts of specification requirements pertaining to quality control are included in Appendix 5D, "Quality Control."

The experience and knowledge gained in the design at Rochester Gas and Electric Company's Brookwood Station No. 1 (AEC Docket No. 50-244), and Metropolitan Edison Company's Three Mile Island Station (AEC Docket No. 50-289), as well as designs by others that are similar in functional requirements, will be utilized in the design of this Reactor Building.

5.1.1 DESIGN BASES

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5.1.1.1 Postulated Accident Conditions

The Reactor Building is to provide biological shielding for normal and accident conditions. It will enclose the reactor and the reactor coolant system and will be designed to insure that an acceptable upper limit of leakage of radioactive material will not be exceeded under the maximum loss-of-coolant accident as described in Section 14, "Safety Analysis."

The accident is based on a double-ended pipe break in the main coolant system and produces pressures and temperatures that are influenced by the safeguard system, heat sinks, and energy sources. This is described in Section 6, "Engineered Safeguards" and Section 14, "Safety Analysis."

5.1.1.2 Energy and Mass Releases

Additional energy and mass will be available for release into the Reactor Building from the following sources:

a. Stored heat in the reactor
b. Reactor decay heat
c. Stored heat in the reactor coolant system
d. Water-Metal reaction
e. Hydrogen combustion
f. Fission products in the core.

The energy released from these sources is discussed in Section 14. The energy contribution from the secondary steam system is not included in the calculations for the Reactor Building design pressure and temperatures. The supports for the reactor coolant system will be designed to withstand the forces associated with a break in the reactor coolant pipe so that a rupture in the secondary system will not be considered to act simultaneously with a reactor coolant pipe break.

5.1.1.3 Contribution of Engineered Safeguards

The contribution of the engineered safeguards is discussed in Section 6. This equipment will be actuated to minimize the accident conditions by removing heat from the Reactor Building and inserting negative reactivity into the reactor.

Engineered Safeguards include:

a. High Pressure Injection Equipment
b. Low Pressure Injection Equipment
c. Core Flooding System
d. Reactor Building Emergency Cooling Equipment
e. Reactor Building Spray System
f. Reactor Building Isolation Equipment.

5.1.2 STRUCTURE DESIGN

5.1.2.1 Design Conditions

The Reactor Building will be a Class I structure as defined in Appendix 5A. The internal free volume will be at least 2,000,000 cubic feet. It will be designed for an internal pressure of 55 psig with a coincident temperature of 281 F at accident conditions and an external pressure differential of 2.5 psi at normal working conditions. Due consideration will be given to the dead load, live load, temperature gradients, and penetrations at accident and working conditions. The functional requirements for the liner are covered in detail in Appendix 5E, "Liner Plate Specification." 0047

The design moments, shears, deflections, membrane forces, and stresses are being developed but have not been completed. For the preliminary structural concept of the Reactor Building refer to Figure 5-1. The design criteria for the Reactor Building is covered in Appendix 5C, "Design Criteria for Concrete Reactor Building." The means for comparing actual response during the structural test to design response is contained in Appendix 5F, "Reactor Building Instrumentation."

5.1.2.2 Design Leakage Rate

The Reactor Building will be designed to limit the leakage rate to 0.25 per cent by weight of air in 24 hours at the design pressure.

5.1.2.3 External Loadings

5.1.2.3.1 External Pressure

The Reactor Building will be designed for an internal pressure of 2.5 psi less than that of the external atmospheric pressure.

5.1.2.3.2 Tornadoes

Structures will be designed to withstand short term tornado loadings including tornado generated missiles where such structures house systems and components whose failure could result in an inability to safely shut down and isolate the reactor. Structures that will be so designed include the following:

a. Reactor Building

b. Control, Relay, and Battery Rooms

c. Designated areas of Auxiliary Building.

The tornado design requirements are:

a. Tangential wind velocity of 300 mph

b. An external vacuum of 3 psig. This is conservative considering that most measured pressure drops are in the order of magnitude of 1.5 psig. 1

c. Missile equivalent to a utility pole 35 ft long, 14 inches in diameter, weighing 50 PCF and traveling at 150 mph

d. Missile equivalent to a one-ton automobile traveling at 150 mph.

Recognition is given to the fact that the steel superstructure of the auxiliary Building and the Spent Fuel Building could be damaged. Class I components and systems will be located below the Operating Floor (top most concrete floor). The Operating Floor will be designed to withstand missiles including the tornado missiles which should constitute a more severe con- 1

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dition than a collapsing steel superstructure. The structural concrete walls and base mat of the Spent Fuel Pit will be designed to resist all tornado loads thereby insuring no excessive loss of water due to structural failure. Therefore there appears to be no reason to classify the superstructure as other than Class II. 1

5.1.2.3.3. Wind

The Reactor Building will be designed to withstand a roof line load of 35 psf and a wind velocity corresponding to a 100 year frequency wind, and will also be analyzed to ensure that it withstands tornado generated winds of 300 mph. (See Sec. 1.3, App. 5B). The application of the wind forces will be in accordance with ASCE Paper No. 3269, "Wind Forces on Structures."

The variation of wind velocity with height will be expressed as:

$$V_z = V_{30} \left[\frac{z}{30} \right] x$$

where V_z = velocity at height z above grade

V_{30} = fastest mile of wind at 30' above grade

z = height above grade

x = exponent for Ekman spiral, ranging from 0.3 to 0.118

For the Crystal River site the above will have the value of:

V_{30} = 156.2 mph (110 mph x 1.42 gust factor)

z = 144 ft.

x = 0.118 (extrapolated)

V_z = 187 mph

The pressure corresponding to the mean velocity will be used:

$$V = \frac{187 + 156.2}{2} = 172 \text{ mph}$$

The dynamic pressure q is given by the formula:

$$q = 1/2 \rho v^2$$

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For air at .07651 lb per cu ft and V in miles per hour:

$$q = .002558V^2 \text{ psf} = .0000177V^2 \text{ psi}$$

or

$$q = 0.524 \text{ psi}$$

The wind pressure distribution in the horizontal plane will be defined by the Fourier series:

$$F(x) = -0.3145 + 0.1568 \cos x + 0.4864 \cos 2x + 0.2158 \cos 3x$$
$$-0.0263 \cos 4x + 0.0071 \cos 5x + 0.0348 \cos 6x$$
$$-0.0044 \cos 7x - 0.0263 \cos 8x - 0.0065 \cos 9x$$

5.1.2.3.4 Groundwater and Floods

The top of the Reactor Building foundation mat will be approximately at Elevation 93 ft. The groundwater elevation will be approximately at Elevation 90 ft. The foundation slab design will take into consideration groundwater pressure. Fluctuations in the groundwater elevation due to flood and normal variations will be considered in designing the walls and foundation slab. Refer to Sections 2.4 and 2.5 for flood and normal water elevations.

5.1.2.3.5 Seismic Conditions

The site seismology and response spectra are described in the Seismology section of Appendix 2I.

The seismic design of the Reactor Building will be based on the response to a maximum horizontal component of ground acceleration of 0.05 g. In addition, the design will be checked to ensure no loss of function based on the response to a maximum horizontal component of ground acceleration of .10 g. The vertical component will be taken as 2/3 of the horizontal component and will be assumed to occur simultaneously with the horizontal component.

5.1.2.4 Codes

The Reactor Building will be designed under the following codes:

a. Southern Standard Building Code, Latest Revision
b. Building Code Requirements for Reinforced Concrete, (latest revision)
c. AISC Manual of Steel Construction
d. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessel, Sections VIII, Unfired Pressure Vessels, Section IX, Welding Qualifications (Applicable Portions).

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For additional design criteria see Appendix 5A, "Structural Design Bases."

5.1.2.5 Drawings

Figures 1-2 through 1-11 and Figure 5-1 are plans and elevations showing principal elevations and dimensions of the Reactor Building.

5.1.2.6 Penetrations

5.1.2.6.1 Piping, Duct, and Electrical Penetrations

Penetrations for process piping, ventilation ducts, fuel transfer, instrumentation lines, and electrical cables will be designed to withstand the following loads:

a. Incident pressure and temperature including response of the Reactor Building shell.
b. Pipe reactions based on thermal flexibility and seismic loads.
c. Expansion of containment shell under test conditions.
d. Pipe thrust loads to ensure the vapor barrier is not breeched due to the rupture of any piping system. 1

All penetrations will be made of material which exhibits by test a transition temperature at least 30 F below the minimum service metal temperature. All piping penetrations will be designed to ensure that the liner is not breeched due to the rupture of any process pipe. The load imposed on the containment shell will be based upon the full plastic moment capability of ultimate strength of the pipe. This load will be considered, provided there is a sufficient moment arm and thrust to potentially develop such a moment. In addition to the foregoing, the pentrations will be designed for those loadings detailed in Item a. through c. of Section 5.1.2.6.1 of the PSAR. Therefore, the piping penetration and its anchorage into the liner will be designed as a stronger element than the piping system.

Typical piping duct, and electrical penetrations are shown in Figure 5-2. All penetrations will be of the double barrier type. Where temperature changes require, the second barrier of piping penetrations will include expansion bellows. The electrical penetrations are welded closures which will have double barriers. All penetrations are designed to provide a captive air space that can be pressurized to the Reactor Building design pressure for leak testing. 1

5.1.2.6.2 Personnel Access Locks

Two personnel access locks are provided, one of which penetrates the dished head of the equipment hatch. Each personnel hatch is a welded steel assembly with double doors equipped with double gaskets to provide an air space that can be pressurized to the Reactor Building design pressure for leak testing. The doors are interlocked to ensure that only one door is opened at a time. Remote indicating lights and annunciators will be provided in the Control Room to indicate if a door is open.

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The personnel locks will be designed and fabricated so as to comply in all respects with the requirements of Section III of the ASME Boiler and Pressure Vessel Code for Class B vessels. Functional requirements and design and quality contr l data are included in Section 5.4 and Appendix 5E.

5.1.2.6.3 Equipment Access Hatch

An equipment hatch with an inside diameter of 22'-4" is provided to enable passage of large equipment and components into the Reactor Building during a plant shutdown. The items brought into the vessel include, in part, the reactor coolant pumps and motors, and the reactor vessel O-rings. 7

5.1.2.7 Missile Protection Features

5.1.2.7.1 Reactor Building

Missile protection for the Reactor Building liner:

a. The building and liner will be protected from loss of function due to damage by such missiles as might be generated in a loss-of-coolant accident for break sizes up to and including the double-ended severance of a main coolant pipe.

b. The engineered safeguard components required to maintain containment integrity and to meet the site criteria of 10 CFR 100 will be protected against loss of function due to damage by the missiles defined below.

During the detailed plant design, the missile protection necessary to meet the above criteria will be developed and implemented using the following considerations:

a. The reactor coolant system will be surrounded by reinforced concrete and steel structures designed to withstand forces associated with double-ended rupture of a main coolant pipe and any missiles that may be generated.

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c. Components of the reactor coolant system will be examined to identify and classify potential missiles according to size, shape, kinetic energy, and driving force for purposes of analyzing their effects.

d. The types of missiles for which missile protection will be provided are:

1. All valve stems up to and including the largest size to be used.

2. All valve bonnets

3. All instrument thimbles

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4. Various sizes of nuts and bolts

5. Reactor vessel head bolts

The missiles within the Reactor Building are relatively small as compared with the mass of the surrounding reinforced concrete structures. The high velocity and small impact area will cause the missiles to penetrate into the protective structures until its energies are dissipated. The analytical procedure in design of missile protection shielding will be divided into three groups: 1

1. The Impact Area
2. The Local Effect
3. The Overall Structural Effect

1. The Impact Area

The depth of penetration will be computed based on an empirical expression contained in the "Design Criteria Structural Theory for Protective Construction," Department of the Navy Bureau of Yards & Docks, April 1951, as mentioned in Section 5.1.2.7 of the PSAR.

The protective reinforced concrete shield must be of a thickness not less than three times the depth of penetration.

2. The Local Effect

Ref. a: "Structural Design for Dynamic Loads," C. H. Norris et al., McGraw Hill

Ref. b: "Plastic and Elastic Design of Slabs and Plates," R. H. Wood, Ronald.

The collapse mechanism that will be investigated has the form of a circular fan, Ref. b, p 29. Energy absorbed by the protective structure, according to the Yield Line Theory, will be based on an ideal condition of plastic impact, Ref. a, p. 128. Mass ratio between the missile and part of the structure that will be considered effective in absorbing the energy during the impact is assumed equal to 0.1. Based on the above condition, the kinetic energy to be absorbed by response of the collapse mechanism can be computed as $W_e = W_2/2g(V_2')^2$, Ref. a, p. 128. The work dissipated in the fan is $W_i 4M$, Ref. b, Eq. 14, p. 28. Then, by equating $W_e = W_i$, the required minimum ultimate moment capacity of the structure, according to ACI 318-63, Chapter 16, can be computed as $M = W_e/4$.

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3. The Overall Structural Effect

The protective structure will be investigated for a concentrated load applied at the point of missile impact. In determining the collapse mechanism according to the Yield Line Theory, the kinetic energy to be absorbed by the structure, will be as computed above. The internal strain energy dissipated by the collapse mechanism will again be equated such: $W_e = W_i$, from which the ultimate moment capacity M can be solved.

5.1.2.7.2 Main Steam Turbine Missiles

The turbine-generator supplier has made a study of potential missiles resulting from the failure of turbine-generator rotating elements operating at design (115 of normal operating) speed or, due to failure of components that control admission of steam to the turbine, operating at excessive overspeed. No Westinghouse turbine-generator has ever experienced a rotor or disc failure. Historical reliability plus continued improvements in design indicate that excessive overspeed is the only envisaged cause of turbine-generator failure. An overspeed condition is very unlikely because of the redundancy and reliability of the turbine control and protection system and of the steam system. Nevertheless, the consequences of a turbine-generator runaway caused by all the steam admission valves stuck fully open upon a full load rejection have been analyzed.

On the low pressure turbine, an analysis was made of the bursting speed of each disc based upon an ultimate tensile strength 20 per cent greater than the minimum guaranteed tensile strength. Each alloy steel disc is accurately machined and shrunk on to an alloy steel rotor. After shrinking on all the discs, the blading is installed, and the finished assembly is balanced to close tolerances. The completed rotor is tested at operating temperature and overspeed. Based upon experience and tests, it is concluded that the mode of failure of a disc, should it occur, is a rupture in two or four parts. The maximum speed at which the unit might run with no disc failure is 175 per cent of nominal speed. At this speed the first discs will rupture. Immediately following the rupture of the first discs, the steam flow between the blades of the remaining discs would be significantly reduced, the turbine-generator would slow down, and further disc failures would not be anticipated. A disc rupture in the low pressure turbine would result in a missile which would strike and deeply deform the inner cylinder No. 1 causing some deformation of the inner cylinder No. 2 and of the outer cylinder but will be contained within the unit. Therefore, no outside missile is anticipated to be generated from the low pressure turbine. Refer to Figure 5-3 for identification of the internals of the turbine.

On the high pressure turbine the maximum theoretical speed at which the unit may run, based on the admission steam thermodynamic properties and blade geometry, is 205 per cent of normal speed. As shown here before, the maximum speed at which the unit may run, based upon the stress analysis of the low pressure unit, is 175 per cent of nominal speed. The minimum bursting speed of the spindle, based o the minimum guaranteed mechanical properties of the spindle material, is 270 per cent of nominal speed. Therefore, it is concluded that no missiles should be generated by the high pressure turbine due to a unit runaway.

A more complete description of the results of the analysis of potential turbine missiles is contained in a report titled Consideration of the Consequences of a Turbine Generator Failure in Which Missiles are Generated, included as Appendix 5G. 1

5.1.2.8 Corrosion Protection

The containment structure is protected against external corrosive influences by the following means:

1. A water-proof tendon access gallery at the base of the cylinder.

2. A retaining wall around the vessel which precludes contact of ground water with the shell.

3. A cover of concrete in excess of that for normal construction as examplified by code requirements.

4. Use of a water-tight steel conduit for tendons with added precaution of a thicker walled, rigid conduit in the base mat and extending immediately above into the cylinder.

5. An inboard-oriented haunch which results in only nominal tensile stresses of the outer fibers; these stresses are within the capacity of concrete in tension.

The Reactor Building will be surrounded by a circular retaining wall extending from top of foundation to elevation 118'-6". The retaining wall is to provide access space for stressing and inspecting tendons. At the base of the retaining wall between the wall and the Reactor Building, a drainage system will be provided to keep water from accumulating in the access space. Therefore, waterproofing of the lower portion of the cylinder is not required. There will be no waterproofing nor membrane used on the base slab.

The retaining wall and drainage system around the Reactor Building provides excellent protection for the liner and tendons against ground corrosion and therefore no cathodic protection system will be provided. All metallic components including the liner plate and tendon conduit will be electrically connected to prevent stray current corrosion. The tendons will be enclosed in a metallic tube so as to isolate them from outside electrical influences.

The concrete cover provisions for reinforcing steel and prestressing will be as shown in Table 5-2. The minimum cover specified by ACI 318 is also tabulated for comparison.

The exposed surface of the liner will be given a protective coating of Carbozinc 11 Gray, as manufactured by the Carboline Company. The outer surface of the steel will be in direct contact with the concrete, which provides adequate corrosion protection because of the alkaline properties of concrete.

The prestressing wire will be kept dry and stored in a dry place at the fabricator's facility. During fabrication the wire will be inspected to ensure that the surface is free from any pitting visible to the naked eye. After fabrication, each wire will be protected with a coating of NO-OX-1D "CM" and enclosed in a container. The tendons will be protected from damage during handling and shipping. 1

The tendons will be inserted in steel conduit embedded in the concrete, which will provide an additional water barrier as well as an electrical shield against stray currents. The inner surface of the steel conduit, as well as the anchorages, will be protected with NO-OX-1D, "CM" Casing Filler, as manufactured by the Dearborn Chemical Division of W. R. Grace and Company. This material is composed essentially of a selected paraffin-base refined mineral oil, blended with a microcrystalline petroleum-derived base (petrolatum) which has a definite melting point and penetration range. Additives consisting of lanolin and sodium petroleum sulphonates are incorporated as water-displacing, surface-active agents and corrosion inhibitors.

The proportion of oil to microcrystalline wax in the formulation is adjusted to give a pour or gelling point within the range of 110 to 120° F. The oil and wax are highly refined long-chain, saturated, parraffins, petroleum derivatives, resistant to oxidation and chemical or physical degredation within the temperature ranges to which they will be exposed in this service. The lanolin is a polar substance which enhances inhibitor performance and wetting of the metal surface by the microwax blend. The petroleum sulphonate is a surface-active, water displacing corrosion inhibitor of long tested merit. The physical properties and tests performed on this material are more completely described in the Fourth Supplement to the Preliminary Facility Description and Safety Analysis Report for Brookwood Nuclear Station Unit No. 1 (Docket No. 50-244). The end anchorages will be covered with a metal container and provision made to control the humidity to a safe limit so that the temperature of the air within the enclosure is maintained above the dew point for all operating conditions. The tendon end anchorages will be protected with a steel container, which may receive a coating. These containers or caps, will be readily accessible, and since they must be removed for anchorage inspection, the frequency of such inspection will be as stated in section 5.6.2.2. 3

(GPU is sponsoring continued research and testing on the use of grout as protection for tendons in which GAI is participating. However, it is not anticipated that this program will be completed in sufficient time to warrant consideration for Crystal River.)

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(Deleted) 1

5.1.2.9 Insulation

The Reactor Building liner plate will not be insulated. Insulation and/or cooling coils will be provided when required for hot line penetrations through the Reactor Building so as to limit concrete temperatures during normal operation to no greater than 150 F.

5.1.2.10 Shielding

The Reactor Building completely encloses the reactor coolant system and is designed to contain all radioactive material which might be released following a loss of coolant accident. The building provides adequate biological shielding for accident and normal operating conditions. The interior walls provide adequate shielding for limited access during normal operating conditions.

5.2 ISOLATION SYSTEM

5.2.1 DESIGN BASES

The general design basis governing isolation valve requirements is:

> Leakage through all fluid penetrations not serving accident-consequence-limiting systems is to be minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss of isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the Reactor Building, and various types of isolation valves.

Reactor Building isolation occurs on a signal of approximately 4 psig in the Reactor Building. Valves that isolate penetrations that are directly open to the Reactor Building, such as the Reactor Building purge valves, will also be closed on a high radiation signal. 1

The isolation system closes all fluid penetrations not required for operation of the engineered safeguards in order to prevent leakage of radioactive materials to the environment. Fluid penetrations serving engineered safeguards also meet this design basis.

All remotely operated Reactor Building isolation valves are provided with position limit indicators in the control room.

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5.2.2 SYSTEM DESIGN

The fluid penetrations that require isolation after an accident may be classed as follows:

Type I. Each line connecting directly to the reactor coolant system has two Reactor Building isolation valves. One valve is external, and the other is internal to the Reactor Building. These valves may be either a check valve and a remotely operated valve or two remotely operated valves depending on the direction of normal flow.

Type II. Each line connecting directly to the Reactor Building atmosphere has two isolation valves. At least one valve is external and the other may be internal or external to the Reactor Building. These valves may be either a check valve and a remotely operated valve or two remotely operated valves depending on the direction of normal flow.

Type III. Each line not directly connected to the reactor coolant system or not open to the Reactor Building atmosphere has one valve, either a check valve or a remotely operated valve. This valve is located external to the Reactor Building.

Type IV. Lines that penetrate the Reactor Building and are connected to either the building or the reactor coolant system, but which are never opened during reactor operation, have provisions for locking in a closed position.

There are additional subdivisions in each of these major groups. The individual system flow diagrams show the manner in which each Reactor Building isolation valve arrangement fits into its respective system. For convenience, each different valve arrangement is shown in Table 5-1 and Figure 5-5 of this section. The symbols on this figure are identified on Figure 9-1. This table lists the mode of actuation, the type of valve, its normal position, and its position under Reactor Building isolation conditions. The specific system penetrations to which each of these arrangements is applied are also presented. Each valve will be tested periodically during normal operation or during shutdown conditions to insure its operability when needed.

The accident analysis for failure or malfunction of each valve is presented with the respective system evaluation of which that valve is a part, for example chemical addition and sampling system, etc.

There is sufficient redundancy in the instrumentation circuits of the safeguards actuation system to minimize the possibility of inadvertent tripping of the isolation system. This redundancy and the instrumentation signals that trip the isolation system are discussed further in Section 7.

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The system abbreviations in column three of Table 5-1 are defined as follows:

MU Makeup and Purification System
DH Decay Heat Removal System
RB Reactor Building Cooling System
SF Spent Fuel Cooling System
WD Waste Disposal System
CA Chemical Addition and Sampling System
BS Reactor Building Spray System
IC Intermediate Cooling System
SW Nuclear Services Cooling Water System
RW Raw Sea Water
SP Secondary Plant

5.3 VENTILATION SYSTEM

5.3.1 DESIGN BASES

5.3.1.1 Governing Conditions

The Reactor Building ventilation system performs the following functions:

a. Removes or adds sensible heat under normal conditions of operation to maintain the building at or below a predetermined maximum temperature and at or above predetermined minimum temperature.

b. Removes sensible and latent heat under emergency accident conditions to maintain the building at or below predetermined maximum values of temperature and pressure.

c. Filters all air.

d. Filters contaminated air through charcoal filters before discharging it to atmosphere.

e. Purges the Reactor Building with outside air whenever desired.

5.3.1.2 System Performance

To provide normal cooling in the Reactor Building, the ventilation system will be sized to control the interior air temperature to 110 F maximum in accessible areas during operation and 60 F minimum during shutdown. Air from the system will be distributed over and around all heat producing equipment. The anticipated total normal cooling load is 4.3×10^6 Btu/hr. [1]

To provide for emergency accident cooling in the Reactor Building, the ventilation system will be additionally selected and sized to meet the abnormal conditions of temperature, pressure, and latent heat which will be imposed upon it is such an event. It is anticipated that these conditions will be 281 F, 55 psig, and 240×10^6 Btu/hr.

0059

To provide a flow of outside air through the Reactor Building, the purge components of the ventilation system will be selected and sized for a purge rate of 50,000 cfm. All air entering the Reactor Building will be filtered and heated as required. All air leaving will be passed through HEPA and charcoal filters before being discharged to the atmosphere through the vent.

The cooling which is required during normal operation will be obtained from water in a closed circuit which is cooled in an evaporative-type industrial cooler to a temperature of 85 F maximum.

The cooling which is required during an emergency condition will be obtained from water from the nuclear services cooling water system which will be available at a design temperature of 95 F.

The normal cooling water circuit will be entirely independent of the emergency cooling water circuit.

5.3.2 SYSTEM DESIGN

Figure 5-6 is a flow diagram of the ventilation system for the Reactor Building showing the cooling and purging components.

The cooling components consist of three fan assemblies, each of which will contain a normal coil, emergency coil, filter, direct-driven fan, and suitable casing. To insure adequate air distribution to the major heat loads, auxiliary fans and ducts will be employed. | 1

The purge supply air component consists of two fan assemblies each of which will contain a dust filter, heating coil, centrifugal fan, and casing. Each assembly will handle one half the total air quantity (25,000 cfm each). | 1

The purge exhaust air component consists of two fan assemblies each of which will contain a HEPA filter, a charcoal filter, fan, and casing. Each assembly will handle one half the total air quantity (25,000 cfm each). | 1

All of the purge components of the ventilation system, except the interior ducts and isolation valves as shown on Figure 5-5, will be located outside the Reactor Building.

The purge discharge to the plant vent will be monitored and alarmed to prevent radioactive release exceeding acceptable limits. | 1

5.3.2.1 Isolation Valves

As the cooling component of the ventilation system is contained completely within the Reactor Building, it will not include provisions for any isolation valves other than on cooling water lines.

0060

The purge component of the ventilation system will be provided with double automatic isolation valves (or dampers) in both the supply and discharge ducts. These valves will be normally closed and will be opened only for the purging operation. They will be electrically actuated inside the Reactor Building and pneumatically actuated outside.

The isolation signal and controls are discussed in section 5.2. The closure times and sequence will be developed during detailed design and safety analyses. Operability testing of the isolation valves is accomplished each time the purge system is put into operation.

5.4 LEAKAGE MONITORING SYSTEM

Provision for monitoring of leakage through the penetrations and penetration sleeve to liner welds will be provided.

The normal barrier to leakage from the Reactor Building will be a steel liner 3/8 in. thick in the cylinder and dome and 1/4 in. thick in the base. All penetrations through the containment shell for pipes, ducts, electrical conductors, and access will be welded to the liner plate before the concrete is placed. Insofar as possible, the penetrations and reinforcing plates will be shop welded assemblies. All penetrations in the Reactor Building will be of the double barrier type and will be equipped for leakage monitoring with means provided to isolate and locate a leaking penetration. All penetrations, as shown typically in Figure 5-2, will be designed, fabricated, and tested so as to ensure leak tightness. All penetration sleeve to liner plate welds will be covered by a test channel to permit verification of leak tightness.

The liner plate will be anchored to the concrete shell so as to ensure elastic stability under all loading conditions and composite action between the liner and the concrete. Elastic stability will be ascertained by analyzing the liner as a flat plate between supports subjected to the biaxial stresses. Should liner stresses under the factored load conditions exceed yielding in compression, the analysis will assume plastic behavior at a stress of 1.2 times the minimum guaranteed yield stress. The Reactor Building will be tested for leak tightness under conditions more severe for possible liner leakage than under the conditions of the hypothetical accident and other simultaneously occurring loads. Surveillance of the quality of all materials and workmanship will be maintained so as to ensure that the containment shell including the liner, penetrations, and reinforced concrete does meet the intent of the design in all respects.

The details of a program to demonstrate the integrity of the structure and the leak tightness of the vapor barrier are described in Section 5.6 and summarized as follows:

a. The test to demonstrate structural integrity will be at 115 per cent of design pressure. This test is established to duplicate, insofar as possible with a pressure load only, the incident loads plus the most severe effects of the design earthquake.

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b. The integrated leak rate test will be at design pressure and will demonstrate that the leak rate is no greater than 0.25 per cent by weight of the contained volume of air in 24 hours at the design pressure. The conditions for this test will impose greater tensile or lower compressive membrane stresses on the liner than would exist under accident conditions, thereby ensuring that the test is performed under the most severe condition insofar as leakage is concerned. The pressure load in combination with the thermal load results in the lesser probability of leakage.

As described in Section 5.1.2.6, the personnel access hatches will be interlocked so as to ensure that one barrier will always remain intact, and that indicators and annunciators will be provided in the Control Room to indicate when a door is open. The permanent equipment access hatch cannot be opened except by extensive deliberate action which cannot be taken except at plant shut down.

All liner seams will be tested individually to demonstrate their leak tightness. Where the liner will be inaccessible for future inspection (i.e., where it will be totally encased in concrete), test channels will be provided on all weld seams. These channels will ensure leak tightness of the plate to plate weld and the channel to plate weld and will be segmented so as to permit pressurization at specified locations. On liner plate weld seams where test channels are not provided, testing will consist of the use of soap film and a vacuum box. The penetration sleeve to liner plate welds are also covered with test channels to permit testing similar to that for inaccessible weld seams. Each shop assembled unit including single and multiple penetration units will be shop tested to demonstrate leak tightness. More complete details of liner installation and testing procedures on the liner and penetrations are included in Appendix 5E.

Where the liner abuts concrete, it will be in an environment which will minimize any possible corrosion. Where the liner is exposed, it will be protected with a coating similar to Carbo Zinc as manufactured by the Carboline Corporation. The concrete shell will protect the liner from weather influences and potential externally generated missiles and insulate the liner plate from low temperatures. The liner surface on the interior of the vessel will be protected from internally generated missiles by concrete and in instances a combination of concrete and steel shields. 1

Because the vapor barrier is protected in this manner, and once the adequacy of the liner has been initially established, there is no reason to anticipate deterioration of the liner, which would jeopardize its effectiveness as a vapor barrier. Nevertheless, periodic inspections will be made of the exterior of the Reactor Building and of interior spaces which are accessible during full power operation.

A significant number of steel lined, reinforced concrete containment vessels are being constructed presently: these will provide a considerable background of experience prior to the construction of this vessel. Full advantage of this knowledge will be taken in all phases of design, fabrication, installation, inspection, testing, and operation.

5.5 SYSTEM DESIGN EVALUATION

The Reactor Building, including extensions of the containment boundary as described in Sections 5.2 and 5.3, will, with the functioning of the additional engineered safeguards, prevent an uncontrolled release of radioactivity to either the plant site or the surrounding areas during normal operation or any accident conditions up to the most severe hypothetical accident.

5.6 TESTS AND INSPECTION

5.6.1 PREOPERATIONAL TESTING AND INSPECTION

5.6.1.1 During Construction

Test, code, and cleanliness requirements will accompany each specification or purchase order for materials and equipment. Hydrostatic, leak, metallurgical, electrical, and other tests to be performed by the supplying manufacturers will be enumerated in the specifications together with the requirements, if any, for test witnessing by an inspector. Fabrication and cleanliness standards, including final cleaning and sealing, will also be described together with shipping procedures. Standards and tests will be specified in accordance with applicable regulations, recognized technical society codes, and current industrial practices.

Inspection will be performed in the shops of vendors and subcontractors as necessary to verify compliance with specifications.

5.6.1.1.1 Concrete

Testing of concrete materials and concrete as placed is described in Appendix 5D, "Quality Control." An experienced full-time concrete inspector will continuously check concrete batching and placing operations.

5.6.1.1.2 Prestressing

Testing and inspection of all prestressing materials and special installation equipment is described in Appendix 5D, "Quality Control." Full-time supervision of the prestressing operations will be provided by an inspector experienced in prestressing as well as by the aforementioned concrete inspector.

5.6.1.1.3 Reinforcing Steel

Testing and inspection of reinforcing steel is described in Appendix 5D, "Quality Control." The concrete inspector will check the condition and placement of the bars in the forms for compliance with drawings and specifications, including welded splices.

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5.6.1.1.4 Liner Plate

Testing and inspection of the liner plate is described in Appendix 5E, "Liner Plate Technical Specifications."

5.6.1.2 Structural Test

Following completion of construction and prior to the initial fuel loading, the Reactor Building will be pressurized at 115 per cent of the design pressure for one hour to establish the structural integrity of the building. The response of the building will be compared with the calculated behavior to confirm the design by means of instrumentation described in Appendix 5F, "Reactor Building Instrumentation."

5.6.1.3 Initial Integrated Leakage Rate Test

The purpose of the integrated leakage rate test is to measure the percentage by weight of air which can leak out of the reactor building per day at the design pressure of 55 psig. The specified design leakage of 0.25 per cent or less per day will be measured using the Absolute Method. During the test, air pressure, water vapor pressure, and air temperature will be measured using high accuracy instruments. The reactor building ventilation system will be used continuously throughout the test to achieve complete air mixing and control of air temperature. Duration of the test will be a minimum of 24 hours.

5.6.2 POSTOPERATIONAL LEAK MONITORING

5.6.2.1 Leakage Monitoring

The penetrations will be periodically pressurized to design pressure to demonstrate their leak tightness. The frequency of testing will be established prior to the submission of the application for the operating license.

5.6.2.2 Tendon Surveillance

The program for future surveillance, to be included as a part of the operating procedures, is under study. Capability will be provided to do the following: 1

a. A periodic visual inspection of tendon anchorages could be made to determine if wire breakage has occurred.

   The tendon is under a stress of approximately 140,000 psi. Should a wire break the released energy will result in the buttonhead noticeably projecting beyond the anchorage. The anchorages could be completely inspected immediately before and after the structural test. Consideration is being given to randomly selecting no less than 10 per cent of the anchorages for inspection six months after the test, and annually thereafter, for a period of five years. Partial inspection could be conducted after this period.

b. The prestress in the tendon could be reconfirmed by a jacking technique similar to that used in the original stressing operation. 1

This phase of the program could provide for obtaining a lift-off reading by using an hydraulic jack to simply lift the anchor head off the ship. This procedure will provide a determination of the stress level in the tendon, and will also confirm previously predicted stress losses including steel relaxation and concrete creep. No less than 1 per cent tendons randomly selected could be so jacked on a biannual basis until losses are stabilized.

c. Removal of a wire, or wire specimen, could be made from a tendon environment at varying time intervals. This wire specimen would undergo inspection and tensile testing if required for evaluation.

A selected number of tendons could include an additional unstressed 7mm diameter wire specimen obtained from a reel represented in the tendon. These specimens which consist of the same material as the tendon and located in the same environment could be removed on an annual basis for examination. For example, four specimens could be removed annually. 7

d. All tendon anchorages will be accessible, and with the unbonded system, the tendons can be removed and replaced. Tendons will be removed for inspection only if a question exists on the well being of the tendons based upon the other surveillance methods.

(Certain aspects of the suggested program are developed for tendons consisting of wires and would have to be modified if strands or bars are used.)

0065

Table 5-1
Reactor Building Isolation Valve Information

Penetration No.	Service	System	Flow Direction	Valve Arrgt.	Location Referred to R.B.	Valve Type	Line Size, in.	Method of Actuation	Signal	Normal Valve Position	Position Indication	Post-Accident Position	
1	Pressurizer and Reactor Coolant Sample Lines	CA	Out	5	Inside	Gate	3/4	EMO*	ES	Closed	Yes	Closed	
						Gate	3/4	EMO	ES	Closed	Yes	Closed	
					Outside	Gate	3/4	Air	ES	Closed	Yes	Closed	
2	Letdown Cooler Cooling Water Supply Line	IC	In	4	Inside	Swing Check	6			Open	No	Closed	1
					Outside	Swing Check	6	--	--	Open	No	Closed	
3	Letdown Cooler Cooling Water Outlet Line	IC	Out	3	Inside	Gate	6	EMO	ES	Open	Yes	Closed	
					Outside	Gate	6	Air	ES	Open	Yes	Closed	
4	Reactor Bldg Sump Drain	WD	Out	3	Inside	Gate	4	EMO	ES	Closed	Yes	Closed	
					Outside	Gate	4	Air	ES	Closed	Yes	Closed	
5	Let Down Line to Purification Demineralizers	MU	Out	5	Inside	Gate	1-1/2	EMO	ES	Open	Yes	Closed	
						Gate	1-1/2	EMO	ES	Closed	Yes	Closed	
					Outside	Gate	1-1/2	Air	ES	Open	Yes	Closed	1
6	Reactor Coolant Pump Seal Return Line	MU	Out	3	Inside	Gate	3	EMO	ES	Open	Yes	Closed	
					Outside	Gate	3	Air	ES	Open	Yes	Closed	
7	Reactor Coolant Pump Seal Water Supply	MU	In	1	Inside	Stop Ck.	1	--	--	Open	No	Closed	
					Outside	Globe	3	Air	ES	Throttled	Yes	Closed**	
						Globe	3	Air	ES	Closed	Yes	Closed**	
8	Normal Makeup to the Reactor Coolant System	MU	In	2	Inside	Swing Check	2½	--	--	--	No	Open	1 2
					Outside	Globe	2	Air	--	Throttled	Yes	Open	
					Outside	Globe	2	Air	--	Closed	Yes	Closed	
9	High Pressure Injection Line	MU	In	7	Inside	Swing Check	2½	--	--	--	No	Open	
					Outside	Gate	2½	Air	ES	Closed	Yes	Open	
9a, 9b, 9c	High Pressure Injection Line	MU	In	7	Inside	Swing Check	2½	--	--	--	No	Open	
					Outside	Gate	2½	Air	ES	Closed	Yes	Open	
10	Fuel Transfer Tubes	SF	In/Out	8	Inside	Special Closure	30	--	--	Closed	--	Closed	
					Outside	Gate	30	Manual	--	Closed	No	Closed	
11	Reactor Bldg Spray Inlet Line	BS	In	7	Inside	Check	8	--	--	--	No	Open	
					Outside	Gate	8	Air	ES	Closed	Yes	Open	

*All valves with electric motor operators are also equipped with hand wheels.

**Postaccident reactor coolant system pressure causes valve to close.

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Table 5-1 (Cont'd)

Penetration No.	Service	System	Flow Direction	Valve Arrgt.	Location Referred to R.B.	Valve Type	Line Size, in.	Method of Actuation	Signal	Normal Valve Position	Position Indication	Post-Accident Position
12	Reactor Bldg Spray Inlet Line	NS	In	7	Inside	Swing Check	8	--	--	Closed	No	Open
					Outside	Gate	8	Air	ES	Closed	Yes	Open
13	Decay Heat Coolant Return	DH	In	7	Inside	Swing Check	12	--	--	Closed	No	Open
					Outside	Gate	12	Air	ES	Closed	Yes	Open
14	Decay Heat Coolant Return	DH	In	7	Inside	Check	12	--	--	Closed	No	Open
					Outside	Gate	12	Air	ES	Closed	Yes	Open
15	Decay Heat Coolant Let-Down	DH	Out	9	Inside	Gate Gate(2)	10 10	Manual EMO	-- Remote Manual	Closed Closed	No Yes	Closed Closed
					Outside	Gate	10	Air	Remote Manual	Closed	Yes	Closed
16	Reactor Bldg Sump Recirculation Lines	DH	Out	10	Outside	Gate	18	Air	Remote Manual	Closed	Yes	Open
17	Reactor Bldg Sump Recirculation Lines	DH	Out	10	Outside	Gate	18	Air	Remote Manual	Closed	Yes	Open
18	Reactor Coolant Drain Tank Vent	WD	Out	15	Outside	Gate	2	Air	ES	Closed	Yes	Closed
19	Reactor Bldg Inlet Purge Line	-	In	11	Inside	Butterfly	42	EMO	ES	Closed	Yes	Closed
					Outside	Butterfly	42	Air	ES	Closed	Yes	Closed
20	Reactor Bldg Outlet Purge Line	-	Out	12	Inside	Butterfly	42	EMO	ES	Closed	Yes	Closed
					Outside	Butterfly	42	Air	ES	Closed	Yes	Closed
21-23 29-31	Reactor Coolant Pump Motors and Lub Oil Coolers	SW	In and Out	13	Outside	Stop Check	6	Manual	-	Open	No	Closed
					Outside	Gate	6	Air	ES	Open	Yes	Closed
24	Reactor Bldg Normal Air Coolers	-	In and Out	13	Outside	Stop Check	8	Manual		Open	No	Closed
					Outside	Gate	8	Air	ES	Open	Yes	Closed
31-33 34-36	Reactor Bldg Emergency Air Coolers	SW	In and Out	6	Outside	Stop Check	8	Manual	-	Closed	No	Open
					Outside	Gate	8	Air	ES	Closed	Yes	Open
37-39	Feedwater and Steam Lines	-	In and Out	14	Outside	Stop Check	18	-	-	Open	No	Closed
		-			Outside	Turbine Stop	24*	Hydr	ES	Open	Yes	Closed
40-42	Feedwater and Steam Lines	-	In and Out	14	Outside	Stop Check	18	-	-	Open	No	Closed
		-			Outside	Turbine Stop	24*	Hydr	ES	Open	Yes	Closed

*Approximate

0067

Table 5-1 (Cont'd)

Penetration No.	Service	System	Flow Direction	Valve Arrgt.	Location Referred to R.B.	Valve Type	Line Size, in.	Method of Actuation	Signal	Normal Valve Position	Position Indication	Post-Accident Position
43	Steam Generator Sample Line	CA	Out	5	Inside	Gate	3/4	EMO	Remote Manual	Closed	Yes	Closed
					Outside	Gate	3/4	Air	ES	Closed	Yes	Closed
44,45	Core Flooding Tank Nitrogen Supply Line	CA	In	4	Inside	Check Valve	3/4	--	--	Closed	No	Closed
					Outside	Check Valve	3/4	--	--	Closed	No	Closed
46	Reactor Coolant System Drain Line	WD	Out	3	Inside	Gate	2-1/2	EMO	ES	Closed	Yes	Closed
					Outside	Gate	2-1/2	Air	ES	Closed	Yes	Closed
					Outside	Gate	3/4	Solenoid	--	Closed	Yes	Closed
47-49	R.B. Fan Motor Cooling Lines	SW	In	6	Outside	Gate	1½	Manual	--	Open	No	Open
50-52	R.B. Fan Motor Cooling Lines	SW	Out	6	Outside	Gate	1½	Manual	--	Open	No	Open

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TABLE 5-1 (Cont'd)

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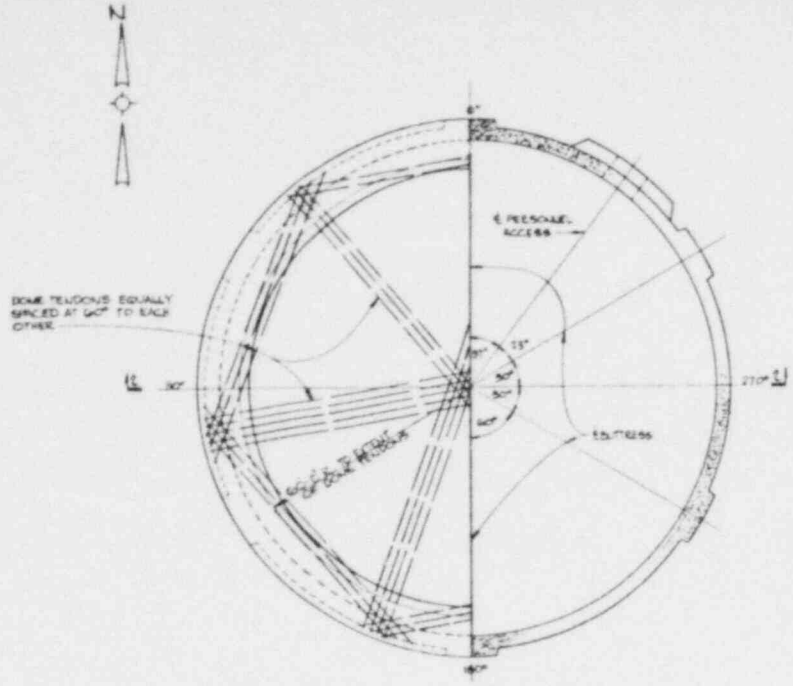
TABLE 5-2

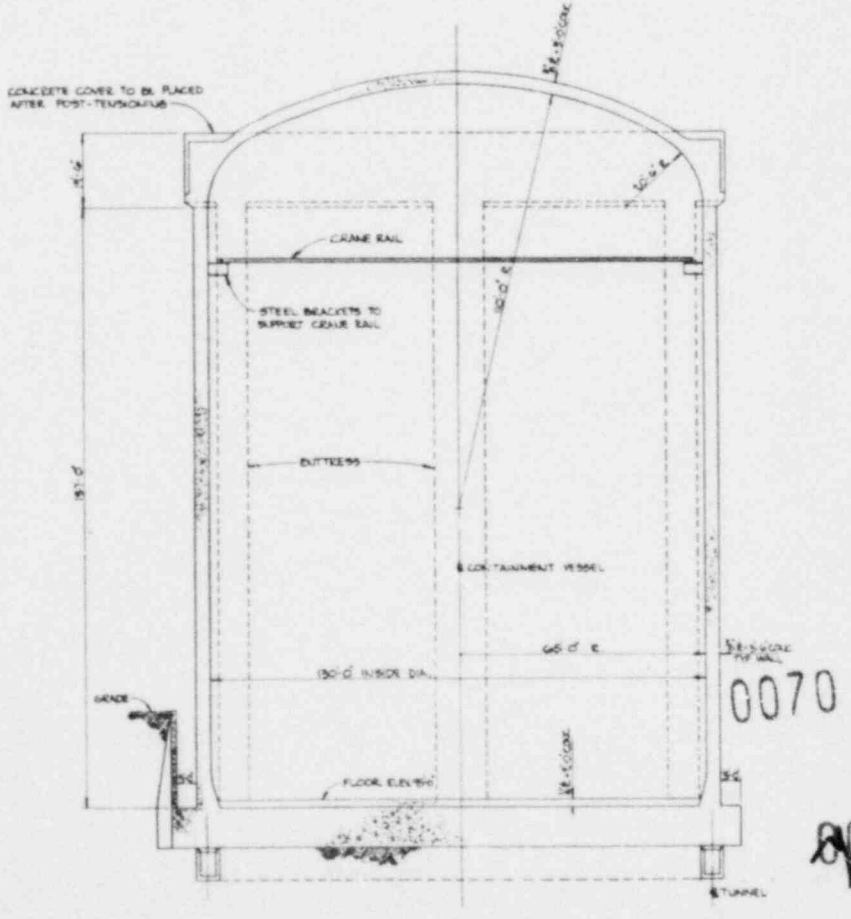
CONCRETE COVER PROVISIONS FOR
REINFORCING STEEL AND PRESTRESSING

LOCATION	TYPE OF STEEL	MINIMUM COVER	
		TO BE USED	BY ACI-318
DOME	REINFORCING	18 & 14S - 2¼ IN. OTHERS - 2¼ IN.	2¼ & 1¾ IN. 1½ IN.
	PRESTRESSING	6 IN.	1½ IN.
CYLINDER	REINFORCING	18 & 14S - 2¼ IN. OTHERS - 2¼ IN.	2¼ & 1¾ IN. 1½ IN.
	PRESTRESSING	6 IN.	1½ IN.
BASE MAT	BOTTOM REINFORCING	18 & 14S - 3 IN. OTHERS - 3 IN.	3 IN. 3 IN.
	TOP REINFORCING	18 & 14S - 2¼ IN. OTHERS - 2¼ IN.	2¼ & 1¾ IN. 1½ IN.

0069



ROOF PLAN FLOOR PLAN



SECTION 3-3

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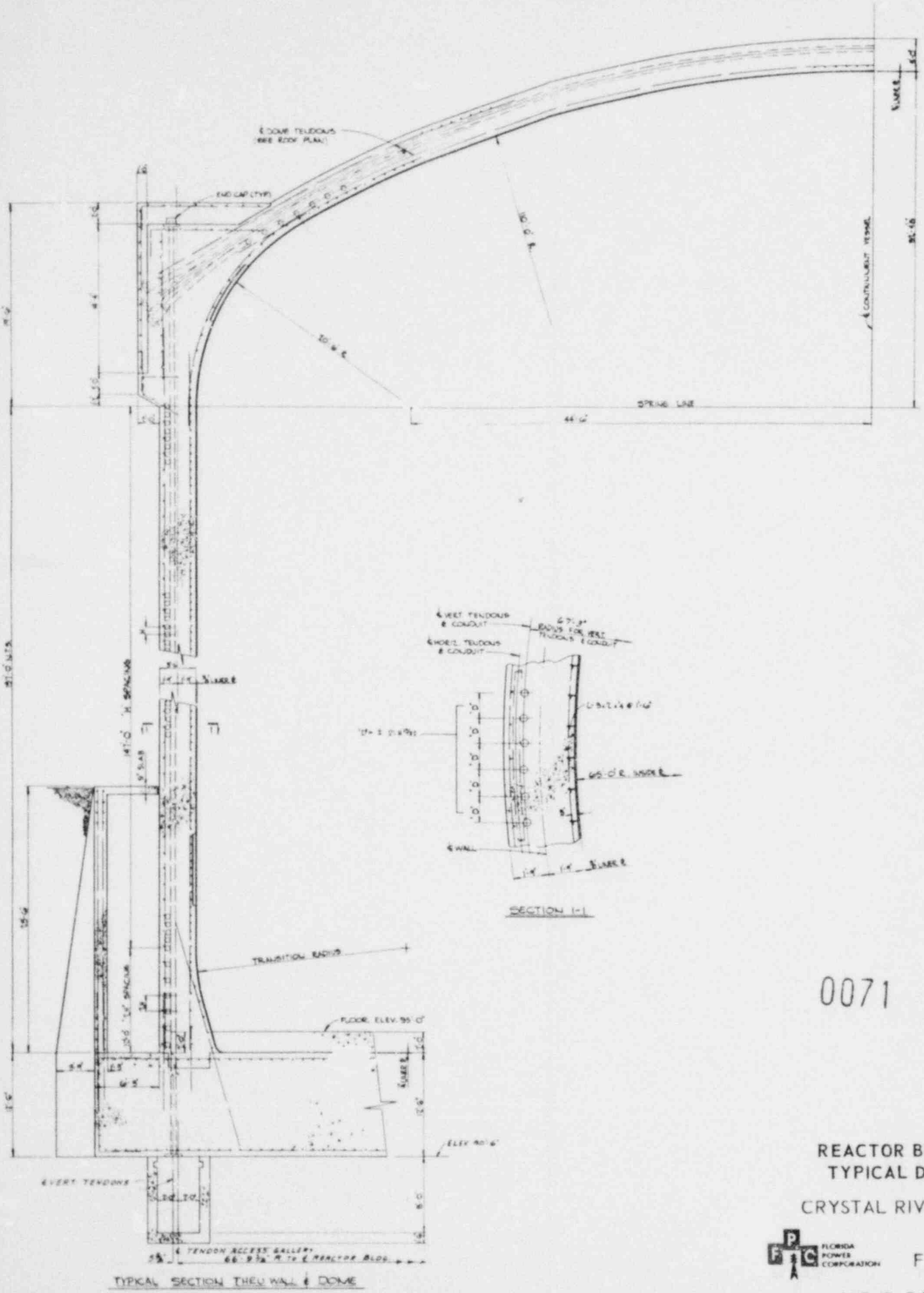


0071

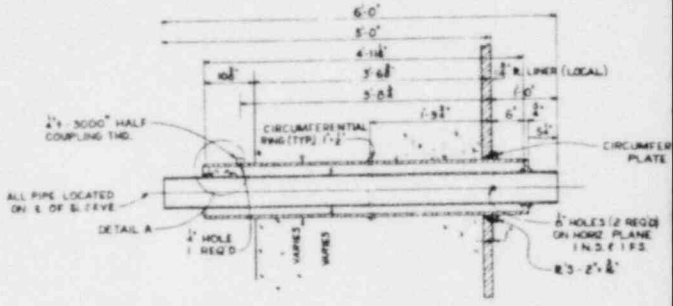
REACTOR BUILDING
TYPICAL DETAILS

CRYSTAL RIVER UNIT 3

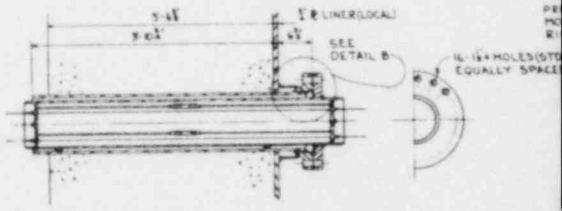
FLORIDA POWER CORPORATION

FIGURE 5-1

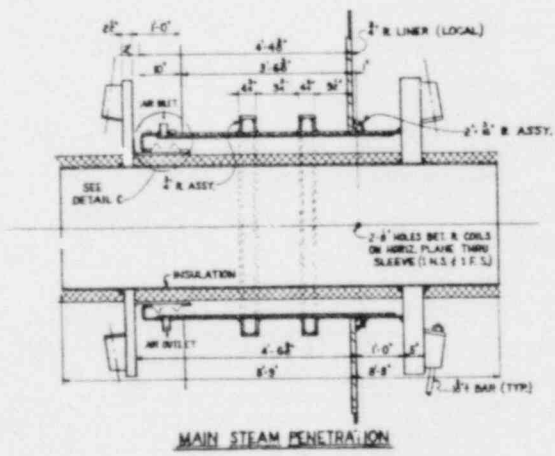
AMEND. 7 (7-15-69)



COLD MECHANICAL PENETRATION

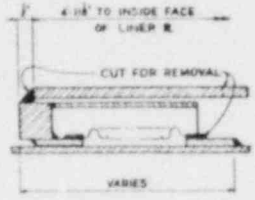


ELECTRICAL PENETRATION

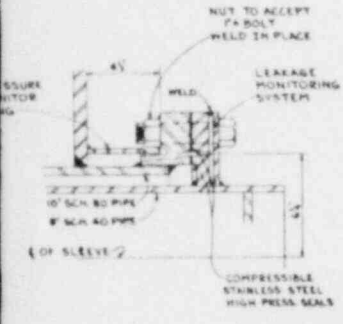


MAIN STEAM PENETRATION

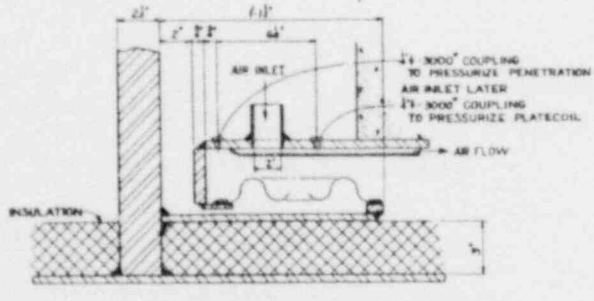
0072



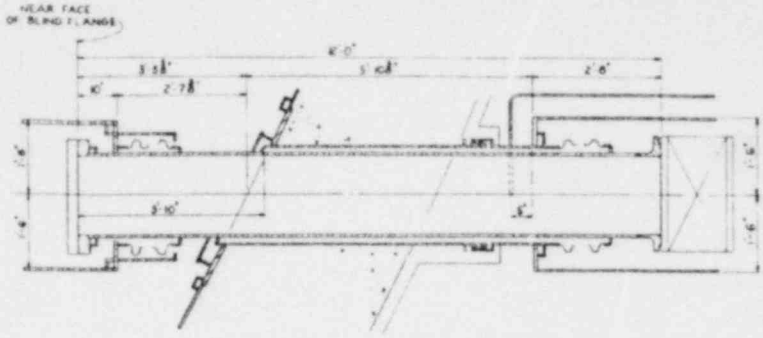
DETAIL A



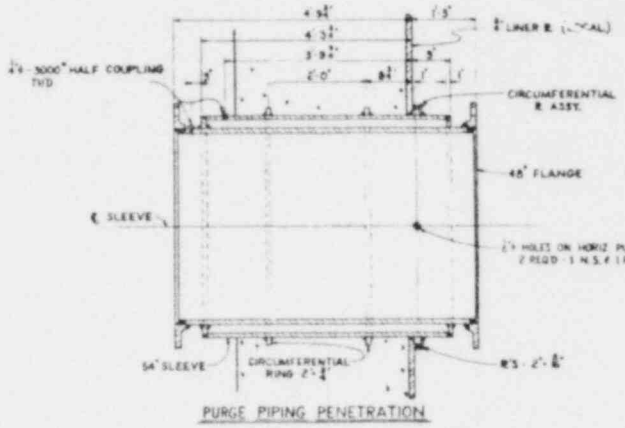
DETAIL B



DETAIL C



FUEL TRANSFER TUBE PENETRATION



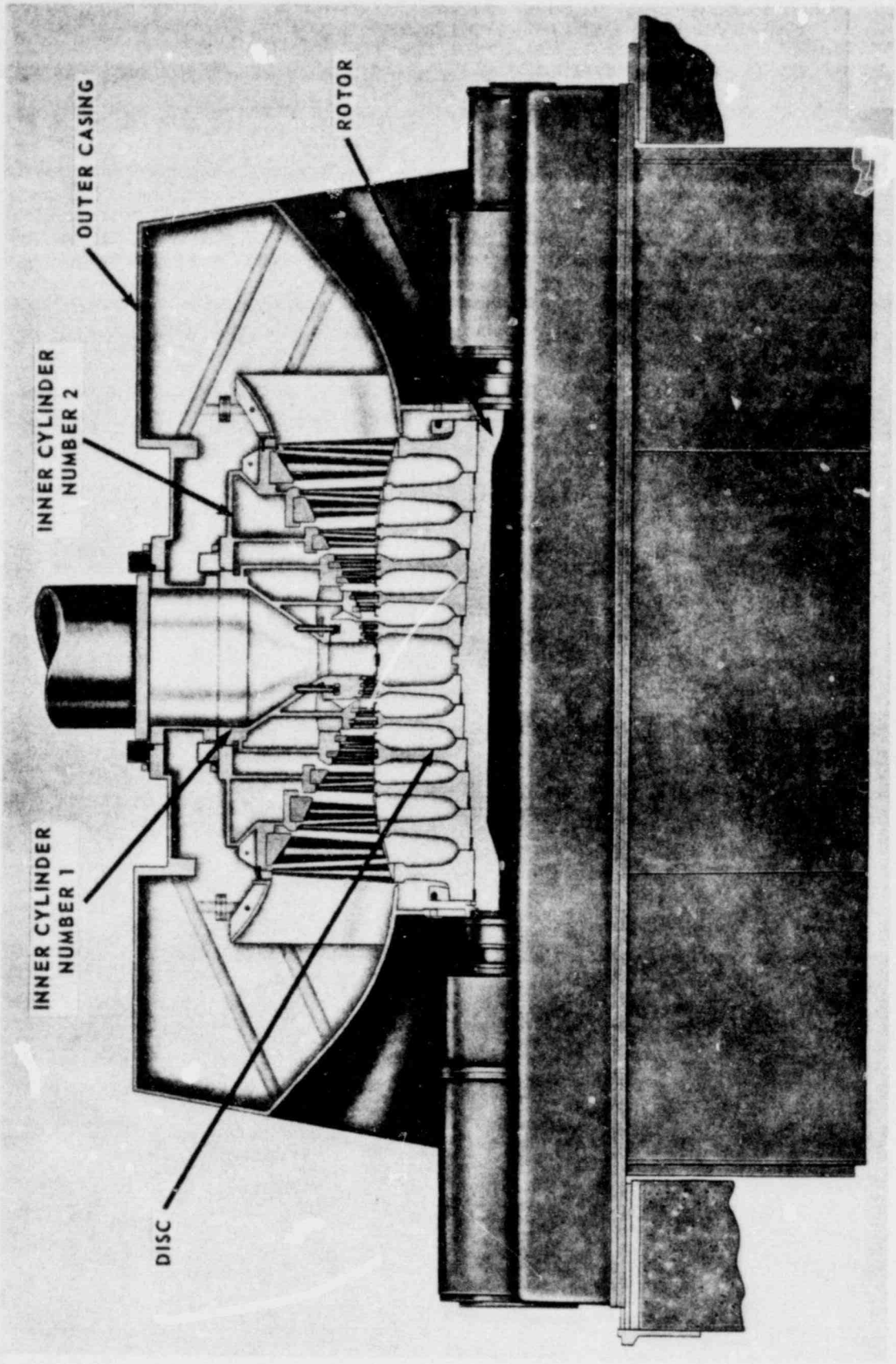
PURGE PIPING PENETRATION

0073

TYPICAL ELECTRICAL
AND PIPING PENETRATIONS
CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 5-2



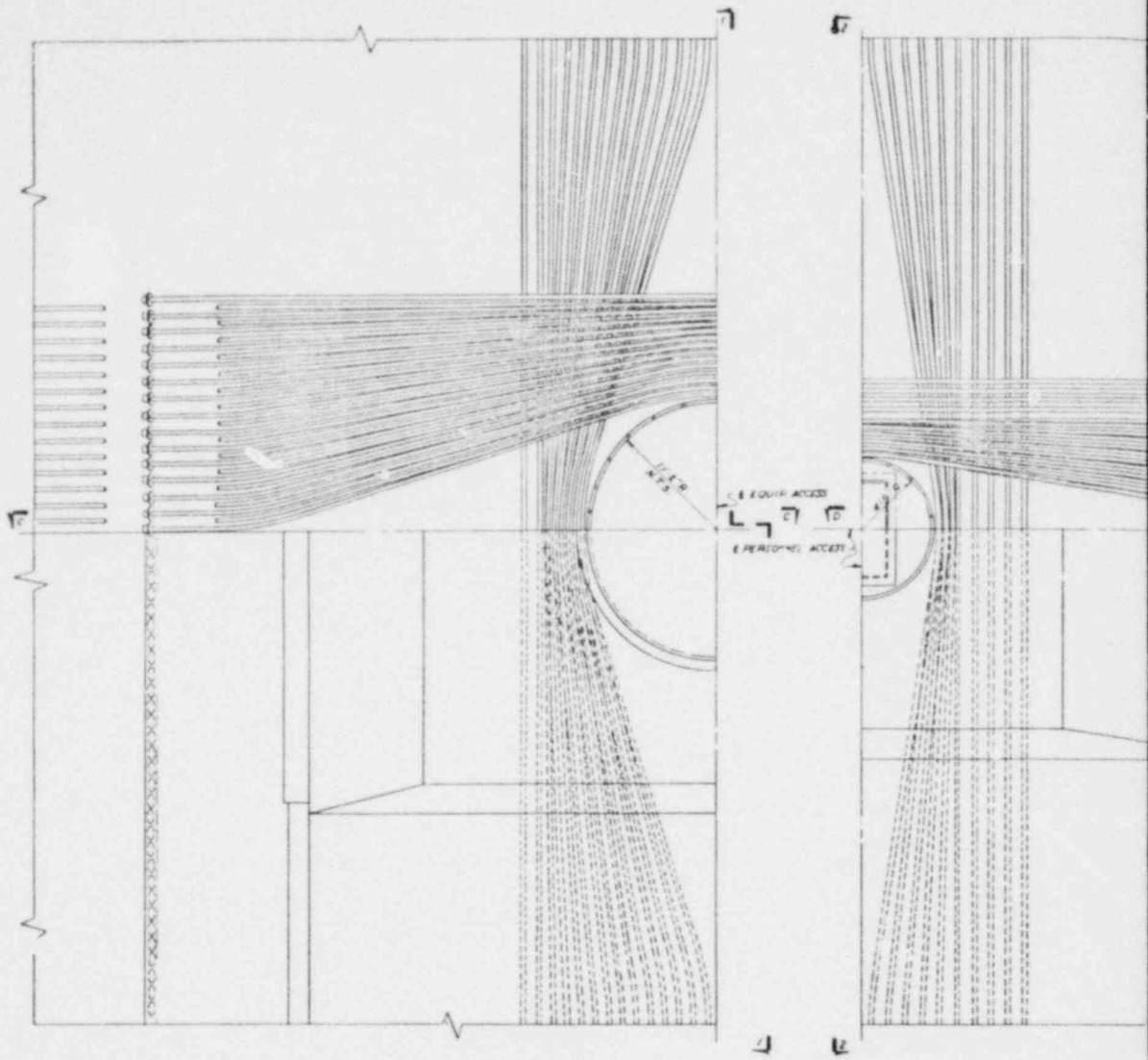
0074

LOW-PRESSURE TURBINE CROSS-SECTION
1800-RPM DOUBLE-FLOW DESIGN

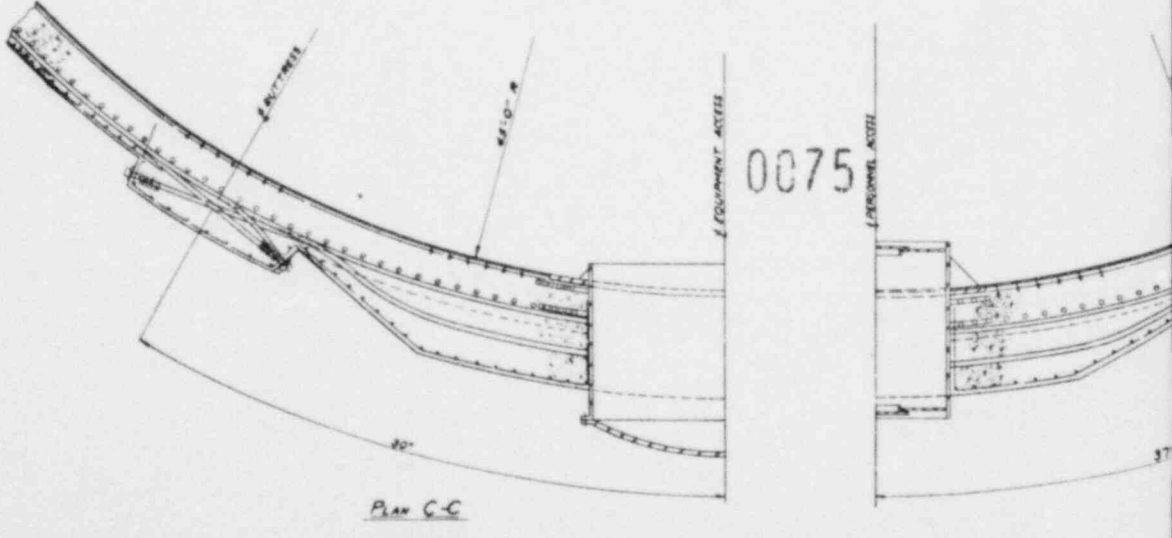
CRYSTAL RIVER UNITS 3 & 4

FPC FLORIDA POWER CORPORATION

FIGURE 5-3

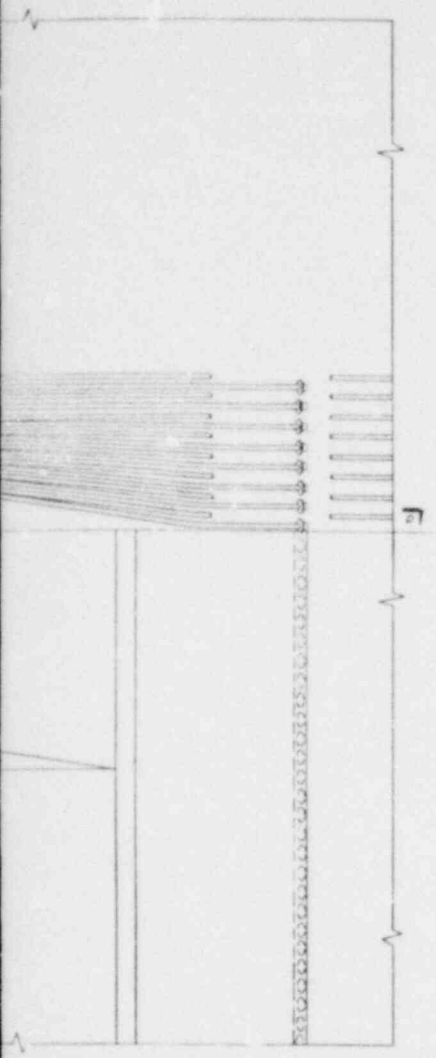


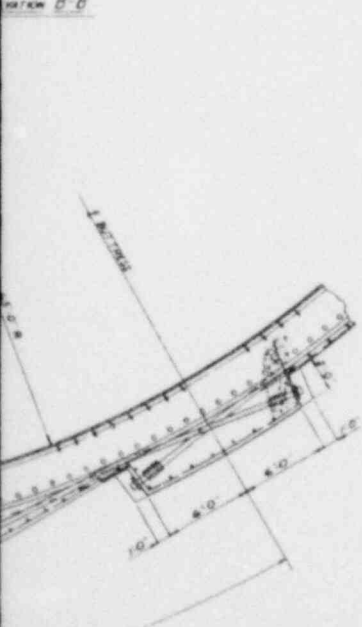
ELEVATION A-A

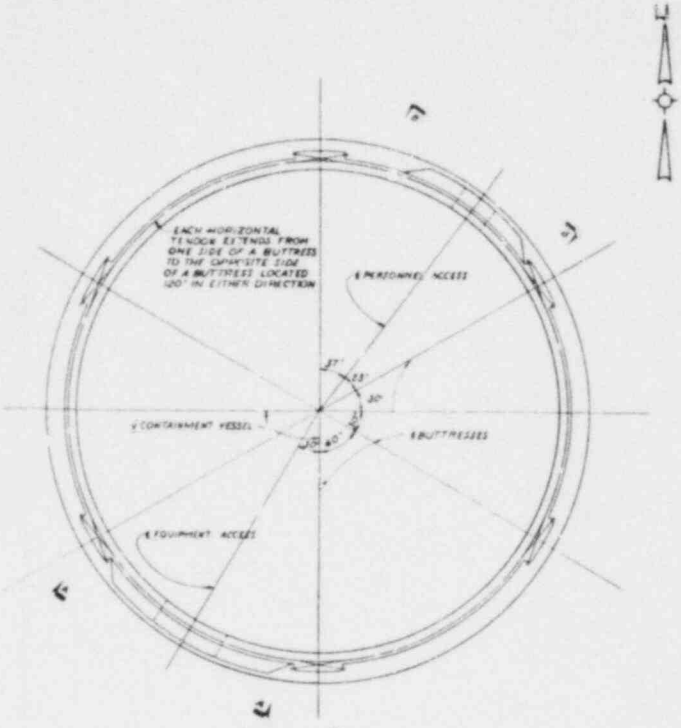


PLAN C-C

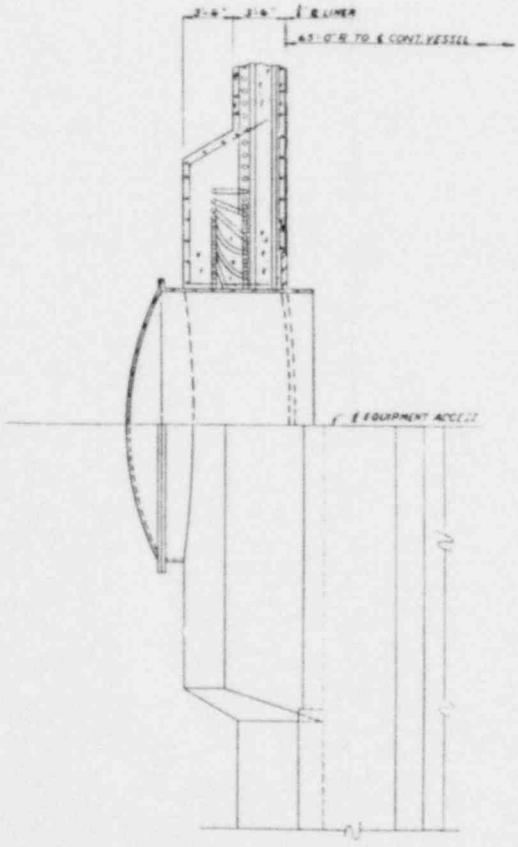
0075



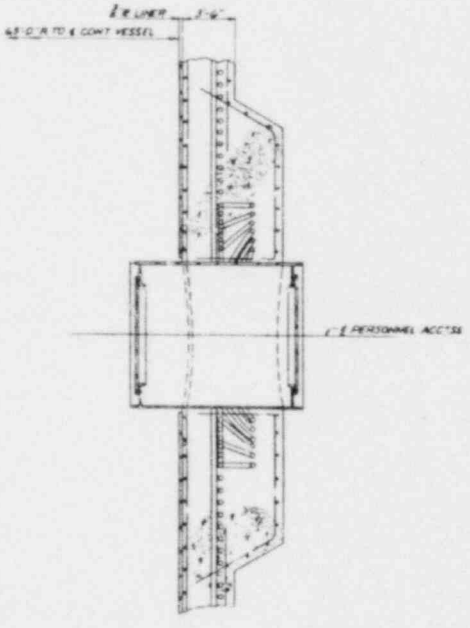




LOCATION PLAN



SECTION 1-1



SECTION 2-2

0076

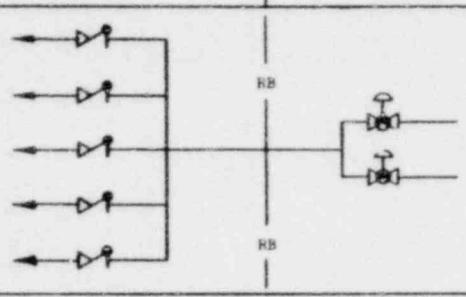
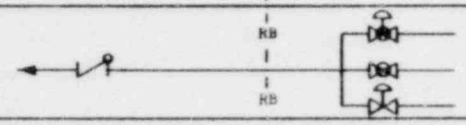
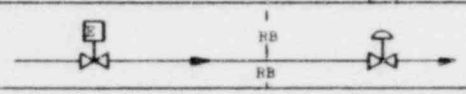
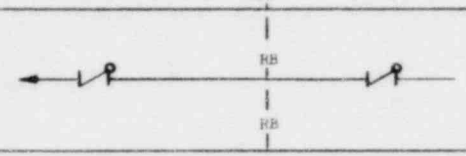
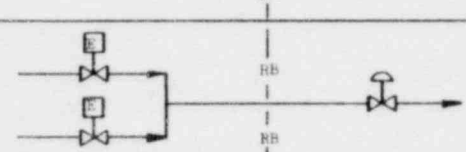
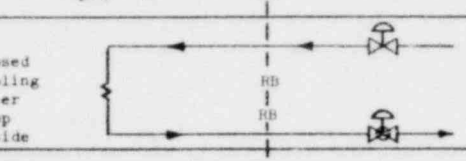
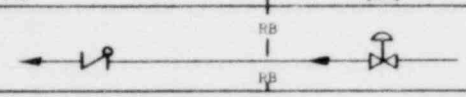
DETAILS OF EQUIPMENT HATCH AND PERSONNEL LOCK

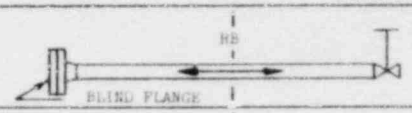
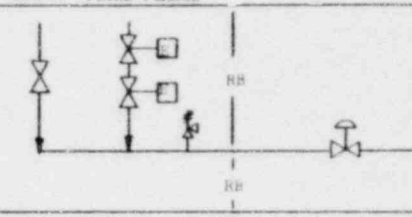
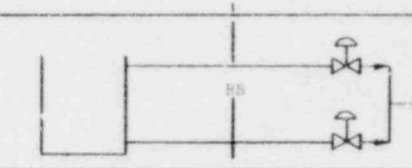
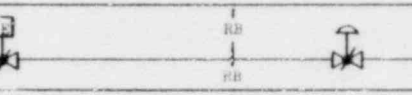
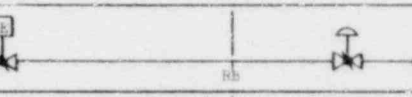
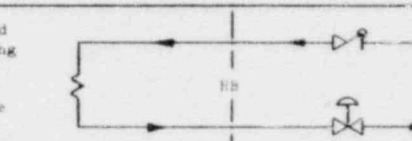
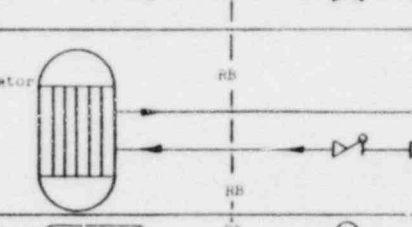
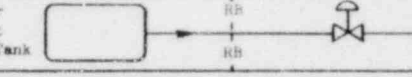
CRYSTAL RIVER UNIT 3

FLORIDA POWER CORPORATION

FIGURE 5-4

AMEND. 7 (7-15-69)

VALVE ARRGT	INSIDE REACTOR BUILDING	OUTSIDE REACTOR BUILDING	PENE-TRATION NO. AND TYPE
1			7 (TYPE I)
2			8 (TYPE I)
3			3,6 (TYPE I) 4, 46 (TYPE II)
4			2 44, 45 (TYPE III)
5			1, 5, 43 (TYPE I)
6		Closed Cooling Water Loop Inside	31 - 36 (TYPE III)
7			9,13,14 (TYPE I) 11, 12 (TYPE II)

VALVE ARRGT	INSIDE REACTOR BUILDING	OUTSIDE REACTOR BUILDING	PENE-TRATION NO. AND TYPE
8			10 (TYPE IV)
9			15 (TYPE I)
10			16, 17 (TYPE II)
11			19 (TYPE II)
12			20 (TYPE II)
13		Closed Cooling Water Loop Inside	21-28 INCL. 29,30 (TYPE III)
14		Steam Generator; Turbine Stop Valve	37-42 INCL. (TYPE III)
15		Reactor Coolant Drain Tank	18 (TYPE III)

REACTOR BUILDING ISOLATION

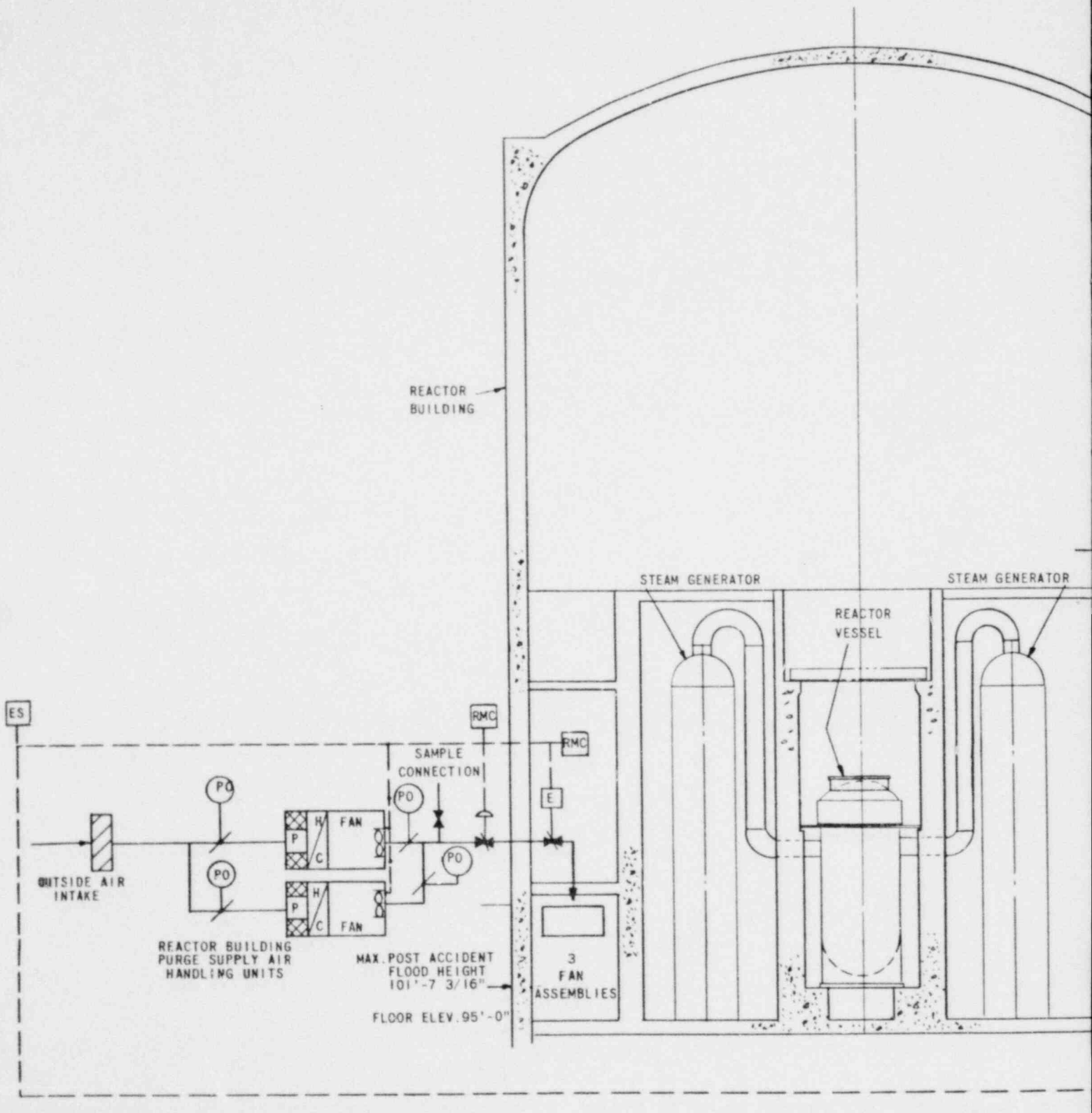
VALVE ARRANGEMENTS

CRYSTAL RIVER UNITS 3 & 4

FIGURE 5-5

AMEND. 1 (1-15-68)

0077



NOTE:
FOR LEGEND NOMENCLATURE SEE

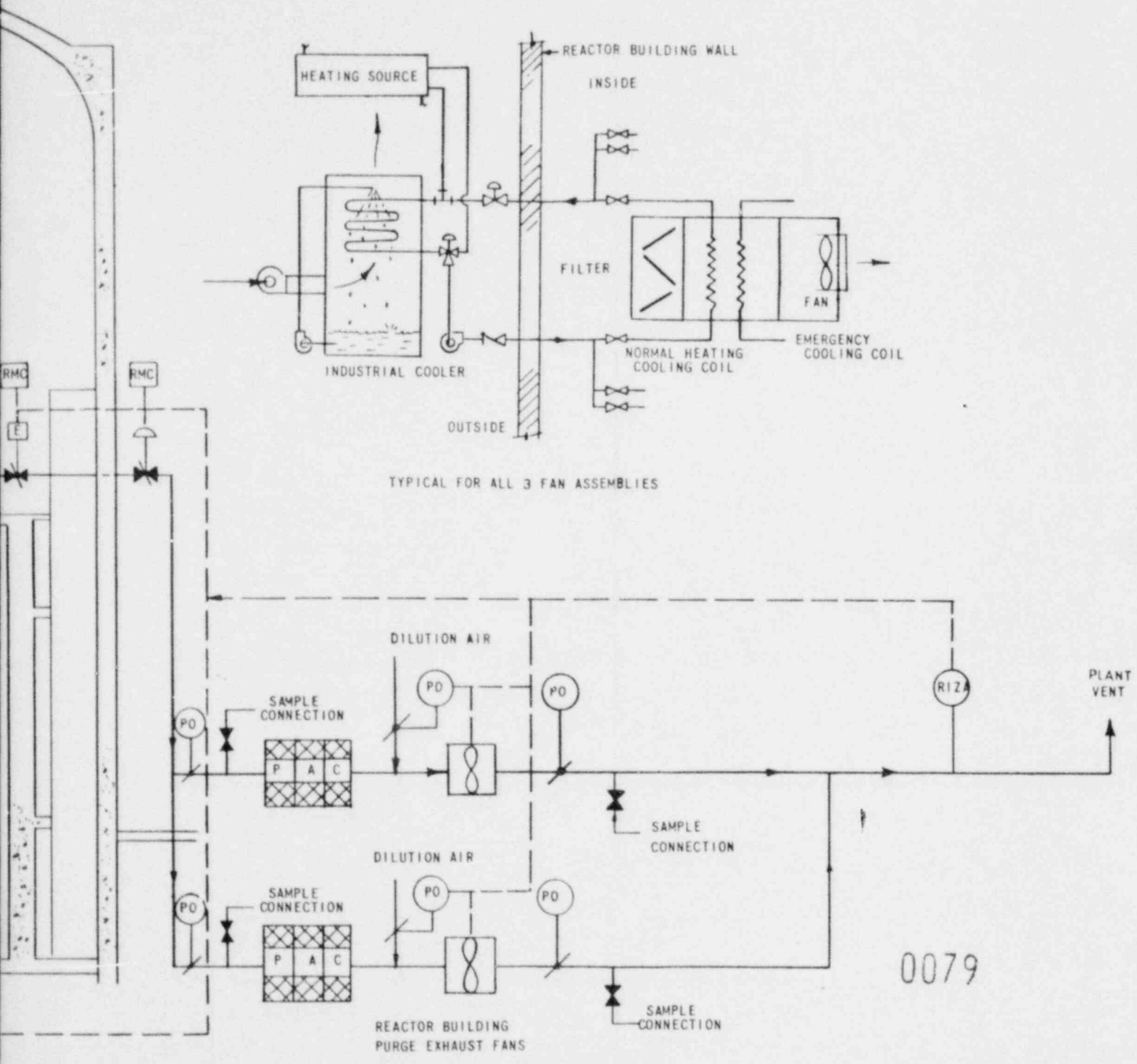


FIGURE 9-1

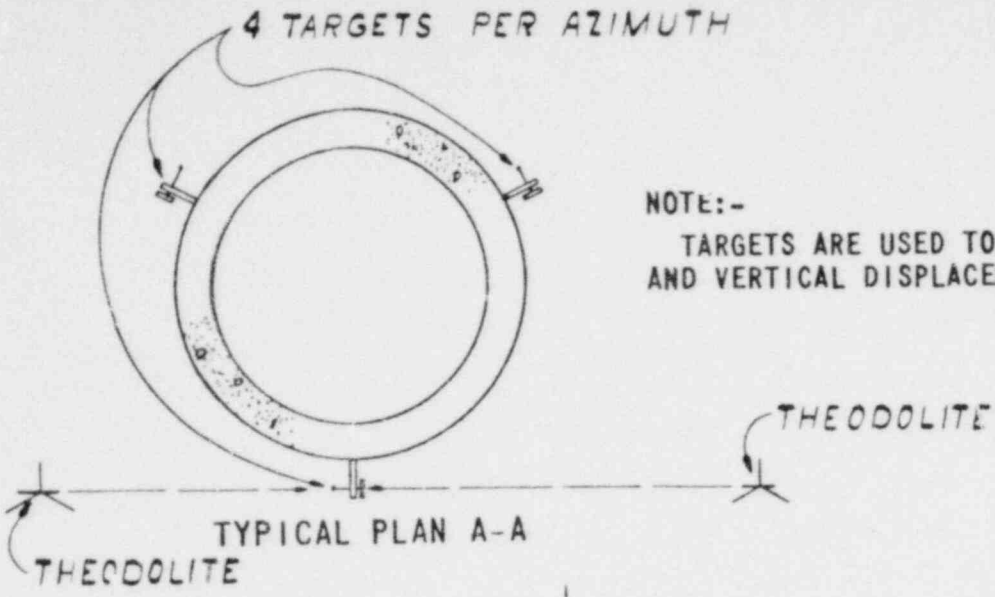
REACTOR BUILDING NORMAL VENTILATION SYSTEM

CRYSTAL RIVER UNITS 3 & 4

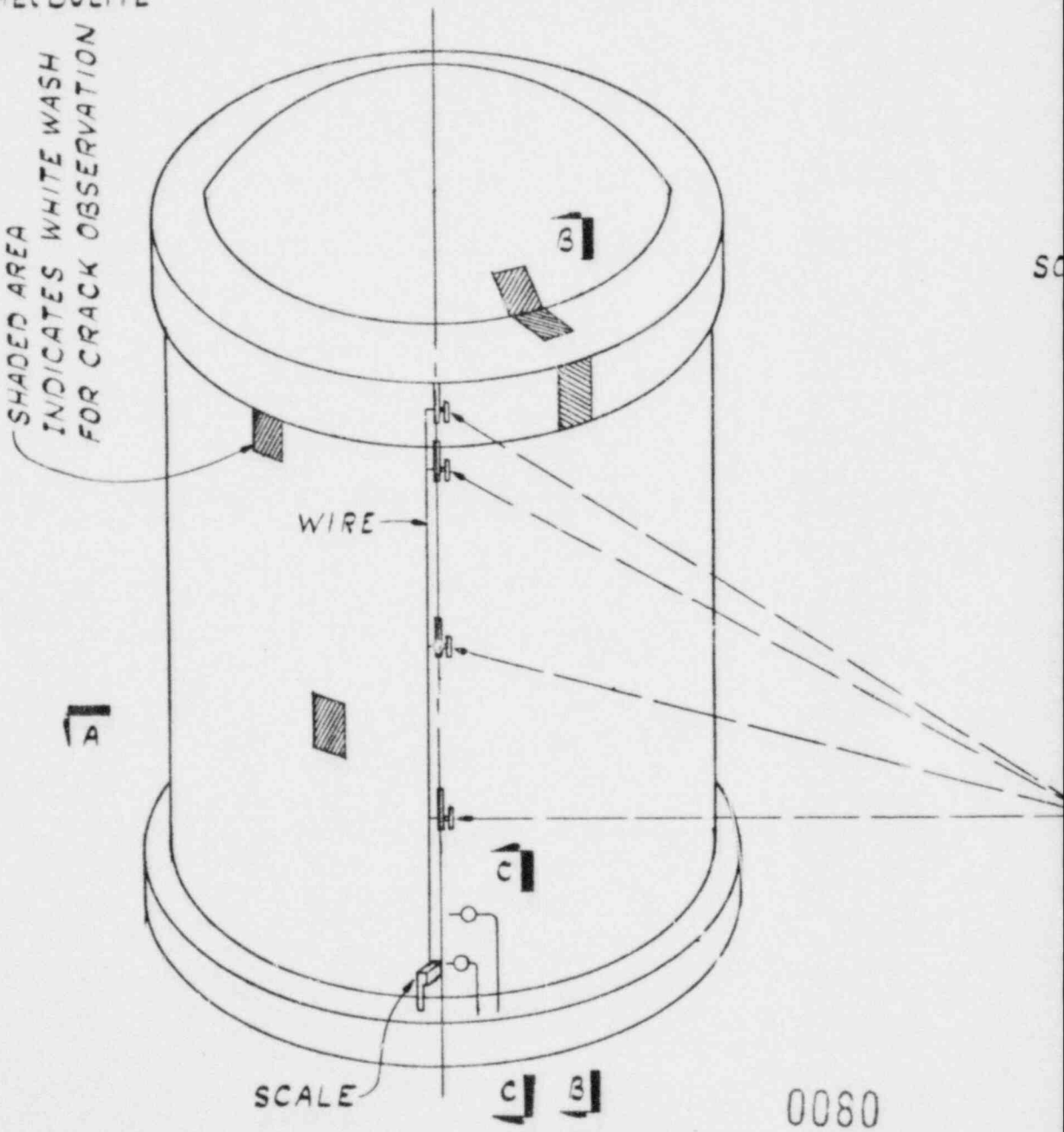
FPC FLORIDA POWER CORPORATION

FIGURE 5-6

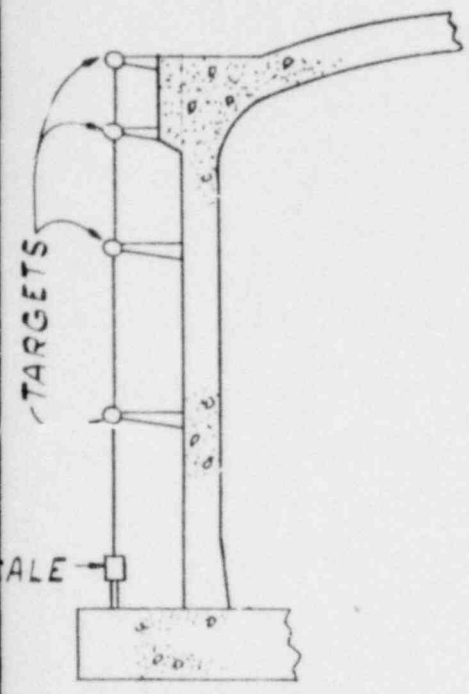
AMEND. 2 (2-7-68)



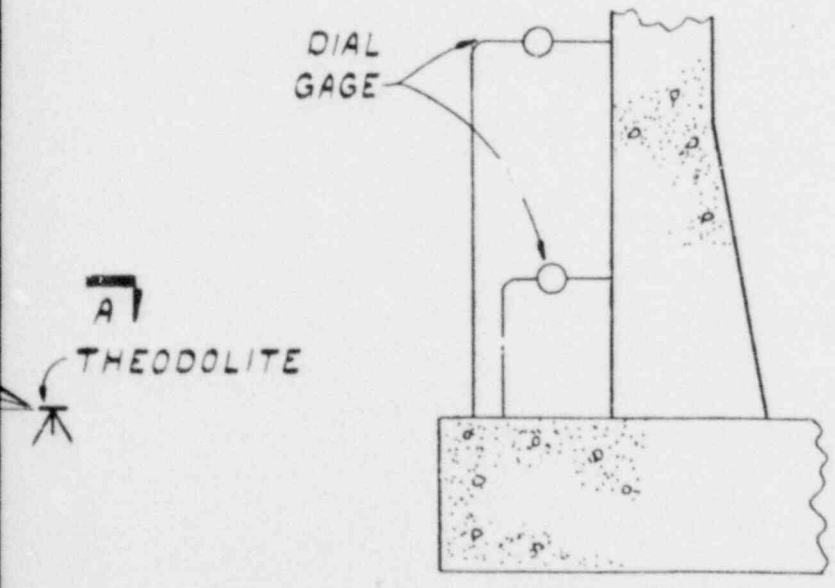
NOTE:-
TARGETS ARE USED TO MEASURE HORIZONTAL
AND VERTICAL DISPLACEMENTS.



0080



ELEVATION B-B
TYP. AT 3 AZIMUTHS



ELEVATION C-C
TYP. AT 3 AZIMUTHS

0081

REACTOR BUILDING INSTRUMENTATION
CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 5-7