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SECTION 5

CONTAINMENT SYSTEMS

5.1 GENERAL DESCRIPTION

The containment system, together with the engineered safety features (Section 6), is designed to limit doses, without normal power supply, at the site boundary and beyond to less than the 10 CFR 50.67, as applicable, criteria under conditions resulting from the Design Basis Loss of Coolant Accident (LOCA) (Section 14.3).

The containment system consists of the following:

1. The steel lined, reinforced concrete containment structure, including foundations, access openings, and penetrations, is shown in Figures 5.1-1, 5.1-2, 5.1-3, 5.1-4, 5.1-5, 5.1-6, and 5.1-7 and described in Section 5.2. It is designed and constructed to maintain full containment integrity when subjected to (1) the temperature, and pressures resulting from the DBA and (2) the earthquake conditions described in Section 2 and Appendix B. Provisions are made for the pressure testing of liner seams, penetrations, and access openings, and for leak testing the containment structure.
2. The containment isolation system, described in Section 5.3, is used to effectively isolate the containment following a DBA. The containment isolation system consists primarily of valves inside and outside the containment structure which are either check valves, normally closed manual valves, or valves which close automatically on a safety injection, containment isolation Phase A or containment isolation Phase B signal.
3. The containment ventilation systems, described in Section 5.4.1, consist of:
 - a. Air recirculation cooling systems
 - b. Control rod drive mechanism shroud cooling system
 - c. Purge exhaust and supply system

The containment ventilation systems are designed to limit ambient air temperature buildup and to provide a suitable environment for personnel and equipment with maximum safety against the spread of radioactive contamination. The only parts of these systems designed to remain functional following a DBA are the containment isolation valves, piping between these valves, and the exhaust duct external to containment isolation valves, VS-D-5-3A, to the supplementary leak collection and release system.

4. The containment vacuum system, described in Section 5.4.2.1, is designed to produce and maintain subatmospheric pressure in the containment during normal operation.
5. The containment leakage monitoring system, described in Section 5.4.2.2, is used to determine the leakage rate into or out of the containment during normal operation and periodic tests.
6. The containment depressurization system, described in Sections 5.4.3 and 6.4, reduces the containment pressure after a DBA.

5.2 CONTAINMENT STRUCTURE

5.2.1 General Description

The reactor containment structure is a totally reinforced concrete, steel-lined vessel with a flat base, cylindrical walls, and a hemispherical dome. The foundation mat is a soil bearing concrete slab about 10 ft thick, without projections below its lower surface. Projections into the foundation mat are provided for sumps. Embedments into the foundation mat are furnished for liner and internal anchorage. Structures contiguous to the containment are not structurally integral with the containment structure. The outside recirculation spray pumps and low head safety injection pump casings are embedded at their lower end in the portion of the containment mat which extends beyond the containment wall and are tied back to the containment wall at their upper end to limit movement of casings relative to containment during a seismic disturbance such that operation of pumps and integrity of piping is not compromised. The structures contiguous to containment are provided with a shake space to accommodate movement relative to containment. These structures are:

1. Fuel building
2. Safeguards areas
3. Main steam valve area
4. Cable vault
5. Pipe tunnel and penetration enclosure.

The integrity of Seismic Category I systems which extend between containment and contiguous structures is ensured by arrangement of piping and supports to accommodate relative movement caused by differential settlement and seismic disturbance. A dynamic analysis is made to confirm that stress values will not exceed those defined in the American National Standards Institute (ANSI) Standard for Power Piping, ANSI B31.1.⁽¹⁰⁾

The concrete vertical cylinder wall has an approximate 126 ft ID with a nominal 4.5 ft thick wall. The spring line of the dome has an inside radius of about 63 ft and is approximately 2.5 ft thick. The steel liner is described in Section 5.2.4.

The concrete foundation mat, cylinder wall and dome are the structural members of the containment structure. The steel liner acts as an impervious membrane, transmits loads to the concrete and was used during construction as the inside form for the reinforced concrete wall and dome. The containment structure does not require the participation of the liner as a structural component. No credit is taken for the presence of the steel liner in designing the containment structure to resist earthquake forces or other design loads.

The 4 inch minimum thick layer of porous concrete below the structural mat is a coarse aggregate bound with a water cement paste as noted in Section 5.2.3.2. This porous concrete sub-base underlies the containment mat.

The exterior surface of the concrete shell and foundation mat, shown in Figure 5.2-1, has a continuous waterproofing membrane to protect the containment structure against water seepage during flood stages resulting from the Standard Project Flood (SPF) and the Probable Maximum Flood (PMF), both of which are described in Section 2.3. As a supplementary safety factor, water relief systems are provided in the floor of the two instrument pits at El. 690 ft-11 inches (Figure 5.1-4). They are located in the mat outside of the containment wall. The sumps extend into the mat to a point above the bottom reinforcement. From the bottom of the sump, a vertical 6 inch pipe projects into the underlying porous concrete. In the event of a flood and unexpected leakage through the membrane, the vertical pipe would allow the water to rise in the sump where it would sound an alarm in the control room after reaching a predetermined height. The water would then be removed by a sump pump to prevent buildup of pressure behind the steel liner. The instrument pits are enclosed by the waterproofing membrane protecting the containment structure.

The areas contiguous to the containment which house safety related Seismic Category I equipment are free standing and are founded on the same soil stratum as the containment mat, with the exception of a narrow strip which is founded on a compact granular fill between the cofferdam and the containment wall. Differential settlement is expected to be small. Loads imposed on Seismic Category I equipment and piping due to differential settlement and relative movement of structures during seismic disturbance is compensated for by design of the piping, hangers, and equipment supports so that the stress in the pipe does not exceed the maximum stress allowed by ANSI B31.1, and thus system integrity is ensured.

There are numerous other structures surrounding the containment that house Seismic Category I equipment. These structures, which are interconnected and are also connected to the containment by piping include:

1. Primary auxiliary building, including pipe tunnel
2. Fuel building
3. River water pump and engine-driven fire pump intake structure
4. Refueling water storage tank
5. Safeguards area
6. Main steam valve area.

The maintenance of integrity of the Seismic Category I piping systems after differential settlement has occurred and while experiencing loading due to differential movement of equipment and structures during seismic disturbance is ensured by designing the piping, hangers and equipment supports to accept these loadings while experiencing stress values that are below those allowed by ANSI B31.1.

Table 5.2-17 provides the structures, systems and components within the containment that are not designated Seismic Category I.

Other structures, systems and components within the containment whose potential failure could compromise the functional capability of surrounding Seismic Category I components are classified as Seismic Category II in Appendix B and are designed to withstand the combined effects of normal operating loads and earthquake loads. By the static analysis methods outlined in Section B.2.2 structural elements, anchorages and restraints are designed to preclude the possibility of Seismic Category I components becoming endangered. Analysis of Seismic Category II components is not intended to assure their functional capability.

5.2.2 Design Basis and Loading Criteria

The structural design of the containment is based upon:

1. Shielding requirements
2. The pressure and temperature generated by the DBA (see Section 14)
3. The Operational and Design Basis Earthquake (see Section 2.5 and Appendix B)
4. The maximum calculated core thermal power level of 2,917.4 (2900 MWt x 1.006).

The temperature and pressure resulting from the DBA are selected as the containment design basis, since the normal operating conditions would result in lower design temperatures and pressures. The containment structure is also designed for atmospheric operation and for a maximum leakage of less than 0.1 percent per day of the weight of containment air at the calculated peak containment pressure as given by the limiting LOCA analysis, as described in Sections 14.2 and 14.3.

During normal operation, the containment air partial pressure is maintained between 12.8 and 14.2 psia and the containment air temperature can range between 70°F minimum and 108°F maximum. The containment is accessible for inspection and minor maintenance during operation.

The design pressure and temperature for the containment are 45 psig and 280°F. The accident pressure is calculated by using the MAAP-DBA computer program (Sections 5.2.2.4 and 14), which calculates the containment pressure transient of the DBA. The results of the pressure transient calculations, with minimum engineered safety features, are described in Sections 14.2 and 14.3. The design pressure for the containment exceeds the maximum calculated pressure.

Localized pressure pulse effects are considered in the design of the containment structure. These effects result from a double ended pipe rupture or split pipe rupture at any location in either the reactor coolant loops or main steam lines. All closed or restricted spaces subjected to these localized pressure pulses are designed with sufficient vented openings to limit the pressure differentials between adjacent compartments. All structural components, walls, floors, and beams in these spaces are designed to withstand the pressure differential as part of their design loading.

The concrete containment structure including the interior compartments and shield wall is designed by ultimate strength methods conforming to American Concrete Institute (ACI) Standard 318-63, Part IV⁽¹¹⁾. Design load criteria, based on ACI requirements and stated below, conform to current containment design practice. The combinations of dead load, pressure, and earthquake or tornado loading, expressed in the design criteria, contain varying load factors for pressure and earthquake forces. The total loading resulting from the summation of any one of the combinations of these loads causes a maximum stress condition, depending upon the type of stress and member under consideration.

In accordance with ACI 318, Section 1506, forces due to wind or tornadoes and those due to earthquakes are not considered to occur at the same time. Normal wind forces, as recommended by ASCE Paper No. 3269, as referenced in Section 2.7, do not govern the design. Buoyance, caused by flooding, is a factor in design only during construction of the containment (Section 5.2.2.5).

The effect of creep, shrinkage and stress concentration on the modulus of elasticity and Poisson's ratio on reinforced concrete is of small consequence during the short term loading incurred as a result of the DBA, and is therefore not included in the analysis of design of the conventionally reinforced concrete containment.

The pressure and temperature conditions within the containment during a LOCA DBA are shown in Figures 14.3-56, 14.3-56a, and 14.3-56b. Temperature conditions within the containment during a MSLB DBA are shown in Figure 14.2-35. The containment liner temperature following the MSLB DBA is shown in Figure 14.2-36. The temperature increase in the rebars will decrease the modulus of elasticity of the rebars only slightly (less than 3 percent). This reduction is within the capacity reduction factors listed below for variation in material strengths and workmanship. In addition, the DBA pressure will be reduced significantly before the rebar is affected by the temperature rise.

The effect of temperature, shrinkage and creep on the concrete modulus does not affect the analysis of the containment shell and dome since the concrete is assumed to be cracked during the DBA loading. The containment mat, 10 ft thick, is insulated by approximately 2 ft of overlying concrete. The DBA produces only short term loading on the mat and creep is not significant.

Concrete is considered to offer resistance only near the base of the containment. The temperature gradient through the concrete shell thickness of 48 inches is shown in Figure 5.2-28. Since the maximum design pressure occurs within the first 20 seconds, the temperature effects are negligible.

The load capacity of concrete members that are subject to tension is based on the guaranteed minimum yield strength of the reinforcement steel.

Load capacities of flexural and compression members are determined in accordance with ACI 318-63. Load capacities so determined are reduced by a reduction factor multiplier, θ , to compensate for small adverse variations in material and workmanship. The reduction factors are provided in Table 5.2-18.

Stress and strain limits resulting from the loading criteria conform to the requirements of ACI 318-63, Part IV-B. Principle reinforcing steel used in the construction of the containment structure has a minimum yield strength of 50,000 psi and a minimum ultimate strength of 70,000 psi. Concrete has a specified 28 day compressive strength of 3,000 psi. The analyses of the structure for static loading and for dynamic loading are covered in Section 5.2.2.5. Design load values relative to the site are summarized in Section 2.7.

Structural steel sections of the containment interior are designed in accordance with the American Institute of Steel Construction, AISC-63⁽¹²⁾, except that the seismic loading referred to in AISC-63, Part 1, Paragraph 1.56 is considered as the OBE. Allowable stresses are factored as follows:

1. OBE: 33 percent increase
2. DBE: 90 percent of the specified minimum yield strength for structural steel.

The designed ultimate load capacity of the containment structure, as modified by the safety provisions of ACI 318-63, Section 1504, is not less than that required to satisfy the structural loading criteria for various components described in the subsequent sections.

5.2.2.1 Containment Shell Structural Loading Criteria

1. Maximum hypothetical loading for the Design Basis Accident (DBA) only with a 1.5 factor for the DBA pressure:

$$D(1.0 \pm 0.05) + 1.5P + T + TL_{(1.5P)} \tag{5.2-1}$$

2. Maximum hypothetical loading for the 1.25 factored DBA and 1.25 factored Operational Basis Earthquake (E):

$$D(1.0 \pm 0.05) + 1.25P + T + TL_{(1.25P)} + 1.25E \tag{5.2-2}$$

3. Maximum hypothetical loading for the unfactored DBA and unfactored Design Basis Earthquake (E')

$$D(1.0 \pm 0.05) + 1P + T + TL_{(P)} + E' \tag{5.2-3}$$

4. Maximum hypothetical loading for tornado only:

$$D(1.0 \pm 0.05) + T + C \tag{5.2-4}$$

Where: D = Dead load of structure and equipment, including effect of earth, hydrostatic pressures, ice, and snow loads, when their effect increases the resultant stresses. To provide for variations in the assumed dead load, the coefficient for the dead load component is adjusted by ±5 percent, as indicated in the above formulas, to provide the maximum stress levels.

- P = Pressure load resulting from the DBA
- T = Load due to maximum temperature gradient through the concrete shell and mat for normal operating conditions
- TL = Load exerted by the liner, when it is exposed to the temperature associated with the pressure resulting from a DBA
- E = Loading from "Operational Basis Earthquake" (Section 2.5)
- E' = Loading from "Design Basis Earthquake" Section 2.5.
- C = Load due to negative pressure and horizontal wind velocity resulting from assumed tornado (Section 2.7).

5.2.2.2 Containment Interior Loading Criteria

The design pressures for the interior compartments of the containment structure are determined by the numerical integration of finite differential equations defining heat and mass flows into and out of the interior compartments, assuming a double-ended rupture of the primary coolant pipe within the compartment.

The time dependent pressure differential curves for the steam generator cubicles are shown on Figures 14.3-86, 14.3-87 and 14.3-88 for a hot leg double-ended rupture. The containment peak pressure curves are shown on Figure 14.3-55 for three types of double-ended ruptures. A comparison of these curves shows that peak containment pressure occurs later than the peak pressure of the steam generator cubicles. It is at this later time, that the foundation mat is under maximum loading from the accident pressure. Force from internal cubicle pressure loading is transmitted to the mat in part by the column under the cubicle floor which supports the steam generator. The loading due to cubicle pressure is well in advance of the maximum foundation mat loading. Maximum cubicle loading and maximum foundation loading cannot occur at the same time. The foundation mat is designed for either the maximum cubicle pressure loading or the factored containment overpressure loading.

The design of the steam generator cubicles, including the discontinuities, utilizes the same ultimate strength design methods described for the design of the exterior structure. The compartments are designed to have a load capacity adequate to resist:

$$D (1.0 \pm 0.05) + R + T + P' + E' \quad (5.2-5)$$

- where:
- D = Dead load of structure and equipment
 - R = Double-ended or longitudinal pipe rupture thrust on structure as a function of time (including jet forces, as applicable)
 - T = Load due to maximum temperature gradient through the concrete from increased temperatures resulting from the pipe rupture and pressure buildup
 - P' = Pressure buildup from the expansion of the fluid released from the ruptured pipe as a function of time

E' = Design basis earthquake loading

For design purposes, the maximum values of R and P' are assumed to occur concurrently.

The strength capacity of reinforced concrete sections is determined by ultimate strength provisions of ACI 318-63 incorporating the load capacity reduction factors used for the containment structure design. An elasto-plastic analysis is conducted on the steam generator cubicle walls to ensure that they have sufficient ductility to absorb the energy of the jet impingement forces.

The steam generator support columns together with other support elements of the structure are designed to carry safely the maximum axial forces, moments and shears which are developed from studies of the postulated pipeline ruptures. These loadings are decaying forces which are maximums at the beginning of the accident when the mat is not loaded by accident pressure and equal to zero when full accident pressure on the mat occurs.

Time dependent temperature rise in the cubicle resulting from accident has negligible effect on the mat.

The radial walls, crane wall and floors which define the steam generator cubicles are designed for the differential pressures and loads stated above.

The temperature differentials are negligible due to the fact that the containment temperature is of the order of 140°F within 30 minutes and as a result only a few inches of concrete would be affected by the transient.

The primary shield wall which houses the neutron shield tank and the reactor is designed for the following loading conditions:

1. Dead load
2. Transient internal pressure due to pipe rupture in the reactor cavity varying from 146.2 psi at the coolant pipe nozzle penetration to 20 psi at the bottom of the shield tank and 0 psi at the top of the primary shield wall
3. Average steady-state temperature rise in wall from 50°F to 120°F
4. Operating gradient temperature across the wall thickness equal to 23°F
5. Accident transient temperature on wall face equal to 160°F
6. Induced forces (moment and shear) due to operating thermal gradient (25°F) and average steady-state temperature rise (30°F) in the foundation mat
7. Seismic shear and moment on the wall
8. Reactions of reactor vessel, steam generator and reactor coolant pump supports due to pipe rupture

9. Load of 2563 K applied to 48 inch grouted area between shield tank and primary shield wall, located at the top of shield tank, due to seismic and pipe rupture loads on reactor. The loading conditions 1 through 8, as pertinent, are combined with the general containment loading conditions, as given in Section 5.2.2.1 to produce the most severe design combinations.

For the design of the foundation mat, the factored loads stated in Section 5.2.2.1 are superimposed upon normal operating loads which include the thermal conditions and the normal and accident forces induced by the primary shield wall, the crane wall, steam generator support columns and the containment wall.

The design of the mat includes the effect of change of distribution of soil pressure due to overturning moment from earthquake loadings.

5.2.2.3 Reactor Coolant System Equipment Supports

5.2.2.3.1 Design Basis

The reactor coolant system includes the reactor vessel, three steam generators, three reactor coolant pumps and a pressurizer. Structures are provided to support this equipment to ensure system integrity during normal operation and Design Basis Accident conditions.

Originally, all supports in the reactor coolant system were designed to withstand dead weight and the Design Basis Earthquake acting simultaneously with an instantaneously applied pipe rupture. If applicable, two types of piping failures are considered separately: (1) a double-ended rupture and (2) a longitudinal rupture on either the horizontal or vertical axis of the pipe. These failures were assumed to occur in the reactor coolant piping, the reactor coolant branch piping including the pressurizer surge line, the main steam piping or the feedwater piping. The value of the pipe thrust for any reactor coolant or reactor coolant branch pipe rupture was based on Westinghouse blowdown analysis. Historical static design loads for the pipe thrust are listed in Table 5.2-1. These loads include a value of two for the dynamic load factor. The supports were initially analyzed for these loadings simultaneously with the Design Basis Earthquake, and all stresses remained within 90 percent of the minimum yield point of the structural material used. A dynamic analysis was then performed to validate the static analysis. Except for the Steam Generator Upper Support Ring this original design basis still provides bounding design basis for the current plant. The Steam Generator Upper Support Ring was redesigned as part of the Steam Generator replacement and considers pipe breaks loads from the reactor coolant loop branch piping (excluding the main reactor coolant piping and the pressurizer surge line) and from the main steam and feedwater piping.

All welding is in accordance with the American Society for Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section IX, and all welds are examined by either radiographic, ultrasonic, dye penetrant, or magnetic particle techniques, depending on the material, and the state of stress at the weld.

5.2.2.3.2 Description

Reactor Vessel Support

The reactor vessel is supported by six sliding foot assemblies mounted on the neutron shield tank as shown in Figure 5.2-2. These foot assemblies were fabricated from modified American Iron and Steel Institute, AISI 4330 forgings. The support feet are designed to restrain lateral and rotational movement of the reactor vessel for simultaneously applied Design Basis Earthquake, pipe rupture loads and dead weight, while allowing thermal expansion. The neutron shield tank is a double walled cylindrical structure of American Society for Testing Materials (ASTM) A-516 steel which transfers the loadings to the heavily reinforced concrete mat of the containment structure and to the primary shield wall. The tank also serves to minimize gamma and neutron heating of the primary concrete shield, and to attenuate neutron radiation outside of the primary shield to acceptable limits.

The shield tank is securely fastened down by anchor bolts. Overturning moments and horizontal forces which are induced on the tank during normal operation or accident condition are taken in part by the reinforced concrete primary shield wall poured locally around the neutron shield tank. Any resulting vertical uplift force is taken by the anchor bolts.

The tank is completely shop fabricated and is constructed in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII. However, a code stamp is not required. All welding, welding procedures, and welding operators' qualifications are in accordance with the ASME Code, Section IX.

All welds are inspected by either radiographic, dye penetrant, or magnetic particle techniques.

After fabrication, the completed tank is subjected in the vertical position to a hydrostatic test of 15 psig as measured on top of the tank. In no case does the hydrostatic pressure exceed 35 psig anywhere in the tank. The tank is also leak tested with dry air at 20 psig, by applying soapsuds to all welds accessible from the outside of the tank.

Steam Generator and Reactor Coolant Pump Supports

The steam generator and reactor coolant pump supports are shown in Figures 5.2-3 and 5.2-4. The materials used are for the most part commercially available structural shapes of ASTM A-36 steel. High strength quenched and tempered alloy steels are used for local attachments at the steam generator and reactor coolant support pads, in the hydraulic snubbing assemblies, in the pump support columns, and in the steam generator struts.

The steam generator support system consists of an upper support ring (Steam Generator Upper Lateral Restraint) and a lower support frame. The upper support ring is shimmed in the cold condition to the steam generator with a design radial gap sized with consideration for the pressure expansion of the steam generator, and insulated so that it expands thermally as the steam generator is brought up to temperature. The upper support ring transmits horizontal forces from the steam generator through four tangential load trains to the reinforced concrete operating floor. The operating floor in turn transmits these horizontal forces to the reactor shield wall, the crane wall, and to the cubicle walls, where, through shearing actions, they are further transmitted downward to the mat. The tangential load trains from the upper support ring to the

operating floor are equipped with hydraulic snubbing cylinders and struts. The snubbers permit limited slow motion of the steam generator to allow for thermal expansion of the reactor coolant piping from the reactor to the steam generator. The cylinders, however, react to resist suddenly applied forces which occur from earthquake or pipe rupture conditions.

The lower support frame is a weldment fabricated of ASTM A-36 structural steel shapes. The support frame slides on lubricated bearing plates located under each corner column in order to permit thermal expansion of the reactor coolant piping from the reactor to the steam generator. The four columns also transmit vertical forces from the steam generator to the cubicle floor. The support frame has large shear blocks on two sides which fit into embedments in the cubicle floor. These shear blocks guide the lower support frame along a direction radial from the reactor and transmit forces perpendicular to this motion into the embedments in the cubicle floor. The attachment of the lower support frame to the four pads on the steam generator bottom head permits radial thermal expansion of the steam generator.

The reactor coolant pump is mounted in a support frame which permits radial thermal expansion of the pump feet. The frame is held above the cubicle floor by three pin-ended columns which provide vertical support while allowing free movement in the horizontal plane.

The vertical forces applied to the cubicle floor are beamed-out in the reinforced concrete to the edges of the floor where the vertical forces are transmitted to the surrounding walls. The vertical forces transmitted to the cubicle walls are in turn beamed-out to the crane wall columns and to the shield wall where they are carried downward to the mat. A 5 ft-0 inch square column between the cubicle floor and the mat beneath the steam generator provides an additional load path which transmits some of the vertical forces directly from the cubicle floor to the mat.

Horizontal forces applied to the cubicle floor act as a torsional moment about the centerline of the reactor. This moment is transmitted to the mat by torsional shearing forces in the shield wall and by shear forces in the crane wall columns and in the column below the steam generator.

The base of the shield wall (Figure 5.2-6) is designed to pass on to the mat the shear stresses initiated on it by mat deflection.

It also has the capability of transmitting horizontal seismic loading on the internal structure to the mat. The 3 inch by 6 inch bridging bars of the crane wall and the shield wall also participate as far as the stiffness and straining of the structure permits.

Pressurizer Support

The pressurizer vessel is mounted to a rigid ring girder which is suspended from the operating floor by four hanger columns as shown in Figures 5.2-7 and 5.2-8. To offset the flexibility of the hanger columns, two brackets welded to the ring girder slide in guides rigidly attached to the bulkhead which restrain all motions except vertical translation. In addition, anti-sway brackets welded to the shell of the pressurizer fit into striker plate assemblies embedded in the concrete floor close to the center of gravity of the vessel. These brackets permit the pressure vessel to expand vertically but restrain horizontal displacements.

The ring girder is fabricated from ASTM A-516 Grade 70 steel, or equivalent,⁽⁴³⁾ of 38,000 psi minimum yield point. The striker plate assemblies are fabricated from AISI 4340 steel, or equivalent,⁽⁴³⁾ heat treated to 120,000 to 140,000 psi yield point. The hanger columns are fabricated from ASTM A-106, Grade B, or equivalent,⁽⁴³⁾ pipe. The majority of the fasteners and shear pins used in the support are fabricated from either ASTM A-193, Grade 7, or AISI 4340, heat treated, depending on the stress level.

5.2.2.3.3 Evaluation

Steam Generator and Reactor Coolant Pump Supports

To determine dynamic loads on the Steam Generator Lower supports and the Reactor Coolant Pump supports, the analytical model in Figure 5.2-25 was employed. This model accounts for the dynamic behavior of the steam generator and reactor coolant pump, as well as associated piping and supports. The mass and stiffness characteristics of each of the major subsystems were accurately transformed into a lumped parameter system. The STARDYNE program, developed by Mechanics Research, was employed to analyze the dynamic response of the system due to pipe rupture and earthquake motions.

Blowdown data supplied by Westinghouse were used to perform the pipe rupture analysis. These data consisted of time history forces which included jet thrust at break location and internal hydraulic forces resulting from the propagation of shock waves and flow inertias. For the seismic analysis, the total response was evaluated by taking the square root of the sum of the squares of the individual modal contributions.

Since the dynamic model shown in Figure 5.2-25 is an idealization of the support structure and equipment shown in Figure 5.2-4, the results of the dynamic analysis could not be used directly. Instead, the time history of displacements obtained from the dynamic analysis was applied to a more detailed static analysis model to obtain internal loads and stresses in the support structure, loads on the equipment support pads, and loads on the reinforced concrete structure which interface with the equipment supports. This stress model which is shown in Figure 5.2-26 was modeled and analyzed using the STRUDL⁽¹³⁾ program (developed by M.I.T. Civil Engineering Department).

The current RCS LOOP analysis using NUPIPE-SW was completed and the results compared to the original static analysis using STARDYNE. This comparison shows that the design loads were conservative for the majority of the cases. Where the dynamic loads were larger than the design loads, detailed analyses were performed and design modifications made where necessary to ensure the integrity of the structure. Tables 5.2-2, 5.2-3, and 5.2-4 contain static and dynamic loads for some of the components. Thermal gradients produced in the supports by pipe rupture jet impingement were also investigated. This and other areas have been examined for the effects of jet impingement thermal gradient and forces.

Reactor Vessel Supports, Neutron Shield Tank and Pressurizer Supports

For the determination of dynamic loads on the reactor vessel supports, neutron shield tank and the pressurizer supports, an analysis technique was applied which is similar to that used for the steam generator and reactor coolant pump supports. However, for the reactor vessel supports and neutron shield tank the STRUDL program was employed.

The dynamic model and the stress model used in the analysis of reactor vessel supports are shown in Figures 5.2-10 and 5.2-27, respectively. The dynamic model employed for the pressurizer support is shown in Figure 5.2-9. For this system no stress model was required since the dynamic model was sufficiently detailed to directly compute stresses and loads.

5.2.2.4 Bases for Containment Analyses

To ensure that the containment performs the desired function of protecting the public against gross equipment failures, analyses have been performed assuming the double-ended displacement ruptures of piping in the reactor coolant system. These ruptures have been assumed to occur in conjunction with loss of all off-site power.

They are defined in the Design Basis Accident (DBA) descriptions in Sections 14.2 and 14.3.

The design of an atmospheric containment is concerned with the peak containment pressure following the rupture of a RCS or Main Steamline pipe.

The BVPS-1 containment, and engineered safety features are designed to:

1. Withstand, without loss of integrity, a maximum pressure of 45 psig and a maximum containment liner temperature of 280°F
2. Reduce the calculated containment pressure to less than half of the peak pressure within 24 hours.

The results presented in Section 14.3 and analyses of other less serious accidents (Section 14.2) indicate that the bases for design are adequate, and that margin is allowed for uncertainties in design.

5.2.2.5 Static and Dynamic Analyses

5.2.2.5.1 Loads

The containment structure is analyzed and designed for the following loads, which are multiplied by load factors and are added together in the manner detailed in Section 5.2.2.1.

Lateral Earth Pressure

The top of the containment mat is approximately 44 ft below ground elevation; therefore, active and passive earth pressures exert an influence during operation and DBA conditions, respectively. The lateral earth pressure due to seismic response is evaluated in Section 2.6.4.4.

Horizontal and Vertical Earthquake

These forces are determined by means of the dynamic analysis described in Appendix B. The stresses resulting from the horizontal and vertical earthquakes are added algebraically.

Hydrostatic Pressure and Buoyancy

Hydrostatic pressure occurs only during flood stages since the groundwater at the site is normally well below the top of the containment mat. The groundwater is at about El. 666 ft when the river is at pond level El. 664.5 ft. Hydrostatic pressure, when combined with the lateral earth pressure, does not cause maximum stress conditions in the cylinder walls. Loading on the containment during the probable maximum flood (PMF), only exerts a maximum hydrostatic pressure of 17.4 psi at the base of the containment. Buoyancy of the containment was not a design problem, as conditions favoring it could occur only during the early construction period. In the unlikely event of a probable maximum flood exceeding the cofferdam height, buoyancy of the partially completed containment would have been averted by flooding the structure with water. When the containment concrete work was completed to about El. 745 ft, the structure had enough dead load to balance the buoyancy due to a probable maximum flood. Flood stages and the wave resulting from a dam break upstream are discussed in Section 2.3.

Dead and Live Load

Dead load includes the total weight of the structures, including equipment. Live loads, such as snow and ice, are included when their effect increases the resultant stresses.

Internal Pressure

During normal operation, pressure within the Technical Specification operating range is maintained inside the containment. The maximum DBA internal pressure is 45.0 psig. The test pressure used during the tests to determine the integrity of the containment structure was 15 percent greater than the design pressure of 45 psig.

Temperature

During normal operation, the average temperature inside the containment can range from 70°F minimum to 108°F maximum. The maximum temperature difference through the containment wall is 85°F.

Wind

The containment site is located in the 80 mph wind velocity area as determined from Figure 1 (b) for 100 year period of recurrence in ASCE Paper No. 3269, (Section 2.7). Wind design pressure, when substituted for the earthquake loading in the Containment Shell Structural Loading Criteria (Section 5.2.2.1), does not cause maximum stress conditions.

Tornado

A tornado is assumed to produce a reduction in the atmospheric pressure outside the containment plus a wind with translational and rotational velocities. The description of the hypothetical tornado is given in Section 2.7. The wind velocity is converted into an equivalent pressure as recommended in Reference 1.

Missiles

The design of the containment structure is checked for potential external missile penetration during a hypothetical tornado. The impact loading is concurrent with the tornado wind loading criterion stated in Section 5.2.2.1. The hypothetical tornado missile is described in Section 2.7.

Missile impinging upon the containment are discussed in Section 5.2.6.

5.2.2.5.2 Foundation Mat Analysis

The circular containment mat is analyzed to determine the effects of loads imposed by the DBA conditions. Analysis is accomplished by means of a digital computer program which has the capability to calculate bending moments, shears and soil pressures for a symmetrically loaded circular plate on an elastic foundation.

The general method used is described in "Practical Methods for Analysis of Beams and Plates on Elastic Foundations," (in Russian) by Boris N. Zhomochkin, which is for a plate on a semi-infinite elastic half space. This method is an adaptation of the Boussinesq approach. Zhomochkin's method is modified to account for a finite depth of elastic foundation, i.e., the distance between mat and underlying rock. In addition, the cylindrical containment wall, crane wall and primary shield wall are rigidly fixed to the mat and therefore produce discontinuity moments and shears which are applied to the mat as external forces. The magnitudes of the discontinuity forces are determined by enforcing deflection compatibility at the wall-mat interfaces by means of flexibility coefficients representing the shield wall, crane wall and containment wall under the action of edge moments and shears. The flexibility of the containment wall is based on a section which is assured to be completely cracked vertically and cracked horizontally to the neutral axis of the transformed section. The containment wall, then, has a hoop stiffness of the circumferential rebars and a meridional bending stiffness of the transformed section. The compatibility determination also accounts for the membrane displacement of the containment wall under internal pressure and includes temperature effects in the liner on this displacement.

Dynamic analysis of the containment structure provides information about seismic loads imposed on the mat. These loads are a base shear and overturning moment. The maximum forces under the action of these loads are determined using a S&W finite difference computer program called SHELL-1 based on an analysis of unsymmetrical bending of plates and shells found in Reference 2. The results of this analysis are combined with forces from other loads given in Section 5.2.2.1 in order to make a complete maximum stress analysis.

Bending moments, shears and soil pressures are determined for the load conditions stated in Section 5.2.2.1.

5.2.2.5.3 Shell Design

The seismic analysis of the containment structure provides the accelerations to which the containment structure will be subjected. These accelerations are applied as static loading to the containment shell which is analyzed by the SHELL-1 computer program. The tangential shear caused by the seismic loading is resisted by the concrete and a system of diagonal rebars.

As the mat was poured prior to completion of tests made to determine the value of shear resistance obtained within the concrete by aggregate surface friction, No. 14 rebar diagonal stubs were provided in the mat to resist the entire DBE seismic force at the base of the wall. The spacing of these rebars is nominally 2 ft-0 inches horizontally, both ways. Above the base diagonal rebars are provided only as required to supplement aggregate surface friction in resisting surface shear. No. 14 diagonal rebars (nominally 4 ft-0 inches o.c. both ways) are installed to El. 711.5 and above that No. 14 diagonals (nominally 8 ft-0 inches o.c. both ways) are installed to the top of the cylinder wall. No diagonal rebars are provided above the spring line.

SWND-5⁽¹⁴⁾ indicates the allowable shear stress carried by the concrete can be as high as 160 psi, due to effects of aggregate interlock. However, to demonstrate the safety of the containment structure under design loading a more conservative value of 40 psi for the OBE condition and 60 psi for the DBE condition will be used. The shear force (V_u in. K/ft) at the base of the wall, El. 695 ft-1 inches, and at El. 711 ft-0 inches under the factored OBE and DBE are provided in Table 5.2-19.

The allowable shear force in the concrete, V_c associated with aggregate interlock for the factored OBE, is 26 K/ft and, for the DBE, is 39 K/ft. The difference between the seismic shear force, V_u , and concrete allowable shear force, V_c , is taken by the diagonal reinforcing steel. The stresses in the diagonal reinforcing steel and shear stresses in the concrete under the DBE and factored OBE loading conditions are provided in Table 5.2-20.

The diagonal rebar spacing is sufficient to keep the rebar stresses within acceptable limits. The higher stress level in the rebar is 47 psi under the factored OBE condition at a point 25 ft above the mat. The average yield stress of all No. 14 diagonal rebars used in the BVPS-1 containment has been determined to be 58,000 psi, significantly above the guaranteed minimum yield stress of 50,000 psi.

The 8 ft horizontal spacing of diagonal rebars is carried to the spring line at El. 811', considerably above the ACI-63 code nominal allowable shear stress elevation at which diagonal bars are no longer required. These elevations are 715 ft for the DBE case and 751 ft for the factored OBE case.

The meridional and circumferential tensile membrane forces due to internal pressure are carried entirely by the horizontal, vertical and diagonal reinforcing steel as the concrete is assumed not to have any tensile membrane forces. Below the aggregate interlock values of 40 psi and 60 psi for the factored OBE and DBE, respectively, the concrete resists the inplane shear force. The diagonal reinforcing, then, is assumed to carry shear forces only above the maximum allowables in the concrete.

It is assumed the liner does not assist in resisting pressure or seismic effects. If the stiffness of the liner is included in resisting the seismic shear, the force in the diagonals due to internal pressure and seismic shear would not exceed 37.2 ksi at any location for the factored OBE case and 30 ksi for the DBE case. The stress in the liner due to pressure and temperature effects and seismic forces would still remain below the liner yield strength.

The bars at 2 feet horizontal spacing are reduced to bars at 4 feet horizontal spacing at an elevation of 695 ft-1 inch. Bars at 2 feet spacing are not required at any elevation within the containment to resist the $(V_u - V_c)$ shear forces. The bars on 2-feet spacing were installed prior to completion of tests that were made to determine the value of shear resistance obtained by aggregate interlock. The cut off at El. 711 ft 5 inches was based on an analysis of the containment using an earlier response spectrum. Therefore the changes in diagonal rebar spacing do not represent true cutoff points. In actuality, the only steel required is No. 14 at 8-foot horizontal spacing, and these bars are carried up to the springline, 60 ft above the point where they are required to resist seismic shear.

The diagonal rebar is designed to resist the difference in shear between the DBE (V_u at any point) and the factored OBE (V_c at the base). The earthquake reinforcement is based on a conservative value of aggregate surface friction which is less than the shear stress associated with the factored OBE at the base. The diagonal rebar shear requirement plus the additional length needed for development of the bar in bond requires steel only to El. 741. However, the diagonal steel is carried to the spring line at about El. 811 as noted above.

Tangential shear resulting from the earthquake loading is resisted by the concrete and diagonal reinforcing bars. The requirements for diagonal reinforcement necessary for carrying the $(V_u - V_c)$ shear force are determined by an analysis based on treating the bar network of vertical, horizontal, and diagonal reinforcing as a continuum. This allows the establishment of a strain relationship between horizontal, vertical and diagonal reinforcement. The total force in the diagonals is taken as the sum of forces determined under a symmetric membrane state, due to internal pressure and vertical earthquake load ($V_u=0$) involving only horizontal and vertical strains, and forces determined under an anti-symmetric membrane state due to seismic shear ($V_u - V_c$) and involving only shear strains. The force calculated by the above includes the temperature effects in the liner.

It is felt that an allowed shear stress value of 53 psi for the aggregate surface friction associated with the factored OBE at the base can be justified by SWND-5. SWND-5 was submitted to the DRL on December 3, 1969. In addition, a supplementary report concerning cycling effect on tensioned rebars, SWND-5S⁽¹⁵⁾, was submitted to the DRL on April 10, 1970. These reports have demonstrated that an acceptable value of shear stress due to aggregate surface friction in the reactor containment shell would be 160 psi.

Figure 5.2-11, Seismic Shear Allowance for Aggregate Surface Friction, shows the recommended allowable unit membrane shear stress in SWND-5 (160 psi) versus the conservative 1.25 OBE shear values (53.5 psi) allowed for the concrete versus the actual calculated values for aggregate interlock required to resist tangential shear.

The diagonal steel reduces liner stresses induced by seismic forces, but slightly increases liner stresses which are caused by positive temperature variations inside the containment. Liner stress is below allowable values under DBA conditions.

As a means of ensuring that the concrete for the BVPS-1 containment is representative of that used for the test covered by SWND-5, specific requirements as hereafter noted were applied. The concrete has a specified strength of 3,000 psi in 28 days. A good quality, No. 467 coarse aggregate, is used. The material is tested for abrasion in accordance with methods in ASTM C131⁽¹⁶⁾ and meets the requirement in ASTM C33⁽¹⁷⁾ for maximum percent loss by weight. Also, the coarse aggregate is tested for soundness in accordance with ASTM C88⁽¹⁸⁾.

In addition to seismic forces, the cylinder and dome are analyzed for internal pressure loads and base bending moments and transverse shears. These bending moments and transverse shears are the discontinuity forces obtained from the mat analysis for each pressure condition. These edge forces are distributed along a shell meridian from formulas found in Article 114 of Reference 3. Membrane hoop and meridional design forces include the effect of the liner temperature increase associated with the DBA condition. This effect is a pressure acting on the concrete shell due to inhibition of liner expansion. Discontinuity forces which exist at the containment springline are determined by enforcing deflection compatibility through the use of formulas found in Article 114 and 116 of Reference 3. Under DBA condition the containment is assumed to be completely cracked at this location.

Tornado loading on the dome is analyzed according to Reference 1. Discontinuity stresses at the junction of the dome and the cylinder are calculated using the wind pressure distribution from this paper.

The tornado loading on the cylinder is investigated in accordance with Reference 4. With this approach, a statically indeterminate circular ring of unit width is analyzed for the wind load on the structure. The wind load is resisted by tangential shear which produces bending moment, axial load and shear around the ring.

No torsional loading on the containment from tornadoes has been investigated, since the necessary friction or surface drag is considered to be negligible. Overturning of the structure due to tornado is not a factor because the dead load of the structure is sufficient to overcome the wind force without the vertical reinforcement being subjected to tension. The equivalent static force of wind is obtained from formulas in Reference 5.

The structure is also investigated for possible penetration by a flying missile during a tornado. The amount of penetration is determined from the modified Petry Formulae in Reference 5.

5.2.2.5.4 Reinforcement Pattern

At the lower part of the wall, additional vertical steel is added to the meridional reinforcement as required by the fixed base moment due to internal pressure resulting from the DBA. Due to the high shear at the base of the wall and the tight condition resulting if individual shear reinforcing bars were used, it was necessary to develop a shear reinforcement design utilizing a truss arrangement. This design permits a minimum of reinforcement interference during construction. Shear trusses are formed by welding flat bars inclined at 45 degrees with the horizontal to the vertical reinforcement at the interior and exterior faces of the cylinder wall, as shown in Figure 5.2-12. The trusses transmit the radial shear due to the DBA into the mat.

Radial shear varies from a maximum at the base where the foundation mat restrains movement of the wall and becomes normally zero about 18 ft above the mat, with minor reversals above that point. About 13 ft above the mat, the shearing requirements diminish to the point that individual Z-type shearing reinforcement can be used. The welded shear bars are terminated and individual shear reinforcement bars are provided as required above this point. Additional vertical steel is provided, as required by the reversed moment on the lower part of the containment structure in a region 30 ft above the mat.

Tests were conducted on six concrete test specimens which contain two full size inclined shear bars and associated reinforcing steel. The purpose of the test was to demonstrate that the shear bar trusses function satisfactorily when subjected to combined axial, shear and moment loading resulting from the DBA, and that straining of the various members are within elastic limits, resulting in nominal cracking during load application and closure of cracks after removal of load.

A report on these tests of the concrete specimens containing the shear assemblies, entitled "Shear Assembly Tests for Reactor Containment Wall - Beaver Valley Power Station - Unit No. 1," was submitted to the AEC during February 1970.

The results of the six tests proved the adequacy of the design and the ability of the shear system and the component parts to resist the design load of 0.9 yield or 45,000 psi as maximum rebar stress without excessive strain. Tests were also carried to the limit of the 200 ton capacity testing machine. For that tensile loading, the maximum rebar stress was about 50,000 psi. There was no evidence of yielding or failure of metal. The general crack width developed was less than 0.02 inches under capacity load and reduced to hairline condition upon relief of load.

The amount of radial reinforcing steel crossing the potential splitting zone between the vertical tension bars at the interior of the containment wall was computed by the formula,

$$v = V \times \frac{Q}{I} \times b \quad (5.2-6)$$

where: v = radial splitting shear intensity

V = total radial shear force

Q = statical moment

I = moment of inertia

b = width of section

The amount of radial rebar dowels was determined and tested during the concrete shear bar tests referenced above. As a practical means during the test, 3/4 inch diameter, Gr. 40 rebar was welded to the liner plate on 6 inch centers.

The tests with this arrangement showed elastic opening of the cracks due to splitting with a maximum stress of 50,000 psi in the tension rebar. Upon removal of load, the cracks closed to hairline widths.

In the BVPS-1 containment structure, 5/8 inch diameter, 70,000 psi yield radial dowels, 3 ft-10 inches long, are stud welded to the liner on 12 inch centers at the base of the wall and supplemented by 3/4 inch diameter, Gr. 40 rebar hairpins 12 inches on centers as shown in Figure 5.2-12 to 7 ft above the mat. The long studs replace the short anchor studs which tie the liner to the concrete shell as described in Section 5.2.4.

Above that point, the Z bar shear reinforcement was provided as required by the radial shear input. The supplementary hairpin steel is required at the base, as it is not practical to utilize the 5/8 inch diameter studs closer than 12 inches on centers.

The radial rebar and studs are fully developed at the splitting plane by bond in the compression area of the concrete at the outer face of the wall.

The specified strength of concrete at 28 days is 3,000 psi. The guaranteed minimum yield point of the No. 14 and 18 reinforcing bars is 50,000 psi. The guaranteed minimum yield point of the reinforcing bars used as stirrups and binders is 40,000 psi.

The stresses of the reinforcement associated with the various loading combinations do not exceed the yield stress multiplied by the load capacity reduction factor, ϕ (Section 5.2.2).

The concrete base mat is reinforced by radial and circumferential patterns of reinforcement rods in the top, and a rectangular pattern in the bottom, as shown in Figure 5.2-12. The reinforcement in the top of the mat is arranged to facilitate the placing of the wall reinforcement which extends into the mat.

The cylinder wall is reinforced by meridional (vertical) and circumferential (hoop) reinforcement, and is shown typically by Section "AA" in Figure 5.2-12.

The dome is reinforced by a pattern of meridional and circumferential bars.

The spacing between the meridional reinforcing bars in the dome constantly decreases as the bars proceed toward the crown. Therefore, more reinforcing bars are used in this area than are needed. Half of the bars are terminated beyond a point where more than twice the number required for design purposes are used. In the vicinity of the crown, the meridional bars are welded to a ring with full penetration butt welds. Anchorage for terminated bars is developed either by bond or by means of a mechanical anchorage. Since the mechanical anchorage depends upon bearing in the concrete there is no problem with the biaxial tension.

For bars terminated in the biaxial tension zone a reduced bond value for reinforcement which terminates in a tension zone is 75 percent of that allowed by ACI 318-63. The bond length for flexural steel anchorage is terminated in the manner which is conventional in reinforced concrete design practice, using the reduced bond value.

The steel liner is not considered as making any contribution to the structural integrity of the containment shell. However, the resultant composite action due to the anchorage of steel liner to the concrete shell does contribute and adds to conservatism of the containment design.

5.2.2.5.5 Penetrations

The penetrations through the containment wall are divided into the following three categories:

1. All Piping Penetrations 9 Inch Diameter or Less

No reinforcing is provided. The principle wall reinforcement is located to avoid interference with the penetration.

2. All Piping Penetrations Larger than 9 Inch Diameter

Reinforcing bars terminated at penetrations are replaced by at least twice the number of bars, one-half of these being placed on each side of the opening. Diagonal reinforcing bars are also provided around openings to take shear and diagonal tension.

The anchorage length of the additional bars that frame the openings is determined by using a conservative value for bond stress of 75 percent of that allowed by ACI-318-63. This method is consistent with the practice established and pressure tested at the Connecticut Yankee Installation at Haddam Neck and the Surry Unit 1 containment at Surry, Virginia.

3. Personnel Access and Equipment Access Hatches

Penetrations for the equipment and personnel hatches are analyzed by means of computer programs which utilize the finite/element method. These programs, because they maintain compatibility between the ring beam and cylinder wall, are used to supplement S&W's computer program, "Nuclear Containment Structure Access Opening," which analyzes by the method of virtual work, an isolated, doubly-curved beam.

The equipment hatch opening (14.5 ft ID) is analyzed using a computer program based upon the Ph.D. Thesis (June 1968) by C. A. Prato entitled "A Mixed Finite/Element Method for Thin Shell Analysis." The personnel hatch opening (7 ft-1 inch ID) which has a projecting ring on the outside of the wall, is analyzed by utilizing the 3-dimensional finite/element capability of STRUDL, since this program could more accurately investigate an eccentric ring beam. Results from both these programs were compared with the S&W computer program.

In order to obtain more meaningful results, the ring beam and cylinder wall for both hatches are assumed to be cracked, and to have the extensional stiffness of the reinforcing bars only. The analyses show that sizeable tangential (in plane) shears exist in the wall near the ring beam. These shears are resisted by special reinforcing bars which are placed parallel to the typical earthquake shear bars.

The ring beam is designed to resist the axial tension and shears resulting from the loading criteria listed in Section 5.2.2.1. The axial tension is assumed to be resisted by the reinforcing bars only. The shears, including torsional shear, are resisted entirely by stirrups placed radially around the penetrations.

In effect, any concrete resistance to tension and shear is neglected. The principle circumferential and meridional reinforcing bars, as designed, are extended to the inner face of the ring beam, hooked 90 degrees and cadwelded to each other, thereby, providing additional shear resistance additive to that provided in the design.

The normal pattern of membrane forces (meridional and circumferential) in the containment wall is disrupted in the region of the hatch openings. The redistribution of these forces is provided by the finite/element computer programs and extra reinforcement added to areas of marked deviation from the normal pattern.

The outline and reinforcement of the personnel and equipment hatches are shown in Figures 5.2-13, 5.2-14, and 5.2-15.

5.2.2.5.6 Reinforcement splices

Structural ductility is maintained by staggering critical splices to ensure that adverse effects of multiple splices in the same plane do not occur. Full-scale pressure tests were conducted in May 1967, on the completed concrete containment structure for the Connecticut Yankee Atomic Power Plant and in May 1972 for Unit 1 of the Surry Power Station, where cadweld splices were used in a similar manner to that used for this project. These tests showed no stress concentrations or lack of structural ductility. Locations of splice groups were not discernible from inspection of the test crack patterns. The locations of the splices for the tested containment structure, which are described below, are generally followed.

Splices in individual reinforcing bars performing the same function in the foundation mat are, in general, not less than 4 ft apart. A minimum spacing of 2 ft is permitted in a limited number of cases where dictated by the reinforcement pattern. Vertical bars in each face of the vertical containment wall and dome are placed in groups of 20 bars of each length. Three groups are employed, each of which has a minimum differential length of 6 ft. Thus, no adjacent groups in the same face or adjacent bars in the opposite face of the wall have splices closer than 6 ft vertically. All individual horizontal circumferential bars, including those in the dome, have their joints staggered by at least 3 ft in all directions. The tangential shear stubs in the mat are designed to have the joints staggered 4 ft minimum in groups of 10. At the temporary construction openings, both the vertical and horizontal reinforcement splices are staggered a minimum of 12 inches.

All splicing is according to ACI Standard with one exception. At penetrations when it is necessary to cut No. 18 bar, additional bars are added to the extremities of the cut-out and a bond value 75 percent of that allowed in ACI-318-63 is used. This applies to all "Class B" category containment penetrations (Section 5.3.2). The bond value is reduced sufficiently to provide a conservative design.

5.2.3 Materials

5.2.3.1 Structural concrete

The structural concrete of the reactor containment structure, including the mat, wall, and dome, is ultimate strength design concrete with a specified 28-day compressive strength of 3,000 psi.

The concrete mix contained fly ash as a replacement of a portion of the Portland Cement, and a water reducing admixture. Concrete exposed to weathering also contained entrained air. Calcium chloride was not permitted in the concrete mix.

The measuring, mixing, and placing of concrete and the cleaning, bending, and placing of reinforcement complied with the standards of the American Concrete Institute except as otherwise stated in ACI-301,⁽¹⁹⁾ ACI-304,⁽²⁰⁾ and ACI-305.⁽²¹⁾

Trial concrete mixtures and the selection of mixes were in accordance with ACI-301 Chapter 3, Method 2. The testing of concrete used in construction was as set forth in Section 5.2.5.2.

All cement was an approved American brand conforming to ASTM C150,⁽²²⁾ Type II low-alkali. This was locally available cement and was suitable for the containment structure because of its moderate heat of hydration, improved resistance to sulfate attack and low alkaline content. The manufacturer furnished certified copies of mill tests, showing that the cement met or exceeded ASTM specification requirements for Portland Cement.

Tests of the mortar strength of the cement was performed by an independent testing laboratory as necessary to establish that it was in compliance with the specification.

Fine and coarse aggregate conformed to ASTM C33. Aggregates were evaluated for potential reactivity in accordance with ASTM C289.⁽²³⁾ Aggregates containing any materials which cause excessive expansion of mortar or concrete were not acceptable. All aggregates were tested for compliance with the above requirements by an independent testing laboratory.

Mixing water was clean and free from injurious oils, acids, alkalies, salts, organic materials, or other substances deleterious to concrete or steel. The quality of water was the equivalent of that suitable for drinking and was tested periodically.

The fly ash was obtained from approved sources and fulfilled the requirements of ASTM C618⁽²⁴⁾ except that the loss on ignition was limited to not greater than 6 percent. Initial and periodic tests were made of the fly ash, in accordance with ASTM C311.⁽²⁵⁾

A chemical admixture meeting the requirements of Specifications for Chemical Admixtures for Concrete ASTM C494⁽²⁶⁾ was used to permit reduction of the water requirements for a given slump. This controls bleeding and drying shrinkage, thereby producing a denser and more durable concrete.

An air-entraining agent was used in all concrete exposed to the weather. The amount used was sufficient to entrain from 4 to 6 percent of air. This agent conformed to the requirements of ASTM C260.⁽²⁷⁾ The air-entraining agent was added in solution with a portion of the mixing water and batched by means of a mechanical dispenser capable of accurately measuring and recording the amounts used. Manufacturer's air-entrained cement was not used.

5.2.3.2 Porous Concrete

Porous concrete used under the base mat was formed by the omission of the fine aggregate from a standard structural concrete mix. For this type of concrete, fly ash was not used as a replacement for Portland Cement. The mix was designed to have a 28-day compressive strength of 1,000 psi minimum.

Water porosity tests of the porous concrete were performed during December 1962 by Thompson & Lichtner Inc. of Boston using 6 inch by 12 inch cylinders prepared in the laboratory by compacting the material in three layers with standard tamping rods. After curing the concrete cylinders, the amount of water flowing through the 12 inch length of concrete cylinders during a 3 minute period with a constant head of 4 inches of water above the top of each cylinder was determined. Results indicated water porosities of from 28 to 47 gpm per sq ft, depending upon the density of the cylinders.

The porosity determined by the laboratory tests indicated that the 4 inch porous concrete layer under the base mat provides adequate drainage for the emergency seepage removal system described in Section 5.2.1.

5.2.3.3 Reinforcement Steel

Reinforcing steel used in the containment structure, consisting of No. 11 bars and smaller, had a 40,000 psi guaranteed minimum yield point. The bars conformed to ASTM A615,⁽²⁸⁾ Grade 40 or equivalent.⁽⁴³⁾

Reinforcing bars, sizes No. 14 and 18, had a 50,000 psi guaranteed minimum yield point, conforming to ASTM A615 as modified by the chemical and physical requirements provided in Table 5.2-21.

A special marking stamp was rolled into all bars conforming to the special chemical requirements for the containment structure reinforcing steel. The stamp identified all bars that conformed to the specified requirements. Particular care was taken to check that all No. 14 and 18 bars placed in the reactor containment structure had the special markings.

All ingots for special chemistry rebars were traced to maintain identity and billets were stamped with identifying heat numbers. Each heat contained about 100 tons of steel. All bundles of bars were tagged with the heat number as they came off the rolling mill, and were put onto the storage racks. Random inspection of not less than 10 percent of the work was made by Quality Control personnel during the pouring of heats. This inspection was made to ensure that those test ingots from which the chemical analysis tests were made had been correctly identified, and to observe the chemical analysis tests. All test results were checked for compliance with the specifications.

All special No. 14 and 18 bars were visually examined for surface discontinuity at the mill.

Physical and chemical tests were made by the Supplier on both the 40,000 and 50,000 psi yield point bars for each heat. The shop tests of the Grade 40 bars were made on full size specimens. At first, the shop tests of the Grade 50 reinforcement bars were made on specimens 0.505 inch in diameter with 2 inch gage length. Later tests for Grade 50 rebars were made on full size specimens. Verified copies of the tests were provided by the Supplier.

The samples of the 50,000 psi bars, supplied by the Supplier for verification of properties, consisted of one, 2 ft long, full size test sample obtained from a finished bar for each heat. Physical and chemical analyses of the samples were conducted by an independent testing laboratory.

A standard 8 inch long tensile test specimen was prepared from the sample tested for physical properties. Machining chips were used for chemical analysis. The results were compared with the specified requirements and the mill test results in accordance with ASTM A29⁽²⁹⁾. These data, the specified requirements, the mill test results and the laboratory check analysis were recorded on a standard form and submitted to the BVPS-1 for permanent records. The fractured tensile test specimens were retained and submitted to BVPS-1 upon request. The remains of the sample were held in case further testing was required.

The mill reports of the Grade 40 reinforcing steel were reviewed and copies provided to BVPS-1.

Quality Control inspectors witnessed, on a random basis, the physical and chemical tests of the Grade 50 reinforcing steel in the mill. The inspectors also made random spot checks of the finished fabricated bars as directed by the Senior Quality Control Engineer (Appendix A).

5.2.4 Liner

The containment liner includes the liner plate, penetrations, insert plates, anchors and access openings.

5.2.4.1 Description

The liner is a continuous steel membrane, supported by and anchored to the inside of the containment structure. Its function is to act as a gastight membrane.

The basic shape of the containment structure consists of a cylindrical portion, anchored at its base to the foundation mat and closing the upper end with a hemispherical dome. The inside diameter of the containment structure is 126 ft and the interior vertical height is 185 ft measured from the top of the foundation mat to the interior apex of the dome.

The cylindrical portion of the liner is 3/8 inches thick, the hemispherical dome liner is 1/2 inch thick, the flat floor covering the mat is 1/4 inch thick, with the exception of areas where the transfer of loads through it requires a reinforced thickness. The bottom mat liner plate is covered with a thick layer of reinforced concrete that will insulate it from temperature effects.

The 1/4 inch thick bottom liner plates are assembled in place and are continuously welded at their periphery to cruciform steel inserts which are cast in the reinforced concrete base mat as shown in Figure 5.2-16.

The 3/8 inch thick liner served as the internal form for the concrete containment during construction. All liner seams are double butt welded, except for the lower 30 ft of the cylindrical shell liner plate where the liner plates are welded using a backing plate. Liner repair plate welds were made using a backing plate. Details are shown in Figure 5.2-17. The liner wall is continuously anchored to the concrete shell with concrete anchor studs.

The 1/2 inch thick hemispherical steel plate dome liner served as an internal form for the containment reinforced concrete dome during construction. All seams in the liner dome are double butt welded. The liner dome is continuously anchored to the reinforced concrete containment dome with concrete anchor studs.

The bottom to wall liner junction is made of 3/8 inch thick plates. The sections of the joint plates that are in contact with the bottom mat are attached to steel plates that are anchored in the concrete mat as shown in Figure 5.2-16.

The wall to dome liner junction is a double butt welded joint as shown in Figure 5.2-17.

The bottom liner is overlaid with an approximate 2 ft thick reinforced concrete slab that is anchored to the bottom concrete mat, through the steel liner, as shown in Figures 5.2-16, 5.2-18, and 5.2-19 except at incore instrumentation area and the sump areas which are covered by grating.

All welded seams in the mat, cylindrical liner wall, hemispherical dome, and liner penetrations were originally covered with continuously welded test channels, as shown in Figure 5.2-17. As discussed in paragraph 5.2.5.4, test channels were not installed on liner plate seams associated with the Steam Generator Replacement Project construction opening. The nondestructive examination of primary containment liner seam welds is described in Section 1.3.3.19.

All bolted closures are double gasketed with means provided for introducing a pressurized halogen gas between the gaskets so that the closures may be examined for leaks, using a halogen gas detector.

5.2.4.2 Liner Materials

The ASME Boiler and Pressure Vessel Code, Section III, is used as a guide in the selection of materials and fabrication of the steel containment liner. Ferritic materials for the reactor containment boundary were specified so that when the liner is exposed to the emergency, test, normal, and severe operational conditions, the corresponding and resultant stress level will be below the maximum stress level, at this temperature, permitted by the "cat" curve of the NRL report 6900.

The liner materials are ASME SA 537 Gr B (quenched and tempered) for the first 30 ft, starting at the mat level of the cylindrical portion. Repaired sections within the first 30 ft use ASME SA 537 Cl.1 plate. The remainder of the liner is built with ASME SA 516 Gr 60 (fine grain practice). The ASME SA 537 Gr B quenched and tempered has a specified minimum tensile strength of 80,000 psi, a minimum guaranteed yield strength of 60,000 psi and a guaranteed minimum elongation of 22 percent in a standard 2 inch specimen.

The ASME SA 537 Cl.1 plate has a specified minimum tensile strength of 70,000 psi, a minimum yield strength of 50,000 psi, and a minimum elongation of 22 percent in a standard 2 inch specimen.

The ASME SA 516 Gr 60 has a specified minimum tensile strength of 60,000 psi, a minimum guaranteed yield strength of 32,000 psi and a guaranteed minimum elongation of 25 percent in a standard 2 inch specimen.

The material nil ductility transition (NDT) temperature, for original materials, was tested not to exceed -20 F. The material NDT temperature for liner repair plate is also tested not to exceed -20 F as an information-only test. The plates of ASME SA 516 Gr 60 are heat treated for improved notch toughness and both materials are certified to the mechanical and chemical limits specified in the ASME code requirements.

The liner plates were ordered to conform with standard mill practice as given in ASTM A20⁽³⁰⁾ with regard to thickness tolerance.

Ferritic steel items, except backing plates and anchors, gas testing channels (formed shape), equipment hatch bolts and equipment hatch nuts were made to fine grain practice and normalized or quenched and tempered to the appropriate material specification. In addition, ferritic steel items, other than the above, were NDT temperature tested in accordance with the following:

1. Material 5/8 inch and thicker was tested by the drop weight test method in accordance with ASTM E208⁽³¹⁾
2. Material less than 5/8 inch thick was tested by the drop weight tear test method as developed by the U.S. Naval Research Laboratory (NRL Report 6300)
3. Material 5/8 inch and thicker has an NDT temperature no higher than -20°F
4. NDT temperature test data are available for information purposes only on material below 5/8 inch thickness
5. Plates from a given heat were heat treated together, have but one set of NDT temperature test data
6. Each plate from a given heat, that were individually heat treated, have one set of NDT temperature test data
7. Heat treatments of each plate or set of plates were described in detail and their NDT temperature were recorded on notarized documents submitted as part of the contract records
8. As part of the welding procedures, NDT temperature determinations were performed on the weld metal and heat affected zone for each different type welding procedure used. NDT temperature determinations were not performed for liner plate repair welding.

The steel liner material was tested and certified so that its mechanical properties met the requirements as specified in ASME Section III.

5.2.4.3 Load Combinations

The function of the liner plate is to act as a gastight membrane under any one of the conditions that can be encountered throughout the operating lifetime of the unit. The liner plate is designed to resist all direct loads and accommodate deformation of the concrete containment without jeopardizing leaktight integrity since stress levels are kept within fracture mechanics limits to minimize the probability of crack propagation. It is anchored to the concrete at sufficiently close intervals so that the overall deformation of the liner under the parameters derived from the Design Basis Accident will be essentially the same as that of the concrete containment.

The containment liner plate, penetrations, insert plates, anchors, and access openings were designed for the load combinations presented in Tables 5.2-13 and 5.2-14. The areas of analysis for penetration load combinations are shown in Figure 5.2-20.

The containment liner plate was designed using the Stone & Webster computer program ST-133 for orthotropic shells of revolution axisymmetrically loaded.

This program which is based on finite element analysis techniques can handle up to 900 elements, each one of which has to be described by its geometry, elastic properties, and loads and/or deformations. The computation results in the stress level for each element, which can be compared with the allowable limits.

The equipment and personnel hatches were designed for the emergency, test, and severe operational conditions as listed in Table 5.2-13.

The minimum design pressure of the containment is 8.0 psia. This pressure is the minimum operating air partial pressure minus the pressure drop due to the inadvertent operation of quench sprays. During this condition, the containment vacuum may be increased over normal operational pressure by the operation of the quench spray subsystem (Section 6.4). Assuming that the containment is at 108°F and is suddenly cooled to the quench spray water temperature of about 42 to 45°F, the vacuum pressure would be increased by about 1.4 psi. This results in the minimum possible containment pressure above 8.0 psia.

The containment liner is designed to withstand the minimum containment pressure within allowable stresses. The bottom mat liner is capable of withstanding an internal pressure of 8.0 psia.

The barometric pressure change due to the maximum hypothetical tornado is stated in Section 2. This atmospheric disturbance causes a decrease in the atmospheric pressure, which decreases the differential between the atmospheric pressure and the containment vessel ambient, thereby decreasing the potential for stresses in the containment.

5.2.4.4 Liner Plate Attachments

All loads, generated within the containment, except pipe loads at containment wall penetrations, are carried by the internal concrete structures. The reinforced concrete overlay slab provides anchorage and support for equipment located in the base of the containment structure.

Internal concrete structures are anchored firmly to the concrete base mat by lengths of 3 inch by 6 inch steel bars which are placed horizontally and extend through the bottom plate liner as shown in Figure 5.2-18. The main vertical reinforcing steel bars are welded to the top and bottom 3 inch faces of these bars, thus providing bar continuity without creating multiple penetrations through the liner. The reactor vessel support in addition to having the vertical reinforcing steel welded to these bars are inclined shear bar welded to individual 3 inch by 8 inch steel bars as shown in Figure 5.2-6. The reinforced concrete overlay slab is anchored to the concrete base mat through the steel liner plate in a similar manner, using 7 inch by 1/2 inch bars, as shown in Figure 5.2-19.

These anchor bars, called "bridging bars" form an integral part of the steel liner plate and conform to the material and workmanship specifications of the steel liner plate. All welded joints are covered by test channels and tested as are all other liner plate joints.

The polar crane is supported by the crane wall which is an integral part of the internal concrete structure.

Electric conduit, lighting fixtures, miscellaneous small piping and scaffold brackets used to support temporary scaffolding for painting and cleaning are attached to, and supported by, the steel liner plate.

5.2.4.5 Anchorage

Steel insert plates are used in the liner plate to attach brackets for the support of miscellaneous small piping, electric conduit and scaffolds. The loads derived from support of piping, scaffolds, or other miscellaneous equipment are transferred to the containment concrete wall through insert plates and its anchors. The anchors were designed in number and size for each insert plate to be within the limits specified for the liner anchor studs as shown in Table 5.2-13. The thickness of the insert plates were chosen to provide a rigid base for the studs and brackets attached to each side of the plates.

The liner plate is anchored to the concrete wall and dome with Nelson concrete anchor studs. The anchor studs are designed for the following two different loading conditions: 1) a subatmospheric containment pressure, which causes a tension load in all studs, and 2) after a DBA when the unbalanced studs, at the lower edge of the cylindrical portion will be subjected to a shear loading derived from the liner computer evaluation. The resultant shear due to earthquake results in negligible loads on the anchor studs. The stud anchors were designed for both loading conditions as shown in Table 5.2-13. The anchorage layout is in a diamond pattern. Pitch dimensions were determined by the procedure set forth in Timoshenko "Theory of Elastic Stability" for a cylindrical shell under combined axial and uniform lateral pressure. Tolerance on location of each stud is 1 1/2 inches in any direction from its theoretical location, as dimensioned on the erection drawings, in order to clear possible interferences with reinforcement bars or other embedded parts.

The bottom mat liner plate is covered with a thick layer of reinforced concrete, that will insulate it from temperature effects.

The design load on each stud is low compared with the stud load capability. Design spacing of these studs is such that it will require, when the lowest pressure is acting within the containment, a set of six adjacent studs not to perform properly, to load the remaining stud at the center of such area to the ultimate strength of the stud material.

To verify the capabilities of the anchor studs, tests were conducted at Northeastern University, Boston, Massachusetts, using 1/2 inch diameter studs and 3/8 inch thick plate. These tests showed that shear failure occurs in the stud adjacent to the weld connecting the stud to the plate. In no instance was the plate damaged. Tests conducted for the stud manufacturer under the direction of Dr. I. M. Viest indicate that, with the manufacturer's recommended depth of embedment of the stud in concrete, the ultimate strength of the stud material can be developed in direct tension.

The liner plate base is anchored to the concrete base mat by continuously welding the plates, at their periphery, to cruciform steel inserts which have been cast in the reinforced concrete base mat.

The shell anchorage attachments, in the first 14 ft from the mat, are deformed anchors 5/8 inch diameter by 46 inch long; above the 14 ft level and in the spherical dome, the anchorage attachment is 5/8 inch diameter by 6-9/16 inch long concrete headed anchor studs. These attachments are placed in the above mentioned diamond pattern, and cast in the containment as the concrete is poured against the liner.

Anchor straps are provided at the spring line of the dome. These straps are welded to the liner shell and cast in the containment concrete during its placement, as shown in Figure 5.2-17.

5.2.4.6 Construction Tolerances

The maximum plus or minus deviation of the containment shell and dome from true circular or spherical form, measured radially on the inside of the vessel does not exceed 3 inches.

The maximum deviation from true circular form measured between any two points 14 inches apart in a circumferential direction does not exceed 1/4 inch. The allowable deviation from true circular form does not affect the elastic stability of the containment liner plate because of the restraint provided by the studs tying it to the reinforced concrete shell.

The maximum deviation from a straight line between any two adjacent points 14 inches apart in the meridional direction does not exceed 1/4 inch.

All measurements are taken on parent metal and not at welds.

All welding procedures and tests required in the ASME Boiler and Pressure Vessel Code, Section IX for welding qualifications were met in the selection of weld rod material, weld rod flux, heat treatment, qualifying the welding procedures, and the performance of welding machine and welding operators who were engaged in the construction of the containment liner. The welding qualifications include 180 degree bend tests of weld material. These procedures ensured that the ductility welded seams were comparable to the ductility of the containment liner plate material.

5.2.4.7 Piping Penetrations

Services and communications between the inside and outside of reactor containment are performed through penetrations. These penetrations are classified as follows:

1. General piping penetrations thermally hot and cold, all non-electrical penetrations up to and including 3 ft 6 inches diameter typical piping penetrations, their part nomenclature and boundaries are shown in Figures 5.2-21 and 5.2-22
2. Electrical penetrations
3. Personnel access and equipment access hatches.

The reinforcement pattern for classes 1, 2, and 3 are discussed in Section 5.2.2.5.4.

General Piping Penetration

All primary containment piping penetrations consist of basic containment inserts, plus additional items as required for the individual services. All primary containment piping penetration inserts are anchored in the reinforced concrete containment wall so that loads can be transferred from the piping to the reinforced concrete.

For cold penetrations, the piping is welded to a plate flange, as shown in Figure 5.2-21, which is anchored to the containment concrete wall, so the loads can be transferred from the piping to the concrete wall.

Each insert carrying a thermally hot fluid in the piping (over 180 F) is equipped with two water cooled, cooling units; one located inside of the penetration (inner unit) encompassing the full length of the sleeve, and the other located on the outside of the penetration sleeve in close proximity to the liner as shown in Figure 5.2-21 and 5.2-22.

Each thermally hot penetration is designed with adequate space between the sleeve and the piping for the required pipe insulation and for the installation of the inner cooling unit.

5.2.4.8 Penetration Design and Thermal Protection

The analytical evaluation of the penetration discontinuities was done using a computer program entitled SHELL-1 a proprietary computer program, by S. L. Coy, which analyses axisymmetric thin shells of revolution under unsymmetric loading. This program was developed at Stone & Webster on the basis of Reference 32.

Temperature distribution at discontinuity areas exposed to operating conditions were evaluated using finite difference or finite elements techniques.

To determine the loading combinations that will act in any given penetration as shown in Table 5.2-14, it was assumed to have a pipe rupture in such line, at such point, that the jet reaction multiplied by the distance to the attachment plate, caused the pipe to completely yield in either bending or torsion. The thickness for the attachment plate and sleeve is designed so that when the assembly is exposed to such load, the stresses in the components will not exceed 90 percent of the yield strength of the material.

The stresses induced in the liner by these loads are only those due to the resulting distortions in the reinforced concrete containment wall and are minor in intensity.

The equipment and personnel hatches were designed to withstand the loading combinations detailed for the liner plate.

Materials for penetrations are listed in Figure 5.2-21. The NDT temperature of all materials used for the piping penetrations is not higher than -20°F.

The cooling units for the thermally hot penetrations will use water from the component cooling water system.

The component cooling water system has two lines supplying the penetration cooling units. One line circulates water through the outer unit; the other circulates water through the inner unit. The cooling water circulation pipes do not require any secondary penetration of the containment structure.

The inner unit limits the radial heat flow, resulting from convection and radiation from the thermally hot pipe penetration, which will keep the temperature of the concrete below 150°F. In addition, it controls the longitudinal heat flow, resulting from conduction from the same heat source, which will keep the temperature of the liner plate and temperature gradient along the sleeve within allowable limits and consequently, will result in values of thermal stress in the liner, plate and sleeve below the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Paragraph N-414.4.

The outer unit limits the longitudinal heat flow, as does the inner unit, providing an independent thermal protection of the penetration sleeve and liner.

In the event that the inner unit fails to operate, the other unit operating alone has the capability of thermally protecting the penetration sleeve and liner. The temperature of the concrete adjacent to the sleeve will slowly increase, but does not exceed allowable limits. In this situation, the temperature of the layer of concrete 4 inches away from the penetration barrel reaches 150°F in 70 hours.

For thermally hot penetrations and multiple penetrations per single sleeve, an attachment plate links the sleeve with the piping as shown in Figure 5.2-21. The sleeve is welded to a plate flange which is anchored to the containment concrete wall so the loads can be transferred from piping to wall through the attachment plate and sleeve.

The circumferential groove in the attachment plate, between the sleeve and penetration with its outside threaded connection, serves as a test chamber for the testing of the welds joining the attachment plate and penetration.

Access Openings

1. Equipment Hatch - The equipment hatch is designed, fabricated, tested and stamped in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class B, 1968 Edition through winter 1970 Addenda.

The equipment hatch (Figure 5.2-23) is a single closure penetration assembly approximately 8 feet - 4 inches in length and 14 feet - 6 inches in diameter (I.D.). The equipment hatch cover is mounted inside the containment structure and is provided with a two point suspension hoist and a sliding rail for storage. A positive locking device is furnished to prevent circular swing. The cover is double-gasketed with a leakage test tap between "O" rings. The "O" rings are pressure tested as required by Technical Specifications. A removable concrete missile shield protects the equipment hatch.

2. Emergency Air Lock - The emergency air lock is designed, fabricated, tested and stamped in accordance with the ASME B&PV Code, Section III, Class MC, 1971 Edition through winter 1972 Addenda. Pressure equalization valves, which are installed on the emergency air lock, are containment isolation valves.

The emergency air lock (Figure 5.2-23) is a subassembly of the equipment hatch, consisting of a double-closure removable penetration approximately 12 feet - 6 inches long and 5 feet in diameter, attached to the removable equipment hatch cover by bolted, flanged connections with double "O" rings. A leakage test tap is located between the "O" rings. A 30 inch diameter opening is located at each end of the lock for personnel access.

The doors are mechanically interlocked, such that one door cannot be operated unless the other is sealed. Provisions exist to override the interlock by utilization of special tools or keys. An electromechanical interlock is provided to prevent operation of a door when a pressure differential greater than 0.5 psi exists across the bulkhead.

The operating mechanism includes pressure switches to indicate the sealed or unsealed status of the doors. These switches will energize indicating lights mounted on each airlock operating station and will provide a remote signal to the control room.

Each door is equipped with a valve for equalizing the pressure across the door. With mechanical linkage, only one valve at a time will be open and in no case can an equalizing valve be open on one door while the other door is operating. The equalizing valves are capable of equalizing pressure differential in the emergency air lock within 2.5 minutes, maximum.

The air lock will be operated manually, including operation of the mechanism and swinging of the doors. In addition to normal operation, the inside door is operable from outside the containment and the outside door is operable from inside the containment (utilizing the applicable remote handwheel).

All shafts penetrating the door or bulkhead have double packing and a test connection to permit periodic leak testing between the seals. Test connections are also provided for periodic leak testing between the double seals on each door. The entire emergency air lock can be independently pressurized for testing.

The emergency air lock will be unbolted from the equipment hatch and secured on a cart for removal prior to utilizing the equipment hatch.

Four pressure gages are provided: Two penetrate the bulkhead at the containment end of the air lock (one to measure containment vessel pressure and the other to measure interior air lock pressure). The other two penetrate the bulkhead at the other end of the air lock (one to measure interior air lock pressure and the other to measure the ambient pressure).

The emergency air lock has a capped emergency air connection which will also permit testing of the lock at any time without interfering with the normal operation of the plant. Two electrical penetration connections with six spare terminals for telephone and other circuits are provided.

The air lock barrel, outside of the equipment hatch, is enclosed full-length in insulation to conserve heat input at the containment end. The heat retained is sufficient to prevent loss of metal ductility during frigid weather.

3. Personnel Air Lock - The personnel air lock is designed, fabricated, tested and stamped in accordance with the ASME B&PV Code, Section III, Class MC, 1971 Edition through winter 1972 Addenda. Pressure equalization valves, which are installed on the personnel air lock, are containment isolation valves.

The personnel air lock (Figure 5.2-23) is a double-closure penetration, 15 feet long and 7 feet in diameter (I.D.). Each closure is flanged and double-gasketed with a leakage tap between "O" rings. The "O" rings are pressure tested as required by Technical Specifications. The entire personnel hatch can be independently pressurized for testing.

Each door (closure) consists of three major components: a nonrotating head, a rotating locking ring, and a fixed shell extension flange. Both doors are hinged and hydraulically latched and manually swung after the latch is released. The doors are interlocked such that if one door is open, the other cannot be unlatched. In addition to normal operation, provisions have been made to allow for opening or closing of the doors remotely from either inside or outside of the containment. A 6-inch sight glass is installed in each door.

Manually operated equalizing valves are installed on each side of both doors which allows pressure equalizing at an adjustable rate (5 minutes, maximum pressure) in the event of a power failure.

Each door of the personnel air lock is provided with an 18-inch diameter, double gasketed, emergency manhole and cover. Each manhole is similar to a navy scuttle hatch (which is operated by handwheels on each side of the door). The design of the emergency manholes includes means of fast pressure equalization across the manhole with appropriate interlocks (i.e., a locking pin) to prevent the cover from blowing open.

Also provided are two electrical penetration connections, a capped emergency air connection, a walkway and two folding waiting benches inside the air lock. Each electrical penetration has at least six spare terminals for telephone and other circuits.

Electrical Penetrations

Electrical conductors penetrating the reactor containment vary in size. Each penetration group passes through 8 inch or 12 inch diameter pipe sleeves. The sleeves are welded into the 1-inch thick containment liner insert plate with a test channel around the seal weld for periodic halogen leak testing, as shown in [Figure 5.2-24](#).

5.2.5 Construction Procedures

5.2.5.1 General

The containment structure was constructed in a cofferdam of interlocking steel sheet piling which was braced by reinforced concrete ring wales. Prior to the construction of the cofferdam, the overburden of entire building area, including that for the containment structure, was removed to the top of the bearing material at approximately Elevation 715 ft. The ground in the area of the cofferdam was then excavated to Elevation 709 ft for the construction of the first concrete wale. Using the wale as a template, the steel sheet piling ring wall was then driven to tip grade of approximately Elevation 671 ft with the top of the cofferdam at about Elevation 716 ft. The earth in the cofferdam was then excavated to a depth sufficient for the installation of the second wale. Excavation continued to about Elevation 679 ft-6 inches to allow installation of the reinforced concrete mat.

A compacted granular fill was next put in place. The fill was shaped to have a conical surface to provide drainage of the waterproofing membrane when later applied. A 2 inch thick working slab of unreinforced lean mix concrete was next installed. The waterproofing membrane was laid on top of the working slab as indicated in Section 5.2.7.3 and a 4 inch minimum unreinforced porous concrete layer with a level top surface was poured on it. The porous concrete, which served as a working surface for placing the mat reinforcement, covers only the area directly beneath the reinforced concrete mat which was poured later. The 10 foot thick mat was poured in six similarly sized segments to reduce cracking due to shrinkage and to facilitate construction.

Steel bridging bars, described in Section 5.2.4.4, and other miscellaneous steel inserts were cast in the concrete mat during construction.

Erection of the steel liner was started upon completion of the concrete mat. The 3/8 inch thick steel wall liner was erected to approximately Elevation 813 ft. Concurrent with the wall liner installation, the 1/4 inch thick mat liner plate was installed on top of the concrete mat. Upon completion of the mat liner and the wall liner to a convenient elevation above ground grade, all welds were checked for compliance with the approved weld inspection and gas test requirements. Work on the liner was then stopped until the interior concrete work in the containment structure was completed, the polar crane was erected, the concrete containment wall was completed to above ground grade, and the compacted granular backfill around the structure was brought to Elevation 735 ft.

The 1/2 inch thick steel dome liner was then completed above Elevation 813 ft. As the dome liner was completed, weld inspections and gas testing were performed.

The reinforced concrete wall above ground grade was completed following the construction of the steel liner. The reinforced concrete dome was constructed upon completion of the concrete wall.

The completed steel liner was tied to the principle wall reinforcement to prevent distortion during concrete placement. The exterior concrete forms were supported from the previously poured concrete.

Cantilevered steel strongbacks were used in the construction of the concrete dome. As a means of supporting the steel formwork, strongbacks were cantilevered from the completed concrete of the walls or the dome to approximately 50 degrees from the horizontal. Above this level the dome was poured without exterior forms, whereby the dome liner was tied to the rebar to support the liner against deformation due to the loading of the freshly deposited concrete.

The placing of concrete material and workmanship for the construction of the containment structure conformed to job specification and the following codes of practice on which the specifications were based:

1. ACI-301
2. ACI-304
3. ACI-305⁽³³⁾
4. ACI-306⁽³⁴⁾
5. ACI-318⁽³⁵⁾
6. ACI-347.

In addition to the above referenced codes, all applicable specifications of the American Society for Testing and Materials, referred to in Section 105 of ACI 301 and elsewhere in the codes referenced, were followed unless stated otherwise herein.

Quality control inspectors were at the job site at all times to ensure that all operations met the requirements of the job specifications and the standard codes of practice specified.

5.2.5.2 Concrete Mixing, Testing, Placement and Curing

Mixing

The concrete was mixed in an automatic batch and mixing plant erected at the site. Batching and mixing conform to the requirements of ACI 301, Chapter 7 for central mix concrete. The mixed concrete was transferred to agitator trucks for delivery to the placement site. Facilities were provided at the concrete plant for supplying, measuring and recording and mixing the concrete ingredients in the mix.

Testing

A credible outside testing laboratory designed the concrete mixes and performed the physical tests on cement, aggregates and admixtures. A field testing laboratory was established and staffed with qualified personnel who prepared and stored samples, performed the gradation tests of aggregates and the strength tests of the concrete cylinders, and reported to the Senior Quality Control Engineer in the field (Appendix A). Statistical quality control of the concrete was maintained (Appendix A) by a computer program. This program analyzed compression test strengths reported by the field testing laboratory in accordance with methods established ACI-214⁽³⁶⁾.

The design of the mixes conformed to ACI 301, Chapter 3 and resultant testing of concrete by the testing laboratory conformed to ACI 301, Chapter 16, supplemented by the three or more trial batches, each 1/3 cubic yard or more, with different water-cement ratios, were prepared for each of the mixes specified, to produce a desired design strength. Six test cylinders were made from each batch, cured and tested. Two of the cylinders were tested from each group at 7 days age and the remaining four specimens in each group at 28 days age. The results of the 7 day tests did not enter into the design of the mix, but were recorded for future comparison with the results of further tests made during the progress of construction. The results of the tests on the fully cured specimens for each water-cement ratio were averaged and curves plotted for the water-cement ratio against 28 day compressive strength. The water-cement ratio adopted for the concrete used in construction was determined from these curves. Upon completion of the preliminary tests and of plotting the curves, the mixes to be used in the various parts of the work were selected. No substitutions were made in the materials used on the work without additional tests and designs in accordance with the procedures outlined above.

The concrete was sampled and tested during construction in accordance with ACI 301 and ACI 318-63 to ensure compliance with the specifications.

Concrete strength tests of the job concrete were performed in accordance with ACI 301, Chapter 16, supplemented by additional procedures. Two or more sets of compression tests specimens were made for each strength of concrete placed during the first two days of placing concrete. Thereafter, one set of test specimens was made for every 250 cubic yard, or fraction thereof, of each mix design of concrete placed in any one day. In addition, one set of specimens was made whenever, for any reason, the material, method of concreting, or proportioning was changed.

The test specimens for compressive strength were 6 inch diameter and 12 inch long cylinders. Each set consisted of five specimens; one was tested at 7 days and three at 28 days age. The remaining cylinder was retained at the laboratory for further tests if the results of the previous tests made such a test desirable. Concrete test cylinders taken in the field were stored and tested on a controlled environment kept at a temperature of approximately 74 F.

The average compressive strength for any three consecutive 28-day cylinders were equal to or greater than the specified strength for the mix tested. Also, considerably fewer tests than the 10 percent allowable had values less than that specified.

If cylinders failed to meet concrete strength requirements at 28 days, then the extra cylinder was tested at 60 days. In case this cylinder failed to develop specified strength, specimens of hardened concrete were obtained and tested in accordance with ASTM C42⁽³⁷⁾.

Slump tests were made each time concrete test specimens were made, and as otherwise required by job conditions. The field test for slump was in accordance with ASTM C143⁽³⁸⁾. When air entrained concrete was being mixed, air content tests, in addition to the slump tests, were made at time of casting test cylinders.

Placement and Curing

As previously stated, the concrete was placed in accordance with ACI 301. However, the maximum slumps for the concrete mixes were greater than those specified in ACI-301 to expedite the placing of concrete in the heavily reinforced containment structure. The normal slump for concrete placed by conveyor was about 3 inches and that for pumped concrete was 4 1/2 to 5 inches. The temperatures of the concrete, when placed, met the requirements of ACI-305 and ACI-306 for hot and cold weather concreting, respectively.

Before placing another lift of concrete onto the previous lift, the surfaces of all construction joints were thoroughly cleaned by blasting with an air or water jet and prepared for the next placement.

Minimum concrete protection of not less than 3 inches was generally provided between principal reinforcing bars and the concrete surface of the exterior containment walls and dome. This protection agrees with the requirements of ACI-318, Paragraph 808 (d), which states that concrete protection for reinforcement shall in all cases be at least equal to the diameter of the bars. The principal reinforcing bar, No. 18, thus required a minimum of only 2 1/2 inches.

No. 9 and similar bars used as binders around main reinforcing steel at hatch openings have a concrete cover of 2 inches minimum thus conforming to ACI-318, Paragraph 808 (a).

In general, the clear distance between parallel reinforcing bars has been limited to one bar diameter 1 1/3 times the size of the coarse aggregate or a minimum of 1 inches as called for in Paragraph 804 (a), ACI-318. However, in certain areas such as the junction of the wall and slab of the steam generator cubicles, heavy intersecting reinforcement required that the clear distances between steel be cut down in localized areas. At such places, grout was used to ensure that maximum concrete coverage was obtained around the bars. As a further precaution, individual cases of bar tightness were reviewed to determine that pertinent bars had sufficient coverage to develop the design stress in the bars using conservative bond values.

Subsurface concrete of the base and the lower walls of the containment that are below probable maximum flood level are protected by a waterproof membrane.

Placing of concrete in the mat was done by conveyors, pumps, buckets or chutes. Close regulation was maintained over the amount of concrete deposited at any one place. It was vibrated and compacted into horizontal layers of about 18 inch thickness. To reduce segregation and voids within the concrete, it was placed so that a minimum of lateral movement was necessary after pouring.

Concrete pumps were used to place the concrete wall and dome of the containment. Distribution points from the pump pipe were located around the periphery of the wall about 5 ft on centers. The concrete was placed successively from distribution points to maintain continuity of the pour. The concrete was placed in approximately 18 inch layers and vibrated. The distribution points allowed minimal lateral movement of the concrete, and segregation of concrete aggregate was prevented.

All concrete placement equipment and methods were subject to quality assurance inspection.

All concrete in the wall and dome of the containment structure was poured horizontally around the entire circumference in uniform lifts of approximately 6 ft.

Each lift was built-up in approximate 18 inch thick layers which were placed at such a rate that the concrete surfaces had not reached initial set before additional concrete was placed.

Concrete forms were used for the exterior of the concrete dome to line 50 degrees above the horizontal. Above the 50 degrees line, the dome concrete was poured in three pours without the use of exterior forms. For the area where exterior forms were used, the concrete joints were in horizontal planes. The permanent steel liner on the inside of the dome served as the inner form for the entire dome.

Curing and protection of freshly deposited concrete conformed to ACI-301, Chapter 12. Where wood forms were used, and remained in place for curing, the forms were kept wet as required to prevent their opening at the joints and drying out of the concrete. The water used for curing was clean and free from any elements which might cause objectionable effects. The use of curing compounds on any surface was strictly controlled. Curing compounds were not permitted on surfaces to which additional concrete was bonded.

5.2.5.3 Welding (Reinforcing Steel Only)

All reinforcing bars larger than No. 11, that were in tension, were spliced to develop the full section as required by ACI-318, Section 805 (b). In general, Cadweld Type "T" full tension splices were used to splice the No. 14 and No. 18 bars.

Individual Cadweld splices, including the sleeves, develop tensile strengths not less than 90 percent of the specified minimum ultimate strength of the reinforcing bar. The average failure value of the splices was found by tests to be well above the specified minimum ultimate strength of the reinforcing bar. These results meet the ACI limitation which requires that a tension joint develop at least 125 percent of the yield strength of the reinforcing bar and satisfy the requirements of Safety Guide 10.

The ends of the reinforcing steel bars to be joined by the Cadweld Process were saw or flame cut square. The end surfaces of the bars were thoroughly cleaned of all rust, scale, grease, oil, water and other foreign matter before the joint was made.

Only qualified Cadweld Process welders were used. For the earlier phases of construction, each welder, for qualification, was required to demonstrate to the Senior Quality Control Engineer his ability to make an acceptable fixed joint using the Cadweld Process. During the latter phases of construction, each welder was required to demonstrate to the Senior Quality Control Engineer his ability to make two acceptable joints using the vertical position. Each welder was, likewise, requalified after every 200 Cadweld splices. The requalification after 200 splices existed during the entire construction period. Requalification consisted of making a suitable weld under simulated field conditions.

All Cadweld Process joints were visually inspected. The visual inspection included inspection of the ends of the bars for dryness and cleanliness prior to fitting the sleeve over the ends, and inspection of the completed joint for properly filled joints to ensure that filler metal was visible at both ends of the sleeve and at the tap hole in the center of the sleeve.

Randomly selected splices, as listed below, were removed from the structure, together with suitable lengths of adjacent bars, and strength tested for compliance with the specification.

1. One production splice for first 10 production welds
2. One production and three sister splices for next 90 production welds
3. Two sister splices and one production splice for next and subsequent units of 100 production welds.

The test cycle was established separately for horizontal and vertical splices of each bar size. Sister splices were made in place next to a production splice under the same conditions.

Splicing was done by butt welding for the No. 14 and No. 18 bars, where space or other unforeseen requirements made tension splices by the Cadweld Process unsuitable and for all smaller bars where positive connections are required.

The ends of the bars to be joined by butt welding were prepared by sawing or flame cutting and dressed by grinding, where necessary to form a full penetration welded joint. Only qualified butt weld process welders were used. For qualification, the welder demonstrated his ability by making a reinforcing bar test weld in the horizontal fixed position, welding vertically up. Each test weld was sectioned through the center of the weld by machining or power sawing and the cross-sectioned surface was etched with a 10 percent solution of nitric acid and water. The evaluation of the acceptability of the test weld will be done by the Quality Control Engineer.

Welded splices conformed to the American Welding Society, AWS D12.1⁽³⁹⁾. Tack welding of bars was not permitted.

Welding of reinforcing bars was performed using the "Metallic Arc Welding Process" with coated electrodes. The filler metal for the Metallic Arc Welding Process was coated arc welding electrodes, Classification E-10016-D2 or E-10018-D2, in accordance with ASTM A316⁽⁴⁰⁾.

Welding inspection of the reinforcement joints was performed by qualified welding inspectors which consisted of visual inspection and nondestructive testing. Visual inspection of rebar welding was performed by a qualified Field Quality Control welding inspector as specified in AWS D12-1. Nondestructive testing consists of radiographic inspection done in accordance with S&W Company Quality Assurance Non-Destructive Testing Manual. Two percent of the butt welds were tested by this method.

5.2.5.4 Steam Generator and Reactor Pressure Vessel Head Replacement

The Steam Generator (SG) and Reactor Pressure Vessel (RPV) Head Replacement Project created and restored a construction opening in the Containment Building in accordance with administrative procedures and the design control program. The opening was used to facilitate the movement of original and replacement SGs and RPV head out of and into the Containment Building. In accordance with the ASME Section XI repair/replacement program, the opening was restored consistent with the original containment liner specification (BVS-136) with any exceptions reconciled to the original specification.

5.2.5.4.1 Codes and Specifications

Restoration of the containment opening was performed as a repair/replacement activity in accordance with the requirements of ASME Section XI, 1992 edition, 1992 addenda.

ACI 318-63 is the design code for the restored Containment Building structure. The restored structure meets all applicable design loads and load combinations required by ACI 318-63.

Concrete placement, curing, and repair were in accordance with ACI 301-99. The use of ACI 301-99 is in accordance with Section 2.2 of ANSI N45.2.5-74.

Concrete mix proportioning was per ACI 211.1-91 (reapproved 1997) in accordance with Table A of ANSI N45.2.5-74.

Bechtel specifications (References 44-52) address:

- Reinforcing steel procurement, testing and placement
- Cadweld[®] reinforcing steel splices procurement, testing and installation
- Concrete mix design, testing and placement
- Structural steel and materials procurement

5.2.5.4.2 Liner Restoration

Before cutting the steel liner plate, existing containment leak test channels running across the cut line were removed. Steel plate seal “dams” were installed over the open ends of the existing leak test channels to ensure that existing test channels were sealed from moisture intrusion. Leak test channels were installed over all existing liner plate welds during original construction of the containment liner plate as an aid to facilitate leak testing of liner plate welds. The test channels allowed the welds to be pressure tested as sections were completed and repairs made prior to pressurizing the containment for the original Safety Guide 18 Acceptance test. The leak test channels are no longer used and no credit is taken for the leak test channels in any containment structural integrity evaluations. The leak test channels perform no safety related function. Therefore, the removed sections of leak test channel were not restored and no leak test channel was installed over the new liner plate seam welds.

The cut section of the Containment Building liner plate was rewelded to the liner plate with a full penetration weld. The new liner plate seam welds were examined using NDE methods. Liner plate seam weld NDE included 100% Magnetic Particle (MT) examination of the welds, as well as spot radiography, in accordance with Reference 44. The use of MT on the new liner seam welds was a departure from the liquid penetrant (PT) examinations used on the original liner seam welds; however, the MT performed is in accordance with the original specification requirements.

A halogen leak test was performed on the original liner plate seam welds. Because the test channels were not installed, the halogen testing performed on the original liner plate seam welds was not performed for the new welds. The liner plate seam welds were leak tested by the vacuum box method. The NDE and testing of the liner plate seam welds was in compliance with the requirements of AEC Safety Guide 19.

Replacement material was purchased for the liner plate, Nelson studs and test channel seal dams. Reference 52 requires the liner plate material to be ASME SA 516, Grade 60 (or better), fine-grained and normalized. ASME Specification SA 516 states that ASME SA 516 material is identical with ASTM Specification A516, which is identified in plant Specification BVS-136.

5.2.5.4.3 Reinforcing Steel Restoration

The reinforcing steel bars cut during the creation of the construction opening were reinstalled using Cadweld[®] splices or welding, as required, in accordance with References 44, 48 and 49. Reinforcing steel bars that were damaged during the creation of the opening were repaired in accordance with References 44 and 47 or were replaced with reinforcing steel procured in accordance with Reference 46. New N14 and N18 reinforcing steel used for the Containment Building wall restoration conforms to ASTM A615 Grade 60 and meets or exceeds the additional physical and chemical composition requirements described in UFSAR Section 5.2.3.3 for the Containment Building structure existing reinforcing steel.

Welded splices were used if space limitations did not permit the use of Cadweld[®] splices. Original welded splices were performed in accordance with AWS D12.1. However, new welded splices were performed in accordance with AWS D1.4, which replace AWS D12.1 in 1979. The use of this later edition of the code was permissible because the later code was based on more comprehensive provisions, additional research and experience including the following specific enhancements:

- AWS D1.4 lists ASTM A615 as an approved material,
- It uses carbon equivalent or bar size to determine preheat requirements,
- It acknowledges that low hydrogen filler materials could be used from heated containers and
- It requires a welding procedure or welder qualification specifically for reinforcing steel.

All welded splices were performed in accordance with qualified welding procedures included in the SPM (Ref. 44).

5.2.5.4.4 Concrete Restoration

The concrete removed from within the construction opening was restored with fresh concrete with a specified 28-day compressive strength of 3000 psi. Fresh concrete was qualified, tested, mixed, and placed in accordance with References 45, 50 and 51.

5.2.6 Missiles and Pipe Rupture

5.2.6.1 Interior Missiles

The systems located inside the reactor containment have been examined to identify and classify potential missiles. The basic approach is to ensure design adequacy against generation of missiles, rather than allow missile formation and try to contain their effects.

Catastrophic failure of the reactor vessel, steam generators, pressurizer, reactor coolant pump casings, and piping leading to regeneration of missiles is not considered credible. Massive and rapid failure of these components is incredible because of the material characteristics, inspections, quality control during fabrication, erection, and operation, conservative design, and prudent operation as applied to the particular component. The reactor coolant pump flywheel is not considered a source of missiles for the reasons discussed in Section 4.2.2.5. Nuts and bolts are of no concern, because of the small amount of stored elastic energy.

Components which, nevertheless, are considered to have a potential for missile generation inside the reactor containment are the following:

1. Control rod drive shaft, and the drive shaft and drive mechanism latched together
2. Certain valves
3. Temperature and pressure assemblies.

Design provisions to preclude missile damage from these sources are discussed later in this section.

Gross failure of a control rod mechanism housing sufficient to allow a control rod to be rapidly ejected from the core is not considered credible for the following reasons:

1. Full length control rod drive mechanisms are shop tested at 3,750 psig.
2. The mechanism housings are inspected and leak tested as they are installed on the reactor vessel to the head adapters.
3. Stress levels in the mechanism are not affected by system transients at power or by thermal movement of the coolant loops.
4. The mechanism housings are made of type 304 stainless steel. This material exhibits excellent notch toughness at all temperatures that will be encountered.

However, it is postulated that the control rod drive mechanism will become loose and it will be forced upward by the water jet. The following sequence of events is assumed. The drive shaft and control rod cluster are forced out of the core by the differential pressure of 2,500 psi across the drive shaft. The drive shaft and control rod cluster, latched together, are assumed fully inserted when the accident starts. After approximately 12 ft of travel, the rod cluster control spider hits the underside of the upper support plate. Upon impact, the flexure arms in the coupling joining the drive shaft and control cluster fracture, completely freeing the drive shaft from the control rod cluster. The control cluster would be completely stopped by the upper support plate; however, the drive shaft would continue to be accelerated upward to hit the missile shield structure provided. The characteristics of the drive shaft (with the disconnect rod) missile are given in Table 5.2-8.

Valve stems are not considered credible sources of missiles.

All the isolation valves installed in the reactor coolant system have stems with a back seat. This effectively eliminates the possibility of ejecting valve stems even if the stem threads fail. Analysis shows that the back seat or the upset end would not penetrate the bonnet. Additional interference is encountered with air and motor operated valves.

Valves with nominal diameter larger than 2 inches have been designed against bonnet-body connection failure and subsequent bonnet ejection by means of:

1. Using the design practice of ASME Boiler and Pressure Vessel Code, Section VIII which limits the allowable stress of bolting material to less than 20 percent of its yield strength
2. Using the design practice of ASME Boiler and Pressure Vessel Code, Section VIII for flange design
3. Controlling the load during the bonnet body connection stud tightening process.

The pressure containing parts except the flange and studs are designed per Code Class 1 requirements established by the ASME Boiler and Pressure Vessel Code. Flanges and studs are designed in accordance with ASME Boiler and Pressure Vessel Code, Section VIII.

The proper stud torquing procedures and the use of torque wrench, with indication of the applied torque, limit the stress of the studs to the allowable limits established in the ASME Boiler and Pressure Vessel Code. This stress level is far below the material yield. The complete valves are hydrotested per the ASME Boiler and Pressure Vessel Code. The stainless steel bodies and bonnets are volumetrically and surface tested to verify soundness.

Valves with nominal diameter of 2 inches or smaller are forged and have screwed bonnet with canopy seal. The canopy seal is the pressure boundary while the bonnet threads are designed to withstand the hydrostatic end force. The pressure containing parts are designed per criteria established by the ASME Boiler and Pressure Vessel Code.

Whereas, valve missiles are not generally postulated due to the above discussion, exceptions are the valves in the region where the pressurizer extends above the operating deck. Valves in this region are the pressurizer safety valves, the motor operated isolation valves in the relief line, the air-operated relief valves, and the air-operated spray valves. Although failure of these valves should also be considered incredible, failure of the valve bonnet-body bolts is, nevertheless, postulated, and provisions made to ensure integrity of the containment liner from the resultant bonnet missile. (To the extent practical, all valves are also oriented such that any missile will strike a barrier.)

The only credible source of jet-propelled missiles from the reactor coolant piping and piping systems connected to the reactor coolant system is that represented by the temperature and pressure element assemblies. The resistance temperature element assemblies can be of two types: 1) "with well" and 2) "without well". Two rupture locations have been postulated around the weld (or thread) between the temperature element assembly and the boss for the "without well" element and the weld (or thread) between the well and the boss for the "with well" element.

A temperature element is installed on the reactor coolant pumps close to the radial bearing assembly. A hole is drilled in the gasket and sealed on the internal end by a steel plate. In evaluating missile potential, it is assumed that this plate could break and the pipe plug on the external end of the hole could become a missile.

In addition, it is assumed that the welding between the instrumentation well and the pressurizer wall could fail, and the well and sensor assembly could become a jet-propelled missile.

Finally, it is assumed that the pressurizer heaters could become loose and become jet-propelled missiles.

The missile characteristics of the bonnets of the valves in the region where the pressurizer extends above the operating deck are given in Table 5.2-10.

The missile characteristics of the piping temperature element assemblies are given in Table 5.2-11. A 10 degree expansion half-angle water jet has been assumed. The missile characteristics of the piping pressure element assemblies are less severe than those of Table 5.2-11.

The missile characteristics of the reactor coolant pump temperature element, the instrumentation well of the pressurizer, and the pressurizer heaters are given in Table 5.2-12. A 10 degree expansion half-angle water jet has been assumed.

The principle design bases are that missiles generated in coincidence with a LOCA shall not cause loss of function of any engineered safety features or loss of containment integrity. (The containment is here defined as the containment structure, liner and penetrations, and the steam generator shell, the steam generator steam side instrumentation connections, the steam, feedwater and blowdown, and steam generator drain pipes within the containment structure.)

The polar crane wall, operating floor and refueling cavity walls serve as missile barriers between the reactor coolant loops and the containment liner. Openings in the charging floor required for ventilation or access are so located to minimize the possibility of missiles striking the area, and they are covered by heavy steel grating for further protection. Other than for the emergency core cooling system lines which must circulate cooling water to the vessel, the engineered safety features are located outside of the crane wall. The emergency core cooling system lines are routed outside of the crane wall so that the penetrations are in the vicinity of the loop to which they are attached. Physical separation together with barrier protection provided by the refueling cavity walls and various structural beams serve to minimize the potential for missile generation in one loop damaging adjacent loops. The barriers are designed to resist the postulated missiles. The portion of the steam and feedwater lines within the containment have been routed behind barriers which separate them from all reactor coolant piping.

The barriers are designed so that they will not be penetrated by the postulated missiles. The steam generator shell is also ample to resist penetration of the postulated missiles. The lower steam generator shell connecting lines are routed so that they are not in the direct path of the postulated missiles.

A missile shield structure is provided over the control rod drive mechanisms to block any missiles which might be associated with a fracture of the pressure housing of any mechanism. The missile shield is a steel plate and beam assembly that is located in the CRDM cooling plenum assembly above the CRDM seismic support platform. The missile shield is located as close as possible to the housing to limit the velocity of the ejected missiles, to minimize the probability of missiles missing the shield and striking the containment liner, and to minimize the probability of missiles ricocheting and damaging other control rod drive mechanism housings. The missile shield is approximately 42 inches above the top of the CRDM drive rod housing.

Analysis of a drive rod missile, which is the limiting CRDM missile, has shown that the drive rod will not penetrate the missile shield. The missile's impact energy is significantly less than the energy required to penetrate the missile shield.

To protect against valve bonnet missiles postulated in the region where the pressurizer extends above the operating deck, a barrier is built surrounding that part of the pressurizer. Penetration calculations are made to ensure that the postulated missiles would be stopped.

The ability of the reactor compartment walls and the operating deck is evaluated for the postulated instrumentation assembly and pressurizer heater missiles. Generally, the minimum thickness of the reactor compartment walls and the operating deck is 2 feet of concrete. Calculations based on this thickness and the given missile characteristics show that the critical velocity required to penetrate is at least twice the maximum anticipated velocity.

All missile barriers are also designed to withstand the dynamic impact loads. The energy method⁽⁷⁾, the momentum method⁽⁸⁾, or an empirical method⁽⁹⁾ is used.

Based on a kinetic energy and impact area comparison of all credible missile, the tornado generated missile (telephone pole) has the highest kinetic energy and relatively small impact area. The telephone pole has a kinetic energy in the order of 2.5×10^6 ft-lb impacting an area of 150 square inches, while all other credible internally generated missiles (such as flanges, bolts, motor/pump couplings) have kinetic energy in the order of 3×10^3 ft-lb impacting an area of 5 square inches. Calculations provided in Reference 41 have been performed for various types of missiles. For a missile with 3×10^3 ft-lb of energy and an impact area of 5 square inches, the concrete penetration is about 11 inches. A minimum of 24 inches is provided for protection against the tornado missile.

The effect of internally generated missiles, therefore, is less than half that of the tornado generated missile. Each piece of safety related equipment with the exception of the following is separated from all other equipment by a missile-proof cubicle:

1. Refueling water tank (QS-TK-1)
2. Leak detection fans
3. Quench spray pumps
4. Auxiliary feed pumps.

For conservatism, the missile protection in each of these cases is 24 inches, the same thickness required to protect against the tornado generated missile, and therefore, each piece of this equipment is protected from internal and other external missiles.

The effect of missiles on pipe runs and cable runs to, or from, safety-related equipment has not been considered because of the physical separation of the redundant trains.

For the equipment not protected by individual concrete cubicles, a review shows that this equipment is situated in large open areas with relatively few missile generating pieces of equipment nearby. In the case of the auxiliary feed pumps, equipment capable of generating missiles is not located in the missile-protected cubicle with the two electric drive and one steam drive pumps. Because of the high quality inherent to safety related equipment, missiles from this equipment are not considered. The two quench spray pumps share an exclusive cubicle and, as such, are not considered susceptible to internally generated missiles for the same reason as the auxiliary feed pumps.

The leak exhaust fans are located on the 768 ft-7 inch elevation of the auxiliary building. The fan location with respect to the missile generating equipment in the area and the low energies of feasible missiles for the surrounding equipment make it impossible to damage more than one of the leak detection fans. The redundancy of the system is considered adequate missile protection.

In the case of the refueling water system, missile protection is considered unnecessary. Since the tank is needed only in the short term after the Design Basis Accident, it is made seismic. It is not required in the long term, and since a tornado is not postulated for 24 hours after the accident, no missile protection is provided.

The possible missile generation (pressure reducing valve) from the propane storage facility and the effect on the safety-related equipment in the area are presented in Table 5.2-15.

A brief description of various types of pressurized gas containers listed in Table 2.1-14 is presented below:

1. The gas bottle storage facility is provided for storage of various gas bottles in a nonfire hazard area. The area is policed by plant security personnel regularly to monitor housekeeping and overall physical security.

All bottles are provided with safety relief devices for overpressure protection to meet Department of Transportation and Compressed Gas Association (CGA) requirements in the event of excess pressurization due to fire.

The safety relief devices fall under three different categories: (a) a frangible Type A rupture disk, usually of metal, which bursts at a predetermined pressure; (b) a fusible plug of suitable low melting material to melt at a predetermined temperature to permit escape of the gas; or (c) a combination frangible disk-fusible plug, designed to rupture at a predetermined pressure when the melting point of the plug has been reached.

The minimum flow capacity of these rupture devices, as specified in Compressed Gas Association pamphlet S-1.1 is to prevent the pressure from exceeding 80 percent of minimum cylinder burst pressure for DOT-42 cylinders and 4,500 psia for DOT-3E or BTC-42 cylinders.

Cylinders below a certain length (usually 65 inches) are required to have only one safety relief device per bottle. Bottles over this length require a minimum of two relief devices at opposite ends to provide adequate response to temperature conditions.

A single failure of the cylinder's overpressurization system has been considered for possible missile generation as presented in Table 5.2-15.

2. Each CO₂ fire protection system is normally maintained at 300 psia. Each system is provided with a bleeder valve which opens at 341 psia. If pressure continues to rise above the bleeder valve capacity, a safety relief valve will open at 357 psia. This pressure relief valve will offset an increase in pressure above 358 psia resulting from high ambient temperature conditions and almost all insulation being removed. Under loss of refrigeration, the bleeder valve can maintain self-refrigeration of the CO₂ unit. The two CO₂ units are located in areas of very low fire hazard.
3. The bulk hydrogen storage vessels and hydrogen bottle storage pad are located in areas not subject to fire hazards. The locations of the hydrogen bulk storage and Hydrogen bottle storage pad conform to OSHA requirements for outdoor storage of H₂ systems. (Table H.2, paragraph 1910.103 of Federal Register, "Occupational Safety and Health Administration," Volume 37, Number 202, Part A.)

The bulk hydrogen storage system is provided with safety valves to prevent pressure buildup in the hydrogen storage vessels beyond 110 percent of design pressure for system fill operations or rises in ambient temperatures.

A description of the approach used for (1) missile identification, (2) location of missile sources with respect to safety-related equipment, (3) size, weight, and kinetic energy of missiles, and (4) protection against missiles, is given as follows:

1. All safety related equipment was identified on a plant layout drawing.
2. It was noted that a large percentage of the equipment was isolated from all other equipment by two foot thick missile proof concrete walls (missile proof wall) capable of stopping a design basis accident (150 mph 1870# phone pole). Experience indicates that missiles from equipment such as pump motor couplings, pump drive turbine blades, compressor heads or valves and gage attachments to pressurized gas bottles do not have kinetic energies/impact areas equivalent to the design basis missile. In light of this experience, no further investigation was considered required for the isolated equipment unless studies of the unisolated equipment indicated that internally generated missiles with penetrating power approaching the design basis missile was possible.
3. For the equipment not isolated by missile proof walls (targets), possible sources of internally generated missiles were postulated. The source missiles were evaluated with respect to weight, impact areas and kinetic energy. Table 5.2-15 gives the postulated missiles for each target. In order to minimize repetitive calculations, the source missiles are typical and kinetic energy data for each type missile represents the maximum possible for that type. For example, on motor couplings, the heaviest weight, smallest size, highest torque and speed were selected and these parameters were used for establishing kinetic energy and impact area of a design missile coupling. These design missile-types represent a conservative, maximum value for missile parameters.

The missile with the highest kinetic energy-to- impact area ratio (KE/A), which is considered the most destructive missile, is a propane bottle relief device (Type 3) with KE/A of 7656 ft-lb per square inch. The KE/A of 9195 ft-lb per square inch for the design basis missile compared to the maximum KE/A from Table 5.2-15 justifies the "exclusion from further analysis" approach of the safety related equipment isolated by missile proof walls.

5.2.6.2 Exterior Missiles

The containment has not been analyzed for exterior missiles generated by hypothetical aircraft accidents, due to the site being located more than 5 miles from any airport (Table 2.1-7).

Tornado generated missiles discussed in Section 2.7 include one potential missile equivalent to a 35-ft long wooden utility pole impacting at a velocity of 150 mph.

5.2.6.3 Criteria for Protection Against Dynamic Effects Associated with a Major Pipe Rupture

The containment vessel and all essential equipment within the containment are adequately protected against the effects of blowdown jet forces and pipe whip resulting from a postulated pipe rupture of main steam, and feedwater (Class 2) lines. The criteria for adequate protection permits limited damage when analysis or experiment demonstrates that:

1. Leakage through the containment will not cause offsite dose consequences in excess of 10CFR 50.67 guidelines.
2. The minimum performance capabilities of the engineered safety systems are not reduced below that required to protect against the postulated break.
3. A pipe break which is not a loss of reactor coolant will not cause a loss of reactor coolant or steam or feedwater line break. Also, a reactor coolant system pipe break will not cause a steam-feedwater system pipe break and vice versa.

This level of protection is assured by adherence to the following design criteria.

Placement of Piping and Components

The routing of pipe and the placement of components minimize the possibility of damage.

The polar crane wall serves as a barrier between the reactor coolant loops and the containment liner. In addition, the refueling cavity walls, various structural beams, the operating floor, and the crane wall, enclose each reactor coolant loop into a separate compartment, thereby preventing an accident, which may occur in any loop, from affecting another loop or the containment liner. The portion of the steam and feedwater lines within the containment have been routed behind barriers which separate these lines from all reactor coolant piping. The barriers described above will withstand loadings caused by jet forces and pipe whip impact forces.

Other than for the Emergency Core Cooling System lines, which must circulate cooling water to the vessel, the engineered safety features are located outside of the crane wall. The Emergency Core Cooling System lines are routed outside of the crane wall so that the penetrations are in the vicinity of the loop to which they are attached.

Supplemental Protection

In those regions where the careful layout of piping and components cannot offer adequate protection against the dynamic effects associated with a postulated pipe rupture, restraints to prevent excessive pipe movement or special shielding is provided.

The careful layout of piping and components offers adequate protection against the dynamic effects associated with a postulated pipe rupture except in the case of the main steam and feedwater lines outside the crane wall.

The basis for selecting break locations in the main steam and feedwater systems, whose piping is similar to ASME Boiler and Pressure Vessel Code, Section III, Class 2 piping, is discussed below and is consistent with Regulatory Guide 1.46 "Protection Against Pipe Whip Inside Containment."

Since the probability of rupture is strongly related to stress, only a limited number of break locations are postulated. Supplemental protection is provided on the main steam and feedwater lines for breaks at locations described below:

1. At the two terminal points
2. At the point of maximum primary plus secondary stress
3. At any other point where the primary plus secondary stress exceeds 80 percent of its allowable; i.e., $0.8 (S_A + S_b)$.

The main steam and feedwater piping (similar to ASME Boiler and Pressure Vessel Code III, Class 2 piping) requires pipe break restraints in order to protect the integrity of the containment lines. There are six piping runs. Each run contains a total of three or more postulated break points.

Table 5.2-16 gives the pipe break locations postulated for the three main steam and three main feedwater pipe runs inside the containment building. Restraint locations are based upon what were, at the time of design fixing, the prevailing criteria for number and type of break.

Restraints offer good supplemental protection since pipe displacements are minimized and large kinetic energies are prevented.

The placement of the restraints will prevent excessive pipe displacements in the event of either a longitudinal split or circumferential break, or both, depending on the state of stress in the line.

In the area where the feedwater and the main steam piping penetrate the containment shell, the liner is also protected by an overlay of 1 1/2 inch thick quenched and tempered steel plate.

Methods of Analysis

Analyses are performed for pipe impact and jet impingement. In addition, major equipment supports are analyzed to ensure adequacy under postulated pipe rupture loads transmitted by attached piping.

For the purposes of design, unless otherwise stated, the pipe break event is considered a faulted condition, and the pipe, its restraint or barrier, and the structure to which it is attached are designed accordingly.

Restraints which require plastic deformation are based on 50 percent of ultimate strain.

The forces associated with both longitudinal and circumferential ruptures are considered in the design of supports and restraints in order to ensure continued integrity of vital components and engineered safety features.

The break area for both postulated break types is the cross-sectional area of the pipe. The break length for the postulated longitudinal breaks is assumed to be equal to twice the pipe diameter.

The analysis takes advantage of limiting factors on the blowdown thrust force, such as line friction, flow restrictors, pipe configuration, etc. A rise time is applied to the thrust force to simulate the crack opening time. A one millisecond rise time is assumed for circumferential breaks. For longitudinal splits, a rise time is computed based on the growth of a crack from a critical length to a length of two pipe diameters at a propagation rate of 500 ft/second.

Pipe Restraints

The restraints are designed with a gap sufficient to prevent interference with the normal thermal dynamic motion of the lines. This permits the pipe to acquire kinetic energy which must be dissipated upon impact into the restraint. This energy was conservatively set equal to the product of peak thrust times displacement. No energy dissipation mechanisms operating prior to impact, such as plastic deformation in the pipe, were considered. Static analyses of the deformation of the restraints and bolts provided the force displacements characteristics of the restraints. The area (energy) under this force-displacement curve was matched to the kinetic energy of the impacting pipe to determine the deformation and load. Based on recent, more detailed analyses, the conservatism of this design approach has been proven.

The restraints consist of a circular arch (or yoke) and a welded base support structure that is bolted to a supporting wall. These restraints are designed so that, by the use of adjustable shims, the gap between the pipe and the inner surface of the restraint is kept as small as practicable while still allowing free thermal expansion of the pipe during plant operation.

The barrier provided near the containment penetration is attached to the pipe penetration sleeve.

Equipment Supports

The internal structural system of the containment is designed to mitigate loading due to rupture in the main reactor coolant lines and the main steam and feedwater lines. Incident rupture is considered in only one line at a time. The support system is designed to preclude damage to or rupture of any of the other lines as a result of the incident. The snubber and key systems are designed to deliver rupture thrusts on the steam generator into the internal structural system. In determining the steam generator support reactions, the system is reduced to a dynamic model consisting of a suitable nuclear of masses and resistance elements. The dynamic problem is solved by numerical methods, using a thrust time history as loading. Resistance, dynamic amplification of the thrust, and rebound forces are calculated as a function of time. The reactor vessel and support system is similarly treated.

5.2.6.4 Pipe Whip Analysis

The analysis of the restrained piping within the containment was completed and the fabrication of restraints begun before any officially acceptable criteria for analysis was published. Subsequent to the completion of the analysis, analytical methods and criteria to be used in determining pipe whip analysis was transmitted to BVPS from the AEC. The analytical methods and criteria are provided in Attachment A to Section 5.2, "Pipe Whip Analysis Guidelines". The analytical methods and criteria used were similar to, but not identical with, those outlined in Attachment A. To facilitate a comparison, the original criteria is provided in Attachment B using the format of Attachment A and a point-by-point comparison is presented. Emphasis is placed on those criteria which differ.

5.2.7 Corrosion Protection and Coatings

5.2.7.1 Steel Liner

The exterior of the steel liner is not coated because it is in intimate contact with the concrete and has adequate protection from corrosion. The interior of the steel liner has an inorganic zinc coating with a white epoxy topcoating which provides protection for both normal operating and accident conditions.

5.2.7.2 Concrete and Structural Steel

All interior concrete and structural steel surfaces in the containment structure were given a coating suitable for service under DBA conditions. The steel floor grating is galvanized.

The coating system used on the preponderance of carbon steel surfaces within the reactor containment consists of a zinc-rich inorganic vehicle primer, topcoated with one or more coats of polyamide epoxy. Some items of equipment and certain other items, with relatively small amount of surface area, were coated with a straight organic system. Concrete surfaces are coated with at least two coats of polyamide epoxy.

The criterion for the selection of the above coating systems was the performance of these systems when subjected to tests simulating the environment anticipated within the containment in the event of a DBA. The coating systems indicated have demonstrated the ability to retain their integrity under DBA conditions, similar to those described in Section 7, in such a way as to ensure that the coating systems used will not compromise the efficiency of the safety systems.

The DBA simulation tests conducted for the purpose of validating the acceptability of the coating systems used were, for the most part, conducted in accordance with Section 4, "Procedures for Testing Coatings at Simulated DBA Conditions" of American National Standards ANSI N101.2.⁽⁴²⁾

Surface preparations for the various substrates within the reactor containment to which coating systems were applied will, for the most part, be in accordance with Section 6 "Surface Preparation for Shop and Field Application" of the proposed ANSI N101.2.

The exterior of the concrete containment was left untreated until the completion of the pressure testing. On completion of the pressure testing of the structure, the exterior concrete surface was given two coats of clear acrylic or silicone. This coating seals any tension cracks remaining in the concrete after the test and waterproofs the concrete surface.

5.2.7.3 Waterproofing Membrane

The waterproofing membrane is a butyl rubber sheet, 1/16 inch thick for walls and 3/32 inch thick for the mat. The finished membrane showed no evidence of ply delamination due to the manufacturing process.

The membrane was first placed on the concrete working surface at the bottom of the cofferdam and covered by a layer of porous concrete to provide a working surface for installing the mat reinforcement, as described in Section 5.2.5.1. After the structural mat was poured and the concrete wall of the containment completed above El. 730 ft, the exposed concrete surfaces were cleaned and filled to provide a smooth backup surface. The waterproofing membrane was joined to the portion of membrane previously laid under the mat to form a continuous watertight sheet. It was carried up to the sides of the mat, across the top of the projecting portion of the mat, and up the cylinder wall to contiguous structures and to approximately El. 730 ft in other areas. The membrane was continuously cemented to the mat and outer wall surfaces by a suitable bonding adhesive and rolled with a metal roller to remove air and to provide good adhesion.

All horizontal joints of the membrane were formed by a 3 inch minimum overlap of material, and all vertical joints by a 6 inch overlap of material. The overlapping material was continuously joined by a splicing adhesive.

The exposed surface of the completed membrane on the mat and the cylinder walls was covered by a fiberboard material to protect it during backfill. The fiberboard was cemented to the membrane, as necessary, to hold it firmly in place.

The butyl rubber membrane below the mat of the containment structure, referred to also as the "waterproof membrane," was sandwiched between two layers of concrete. To prevent sliding, porous concrete fill was laced between the mat of the containment structure and the sheet pile cofferdam to El. 686 ft 5 inches (Figure 5.2-1).

5.2.8 Containment Structural Instrumentation

5.2.8.1 Seismic Measurement

Instrumentation is provided to monitor input motion and behavior of critical elements of BVPS-1 under earthquake in accordance with Safety Guide 12 (Section 1.3). The instrumentation includes four triaxial force balance accelerometers, four recorders (one for each accelerometer) and a response spectrum analyzer (data acquisition and readout system). Each triaxial force balance accelerometer contains three accelerometers mounted on mutually orthogonal axes. The accelerometers are the transducer type, sensitive to a frequency range of 0.1 Hz to 25 Hz and capable of recording a maximum acceleration of 1.0 g full scale. Recording of the electric signals from the acceleration is by a data acquisition system. Recording of the accelerometers is initiated by a common trigger, set for a 0.01 g acceleration, at the data acquisition system so that whenever one interconnected recorder triggers, the other three recorders will trigger. This also guarantees time-synchronized event-data files. Recording continues until the level of motion drops below 0.001 g.

Two triaxial force balance accelerometers are installed in the reactor containment structure. One triaxial force balance accelerometer is located on a rigid support bracketed off a concrete column near the 4 ft-6 inch shield wall within 2 ft of the basement floor and the second triaxial force balance accelerometer is in a pocket in the operating floor slab. These locations were selected as giving information regarding the motion of the basement floor as near its center as practicable and information regarding motion of the superstructure which supports the principal equipment. The third triaxial force balance accelerometer is located at the foundation mat near the center of the auxiliary building, since this structure has significantly different period and dynamic characteristics from the reactor containment structure. The fourth triaxial force balance accelerometer is located in the main control room mounted on a raised seismic platform near the seismic monitoring cabinet. All instruments are oriented to the same azimuths. Seismic instrumentation packages are located in areas where they can be serviced during BVPS-1 shutdown. The recorders are located in the main control room where they are readily available. Provision is made for printout of the records so that after an incident the operator can determine the maximum level of acceleration experienced at the basement level of the reactor containment structure.

A visual readout will be made of all earthquakes of such intensity that they are recorded. The chart record is evaluated to assess whether the acceleration at foundation level exceeded 0.05 g.

A major service Seismic Category I structure is mathematically modeled and appropriate eigenvectors and eigen values tabulated. These models are so developed that, rising digitalized records for the earthquake from the foundation of the containment structure Accelerometer A, time-motion and response spectra various levels of equipment damping can be computed for the charging floor Accelerometer B at other levels within the containment structure. Comparison can thus be made between computed and recorded values for the charging floor which will permit re-evaluating the damping and elastic properties of the mathematical model. A similar comparison can be made from Accelerometer C in the auxiliary building with the peak motion recorded on the top floor and Accelerometer D for the control room building. Based on these several studies and using the revised elastic and damping properties of the several structures,

response spectra can be computed for other selected locations for comparison with amplified shock spectra used on the design of equipment, piping and other components.

In addition, a stand-alone high dynamic range strong motion accelerograph is provided on the top floor of the auxiliary building. This accelerograph includes a recorder, an internal triaxial force-balanced accelerometer, and a built-in timing system. As the sensors pick up ground acceleration or velocity signals, the recorder continuously monitors those signals to see if they satisfy seismic event detection criteria. When the signals satisfy these criteria, the recorder stores event data for later retrieval.

The accelerograph is oriented to the same azimuth as the four interconnected triaxial force balance described above.

Two peak recording accelerometers are installed in the containment structure. One is at the top of a recirculation spray cooler and one is at the top of a residual heat removal heat exchanger. Two additional peak accelerometers are installed: one on a recirculation spray pump in the Safeguards Structure and one on a primary plant component cooling water heat exchanger in the Auxiliary Building.

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ATTACHMENT A TO SECTION 5.2PIPE WHIP ANALYSIS GUIDELINES

This attachment consists of the acceptable criteria for pipe whip analysis as recommended by the AEC during the round of AEC Questions on the original FSAR. The criteria presented here are retained in the Updated FSAR to locate previously submitted information in one document. Attachment B to Section 5.2 discusses the application of these criteria to BVPS-1.

Analyses are required to assure that pipe motion caused by the dynamic effects of postulated design basis breaks within containment will not impact or overstress any structures, systems or components important to safety to the extent that their safety function is impaired or precluded. The analysis methods used should be adequate to determine the resulting loadings in terms of:

- a. The kinetic energy or momentum induced by the impact of the whipping pipe, if unrestrained, on a protective barrier or a component important to safety.
- b. The dynamic response of the restraints induced by the impact and rebound, if any, of the ruptured pipe.

The basis used to determine the magnitude of jet thrust force as required in dynamic analysis should be provided.

The methods of dynamic analysis specified in II and III are acceptable provided the following associated criteria are met:

I. Pipe Whip Dynamic Analysis Criteria

- a. An analysis of the pipe run or branch should be performed for each longitudinal and circumferential postulated rupture at the design basis break locations.
- b. The loading condition of a pipe run or branch prior to postulated rupture in terms of internal pressure, temperature, and stress state should be those conditions associated with reactor operating condition (normal and upset).
- c. For a circumferential rupture, pipe whip dynamic analysis need only be performed for that end (or ends) of the pipe or branch which is connected to a contained fluid energy reservoir having a sufficient capacity to develop a jet stream.
- d. Dynamic analysis methods used for calculating the piping or piping/restraint system response to the jet thrust developed following postulated rupture should adequately account for the effects of:
 - (1) Mass inertia and stiffness properties of the system,
 - (2) Impact and rebound (if any) effects as permitted by gaps between piping and restraint,

- (3) Elastic and inelastic deformation of piping and/or restraint and
- (4) Limiting boundary conditions.
- e. The allowable design strain limit for the restraint should not exceed 0.5 ultimate uniform strain of the materials of the restraints. The method of dynamic analysis used should be capable of determining the inelastic behavior of (piping/restraint) system response within these design limits.
- f. A 10% increase of minimum specified design yield strength (S_y) may be used in the analysis to account for strain rate effects.
- g. Dynamic analysis methods and procedures should consist of:
 - (1) A representative mathematical model of the piping system or piping/restraint system,
 - (2) The analytical method of solution selected,
 - (3) Solutions for the most severe response among the design basis breaks analyzed,
 - (4) Solutions with demonstrable accuracy or justifiable conservatism.
- h. The extent of mathematical modeling and analysis should be governed by the method of analysis selected among those specified by these criteria.

II. Acceptable Dynamic Analysis for Restraint Piping Systems

- a. Acceptable Models for Analysis for ASME Boiler and Pressure Vessel Code, Class 1, 2 and 3 piping systems are:
 - (1) Lumped-Parameter Analysis Model; Lumped mass points are interconnected by springs to take into account inertia and stiffness effects of the system, and time histories of responses are computed by numerical integration to account for gaps and inelastic effects.
 - (2) Energy-Balance Analysis Model; Kinetic energy generated during the first quarter cycle movement of the ruptured pipe as imparted to the piping/restraint system through impact is converted into equivalent strain energy. Deformations of the pipe and the restraint are compatible with the level of absorbed energy. For applications where pipe rebound may occur upon impact on the restraint an additional amplification factor of 1.5 should be used to establish the magnitude of the forcing function in order to determine the maximum reaction force of the restraint after the first quarter cycle of response. Amplification factors other than 1.5 may be used if justified by more detailed dynamic analysis.

- (3) Static Analysis Model - The jet thrust force is represented by a conservatively amplified static loading, and the ruptured system is analyzed statically. The amplification factor used to establish the magnitude of the forcing function should be based on selection of a conservative value as obtained by comparison with the factors derived from detailed dynamic analysis performed on comparable systems.

III. Acceptable Dynamic Analysis for Unrestrained Pipe Whip

- a. Lumped-Parameter Analysis Model as stated in II.a(1) is acceptable.
- b. Energy-Balance Analysis Model as stated in II.a(2) is acceptable. The energy absorbed by the pipe deformation may be deducted from the total energy imparted to the system.
- c. The assumptions used to guide the mechanism of pipe movement should be justified to be conservative.
- d. The results of analysis should be expressed in terms compatible with the approach used for verifying the design adequacy of the impacted structure.

IV. Flow Thrust Force

- a. The time function of the thrust force induced by jet flow at the design basis pipe break location should consider: (1) the initial pulse, (2) the thrust dip, and (3) the transient function.
- b. A steady state forcing function can be used when conditions as specified in e below are met. The function should have a magnitude not less than

$$T = K_p A \quad (5.2A-1)$$

where:

p = system pressure prior to pipe break

A = pipe break area

K = thrust coefficient.

Acceptable K values should not be less than the following:

- (a) 1.26 for saturated steam, water and steam/water mixture
 - (b) 2.00 for subcooled water-nonflashing.
- c. A pulse rise time not exceeding one millisecond should be used for the initial pulse, unless longer crack propagation times or rupture opening times, can be substantiated by experimental data or analytical theory.

- d. The transient function should be provided and justified. The shape of the transient function, IV a. (3) above, should be related to the capacity of the upstream energy reservoir, including source pressure, fluid enthalpy, and the capability of the reservoir to supply high energy flow stream to the break area for a significant interval. The shape of the transient function may be modified by considering the break area and the system flow conditions, the piping friction losses, the flow directional changes, and the application of flow limiting devices.
- e. The jet thrust force may be represented by a steady state function provided the following conditions are met:
 - (1) The transient function, IV a. (3) above, is monotonically diminishing.
 - (2) The energy balance model or the static model is used in the analysis. In the former case, a step function amplified to the magnitude as indicated in II.a(2) is acceptable.
 - (3) The energy approach is used for the impact effects of the unrestrained piping.

ATTACHMENT B TO SECTION 5.2PIPE WHIP ANALYSISI. Introduction

This attachment consists of a comparison of the actual criteria used (called the original criteria) to those recommended by the AEC and discusses the application of the criteria, to BVPS-1. This attachment also includes two appendices, as in the original FSAR, which discuss the Limita II and MARC computer codes.

II. Comparison of Criteria

In many respects, the original criteria provided herein are more conservative than those proposed in Attachment A, while in others they are less conservative. Since it is difficult to judge the total effect of these differences, one restraint, indicated by the original analysis to be the one most highly deformed, has been reanalyzed using the "Lumped Parameter Analysis Model" of Attachment A. The original and updated analyses are described and the results compared. On this basis it is shown that the restraints are not only acceptable, but provide a large margin of safety, when analyzed by acceptable criteria.

II.A Pipe Whip Dynamic Analysis Criteria

The pipe whip dynamic analysis criteria are Provided as Follows:

- a. Each longitudinal and circumferential postulated rupture at the design basis break location was analyzed. This is identical to the acceptable criteria recommended by the AEC.
- b. The loading conditions of a pipe run or branch prior to postulated rupture, in terms of internal pressure, temperature, and stress state were the design conditions for the piping. This is in excess of the requirements of Attachment A.
- c. For a circumferential rupture, both ends of the pipe or branch were investigated to determine whether the applied blowdown forces would produce significant displacements. If uncontrolled (plastic) motion resulted, pipe whip dynamic analyses were performed. This approach is in conformance with Attachment A.
- d. Dynamic analysis methods used for calculating the piping or piping/restraint system response to the jet thrust developed following postulated rupture adequately accounted for the effects of:
 1. The stiffness of the restraint. The stiffness of the pipe was ignored to conservatively ensure that all input energy could be absorbed by the restraint alone. Since the energy balance analysis model was used, intermediate inertia effects could be ignored. This approach is entirely consistent with, and possibly more conservative than, the acceptable criteria of Attachment A.

2. Gaps between the piping and restraint were considered in the analysis. Energy acquired by the pipe and imparted to the restraint upon impact was equated to the product of blowdown force times the gap. Rebound effects do not exist, as demonstrated by the analysis presented in Section III.B. These methods conform to the acceptable criteria.
 3. The elastic and inelastic deformation of the restraint was computed, as described in Section III.A. The resulting force-deflection curve was integrated to obtain the energy absorption characteristics of the restraint. While elastic and inelastic deformation of the piping was presumed, energy absorption resulting from this inelastic deformation was conservatively ignored. These criteria are thus more stringent than the acceptable criteria.
 4. Limiting boundary conditions were considered as required by Attachment A.
- e. Although the original criteria permitted up to 50 percent uniform ultimate strain of the materials of the restraints, this limit was not approached in the restraints as actually designed. The method of dynamic analysis (Section III.A) was capable of determining the inelastic behavior of the piping-restraint system response within these design limits. The low values of plastic strain computed for the restraints represents a very large margin of safety compared to the acceptable criteria of Attachment A.
- f. The specified design yield strength (S_y) was used in the analysis. Since the energy balance analysis method was used, this criteria alone provides approximately 10 percent more conservatism than the acceptable criteria.
- g. Dynamic analysis method and procedures consisted of:
1. A representative mathematical model of the pipe restraint system.
 2. An analytical method based on energy balance (Section III.A).
 3. Solutions for the most severe response among the design basis breaks analyzed.
 4. Solutions with demonstrable accuracy and justifiable conservatism.
- h. The extent of the mathematical modeling was suitable for the energy balance analysis.

All of the above criteria are equal to, or more conservative than, the corresponding requirements of Attachment A.

II.B Dynamic Analysis for Restrained Piping Systems

Of the three acceptable Models for Analysis presented in Attachment A, (Section II), type (2): Energy-Balance Analysis Model was used in the original analysis. Analysis types (1) Lumped-Parameter Analysis Model and (3) Static Analysis Model were not used and will not be discussed here. However, the Lumped-Parameter Analysis Model was used for a new analysis, conforming to Attachment A and justifying the restraint design. This is presented in detail in Section III.B.

(2) Energy-Balance Analysis Model.

The fluid blowdown force, considered as a constant, was multiplied by the pipe displacement to determine the input energy. Until the time of restraint impact this energy is stored as a kinetic energy and after impact it takes the form of both kinetic and strain energy. The input energy is:

$$E = F \cdot (g+x) \quad (5.2B-1)$$

where: F = blowdown force
 g = pipe-restraint gap
 x = restraint deflection

This is not a single valued function since x depends on the restraint characteristics.

The elastic-plastic force-displacement curve for the restraint was integrated to develop an energy absorption curve

$$E = f(x) \quad (5.2B-2)$$

The input energy, was equated to the strain energy to give the restraint deflection (x). From this, the reaction load and strains in the restraint were determined. A more detailed explanation of the analytical method is provided by a sample problem (Section III.A).

As demonstrated by a lumped-parameter analysis (Section III.B), rebound does not occur subsequent to impact so amplification factors are not required on the forcing function. The above method complies fully with the criteria of Attachment A.

II.C Dynamic Analysis for Unrestrained Pipe Whip

The main steam and feedwater lines within the containment are restrained for all postulated breaks. Therefore, this section is not applicable.

II.D Flow Thrust Force:

- a. The option of using a time function for the thrust force was not utilized. Thus this section is not applicable.
- b. A steady state forcing function was used. The value of the force was

$$F = pA \quad (5.2B-3)$$

where: p = design pressure

A = pipe break area

This force is the only significant variance between the original criteria and Attachment A. The acceptable criteria require amplification factors of:

- 1. 1.26 for saturated steam, water and steam/water mixture
- 2. 2.00 for subcooled water-nonflashing

The following factors prevent a non-conservative design:

- 1. The above amplification factors are ideal theoretical values. In practice, friction reduces their value.
 - 2. The amplification factors are for long term steady state. During the period of pipe acceleration and restraint impact, the force has not built up to the above values. Since the peak reaction loads during impact are well in excess of the above forces, the restraints will support these forces when they finally build up to steady state values.
 - 3. The restraints, as designed, have a large margin of safety since the permissible limits on strain have not been approached.
- c. No rise time was applied to the forcing function. This is in accordance with Attachment A.
 - d. Transient functions were not used so this criteria is not applicable.
 - e. The jet thrust force was considered as a steady state function (part b, above). The justifications are:
 - 1. During the period during which impact phenomena occur, the jet thrust is either steady or drops.
 - 2. The energy balance model was used. Justification is provided for not using an amplification factor. This is in accordance with II.a(2) of Attachment A.
 - 3. Not applicable.

III. Sample Problem

The original analyses of the main steam and feedwater pipe whip restraints inside the containment were based on the energy balance method. The results of those analyses indicated that the restraint which would be most highly deformed as a result of pipe impact was attached to the top of the crane wall near an elbow in a main steam line. The analytical method and results for this restraint are provided in Section III.A. A new analysis, using a lumped-parameter analysis model and conforming to Attachment A, is also presented to prove that this "worst case" restraint is satisfactory when analyzed to the new criteria.

The physical arrangement of the restraint and pipe is shown in Figure 5.2-54. For a circumferential break at one end of the elbow, the pipe is thrust against the restraint, pulling it away from its embedment. A one-inch gap between the pipe and restraint is assured by the placement of shims while the pipe is in the hot position.

III.A Energy Balance Model (Original Method)

The pipe-restraint interaction was analyzed using an energy balance method in which the work done by the blowdown thrust was equated to the strain energy of the deformed restraint. The solution provided the peak reaction load in the restraint and the strains in the component parts of the restraint.

The work done by the blowdown thrust is the product of force times distance. As described in Section III.B:

$$E_i = F \cdot (g+x) \quad (5.2B-4)$$

where: E_i = Energy input

F = Blowdown Thrust

g = pipe restraint gap

x = restraint deflection at impact point.

Section II.D gives for the blowdown thrust:

$$F = p_o A \quad (5.2B-5)$$

where: p_o = design pressure

$A = \pi r_i^2$ = break area

Thus:
$$E_i = \pi p_o r_i^2 (g+x) \quad (5.2B-6)$$

This is shown in Figure 5.2-55 for several gap dimensions. Since shims were used to ensure a one-inch gap between the restraint and the pipe in the hot position, only one of these curves is applicable to the actual design.

The basis for the energy absorption characteristics of the restraint was a multi-stage static stress analysis. The force-deflection properties of the restraint were determined using a mathematical model. Initially, all members were considered elastic, and a load was applied in the radially outward direction. The first region to yield was the arch structure at the point of load application. The restraining structure remained fully elastic up to 900 kips.

At yield, the mathematical model was modified by placing a pin at the node where the plastic hinge had formed. A moment, corresponding to the fully plastic moment across this section, was applied across the pin. This moment remained constant throughout the remaining analysis, so strain hardening was not considered. The load applied to this model was gradually increased until the bolts holding the restraint to the embedment yielded at 1300 kips.

The mathematical model was again modified to reflect the plastic properties of these long stainless steel bolts. For the bolts, which deform in simple tension, strain hardening was considered. These are the only components in which strain hardening was considered during the analysis of all the restraints.

The result of this multi-stage analysis for a radially outward load applied to this main steam pipe whip restraint is the force-deflection curve of Figure 5.2-57. Other curves were derived in a similar manner for all the restraints. In each case three loading conditions were considered: radially outward, tangential to the base, and outward to 45 degrees to the base.

By integrating the force-deflection curve, the strain energy-deflection relationship for this restraint was determined (Figure 5.2-58). Superimposed on this figure is the energy input curve for a 1 inch gap which was shown in Figure 5.2-55. From this graphical presentation it is readily seen that the energy absorbed by strain energy in the restraining structure equals the energy input by the blowdown thrust at a deflection of 1.6 inches. For this deflection, the maximum strain in the arch portion of the restraint is 0.004 and in the bolt is 0.044. These are 2 percent and 10 percent of the uniform ultimate strain of the materials used in these two locations. This is well below the allowable limit of 50 percent of uniform ultimate strain and indicates the large degree of conservatism even in this "worst case" restraint.

III.B Lumped-Parameter Model

The same restraint was reanalyzed using the lumped-parameter analysis model of Attachment A, Section II.a.1. The solution involved a three step analysis. The first analysis determined the time history of the blowdown forces. The next computed the local crushing resistance of the pipe at the restraint and the last step of the analysis involved the elastic-plastic dynamic analysis of the pipe-restraint system.

a. Time Dependent Blowdown Forces

The blowdown forcing function for use with the lumped parameter elastic-plastic dynamic analysis was derived for the specific break being analyzed (Figure 5.2-59).

Steam was treated as an ideal, single phase gas with a constant specific heat ratio of 1.3. Except for the case of steady state blowdown flow, the flow was assumed to be isentropic with negligible pipe friction for the prediction of the transient state forcing function. A graphical characteristic method was used to construct the state (u, c) and physical (x, t) diagrams, and the result was then used to calculate the forcing functions.⁽¹⁾

For the steady state blowdown forcing function calculation, the friction losses, such as pipe friction, were taken into consideration.

Although the pipe has numerous bends and straight segments, it was regarded as a straight pipe of total length L (which is the sum of lengths of the bends and straight segments) for the one-dimensional fluid mechanics analysis. The corresponding blowdown and wave reaction forces may be superimposed on the actual pipe layout to provide segmented time dependent loads for pipe dynamic analysis.

The analysis of the transient state forcing function was based on the method of characteristics. A general description of the method can be found in most gas dynamics textbooks. A graphical method was used to construct the state and physical diagrams for steam discharge via a pipe from the steam manifold⁽¹⁾⁽²⁾⁽³⁾. The result was then used to calculate the transient state forcing functions.

Immediately following the break, a decompression wave travels into the pipe towards the manifold. The fluid in front of the wave is at a state of:

$$u = 0 \tag{5.2B-7}$$

$$c = c_o,$$

where: u = velocity of the fluid,

c = speed of sound.

The fluid state behind the wave is at the sonic condition since the initial pressure was sufficiently high⁽⁴⁾:

$$\frac{u}{C_o} = \frac{C}{C_o} = \left(\frac{2}{\gamma+1}\right) = 0.8695 \quad \gamma = 1.3 \tag{5.2B-8}$$

The blowdown force can be calculated as:

$$\frac{F_B}{P_o A} = \left[P + \frac{\rho u^2}{g} \right] \frac{1}{P_o} = \frac{P}{P_o} + \frac{\rho C^2}{g P_o} = \frac{P}{P_o} + \frac{\rho}{\rho_o} \left(\frac{C}{C_o}\right)^2 \frac{\rho_o C_o^2}{a P_o} \tag{5.2B-9}$$

The pressure ratio across the wave is:

$$\frac{P}{P_o} = \left(\frac{T}{T_o}\right)^{\frac{\gamma}{\gamma-1}} = \left(\frac{C}{C_o}\right)^{\frac{2\gamma}{\gamma-1}} = \left(\frac{2}{\gamma+1}\right)^{\frac{2\gamma}{\gamma-1}} = 0.298 \tag{5.2B-10}$$

and the density ratio is:

$$\frac{\rho}{\rho_o} = \left(\frac{P}{P_o}\right)^{\frac{1}{\gamma}} = \left(\frac{C}{C_o}\right)^{\frac{2\gamma}{\gamma-1}} \cdot \frac{1}{\gamma} = \left(\frac{C}{C_o}\right)^{\frac{2}{\gamma-1}} = \left(\frac{2}{\gamma+1}\right)^{\frac{2}{\gamma-1}} \tag{5.2B-11}$$

Therefore, the blowdown force can be reformulated as:

$$\frac{F_B}{P_o A} = \left(\frac{2}{\gamma+1}\right)^{\frac{2\gamma}{\gamma-1}} + \left(\frac{2}{\gamma+1}\right)^{\frac{2}{\gamma-1}} \cdot \left(\frac{2}{\gamma+1}\right)^2 \cdot \gamma \quad (5.2B-12)$$

Equation 5.2B-12 reduces to:

$$(1+\gamma) \left(\frac{2}{\gamma+1}\right)^{\frac{2\gamma}{\gamma-1}} = 0.685 (\cong 0.7)$$

For frictionless flow, the blowdown force is constant until a return signal from the pressure source reaches the break. The approximate duration for this initial blowdown force extends from L_B/C_o to $1.6 L/C_o^{(5)}$. After that time interval, the fluid state at the exit changes gradually to its steady state value.

When the wave reaches the manifold, it is reflected as a compression wave. The boundary condition at the pressure reservoir on the steady-state ellipse is:

$$\left(\frac{C_i}{C_o}\right)^2 + \frac{\gamma-1}{2} \left(\frac{U_i}{C_o}\right)^2 = 1 \quad (5.2B-13)$$

which is the energy equation applying across the vessel pipe inlet. The boundary condition for this case is:

$$T_o = T_i + \frac{u_i^2}{2c_p} \quad (5.2B-14)$$

where i refers to the state at the inlet to the pipe.

If the steady state is reached, the flow in the pipe is uniform, and if the pressure in the pressure vessel remains high, then, the boundary condition at the break always lies on the sonic line:

$$\frac{u}{C_o} = \frac{C}{C_o} \quad (5.2B-15)$$

Then, from the critical flow condition, one has

$$\frac{U^*}{C_o} = \frac{C^*}{C_o} = \sqrt{\frac{2}{\gamma+1}} = 0.9325 \quad (5.2B-16)$$

and steady state blowdown force with $\frac{fl}{D} = 0$ is:

$$\frac{F_B}{P_o A} = \frac{P^*}{P_o} + \frac{\rho^* U^{*2}}{g} = \frac{P^*}{P_o} + \frac{\rho^*}{\rho_o} \left(\frac{U^*}{C_o}\right)^2 \rho_o C_o^2 / P_o =$$

$$\left(\frac{2}{\gamma+1}\right)^{\frac{\gamma}{\gamma-1}} + \left(\frac{2}{\gamma+1}\right)^{\frac{1}{\gamma-1}} \cdot \left(\frac{2}{\gamma+1}\right) \cdot \gamma \tag{5.2B-17}$$

Equation 5.2B-17 reduces to:

$$(1+\gamma) \left(\frac{2}{\gamma+1}\right)^{\frac{\gamma}{\gamma-1}} \cong 1.255$$

In actual application, the friction loss, is taken into account for predicting the steady state blowdown force. For most cases friction losses severely affect the steady state blowdown thrust. A curve for steady state blowdown with friction (Figure 5.2-60) was derived as follows:

Assuming a dimensionless pipe length $\frac{L_{max}}{D} = 54.0$ and with friction factor $f = 0.01$ then:

$$f \times \frac{L_{max}}{D} \cong 0.54 \tag{5.2B-18}$$

From tables for Fanno line⁽⁶⁾ with $\gamma = 1.3$, the inlet Mach Number is:

$$M = 0.6$$

With inlet Mach Number = 0.6, the stagnation pressure at the exit plane where the flow is accelerated to a Mach Number of unity due to friction effect is:

$$\frac{P_o^*}{P_o} \cong \frac{1}{1.193} \quad \text{or} \quad P_o^* \cong \frac{1050}{1.193} \cong 880 \text{ psia}, \tag{5.2B-19}$$

and the critical pressure at the exit plane where the flow is choked due to friction is

$$P^* = \left(\frac{P^*}{P_o^*}\right)_{M=1} P_o^* = 0.5457 \times 880 = 480 \text{ psia} > 14.7, \tag{5.2B-20}$$

where: $\left(\frac{P^*}{P_o^*}\right)_{M=1}$ (5.2B-21)

was obtained from the isentropic flow table (Reference 6). The blowdown force at the exit plane can be obtained from the impulse function of the table:

$$\frac{F_1}{F^*} = 1.110$$

and the impulse function F_1 is: (5.2B-22)

$$F_1 = \left[P_1 + \frac{\rho_1 U_1^2}{g_1} \right] A = P_1 A (1 + \gamma M_1^2) \tag{5.2B-23}$$

Therefore, the blowdown force is:

$$\begin{aligned} \frac{F_B}{P_o A} &= \frac{F^*}{P_o A} = \frac{1}{1.110} \frac{P_1}{P_o} (1 + \gamma M_1^2) \\ &= \frac{0.7962}{1.110} \times (1 + 1.3 \times (0.6)^2) = 1.053 \end{aligned} \quad (5.2B-24)$$

Repeated use of the above formulae result in the steady state blowdown curve (Figure 5.2-60).

Based on the above analysis, the blowdown forcing function at bend number 1 was calculated (Figure 5.2-61). Since the line is fixed at the penetration, forcing functions at the other elbows were not needed to compute the pipe whip.

b. Local Pipe Indentation

The local stiffness of the pipe was obtained by means of a large displacement, elastic-plastic analysis using the MARC program (Appendix 2). MARC library Element 8, which was used in this analysis is an isoparametric curved triangular shell element based on Koiter-Sanders shell theory. This shell element, developed by Dopuis⁽⁵⁾⁽⁷⁾⁽⁸⁾ is a generalization of shell elements derived from basis functions of polynomial form corrected by rational functions. The nine degrees of freedom associated with each node are three global displacements and their six derivatives with respect to two Gaussian coordinates. MARC Element 14, which is used to obtain the beam deflection, is a simple, closed-section, straight beam element. Six degrees of freedom; three global displacements and three rotations, are associated with each node. The elastic-plastic analysis follows the Prandtl-Reuss equations with isotropic strain hardening. The large displacement analysis makes use of a Lagrangian (initial coordinate) frame of reference and, therefore, the fundamental stress and strain measures are Kirchoff stress and Lagrange strain.

When the pipe impacts on the restraint, the contact area is constant in length in the longitudinal direction (Figure 5.2-62) and propagates in the circumferential direction as the applied load increases. To stimulate this contact type of loading (with spreading load area):

1. Analyses were done for the pipe with several different loading areas
2. Each loading was assumed to have a cosine distribution in the circumferential direction and a $q_o x^\alpha$ distribution in the longitudinal direction as shown in Figure 5.2-62 (α may range from $1.5 \leq \alpha \leq 1.8$).

The coordinate systems and the mesh for the quarter of the pipe (ABCD) are shown in Figures 5.2-62 and 5.2-63 with symmetric boundary conditions imposed along sides AB, BC, and CD.

For each different loading area, the nodal displacement in the loading direction (Z direction) was plotted against the increasing total load. A special loading (P_c) obtained from this figure is said to be the contact loading which produces a contact area the same as this given loading area, which minimizes the quantity e , where:

$$e = \sum \frac{|\bar{Z}^* - Z^*|}{\bar{Z}^*} \quad (B5.2-25)$$

where: $Z^* = Z + W$

$Z = r(1 - \cos\theta)$
is the distance of any point in the loading area (such as Q) to the vertex point (such as B)

$W =$ the displacement of that point in the loading direction

$\bar{Z}^* =$ average of Z^*

$r =$ diameter of pipe

The average displacement (\bar{w}) for this P_c is then obtained from the energy principle, i.e.:

$$P_c \bar{w} = \int A_p W dA \quad (B5.2-26)$$

The (P_c, \bar{w}) pairs obtained for different loading areas can then be plotted to obtain the PW curve for this pipe. To proceed further, the beam deflection which can be obtained using MARC library Element 14 may be subtracted from \bar{w} to obtain a force-local deformation relation for the pipe (Figure 5.2-64). An element with these properties was used to join the beam-element pipe with the restraint in the Lumped-Parameter Analysis model to account for the local pipe flexibility.

c. Lumped-Parameter Dynamic Analysis

The mathematical model of the piping system is shown in Figure 5.2-63.

Initially, the pipe is stressed by internal pressure, but remains in static equilibrium. This is simulated in the mathematical model by applying forces where the pipe is curved, such as at the elbows.

For a circumferential break, as the crack propagates, the load carrying metal area of the pipe decreases, so a force unbalance results. The load in the pipe at the break is assumed to drop linearly to zero in one millisecond. After the break, the forces exerted on the pipe by the fluid are determined by the time dependent blowdown force derived in Section III.B.a. The force was applied to the mathematical model as shown in Figure 5.2-66.

Pipe motion subsequent to rupture is analyzed by the use of an elastic-plastic lumped mass beam element computer code called LIMITA II, (Appendix 1). The analysis is divided into two stages, the first being the free motion of the pipe through the gap. The mathematical model is then modified to include the restraint and the connecting member simulating the local crush resistance of the pipe. (Section III.B.b)

The rebound of the pipe is determined by the sign of the force in the member connecting the pipe and restraint in the mathematical model. Therefore, the rebound effects are considered by connecting and disconnecting that member for impact and rebound respectively. Most of the analyses which have been done so far indicate no rebound occurred. Pipe positions before break, and at maximum deflection are shown in Figure 5.2-65. At the time of impact the pipe line is plastic from joint 4 to joint 8. The velocity of the impact point is 30 ft/sec, while the kinetic and strain energies of the pipe are 840 in-kips and 228 in-kips respectively. After impact the loading history of the pipe indentation member is shown in Figure 5.2-67. The impact point becomes hinged at 820 kips and the bolts yielded at 1060 kips. The pipe displacement at the impact point reaches its maximum (1.35 inch) at 1180 kips. It is noted that no rebound of the pipe occurred throughout this period. The maximum strains in the arch portion and in the bolts were 0.003 and 0.006 respectively, corresponding to 1.5 percent and 1.3 percent of uniform ultimate strain.

V. Conclusions

The criteria used for the original analysis of pipe whip and restraint impact agree in most respects with the criteria of Attachment A. For those criteria which differ, the original criteria are more conservative in most cases. Furthermore, with respect to the strain limit, the restraints, as designed, did not fully exploit the criteria. This assured a large margin of safety. Consequently, the restraints are adequate when judged against the criteria of Attachment A, as indicated by the sample "worst case" analysis.

References for Attachment B

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APPENDIX 1

LIMITA II CODE

Mathematical Model

LIMITA II is a 2-dimensional, nonlinear, transient dynamic analysis computer code. A structure is simulated as an assembly of discrete lumped masses connected by beam elements. Under any loading the equilibrium at each mass point is ensured by the equation of motion:

$$[m] \{\ddot{u}\} + [c] \{\dot{u}\} + [k] \{u\} = \{f(t)\} \quad (1-1)$$

Where:

$[m]$	=	Mass matrix
$[c]$	=	Damping coefficient matrix
$[k]$	=	Stiffness matrix
$\{u\}$	=	Displacement vector
$\{\dot{u}\}$	=	Velocity vector
$\{\ddot{u}\}$	=	Acceleration vector
$\{f(t)\}$	=	External load vector

The displacement, velocity, and acceleration vectors are comprised of all the nonrestrained movements of each mass point. The external load vector $f(t)$ is comprised of the external loads applied to the mass points in all nonrestrained directions of movement. The mass matrix is a diagonal matrix. An element of the matrix, m , is the mass associated with the i degree of freedom. An element of the damping coefficient matrix, c , is applied to the j velocity in the i equation of motion. An element of the stiffness matrix, k , is defined as the force necessary to hold the structural element from moving in the i degree of freedom when the j degree of freedom is given a unit displacement and all of the other degrees of freedom of the structural element are restrained from moving.

For the total structure, the governing equations of motion are:

$$[M] \{\ddot{u}\} + [c] \{\dot{u}\} + [k] \{u\} = \{f(t)\} \quad (1-2)$$

Where $\{\ddot{u}\}$, $\{\dot{u}\}$, $\{u\}$, and $\{f(t)\}$ are vectors comprised of all the degrees of freedom in the total structure and $[M]$, $[c]$, and $[k]$ are the assembly matrices of all the element matrices.

This system of second order differential equations is solved by a linear acceleration integration method, starting from some known initial state of the system at time 0.0. The nonlinear effects, such as plasticity and large deflections are included by varying $[k]$ and $[c]$ at each necessary time step.

In the numerical integration procedure, the following relations are used:

$$\begin{aligned} u &= f(u_t, u_{t-1}, u_{t-2}\dots) \\ u &= g(u_t, u_{t-1}, u_{t-2}\dots) \\ u &= h(u_t, u_{t-1}, u_{t-2}\dots) \end{aligned} \tag{1-3}$$

where f is a cubic function and the acceleration is a linear function across the time interval. Making these substitutions into equations (3) gives:

$$c_1[M] + c_2[C] + [K] \{u\} = \{F(t)\} + \{f[c],[M],\{u_{t-1}\},\{u_{t-2}\},\dots\} \tag{1-4}$$

where c_1 , and c_2 are functions of $(t-t_1)$ and $(t_1 - t_2)$, etc.

In LIMITA II, equation (4) is solved at each time point in the dynamic transient. Since $[M]$, $[C]$, and $[K]$ can be recalculated at each time point, they can vary with time in any desired fashion.

The Von Mises yield surface is used along with the Prandtl-Reuss flow relations. The stress strain curve is assumed isothermal bi-linear with isotropic hardening of kinematic hardening models.

For large deflection analysis, the geometry is modified (if necessary) at the end of each load increment so that the total loading is applied to the deformed structure at the next load increment. This procedure thus follows the large deflection load-deflection curve.

The computation procedures of the LIMITA II program are given in a flow chart (Figure 5.2-45).

Example and Comparison

The following is an example using the dynamic response of a 24 inch steam pipe with a circumferential break at an elbow.

Figure 5.2-46 shows the LIMITA II mathematical model; the whole pipe is divided into 11 elements with 12 joints. The mass of the pipe is lumped at each joint; forces as shown in Figure 5.2-47 are applied normally to the pipe at joints 2 and 3. The material properties of the pipe follow the stress strain curve shown in Figure 5.2-48.

The displacement and velocity responses at Joint 1 are plotted in Figures 5.2-49 and 5.2-50. The deformation shape of the entire pipe at 13.2 milliseconds is shown in Figure 5.2-51 at this time members 5, 6, and 7 are plastic in bending.

An identical model was analyzed by ANSYS computer code. The displacement responses (velocity output not obtained) were plotted against that obtained by LIMITA II. It is seen that the deviations are very small. At time, $t = 13.2$ milliseconds, there are also plastic strains in members 5, 6, and 7.

The ANSYS (Engineering Analysis System) computer program is a large scale general purpose computer program for the solution of several classes of engineering analysis problems. The program was developed and is being maintained and advanced by the personnel of SWANSON ANALYSIS SYSTEMS, INC. Analysis capabilities include static and dynamic; plastic, creep, and swelling; small and large deflections; steady state and transient heat transfer and steady state fluid flow. The matrix displacement method of analysis based upon finite elements idealization is employed throughout the program.

References to Appendix 1

1. H. C. Martin, Introduction to Matrix Methods of Structural Analysis, McGraw-Hill Book Co., New York (1966).
2. W. Weaver, Jr. Computer Programs for Structural Analysis, D. Van Nostrand Co., Princeton, New Jersey (1967).
3. J. M. Biggs, Introduction to Structural Dynamics, McGraw-Hill Book Co., New York (1964).
4. P. G. Hodge, Jr. Plastic Analysis of Structures, McGraw-Hill Book Co. (1959).

APPENDIX 2MARC STATIC CODE

The MARC nonlinear finite element analysis program, which is used to obtain the local pipe indentation stiffness, came into the public domain in December, 1971. It is written in FORTRAN IV in a general form with variable dimensions passed down to the subroutines. A library of elements is available directly in the program.

The elastic-plastic and large displacement analysis is effected in a series of piecewise linear increments. Creep and thermal effects which cause initial strains are analyzed as a series of steps in which an increment of initial strain occurs at the start of each step. Optional facilities enable the lowest eigenvalue to be obtained after each applied increment of load. This eigenvalue furnishes the factor which must be used to scale the next increment of load to cause collapse.

Controls have been added which allow the specification of loading or creep for a total number of increments or time steps, respectively. These controls are referred to as automatic load controls. The automatic load control for creep selects the time step for each increment so that the resulting stress and strain changes remain within a specified limit. A higher order step-by-step integration in time, known as the residual load correction, may be specified for creep problems. This residual load correction feature stabilizes creep solutions.

The behavior is the classical theory of isotropic, elastic-plastic time-independent materials, with a Mises yield criterion, isotropic strain hardening, temperature-dependent elastic properties, and equivalent yield stress. Perfect plasticity is assumed when no strain hardening is specified.

The theoretical basis of the computer code has been presented in References 1 through 9. The accuracy of the code has been demonstrated by comparison with both Theoretical and experimental results. A typical example is shown in Figures 5.2-52 and 5.2-53.

References to Appendix 2

1. P. V. Marcal, "Large Deflection Analysis of Elastic-Plastic Plates and Shell," Proceedings of the 1st International Conference on Pressure Vessel Technology, American Society of Mechanical Engineers and Royal Netherlands English Society, Delft (Sept. 1969).
2. P. V. Marcal, "Finite Element Analysis of Combined Problems of Nonlinear Material and Geometric Behavior," Proceedings American Society of Mechanical Engineers Computer Conference on Computational Approaches in Applied Mechanics, p. 133 (June 1969).
3. P. V. Marcal, "A Stiffness Method for Elastic-Plastic Problems," International Journal of Mechanical Science, Pergamon, Press Ltd., Vol. 7, pp. 229-238 (1965).
4. P. V. Marcal, and J. P. King, "Elastic-Plastic Analysis of Two-Dimensional Stress Systems by the Finite Element Method," International Journal of the Mechanical Sciences, Vol. 9, Pergamon Press Ltd., pp. 142-155 (1967).
5. P. V. Marcal, "Elastic-Plastic Behavior of Pipe Bend With In-Plane Bending", J. Strain Analysis, Vol. 2, No. 1, p. 86. (1967)
6. H. D. Hibbich, and P. V. Marcal, "Finite Element Analysis of Stiffened Shells Using a Combination of Shell and Beam Elements," Report TR72-1, Naval Ship Research & Development Center, U.S. Department of Defense.
7. P. V. Marcal, "Comparative Study of Numerical Method of Elastic-Plastic Analysis," AIAA Journal Vol. 6, No. 1, p. 157-158 (1967).
8. P. V. Marcal "Elastic-Plastic Analysis of Pressure Vessel Components," Proc. 1st Pressure Vessel and Piping Conf., ASME Computer Seminar, Dallas, Texas (September 1968).
9. J. S. Stoddart, and B. S. Owen, Meeting on Stress Analysis Today, Stress Analysis Group, Institute of Physics (1965).

5.3 CONTAINMENT ISOLATION SYSTEM

5.3.1 Design Bases

Except where noted, BVPS-1 conforms to Appendix A of 10CFR50, General Design Criteria for Nuclear Power Plants, Criteria 55 through 57 (Section 1.3 and Appendix IA).

The following are the design bases for containment isolation:

1. During accident conditions, at least two barriers are provided between the atmosphere outside containment and:
 - a. The atmosphere inside the containment
 - b. The reactor coolant system
 - c. Systems which could become connected to either the containment atmosphere or the reactor coolant system as a result of, or subsequent to, a loss-of-coolant accident (LOCA).
2. The two barriers consist of one of the following arrangements:
 - a. One normally closed, administratively controlled isolation valve inside, and one normally closed, administratively controlled isolation valve outside containment; or
 - b. One automatic isolation valve inside and one normally closed, administratively controlled isolation valve outside containment; or
 - c. One normally closed, administratively controlled isolation valve inside, and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or
 - d. One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.
 - e. A sealed system inside containment and one isolation valve outside containment which is either automatic or normally shut and administratively controlled, or capable of remote manual operation. A sealed system is one which is connected neither to the atmosphere inside the containment nor the reactor coolant system during normal conditions or following LOCA.

- f. In the case of the containment sump suction pipe and valve arrangements, a conservatively designed and fabricated single valve and suction pipe arrangement to prevent gross system leakage. A major portion of this special class piping is encased in the reinforced concrete containment. Minimum lengths of suction piping are employed between the single isolation valve and the point where the piping exits the concrete to ensure maximum integrity. This valve is equipped with a highly reliable remote operator. Provisions for detecting leaks in these lines and valves are provided.

The design of this portion of the installation is compatible with letters from the Advisory Committee on Reactor Safeguards to the AEC.⁽¹⁾⁽²⁾

- g. Details of containment isolation arrangements which differ in some manner from the specific arrangements described in 5.3.1.2 (a) through (f) above, such as instrument lines included in the exception to General Design Criteria 55 and 56, are discussed in Section 5.3.3.
3. The design pressure of all piping and connecting components within the isolated boundary is equal to, or greater than, the design pressure of the containment.
 4. The containment isolation system meets the single failure criteria described in Section 1.3.1.
 5. Operation of the containment isolation system is automatic.
 6. The containment isolation system components (piping, valves, penetrations, etc.) are protected from internally or externally generated missiles and water jets.
 7. All remotely actuated valves of the containment isolation system have their positions indicated in the main control room by separate limit switches installed directly on the valve actuator.
 8. Containment isolation system valves outside containment are located so as to require a minimum length of piping between the isolation valves and their penetrations. All outside containment isolation piping and valves are located in missile protected structures located contiguous to the containment structure.
 9. The containment penetrations are designed in a manner such that special operational test procedures, when used in conjunction with test connections (where required) can be used to provide the capability to periodically test the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.
 10. Instrumentation and adjunct control circuits associated with automatic valve closure shall fail in the position that provides greater safety upon loss of voltage or control air. Circuits which control redundant automatic valves are redundant to the extent that no single failure will preclude isolation.

11. Penetrations conform to the safety classes and codes discussed in Section 6.2.2. Note that those portions of any systems used to effect isolation of containment are piping Class I (Q1) or piping Class II (Q2) (Section 6.2.2).

5.3.2 Penetration Classifications and Isolation Valve Arrangements

The penetrations are classified according to whether the piping is connected to the reactor coolant system, the containment atmosphere, a sealed system, or whether the line is open during normal unit operation.

Class A Penetrations

Class A penetration piping is connected to the reactor coolant system (or connecting systems), or is open to the containment atmosphere and is used during unit operation. Any normal operating systems piping which could become connected to either the reactor coolant system or the containment atmosphere as a result of LOCA are also classified as Class A. Class A penetrations are provided in accordance with the arrangements described in Section 5.3.1.2(d) except as noted in Section 5.3.3.

Class B Penetrations

Class B penetration piping is separated from the reactor coolant system (or connecting systems) and the containment atmosphere by a membrane barrier (i.e., sealed inside containment) and is used during normal plant operation. Class B penetrations are provided in accordance with the arrangements described in Section 5.3.1.2(e).

Class C Penetrations

Class C penetration piping must remain open after a LOCA; accordingly, associated valves are not made to close upon failure. Class C penetrations, where they differ in some respects from the arrangements described in Section 5.3.2, are described in detail in Section 5.3.3.

Class D Penetrations

Class D penetration piping has a normally closed valve outside the containment, and is separated from the reactor coolant system by a closed valve or a membrane barrier. These penetrations are, therefore, closed during normal operation. Class D penetrations are provided in accordance with the arrangements described in Section 5.3.1.2(a), (b) and (d) except as noted in Section 5.3.3.

5.3.2.1 Conformance to the 1971 AEC General Design Criteria

Those AEC general design criteria covering isolation of lines penetrating containment are discussed in Sections IA.54 through IA.57. The penetration classifications specified in Subsection 5.3.2 conform with the following 1971 General Design Criteria:

- a. Lines in Class A and Class C, which are connected to the reactor coolant pressure boundary, are in conformance with General Design Criteria 55

- b. Lines in Class A and Class C, which are connected to the containment atmosphere, are in conformance with General Design Criterion 56
- c. Lines in Class B are in conformance with General Design Criteria 57
- d. Lines in Class D are in conformance with General Design Criterion 56
- e. All penetrations conform with General Design Criterion 54.

In order to provide the greatest degree of overall unit safety, it is necessary in certain cases to provide containment isolation arrangements which differ in some manner from the specific arrangements listed above. Such cases are described in detail in Section 5.3.3.

5.3.3 Description

Table 5.3-1 lists each line penetrating the containment structure and indicates the isolation criterion to which it conforms. As indicated, most isolation arrangements conform exactly with the 1971 General Design Criteria. The details of containment isolation arrangements which differ in some manner from the specific arrangements allowed by the General Design Criteria are indicated below:

1. Safety Injection Pump Discharge (Penetration Nos. 7, 33, 60, 61, 62, 96, and 113)

The safety injection system (Section 6.3) must be operated after a DBA to keep the reactor core covered with water following the accident (refer to Figure 6.3-8 for a diagram of the following valve arrangements). The boron injection (high head safety injection to reactor coolant cold legs) containment isolation valves are designed to be opened upon receipt of a safety injection signal. The remote manual valve affecting containment isolation in the low head safety injection header to the reactor coolant cold legs is normally open and normally remains open during the accident (refer to Table 5.3-1). The other valves affecting high head safety injection headers and in the low head safety injection headers to the reactor coolant hot legs are administratively controlled closed in the injection mode. The remote manual valve affecting containment isolation in the high head safety injection lines to the reactor cold legs (Penetration No. 96) is administratively opened in the cold leg recirculation mode. The remote manual valves affecting containment isolation in the low head safety injection to the reactor hot legs (Penetration Nos. 60, 62) are administratively opened in the simultaneous cold leg and hot leg recirculation mode.

The high head safety injection lines to the reactor coolant hot legs and the high head safety injection line to the reactor coolant cold legs are each provided with a normally closed, remotely controlled, motor-operated isolation valve located outside the containment, and a check valve inside the containment.

The boron injection line (high head safety injection to cold legs) is provided with two normally closed, remotely controlled, motor-operated isolation valves located in parallel in the line outside the containment and a check valve inside the containment.

Two of the low head safety injection penetrations are provided with check valves inside the containment in the lines leading to the reactor coolant hot legs. These valves are located downstream of the point at which the two lines form a common header and split into three lines, but upstream of the point where the three lines connect to the high head safety injection line to each of the reactor coolant hot legs. The third low head safety injection line penetrating the containment branches into three lines leading to the reactor coolant cold legs, each of which is provided with a check valve inside containment. These valves are located upstream of the point at which each line connects to the high head safety injection line to each of the reactor coolant cold legs. Outside the containment, the three low head safety injection lines are connected to the discharge lines from the two low head safety injection pumps. The two discharge lines leading to the reactor coolant hot legs are provided with normally closed, remotely controlled, motor-operated isolation valves. The discharge line leading to the reactor coolant cold legs is provided with a normally open, remotely controlled, motor-operated isolation valve. Before the refueling water storage tank (RWST) is empty following a DBA, valves in the low head safety injection (LHSI) system are closed to isolate the RWST from the containment. The safety injection pump discharge conforms to the intent of General Design Criterion 55. The only difference being the isolation valves located outside containment are opened during containment isolation either automatically or administratively to perform a post DBA safety injection and containment depressurization function.

These containment isolation arrangements conform with the design bases specified in Section 5.3.1, and also allow the safety injection system to perform its designed post DBA function.

2. Low Head Safety Injection Pumps and Outside Recirculation Spray Pumps Suction Lines (Penetration 66, 67, 68, and 69)

The suction lines for the low head safety injection pumps and the recirculation spray pumps are very conservatively designed to prevent gross system leakage. The major portion of this special class piping is buried in the reinforced concrete base mat and only a short length of piping exists between the mat and the isolation valves.

The motor-operated isolation valves (one in each line) at the suction of the outside recirculation spray pumps are normally open and remotely controlled. The motor-operated isolation valves (one in each line) for the low head safety injection pump suction lines are normally closed and remotely controlled. The remote operators used for these valves are designed to be highly reliable.

Assuming the worst possible single passive failure occurs to any suction line, as postulated in Section 1.3.1, the safeguards area suction valve pit becomes flooded. This provides a water seal between the containment and the outside atmosphere which prevents leakage into or out of the containment.

The low head safety injection pumps and outside recirculation spray pump suction lines connect directly to the containment atmosphere and are required to have two barriers for containment isolation, as described in General Design Criteria 56. The use of only one containment isolation valve, located outside containment provides a greater degree of reliability for supply of water to these pumps for operation following a DBA.

The design of this portion of the installation ensures a reliable source of water for the low head safety injection pumps, which are required as engineered safety features after a DBA, and meets the containment isolation design basis described in Section 5.3.1. This design is compatible with letters from the Advisory Committee on Reactor Safeguards to the USAEC.(1)(2)

3. Fuel Transfer Tube (Penetration 65)

The fuel transfer tube penetrates containment and is used to transfer fuel elements between the reactor and the fuel pool (Section 9.12.2). The tube is provided with a blind flange with dual O-rings inside containment and a normally closed, administratively controlled, manually operated valve outside containment.

The blind flange inside containment provides the containment isolation function for this penetration except when the plant is shut down for refueling. Blind flange testing is conducted in accordance with the Type B leak test requirements of 10 CFR 50 Appendix J (Section 5.6).

The gate valve outside containment is not required to be Type C leak tested due to the double barrier seal arrangement on the fuel transfer tube inside containment isolation flange (described above).

A test port and associated fitting are provided on the blind flange to ensure the space between the dual O-rings is successfully pressurized during Type B testing. During normal operation a plug is installed in the fitting to prevent potential leakage.

4. Pressurizer Dead Weight Calibrator (Penetration 110) Containment Leakage Monitoring and Containment Wide Range Pressure Monitoring System Open Taps (Penetration 55-2, 57-1, 97-3, and 57-2)

The one-eighth inch pressurizer dead weight calibrator line (Section 4.2) is used only infrequently during unit operation. This instrument line (Penetration 110) has no isolation valve inside the containment, but has two isolation valves in series outside the containment. The outside isolation valves are normally closed, administratively controlled, manually operated isolation valves. The pressurizer dead weight calibrator system is isolated from the reactor coolant system and the containment atmosphere inside containment by a pressure transmitter and capillary tubing. The diaphragm in the transmitter will withstand full reactor coolant system pressure in either direction, and the transmitter body and tubing will withstand external pressure equal to the containment design pressure. Since the diaphragm is flexible, any possible thermal expansion of the fluid in the line will deflect the diaphragm and not result in over pressurization. The containment

isolation barrier consists of a sealed shut manual valve outside containment located as close to the containment as practical.

The containment leakage monitoring system (Section 5.4.2.2) is not used during normal operation. Leakage monitoring equipment is only connected to instrument tubing (Penetrations No. 55-2, 57-1, 97-3, and 57-2) when needed for containment leak testing. The four instrument lines connect directly to the containment atmosphere. The containment leakage monitoring instrument lines are isolated outside containment and downstream of containment wide range pressure monitoring connections by a normally closed administratively controlled valve and pipe cap. This configuration meets the intent of Safety Guide 11.

The containment wide range pressure monitoring instruments (Section 5.4.2.3) are connected outside containment to the containment leakage monitoring system instrument lines (Penetrations No. 55-2, 57-1, 97-3, and 57-2). The instruments are used to monitor containment pressure for actuation of containment isolation signals. Due to the safety related function of these instruments and associated instrument lines, the containment isolation valve inside containment has been eliminated. Each of these instrument lines is three-eighths inch nominal tubing and has a one-eighth inch orifice as near as practical to the inside containment wall, consistent with the requirements of Safety Guide 11.

5. Containment Depressurization System Pump Discharge Lines (Penetrations 63, 64, 70, and 71)

The containment depressurization system (Section 6.4) must be operated after a DBA to limit post accident pressure and temperature in containment to less than design values and provide a means for long term containment heat removal. The valves affecting containment isolation in the system are therefore designed to be opened upon receipt of a containment isolation Phase B (CIB) signal, if not already opened.

Containment depressurization system pump discharge containment isolation valve arrangements conform to the intent of Criterion 56 (4), the only difference being the isolation valves located outside containment are opened during containment isolation either automatically or administratively to perform a post DBA safety injection and containment depressurization function.

Each of the quench spray pump discharge lines is provided with two CIB opened, motor operated, isolation valves arranged in series outside containment and a weight-loaded check valve inside containment.

Each of the outside recirculation spray pump discharge lines is provided with a normally open motor-operated isolation valve (which opens with a CIB signal if not already open) outside containment and a weight-loaded check valve inside containment.

These arrangements conform with the design bases described in Section 5.3.1, and also allow the containment depressurization system to perform its designed post DBA function.

6. Seal Injection Water to Reactor Coolant Pumps (Penetrations 35, 36, and 37)

The reactor coolant pumps seal water injection lines are each provided with a normally open, remotely controlled, motor-operated isolation valve located outside the containment and a check valve inside the containment. Motorizing the outside containment isolation valves minimized the possibility of accidental closure due to loss of air in the valve which would result in a loss of injection flow to a reactor coolant pump. To preclude the possibility of a spurious signal causing closure and, therefore, loss of injection flow to the reactor coolant pumps, automatic closure signals are not provided to the motor-operated valves. There is no need to automatically close these valves since:

- a. To do so unnecessarily could cause damage to the reactor coolant pump
- b. The fluid entering the containment via these paths either enters the RCS or the containment sump if the line is ruptured inside containment and is not irretrievably lost
- c. The operator can remote manually close the valves at any time.

Also, in determining the high head injection curves for a loss of coolant accident for the safety injection analysis, no credit need be taken for any injection flow through the reactor coolant pumps seal injection lines, i.e., the fluid injected to the reactor coolant system during a loss of coolant accident with minimum engineered safety features can be shown to be adequate if the charging/injection flow of one pump minus the reactor coolant pump seal injection flow is used. However, analytical techniques exist which permit taking credit for the flow injected through the reactor coolant pump seals and, hence, strengthen the position for not automatically closing the seal water injection isolation valve.

Upon loss of actuating power, these valves fail "as is" which is the position that provides greater reactor coolant pump safety during normal operation and has been shown above to be satisfactory during a LOCA. Failure of these valves "as is" is also acceptable because the seal injection lines:

- a. Do not communicate with the atmosphere outside containment
- b. Have safety classifications the same as those for engineered safety systems
- c. Have an internal design pressure and temperature rating at least equal to containment design pressure and temperature
- d. Have check valves provided inside containment for isolation should a single failure prevent the outside motor operated valves from operating.

7. Reactor Vessel Level Instrumentation System (Penetrations 95 and 109)

The Reactor Vessel Level Instrumentation System (RVLIS) provides indication of the reactor vessel fluid level or relative void content. Connections at the reactor vessel head, hot legs A and B, and the seal table provide the RVLIS sensing points. Tubing from these connections runs to high volume sensors which isolate the reactor coolant system from the remainder of the RVLIS tubing. The remainder of RVLIS tubing is filled with deaerated demineralized water.

Capillary tubing runs from the high volume sensors, penetrates the containment, and runs to the hydraulic isolators. Tubing then connects the hydraulic isolators to differential pressure transmitters.

Containment isolation is provided by the hydraulic isolator and connecting capillary tubing. The hydraulic isolator located outside containment serves as an air isolation valve on the containment building.

Within the high volume sensor is a check valve which will close under reactor coolant system pressure if the connecting capillary tubing fails.

Containment Hydrogen Monitoring System lines and RVLIS lines pass through containment penetrations 95 and 109. RVLIS lines that penetrate the containment structure conform to the intent of General Design Criterion 57.

8. Post DBA Hydrogen Control, Suction and Discharge to Containment (Penetrations 87, 88, 92 and 93)

The hydrogen recombiner suction lines have 2 inch manual ball valves in series outside containment which are normally closed and administratively controlled. These lines connect to the containment vacuum pump suction lines between the two in series auto trip isolation valves and containment penetrations 92 and 93. The hydrogen recombiner discharge lines have two 2 inch manual ball valves in series outside penetrations 87 and 88 which are normally closed and administratively controlled. The valves are located as close as possible to the containment wall. The piping meets the break/crack exclusion criteria set forth in Branch Technical Position MEB 3-1, Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment ^(15, 16).

9. Auxiliary Feedwater (AFW) Lines (Penetrations 76, 77 and 78).

Each of the three main feedwater lines has an AFW system branch connection located downstream of the main feedwater isolation valve and outside of containment. These AFW branch lines contain a simple check valve located upstream of another motor-operated valve. These motor-operated valves (MOV-FW-158 A,B,C) were previously credited as the containment isolation valves for the AFW system until it was determined that they did not meet the single (electrical) failure criterion, as discussed in AEC Regulatory Staff Position 36. When the power supplies to their motor operators were removed to satisfy

Position 36, the valves no longer met the criterion set forth for containment isolation.

These AFW system branch lines do not form a path for radioactivity release due to the existence of two physical barriers: the steam generators inside containment, and the check valves described above. Local manual operation of these check valves would not be needed to ensure their closure in the event of an accident, and only a differential pressure of about 2 psi on the containment side of the valve would be needed to shut them. Further, the AFW piping downstream of each check valve is usually full of water during normal operations, thus providing an additional barrier against the release of radioactivity to the environment.

There are other check valves and motor-operated valves (located in parallel) upstream of these single check valves which could provide backup isolation capability, if needed. Also, because of the need to assure secondary system integrity inside containment in the event of a LOCA the system has been seismically designed and missile-protected so that the rupture of the AFW system is not postulated to occur either concurrent with or as a result of a loss-of-coolant accident (LOCA). Additionally, a similar AFW system configuration was approved by the Staff for the North Anna Power Station, Units Nos. 1 and 2, on the same bases.

The containment isolation features described above are sufficient to achieve the underlying purpose of GDC 57.

10. Recirculation Spray Heat Exchanger River Water Radiation Monitor Sample Lines (Penetrations 83, 84, 85 and 86)

Radiation monitor sample lines are provided to sample each river water line at a location down stream of the recirculation spray heat exchangers, outside containment, and upstream of the containment isolation valve for the river water line. The radiation monitor sample lines are normally open and remain open (unisolated) following a design basis accident to allow rapid detection and isolation of any radioactive releases resulting from a recirculation spray heat exchanger tube leak.

In the event radioactive leakage is detected, the radiation monitor and associated high radiation alarm would provide indication of a recirculation spray heat exchanger tube leak, and alert the operator to take corrective action. Manual valves at the radiation monitor skids are accessible for local isolation of a sample line during accident conditions.

The use of locked-closed valves to isolate the sample lines would delay isolating a radioactive release due to a leaking recirculation spray heat exchanger tube, and the use of local manual valves will not result in a significant increase in the total offsite radioactivity release. In addition, the use of automatic or remote-manual valves would result in undue cost in comparison to the safety benefit to be derived. Thus, application of GDC 57 in this instance is not necessary to achieve its underlying purpose.

Shortly after a DBA MSLB, the temperature of the containment liner may be as high as 280°F. Although such high temperatures are short lived, it is possible that water trapped in the lines of the systems isolated by the containment isolation system may expand more rapidly than the associated piping. This could result in pressures exceeding the design pressure of the piping. To ensure that such overpressurization of isolated piping cannot adversely affect containment isolation integrity, a relief valve set to relieve at a pressure at or below the design pressure of the associated piping is installed in the few affected lines inside the containment between the containment wall and the inside isolation valve. These relief valves are designed to reseal when overpressure conditions subside.

5.3.4 Containment Isolation Valves

The containment isolation system conforms to Appendix A of 10 CFR 50, General Design Criteria for Nuclear Power Plants, Criteria 55 through 57, as modified by Section 5.3. Valve design and application details are provided in Table 5.3-1. The valves supplied by Stone & Webster were analyzed for stresses due to operating and seismic loads as discussed in Appendix B.2.2. The analysis of the Westinghouse supplied valves is discussed in Appendix B.3.

Containment isolation trip valves are designed to be operable under normal operating environmental conditions during the life of the plant and during seismic conditions. Those with a solenoid or air operator are designed to trip shut at the onset of a DBA and remain closed during post accident environmental conditions. Those valves with a motor operator are designed to "fail as is" at the onset of a DBA.

To ensure reliability, containment isolation motor-operated and trip valves were designed and procured to meet the minimum design requirements for containment isolation valves as specified in American National Standards Institute, ANSI B31.1, 1967⁽³⁾, ANSI B16.5⁽⁴⁾, and Manufacturers Standardization Society Standard Practice, MSS-SP-66⁽⁵⁾, with additional nondestructive testing of pumps and valves in accordance with ASME Boiler and Pressure Vessel Code (Draft issue) dated November 1968. The main feedwater isolation valves were designed to ASME III, Class 2, requirements. The selection of valve operators and types was made on the basis of past experience and best practices.

To ensure safe and reliable system operation, the following design features have been incorporated into the Containment Isolation Valves.

1. The design pressure of all the isolation valves are in excess of the containment design pressure.
2. Check valves, when used as containment isolation valves, are loaded to close against a 2 psi positive differential pressure.
3. Circuits which control redundant automatic valves are redundant to the extent that no single failure will preclude isolation.
4. The closure time for valves in the pipelines, which might have the potential of releasing radioactive elements to the atmosphere, have been limited to as small a period as possible consistent with the design of valves and operators.

Gate valves are used extensively for remote operated containment isolation valves because of their tight seating characteristics, essential deep stuffing box features for handling radioactive fluid, and availability in a larger size of pressure ratings and sizes.

Butterfly containment isolation valves are selected in lieu of gate valves in certain applications for their use in low pressure, large line sizes.

Globe valves are used exclusively on closed conduit systems as the type of containment isolation trip valves because of the short stem travel, and therefore, quick closing capability. This requirement ensures integrity of containment isolation to limit release of radioactivity to the environment. These valves provide overpressure relief protection by opening due to liquid pressure buildup under the seat as a result of transient temperature conditions following a DBA. External relief is not required unless the relief pathway can be isolated. In such cases, a relief valve is installed between the isolation valves. Refer to Table 3.6.1-1 of the [Licensing Requirements Manual](#) for the description of containment penetration relief protection requirements. In addition to the code requirements, additional quality assurance and test programs were imposed on these valves to achieve optimum reliability.

Based on previous successful operating experiences of Limitorque operators in other nuclear plants, Limitorque motor operators have been accepted for BVPS-1. These operators are used extensively in nuclear applications. Limitorque operators are designed to be operable during the life of the plant and are tested to operate during DBA conditions. Motor operators supplied by Limitorque conformed to all applicable National Electrical Manufacturers Association (NEMA) and Institute of Electrical and Electronic Engineers (IEEE) Standards and were tested to NEMA MG-1-10.35⁽¹¹⁾. Motor operators were tested in accordance with IEEE Std. 382,⁽¹²⁾ to demonstrate satisfactory operation in the combined pressure, temperature, atmospheric and radiation conditions. These tests were conducted by Franklin Institute Research Laboratories and are documented in their final report⁽¹³⁾.

In order to ensure the operability of motor-operated valves, manufacturers have performed the operability test in a dry condition in their plant, have certified that valve and motor operator will withstand the specified environmental conditions and have submitted a static seismic analysis which was approved by Stone & Webster.

Prior to their installation in the plant, motor-operated isolation valve bodies were hydrostatically tested at a pressure equal to twice the nominal pressure rating, with a test duration of 10 minutes for valves with minimum wall thickness up to and including 1 inch and 30 minutes for valves with larger wall thicknesses. Motor-operated valves were tested for seat tightness at the nominal pressure rating of the valve for not less than 5 minutes. The permissible seat leakages were in accordance with MSS-SP.61.⁽⁶⁾

Valves listed below were subjected to special preoperational testing requirements:

1. MOV-RS-156A and MOV-RS-156B in the recirculation spray system were subjected to zero seat leakage and backseat tests.
2. Weight loaded check valves QS-3, QS-4, RS-100 and RS-101 in quench spray and recirculation spray systems were subjected to seat tightness test in accordance with MSS-SP61 with differential pressure of 9 to 45 psi in the reverse flow direction and 2 psi in normal flow direction.
3. MOV-FW-156A, MOV-FW-156B, and MOV-FW-156C - motor-operated check valves in the main feedwater system were originally designed to seal against 5.5 psi pressure differential in the normal flow direction. The motor-operators have been removed and the feedwater isolation valves (HYV-1FW-100A, B, C) provide containment isolation.
4. Ventilation isolation valve (dampers), which serves as containment isolation, were subjected to the following tests:
 - a. Shell test - valves were tested at 70 psig for a period of at least 10 minutes (during which there was no leakage allowed)
 - b. Seat test - the valves were air tested under water at 70 psig for a period of 15 minutes, during which time there was no visible leakage
 - c. Steam test - the valves were steam tested at 45 psig saturated for 30 minutes during which time there was no leakage through any part of the valve allowed.

All parts of containment isolation trip valves, subject to line pressure were tested both for mechanical functioning and tightness of the valve seat and shell. Hydrostatic test pressures were equal to or greater than required by ANSI B16.5 and MSS-SP-61 for the conditions of service. In addition to hydrostatic tests, maximum operating pressure backseat leakage was less than 5×10^{-4} cubic centimeters per minute per PSI differential pressure per inch of port diameter. Maximum leakage below valve seats was limited to 6×10^{-7} standard cubic centimeters per second per inch of valve diameter.

All containment isolation valve radiographic examination was in accordance with American Society for Testing Materials ASTM Specifications E-94⁽⁷⁾ and E-142⁽⁸⁾.

In addition to radiographic examination requirements, magnetic particle inspection on ferrous isolation valves was performed in accordance with ASTM E-709,⁽⁹⁾ previously ASTM E-109, and stainless steel isolation valves were liquid penetrant inspected in accordance with ASTM E-165.⁽¹⁰⁾

References for Section 5.3

1. Letter from Stephen H. Hanauer, Chairman, Advisory Committee on Reactor Safeguards, to Honorable Glenn T. Seaborg, Chairman, USAEC, dated May 15, 1969, subject: "Report on Edwin I. Hatch Nuclear Plant."
2. Letter from Stephen H. Hanauer, Chairman, Advisory Committee on Reactor Safeguards, to Honorable Glenn T. Seaborg, Chairman, USAEC, dated May 15, 1969, subject: "Report on Brunswick Steam Electric Plant."
3. "Power Piping Code," ANSI Std. B31.1-1967, The American National Standards Institute.
4. "Steel Pipe Flanges, Flanged Valves and Fittings," ANSI Std. B16.5, The American National Standards Institute.
5. "Pressure-Temperature Ratings for Steel Butt-Welding End Valves," MSS-SP-66, The Manufacturers Standardization Society Standard Practice.
6. "Hydrostatic Testing of Steel Valves," MSS-SP-61, The Manufacturers Standardization Society Standard Practice.
7. "Recommended Practice for Radiographic Testing," ASTM E-94, The American Society for Testing Materials.
8. "Controlling Quality of Radiographic Testing," ASTM E-142, The American Society for Testing Materials.
9. "Practice for Magnetic Particle Examination," ASTM E-709, The American Society for Testing Materials.
10. "Liquid Penetrant Inspection Method," ASTM E-165, The American Society for Testing Materials.
11. "Motors and Generators," NEMA MG-1, The National Electrical Manufacturers Association.
12. "Guide for Type Test of Class 1 Electric Valve Operators for Nuclear Power Generating Stations," IEEE Std. 382, The Institute of Electrical and Electronic Engineers, Inc.
13. "Qualification Test of Limitorque Valve Operation, Motor Brake, and Other Unit in a Simulated Reactor Containment Post-Accident Environment," F-C3327, Franklin Institute Research Laboratories.
14. "American Standard for Industrial Control Apparatus," C19.1, The American National Standards Institute.

References for Section 5.3 (CONT'D)

15. Branch Technical Position MEB 3-1, Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment from NUREG-0800, Section 3.6.2 titled Determination of Rupture Locations and Dynamic Effects Associated with Postulated Rupture of Piping, Rev. 1, July 1981.
16. NERU Calculation No. 8700-46.4-1 and 8700.46.7
17. Technical Specification Change Request No. 130
18. "Boiler and Pressure Vessel Code, Nuclear Power Plant Components," 1971, including Addenda through Winter 1972, ASME Section III, Class 2.

5.4 INTERNAL CONTAINMENT SYSTEMS

5.4.1 Ventilation Systems

The containment ventilation systems consist of containment air recirculation cooling systems, control rod drive mechanism shroud cooling systems, and a containment purge exhaust and supply system, as shown in Figure 5.4-1.

5.4.1.1 Design Bases

The containment ventilation systems and air quantities are designed and sized to limit ambient air temperature buildup and to provide suitable environment for personnel and equipment with maximum safety against the spread of radioactive contamination.

The design of the air recirculation cooling system is based on a total air flow rate of 450,000 cfm with three fan operation. The design of the control rod drive mechanism shroud cooling system is based on an air flow rate of 66,000 cfm with two fan operation.

The air recirculation cooling system is designed so that under full power operation with one unit out of operation, the maximum average temperature will be limited to 108°F within the containment with cooling water at 45°F.

In the event of the loss of all normal onsite and offsite power, the containment average temperature should not exceed 135°F. This temperature limit is established to prevent the damage of instrumentation within the containment which would prevent the plant from being returned to operation once normal power was restored. In the event of loss of all normal onsite and offsite power, when the emergency busses are loaded on the emergency diesel generator (EDG), the containment air recirculation fans do not start on the load sequencer and will not be loaded manually on the EDG.

Normally water is supplied to the cooling coils from the chilled water system at 45°F. An alternate source of water is also provided by piping from each river water header. Component cooling water is supplied to the cooling coils of the control rod drive mechanism shroud cooling system.

The air recirculation cooling systems and control rod drive mechanism shroud cooling systems are designed to operate in both subatmospheric and atmospheric conditions.

No part of the containment ventilation systems is designed for operation during a DBA.

Purge exhaust is designed in the normal mode for a one air change flow rate of 30,000 cfm with connections to supplemental leak collection and release filter banks and duct system dampers arranged to reduce the concentration of any possible airborne radioactivity to levels acceptable for atmospheric discharge at an elevated release point meeting the requirements of 10 CFR 20 and 10 CFR 50. During containment refueling activities, the purge exhaust may be lined up to the supplementary leak collection and release system through seismically designed ducts and dampers. The purge exhaust system is also designed with radiological monitoring to meet the requirements of 10 CFR 20 and 10 CFR 50. The purge exhaust is designed with the capability of diverting a reduced flow rate of 1,000 cfm to the process vent filters and gaseous waste blowers discharging at the top of the cooling tower in the event of a high degree of activity.

The common ventilation vent, located on top of the auxiliary building is designed to discharge low level and nonradioactive purge exhaust air at a point 10 ft higher than the turbine building roof.

Purge supply is designed for a flow rate of 27,000 cfm which is less than the normal exhaust rate to maintain a slight negative pressure in the containment. During containment refueling, air infiltrates into the containment through the purge supply system.

Reactor coolant pump motors do not constitute a major heat source since they are cooled by self-contained water coils fixed to the motor housings.

5.4.1.2 Description

Bulk air cooling of the containment is achieved by three air recirculation cooling systems with the recirculated air being cooled on passing through chilled water or river water coil banks. Cooled air is circulated by three 50 percent design capacity fans, each with a capacity of 150,000 cfm, discharging into common ductwork supplying the ventilated spaces. Air leaving the ventilated spaces is recirculated back to the supply fans via the annular space between the crane wall and containment outside wall. All three fans are normally operable; however, bulk air temperature is limited to a design maximum of 108°F with any two fans in operation. Gravity actuated back draft dampers are installed at the discharge of each fan to prevent reverse flow through an idle fan.

Cooling water counter current to the flow of air to each of the three banks of cooling coils is normally supplied at a rate of 470 gpm from the chilled water system.

The three control rod drive mechanism (CRDM) shroud cooling fan motors are supplied with emergency electrical power in order to maintain the CRDM in a safe condition following the loss of all normal power. One fan motor of the three shroud cooling system units is connected to one emergency bus and the second fan motor in these systems is connected to the other emergency bus. The third fan motor in these systems has the capability of manually being switched to either of the two emergency buses. Under a loss of normal power situation, circuitry is incorporated to ensure the automatic restart of only one CRDM fan on each emergency bus.

The air recirculation system fans and the shroud cooling system fans are capable of operating under both normal atmospheric pressure and subatmospheric pressure conditions. Whereas the fans can operate at normal atmospheric pressure, then due to the lower density of a steam air mixture following an accident, they will be capable of operating up to a pressure of about 10 psig in the containment, assuming that the pressure rise started at full subatmospheric conditions. This capability is desirable for small reactor coolant leaks when it is not desirable to initiate the containment spray system. However, the fans are not considered as engineered safety features equipment and are not required to reduce the containment pressure following an accident.

Cooling of the control rod drive mechanism shroud is provided by three ventilation systems drawing containment ambient air through the shroud and detachable ductwork to three 50 percent design capacity fans, each fan with a capacity of 33,000 cfm. The fans discharge through component cooling water coil banks and gravity actuated backdraft dampers before returning the air to containment ambient. All three fans are considered operable; however, only two are normally used and are sufficient for design heat removal.

During shutdown periods, containment purging ventilation is provided by an exhaust system and a supply system designed to maintain a slight negative containment pressure, with provisions to handle variable flow rates up to one air change per hour, or 30,000 cfm. During unit shutdown, these systems are manually actuated if activity levels within the containment are high enough to require purging before personnel entry. This system also functions as a heating and ventilation system during periods of maintenance.

During refueling, the purge system exhaust duct may be lined up to the supplementary leak collection and release system. The air may be exhausted through the filters with make up air infiltrating into the containment.

The purge system exhaust circuit is provided with connections to dual filter banks consisting of prefilters, particulate-filters and charcoal filters to ensure that any radioactivity is within allowable offsite limits, as required in 10 CFR 20 and 10 CFR 50.

This circuit consists of a valved outlet from the containment, with ductwork extending to the filter banks and connected to the supplementary leak collection and release system. Filter characteristics and the supplementary leak collection and release system are described in Section 6.6. The purge exhaust air bypasses the filters and discharges to the monitored common ventilation vent when filtration is not required. Under conditions of high activity in the containment, a small quantity, 1,000 cfm of the purge exhaust air is also capable of being diverted to the process vent filters and gaseous waste blowers for discharge at the top of the cooling tower. The gaseous waste disposal system is described in Section 11.2.3. In normal unit operation, the purge exhaust and refueling circuit is inoperative and the butterfly isolation valves at the containment barrier are closed. Ventilation vent instrumentation, including radiation monitors and flowmeters, and containment radiation monitors furnish the necessary information for control of the filtered air and bypass air dampers.

The purge system supply circuit furnishes fresh filtered air to the interior of the containment to replace the air removed by the exhaust circuit. Heating coils provide a means of tempering -5°F outside air to maintain an ambient temperature of 60°F during winter shutdown. This supply circuit consists of outdoor air filter heating coil and supply fan of 27,000 cfm design capacity with discharge ductwork through butterfly isolation valves at the containment barrier. During normal unit operation, the purge supply circuit is inoperative and the butterfly isolation valves are closed.

5.4.1.3 Evaluation

The containment air recirculation cooling system is designed so that in the event of the loss of all normal onsite and offsite power, fans are not started automatically or manually on the emergency diesel generator. Under normal operating conditions in the event of a failure of a single fan coil unit the remaining two of the three fan-coil units are sufficient to limit the average containment ambient temperature at 108°F . This limit provides the static heat sink capability to aid in reducing containment pressure following a DBA. Under the condition of a loss of all normal power, the reactor would be shut down. After the reactor is shut down the energy inventory in the reactor coolant system drops to the point where the static heat sink is not as important. Therefore, the containment temperature can be allowed to rise above 105°F when the reactor is not operating.

It is expected that BVPS-1 could have extended operation with one of the air recirculation system cooling units inoperative. Therefore, the design of the emergency power system considers that one fan-coil cooling unit is inoperative when a loss of normal power occurs and it is not necessary to restart a containment air recirculation fan. This requirement is met in a three-unit two-bus arrangement in that a minimum of one operable fan-coil unit is always connected to each emergency bus.

The containment air recirculation cooling system is not considered to be engineered safety features equipment. However, it is capable of removing heat from the containment in the event of a small reactor coolant leak which is not of sufficient magnitude to initiate operation of the containment depressurization system.

Ventilation equipment is located in areas that are accessible for repair or maintenance. Redundant fans and cooling coils are provided for the control rod drive mechanism shroud cooling system.

Radiation monitors are provided for automatically isolating the purge exhaust air with the capability of manually diverting contaminated air through the main filter banks in the supplementary leak collection and release system, and then to the elevated release point. Annunciators and indicators in the main control room permit the operator to determine that the diverting dampers are properly positioned. Purge exhaust air can also be diverted in reduced quantities to the gaseous waste system blowers for discharge at the top of the cooling tower.

The control rod drive mechanism shroud cooling system fans are emergency powered, but are not required to operate during a DBA.

None of the ventilation systems discussed in this section are designed as Seismic Category I, with exception of portions required for containment isolation and containment refueling ventilation.

5.4.1.4 Tests and Inspections

All systems are tested and inspected as separate components and as integrated systems. Anemometer or velometer readings are taken to ensure that all air systems are balanced to deliver the required air quantities. All water coils are hydraulically tested for leakage prior to being placed in service.

Capacity and performance of fans are tested or rated in compliance with Air Moving and Conditioning Association (AMCA) test codes.

5.4.2 Containment Vacuum and Leakage Monitoring Systems

5.4.2.1 Containment Vacuum System

Design Bases

The containment vacuum system is designed to perform two functions, as follows:

1. Evacuation of the containment to within the Technical Specification operating range
2. Removal of air from the containment to compensate for containment inleakage during normal operation

The ejector is designed to reduce the containment pressure from atmospheric pressure to a pressure within the Technical Specification operating range in less than 5 hours. Each of two mechanical vacuum pumps is designed to compensate for inleakage.

Description

The containment vacuum system consists of a containment vacuum ejector and two mechanical containment vacuum pumps, with the required piping, valves and instrumentation.

The containment vacuum ejector removes air from the containment initially and operates on 150 psig steam (nominal) provided by the auxiliary steam system (Section 10.3.2). The containment vacuum ejector discharges to the atmosphere.

Two electrically driven mechanical vacuum pumps of about 70 cfm capacity each are provided. Operation of the vacuum pumps is manual. Redundant Air Pressure alarms on the control board require operator action to manually start a pump if the containment air pressure exceeds the set point. The pumps are capable of being operated from the emergency diesel generators. They discharge to the tank vent filter in the gaseous waste disposal system (Section 11.2.3). The Containment Total Air Pressure indicator alarm will provide sufficient warning of changing conditions within containment.

Evaluation

The containment vacuum pumps are not an engineered safety feature and do not prevent or mitigate design basis accidents. During design basis accident conditions, the containment vacuum pumps are isolated and are not used to depressurize the containment. The containment vacuum pumps were purchased specifically to be of standard design, without special requirements for exotic, long lead time replacement parts. If both pumps were to malfunction when it is determined that containment vacuum pump operation is required, system design allows time for defective pump replacement or repair. The ability to withstand the loss of both vacuum pumps under such conditions exceeds the requirements of the single failure criterion described in Section 1.3.1. Such a loss is particularly unlikely considering the frequent observation of the pumps under operational conditions, the low temperature and pressure service conditions and the inherent reliability of these proven-design pumps.

The operation of the containment vacuum pumps is strictly manual except that a containment isolation phase A (CIA) signal (Section 7) automatically trips the two containment isolation valves in the suction line and stops any operating vacuum pump(s). The vacuum pumps cannot be started again until the isolation signal is cleared. Thus, improper operation of a containment vacuum pump following a LOCA would require overriding both channels supplying the CIA signal to the vacuum pump and both suction line containment isolation valves, and then manually starting the pump and opening the suction valves in violation of administrative procedures and alarms.

The vacuum pumps have a relatively small capacity when compared to the containment free volume. Uninterrupted operation of one vacuum pump for approximately 3 days in violation of alarms and administrative controls is required to effect a 1 psi decrease in the containment pressure. Since the containment is designed to withstand a decrease in pressure of 1.0 psi below the lowest normal operating pressure, the vacuum pumps are not considered capable of accidentally lowering the containment pressure below minimum design pressure (8.0 psia). The containment steam vacuum ejector could establish a vacuum beyond the design loadings of the containment, but only if operated in violation of alarms and administrative controls for over six hours.

The steam ejector is used for evacuating the containment from atmospheric pressure to a pressure within the Technical Specification operating range during start-up operations. The system is designed to reduce the containment pressure to within the Technical Specification range in approximately four hours, compatible with the normal start-up schedule. Unit start-up is performed in accordance with detailed written procedures, which include the steam ejector system. The establishment of containment pressure is performed in accordance with operating procedures and is closely supervised by personnel responsible for BVPS-1 start-up. Pressure indicators are located in the control room to provide the operator with a continuous indication of containment pressure. This close supervision and monitoring assures that the normal operating pressure is not reduced below that permitted by the Technical Specifications. In the unlikely event containment pressure is reduced below the value defined by the Technical Specifications, a low pressure alarm will annunciate in the main control room, notifying the operator that the low pressure condition exists. Because of the slow rate of depressurizing the containment, there is sufficient time to take corrective action, i.e., take the ejector out of service. When the normal containment operating pressure is reached, the steam jet ejector is secured under administrative control and is not used during normal unit operation.

The minimum pressure to which the containment could be depressurized assuming infinite vacuum ejector operating time is 6.6 psia. However, operation of the ejector is entirely manual.

The containment isolation valves in the suction line can only be operated using the hand operators on the valves, one inside and one outside containment. This operation is performed only under strict administrative control and Technical Specifications guidelines. Alarms are provided in the main control room to indicate that the containment pressure has dropped below the normal operating band which is ± 0.1 psi of the setpoint. This allows sufficient time for the operator to take corrective action before the pressure drops below the design loading of the containment. Administrative procedures require that the steam ejector be separated from the containment atmosphere by normally closed, manual, administratively controlled valves to prevent its operation at any time other than that required during startup of the unit.

The minimum pressure to which the containment might be expected to be depressurized, assuming inadvertent operation of the steam ejector, is 0.1 psi below the containment vacuum system setpoint. Assuming the lowest expected setpoint of 13.0 psia and based on the alarm setpoint, the lowest pressure in the containment would be 12.9 psia.

Containment entry may be made at all times during normal power operation, during hot shutdown or at any time while placing the plant in a cold shutdown condition.

Tests and Inspections

The containment vacuum ejector is not considered a part of the engineered safety features and, since it is such a simple mechanical device, periodic tests are not required. The mechanical containment vacuum pumps are operated during the initial containment leakage rate test (Section 5.5) and demonstrated to have adequate capacity to remove inleakage. During normal unit operation, they are alternated in service, thus providing periodic testing of each containment vacuum pump.

5.4.2.2 Containment Leakage Monitoring System

Design Bases

The containment leakage monitoring system is used to determine the leakage rate of the containment under periodic test conditions. The containment leakage rate is determined using the absolute test method, and either the Mass Point or Total Time data analysis method is used to calculate the containment leakage rate.

The system provides for measurement of containment leakage rate of less than 0.1 percent of the contained volume in 24 hours with an accuracy sufficient to meet the requirements of Appendix J, 10 CFR 50. The system is designed in accordance with ANSI N45.4, American National Standard, Leakage Rate Testing of Containment Structures for Nuclear Reactors, March 17, 1972. Containment leakage rate testing is conducted in accordance with 10 CFR 50, Appendix J with certain exceptions as noted in the Technical Specifications.

Description

The absolute method utilizes a data acquisition system to measure internal containment pressure, temperature, and dewpoint temperature. A quartz manometer measures pressure, 16 resistance temperature detectors measure temperature, and 5 moisture analyzers determine dewpoint temperature. Data is recorded and the mass of air in the containment is determined at regular intervals by the ideal gas law. This data is fitted to a linear equation relating time and mass in the containment to leakage rate.

In addition, a statistical analysis using linear regression and a Students' T-distribution is performed on the calculated leakage rate data to establish a 95 percent confidence limit. This confidence limit is added to the calculated leakage rate to account for uncertainties, this sum is the upper confidence limit.

A supplemental verification is performed after the completion of the leakage rate measurement to confirm the accuracy of the data acquisition system. Two verification methods are available for use; the Superimposed Leak, and the Mass Step Change.

The Superimposed Leak verification consists of superimposing a known leak onto the existing leaks in the containment system equal to approximately the maximum allowable containment leakage rate. The superimposed leak measures the air bleed from containment using a variable area flowmeter (rotameter). Again data is obtained and a containment leakage rate calculated. The leakage rate calculated by the absolute data analysis method when summed with the measured superimposed leak must agree with the calculated leakage rate, obtained with the superimposed leak on the containment system, within ± 25 percent of the maximum allowable containment leakage rate.

The Mass Step Change verification consists of injecting a mass of air into the containment system. The quantity of air injected into the containment is measured using a mass flowmeter. The change in mass measured by the flowmeter must agree with mass calculated by the absolute method within ± 25 percent of the maximum allowable containment leakage rate. The mass of air injected into the containment is approximately the mass equivalent of the maximum allowable containment leakage rate.

These two verification methods described above serve to verify the leakage rate results obtained from the data analysis methods.

The quartz manometer used for measurement of change in containment pressure is sensitive to a change of ± 0.001 inch mercury. The moisture analyzers sense changes in dewpoint temperature of 0.5°F . All temperature measuring instruments sense temperature changes of 0.1°F .

Partial pressure of water vapor (i.e., saturation pressure) is determined by measurement of the dewpoint temperatures and the use of saturated steam tables as required by surveillance verification log.

There are two independent air pressure transmitters, indicators, and alarms used to monitor changes in containment.

Design Evaluation

Periodic leakage monitoring is performed by the absolute method and data analysis can be performed by two different methods; Total Time, and Mass Point. Either analysis method can be used, and is sufficiently accurate to establish that the containment leakage rate is less than 0.1 percent of the containment volume per day.

The containment leakage monitoring system and containment wide range pressure monitoring system tubing, as an extension of the containment, is designed to withstand the pressure and temperature expected during a DBA.

Tests and Inspections

All instruments including resistance thermometers, quartz manometers, and moisture analyzers are calibrated prior to each periodic containment integrated leakage rate test. In addition an insitu check of at least one point is performed on all associated resistance thermometers and moisture analyzers.

5.4.2.3 Containment Wide Range Pressure Monitor

The containment wide range pressure monitoring system is used to provide a continuous indication of containment pressure (0 to 200 psia) in the control room. The containment wide range pressure indicator is provided on the main control board with one channel being recorded. The containment wide range pressure instrumentation is connected to the containment leakage monitoring system containment penetration lines (Section 5.3.3).

5.4.3 Containment Depressurization System

The containment depressurization system is described in detail in Section 6.4.

The quench spray subsystems of the containment depressurization system transfer chilled water from the refueling water storage tank to the containment through the quench spray headers. The chilled quench spray water cools the containment atmosphere, resulting in a decrease in containment temperature and pressure. The quench spray subsystems operate principally during the depressurization period after a LOCA.

The recirculation spray subsystems of the containment depressurization system recirculate water from the containment sump through river water cooled heat exchangers to the recirculation spray headers. These subsystems provide the means of heat removal from the containment. They are designed to aid in lowering the temperature and pressure of the containment atmosphere during depressurization and provide a means for long term containment heat removal.

The recirculation spray pumps and the quench spray pumps are electric motor driven from either normal or emergency power sources (Section 8).

Reference for Section 5.4

1. Reference deleted by Rev. 5 (1/87).

5.5 DESIGN EVALUATION

The reactor containment concept is based upon the use of a dry containment maintained within the pressure range listed in the Technical Specifications during normal operation. Pressure limitations allow restricted personnel access to the containment.

The containment depressurization system is considered to be an engineered safety features system. The containment vacuum system (Section 5.4.2) is not considered to be an engineered safety features system.

Containment isolation features, such as penetrations, access hatches, and isolation valves, meet the requirements of 10 CFR 50 Appendix A (Section 1.3 and Appendix 1A).

The containment structural design is in accordance with the best current design practices for steel lined reinforced concrete reactor containment structures. The design procedures incorporate accepted analytical methods. Rigid controls were maintained for all materials and construction practices as indicated in Section 5.2 and 10 CFR 50 Appendix A.

5.6 TESTS AND INSPECTIONS

A testing and surveillance program is in effect during operation to establish that the containment can perform its intended function. The program consists of structural acceptance test, an initial leakage rate test, periodic leakage rate retesting, continuous containment pressure monitoring, and periodic surveillance tests. Material and fabrication inspections and tests are described in Section 5.2.

All applicable welding procedures and tests specified in the ASME Boiler and Pressure Vessel Code, Section IX, 1968, for welding qualifications, were adhered to for qualifying the welding procedures and the performance of welding machines and welding operators who were engaged in the construction of the containment liner. The welding qualification included 180 degrees bend tests of weld material. These procedures ensured that the ductility of welded seams was comparable to the ductility of the containment liner plate material.

Production quality control was exercised through random radiography per the ASME Boiler and Pressure Vessel Code, Section VIII, paragraph UW-52, 1968.

All highly stressed insert plates, which were welded into the liner to transfer loads into the concrete, were ultrasonically tested to check for possible laminations. Tests were conducted on all plates where stress analysis indicated a higher than average stress field, although all such plates were stressed well below the allowable limits for the materials. These tests showed that no laminations exist in the insert plates.

The ASME Boiler and Pressure Vessel Code, Section III, 1968, was used as a general guide in the selection of materials, design stresses and fabrication of the steel containment liner. Paragraphs N-511, N-512, N-513, N-514, N-515, N-516, N-517, N-518, N-519, N-521, N-522, N-523, N-524, N-525, N-526, N-527, N-528, N-531, N-541, N-611, N-612, N-613, N-614, N-615, N-616, N-620, N-622, N-623, and N-627 of the ASME Boiler and Pressure Vessel Code, Section III were adhered to in the liner construction. These paragraphs were subject to both clarification and interpretation as given below. Each interpretation or clarification was independent of any other; compliance with the requirements for a Class A vessel in a specific paragraph did not classify this liner as a Class A vessel. Also, compliance with the requirements of paragraphs referencing Article 3 were not to be construed to mean compliance with Article 3 in its entirety. When V-notch Charpy impact tests were made, the specimens are at -20 F and meet 15 ft-lb minimum. Paragraphs which were subject to clarification are listed in Table 5.6-1.

Construction testing included provisions for testing the leaktightness of all penetrations and liner welds during construction and an air pressure test to ensure the structural integrity of the containment. Electrical penetrations were assembled and tested as a unit for leaktightness following installation in the containment (Section 5.2).

As an aid to testing weld seams in the steel liner, steel test channels were welded over all weld seams. These channels were not continuous, and leak tests were performed section by section. On the bottom and vertical portions of the liner, the test channels were placed on the inside of the liner. On the dome portion of the liner, the test channels were on the outside (concrete side) of the liner. These channels formed a space into which pure Freon R-22 at 50 psig could be injected. Before halogen leak testing, all the test channel welds were soapsud tested by pressurizing the void with air to 50 psig and checking for visible leakage.

After the air test and any subsequent repairs, if required, the test channels were evacuated to a pressure of 1.0 to 0.5 psia by utilizing a vacuum pump. This was to ensure a homogeneous test gas throughout the channel when the channel was pressurized at 50 psig with Freon R-22. For the bottom and vertical portions of the liner, where the test channels were placed on the inside, all tests channel seal welds were leak tested using a halogen leak detector. For the dome portion of the liner, where the test channels were on the outside, the liner weld seams are leak tested using a halogen leak detector.

After testing, the gas was purged from the channels with air and the threaded connections were plugged. The sole purpose of the test channels was additional insurance that the completed containment was capable of meeting the integrated leak test requirements.

The containment structure was subjected to an acceptance test in accordance with Safety Guide 18 as specified in Section 1.3.3.18, during which the containment internal pressure was 1.15 times the containment design pressure. This was performed after the liner was completed, the last concrete placed, and all penetration sleeves and hatches installed and closed or blanked off. The structure was surveyed, measured, and inspected for concrete cracks prior to the test.

The pressure was increased to peak value in four increments and decreased after reaching peak value in four increments, to allow an evaluation at each step.

The radial and vertical deflections of the containment were measured at several locations at each increment of pressure, and at peak pressure.

Deformations around the largest opening in the containment were adequately measured.

The environmental conditions (atmospheric pressure, temperature, and humidity) inside and outside the containment structure were continuously monitored during the test to evaluate their contribution to the response of the containment. The test was not conducted under extreme weather conditions such as snow, heavy rain, or strong wind.

When the containment structure was subjected to the peak test internal pressure, the maximum radial growth was expected to be approximately three-quarters inch and the maximum vertical deformation at the dome apex to be approximately one and one-half inch. These deformations were calculated for the analytical stress evaluation of the containment liner. Strain measurements were made on the steel liner using conventional strain gages at adequately selected points.

During the acceptance test, visual examination and instrumentation were used to record cracking and changes in measurements, both vertically and radially, due to the response of the concrete containment structure to the air pressure test of the liner. Prior to testing, the outside of the concrete structure was surveyed, measured, and inspected for cracks, and all pertinent information recorded. During the test, measurements were made of the radial deflections at various locations on the wall from the top of the mat to the spring line of the dome. Two permanent pits located approximately 90 degrees apart were provided for access to the containment wall below ground grade. These pits allowed localized visual inspection and measurements of the lower part of the wall.

Vertical deflections were measured at the apex and spring line of the dome. Additional strain measurements were made on the surface areas adjacent to the equipment access hatch and in other areas where stresses were critical.

Deformations were measured by linear variable differential transducers (LVDT's) mounted at the internal surface of the liner plate. LVDT's were also used to measure displacements of the concrete ring around the equipment hatch. Cracks larger than 0.01 inch which occur during the test were recorded. They were measured by an optical comparator and checked with feeler gages. After the completion of the test recovery of the structure was recorded. The crack pattern was again inspected and recorded.

The containment concrete surface was whitewashed in areas of high stress and at openings to chart crack patterns. Photographs were taken of the crack patterns to provide permanent records.

Temperature, barometric pressure and weather conditions were recorded hourly during the test period.

Containment Leakage Rate Tests

The containment leakage rate tests are performed in accordance with the guidelines of Appendix J of 10CFR50, "Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors."

The containment leakage testing program includes the performance of Type A tests, to measure the containment overall integrated leakage rate, Type B tests, to measure leakage of certain containment components, and Type C tests, to measure containment isolation valve leakage rate. Containment leakage rate tests are based upon the limiting radionuclide releases due to a LOCA.

The preoperational Type A test was conducted according to the rules of Appendix J, Option A.

Periodic Type A tests are conducted in accordance with Appendix J, Option B (with the exemption noted in the Technical Specifications). These tests are performed using the leakage monitoring system (described in Section 5.4.2.2).

The measured leakage rate does not exceed the design basis accident leakage rate (L_a) of 0.1 percent per 24 hours of the weight of containment air at the calculated LOCA peak containment P_a . The remaining leakage characteristics are determined in accordance with Appendix J, Option B as documented in the Containment Leakage Rate Testing Program (CLRTP).

Type B tests are carried out to monitor the principal sources of leak development in accordance with Appendix J. These tests are performed to measure leakage originating at containment penetrations, such as electrical penetrations, air lock door seals, and equipment and personnel access hatches, which may develop leaks. Refer to the CLRTP for more information regarding the containment components to be tested.

The preoperational Type B tests were conducted according to the rules of Appendix J, Option A by local pneumatic pressurization of the containment components at a pressure not less than Pa.

Periodic Type B tests are conducted in accordance with Appendix J, Option B as documented in the CLRTP. The acceptance criterion for periodic Type B tests is given in the CLRTP.

The periodic Type B tests are scheduled according to the guidelines of the CLRTP.

The Type C tests are performed on the isolation valves to verify their sealing capability and leak-tightness according to Appendix J. The test includes valve closure and leakage tests. A valve closure test is conducted prior to a valve leakage test to demonstrate the proper sealing capability of a valve upon receipt of an isolation signal. Those isolation valves which are normally closed are exercised to verify closure and sealing capabilities. Those containment isolation valves which are in a system that is expected to be filled with water for 30 days following a LOCA and therefore do not represent a containment atmosphere leak path are not subject to the Type C test requirements of 10CFR50 Appendix J.¹

The preoperational Type C tests were conducted according to the guidelines of Appendix J, Option A by local pneumatic pressurization at a pressure Pa.

The periodic Type C tests are conducted according to the guidelines of Appendix J, Option B as documented in the CLRTP.

The structural integrity of the containment will be determined during the shutdown for each Type A containment leakage rate test in accordance with the CLRTP.

[Licensing Requirements Manual](#) Table 3.6.1-1, "Containment Penetrations" lists the containment isolation valves which can be individually leak tested.

References to Section 5.6

1. Amendment No. 65 to Technical Specification 4.6.1.2d and Technical Specification Table 3.6-1.

BVPS UFSAR UNIT 1

TABLES FOR SECTION 5

Table 5.2-1

STATIC DESIGN LOADS FOR PIPE RUPTURE

<u>Type of Break and Location</u>	<u>Loads, Kips</u>
Main Steam Guillotine	1,000
Main Steam Split	1,000
Safety Injection (Accumulator)	474
Residual Heat Removal	573
Main Feedwater	294

Table 5.2-2

MAXIMUM STEAM GENERATOR AND REACTOR COOLANT PUMP
FOOT LOADS, KIPS

<u>Component</u>	<u>Direction</u>	<u>Static</u>	<u>Dynamic</u>
Steam Generator	Horizontal Shear	373	390
	Vertical Compression	1,143	1,153
	Vertical Tension	629	639
Reactor Coolant Pump	Horizontal Shear	277	278
	Vertical Compression	1,161	1,287
	Vertical Tension	1,121	855

Note: Static loads include pipe rupture plus dead weight, thermal and pressure.

Dynamic loads include dead weight, thermal, pressure plus
 $\sqrt{(\text{pipe rupture})^2 + (\text{seismic})^2}$

Table 5.2-3

STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORTS
MAXIMUM LOAD, KIPS

<u>Concrete Reactions From</u>	<u>Node</u>	<u>Direction</u>	<u>Static</u>		<u>Dynamic</u>	
			<u>Condition</u>	<u>Load</u>	<u>Condition</u>	<u>Load</u>
Steam Generator Upper Snubbers & Struts	110 (Strut)	± Y	Main Steam Line Split	1,008	Main Steam Line, Split	1,075
	112 (Snubber)	± X	Main Steam Line Split	1,091	Main Steam Line, Split	1,103
	115 (Strut)	± Y	Main Steam Line Split	1,008	Main Steam Line, Split	1,075
	118 (Snubber)	± X	Main Steam Line Split	1,091	Main Steam Line, Split	1,103

Note: Static loads include dead weight, thermal pressure and pipe rupture.

Dynamic loads include dead weight, thermal pressure plus $\sqrt{(\text{pipe rupture})^2 + (\text{seismic})^2}$

All loads are in units of kips.

X-Direction is parallel to the RCS hot leg; Y-Direction is perpendicular to the RCS hot leg.

Table 5.2-4

STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORTS
MAXIMUM LOAD, KIPS

Concrete Reactions From	Node	Direction	Static		Dynamic		
			Condition	Load	Condition	Load	
Steam Generator Supports	52	Z	Main Steam Guillotine	395	Main Steam Guillotine	412	
		-Z	Main Steam Guillotine	825	Main Steam Guillotine	842	
	61	Z	Main Steam Guillotine	260	Main Steam Guillotine	282	
		-Z	Main Steam Guillotine	865	Main Steam Guillotine	887	
	71	Y	RC Branch Line Break ⁽¹⁾	446	RC Branch Line Break	567	
		-Y	RC Branch Line Break	446	RC Branch Line Break	567	
	82	Y	RC Branch Line Break	330	RC Branch Line Break	520	
		-Y	RC Branch Line Break	330	RC Branch Line Break	520	
	88	Z	Main Steam Guillotine	756	Main Steam Guillotine	770	
		-Z	Main Steam Guillotine	934	Main Steam Guillotine	948	
	97	Z	Main Steam Guillotine	625	Main Steam Guillotine	643	
		-Z	Main Steam Guillotine	851	Main Steam Guillotine	869	
	Pump Columns	9	Z	RC Branch Line Break	214	RC Branch Line Break	4
			-Z	RC Branch Line Break	1,340	RC Branch Line Break	1,456
13		Z	RC Branch Line Break	0	RC Branch Line Break	0	
		-Z	RC Branch Line Break	930	RC Branch Line Break	933	
16		Z	RC Branch Line Break	45	RC Branch Line Break	0	
		-Z	Main Steam Guillotine	992	Main Steam Guillotine	1,117	

Note: Static loads include dead weight, pressure, thermal and pipe rupture.
 Dynamic loads include dead weight, pressure, thermal plus (pipe rupture)² + (seismic)²
 All loads are in units of kips.
 (1) RC Branch Line Breaks include residual heat removal, safety injection, surge line.

Table 5.2-6

STRESSES IN CONTAINMENT LINER

Elevation, In.	Stresses in Liner (1)	Stress for 1P + 1T + DBE Loading (kips)	Stress for 1.15P Loading (kips)	Cyclic Pressure			Cyclic Temperature ($\Delta = 35F$)			Cyclic DBE			Total Usage Factor
				Stress	No. of Cycles	Usage Factor	Stress	No. of Cycles	Usage Factor	Shear Stress (kips)	No. of Cycles	Usage Factor	
0	S (Mer. Ins.)	58.722	4.54	4.55			1.09						
	S (Tang. Ins.)		1.167	1.25	∞	0	9.85	∞	0	10.095	∞	0	0
	S (Mer. Out.)	61.923	3.974	4.26			1.2						
	S (Mer. Out.)		1.385	1.39			9.9						
360	S (Mer. Ins.)	28.889	18.46	11.64			12.88						
	S (Tang. Ins.)		26.81	9.87	10^6	0.005	10.3	3×10^5	0.0167	8.384	∞	0	0.0217
	S (Mer. Out.)	30.289	21.96	12.98			14.2						
	S (Tang. Out.)		26.78	9.84			10.3						
1,465	S (Mer. Ins.)	28.999	23.61	4.03			7.23						
	S (Tang. Ins.)		32.535	1.86	∞	0	7.2	∞	0	3.52	∞	0	0
	S (Mer. Out.)	29.165	27.77	4.17			8.04						
	S (Tang. Out.)		32.526	1.87			7.2						
2,130	S (Mer. Ins.)	28.493	21.69	3.91			6.55						
	S (Tang. Ins.)		18.13	3.00	∞	0	7.83	∞	0	0.663	∞	0	0
	S (Mer. Out.)	34.144	3.4	4.2			7.29						
	S (Tang. Out.)		18.12	3.0			7.83						

(1) Mer. : Meridional
 Tang.: Tangential
 Ins. : Inside face of liner
 Out. : Outside face of liner

Table 5.2-8

CONTROL ROD DRIVE SHAFT, MISSILE CHARACTERISTICS

Diameter = 1.75 in.
 Weight = 157 lb.

<u>Drive Shaft Travel Outside Housing* (inches)</u>	<u>Drive Shaft Impact Velocity (ft/sec)</u>	<u>Drive Shaft Kinetic Energy (ft-lb)</u>
42	111	30,037

* Distance from top of rod travel housing to bottom of missile shield.

Note 1: The critical CRDM missile is the drive shaft alone. It is the limiting case and envelops the drive shaft latched to the drive mechanism case.

Table 5.2-10

VALVE, MISSILE CHARACTERISTICS

<u>Missile Description</u>	<u>Weight (lb)</u>	<u>Flow Discharge Area (in.²)</u>	<u>Thrust Area (in.²)</u>	<u>Impact Area (in.²)</u>	<u>Weight to Impact Area Ratio (psi)</u>	<u>Velocity (fps)</u>
Safety Relief Valve Bonnet (3 in. x 6 in. or 6 in. x 6 in.)	350	2.86	80	24	15.6	110
3 in. Motor-Operated Isolation Valve Bonnet (plus motor and stem) (3 in.)	400	5.5	113	28	14.1	135
2 in. Air-Operated Relief Valve Bonnet (plus stem)	75	1.8	20	20	3.75	115
3 in. Air-Operated Spray Valve Bonnet (plus stem)	120	5.5	50	50	2.4	190
4 in. Air-Operated Spray Valve	200	9.3	50	50	4	190

Table 5.2-11

PIPING TEMPERATURE ELEMENT ASSEMBLY, MISSILE CHARACTERISTICS

1. For a tear around the weld between the boss and the pipe:

<u>Characteristics</u>	<u>"Without Well"</u>	
Flow Discharge Area, in. ²	0.11	0.60
Thrust Area, in. ²	7.1	9.6
Missile Weight, lb	11.0	15.2
Area of Impact, in. ²	3.14	3.14
<u>Missile Weight</u>		
Impact Area, psi	3.5	4.84
Velocity, ft/sec	20	120

2. For a tear at the junction between the temperature element assembly and the boss for the "Without Well" element and at the junction between the boss and the well for the "With Well" element.

<u>Characteristics</u>	<u>"Without Well"</u>	<u>"With Well"</u>
Flow Discharge Area in. ²	0.11	0.60
Thrust Area, in. ²	3.14	3.14
Missile Weight, lb	4.0	6.1
Area of Impact, in. ²	3.14	3.14
<u>Missile Weight</u>		
Impact Area, psi	1.27	1.94
Velocity, ft/sec	75	120

Table 5.2-12

CHARACTERISTICS OF OTHER MISSILE
POSTULATED WITHIN REACTOR CONTAINMENT

	<u>Reactor Coolant Pump Temperature Element</u>	<u>Instrument Wall of Pressurizer</u>	<u>Pressurizer Heaters</u>
Weight, lb	0.25	5.5	15
Discharge Area, in. ²	0.50	0.442	0.80
Thrust Area, in. ²	0.50	1.35	2.4
Impact Area, in. ²	0.50	1.35	2.4
<u>Missile Weight</u>			
Impact Area, psi	0.50	4.1	6.25
Velocity, ft/sec	260	100	55

Table 5.2-13

LOAD COMBINATIONS FOR LINER PLATE

<u>Stress Allowables</u> (Per ASME III Nomenclature)	<u>Load Combinations</u>	<u>Category</u>
$P_m + P_b + Q < 3 S_m$	$D + P_d + T_d + DBE$	Emergency
$P_m < 0.9 S_y$		
$P_m + P_b < 1.35 S_y +$	$D + 1.15 P_d$	Test
"CAT" Curve Considerations		
N-415.1 or N-415.2	150 cycles of ΔP_o	Normal (Anticipated Cycle)
	600 cycles of ΔT	
	150 cycles of OBE	
$P_m (D + P_{min} + OBE) < S_m$		
$P_m + P_b + Q < 3 S_m$	$D + P_{min} + T_{min} + OBE$	Severe Operational
$P_m + P_b (D + P_{min} + OBE)$		
$< 1.5 S_m$		

ANCHORS

$0.425 S_u$	$D + P_d + T_d$	Emergency
$0.45 S_u$	$D + P_{min} + T_{min}$	Severe Operational

Where:

- D = Dead load effect of reinforced concrete structure acting on the liner plus dead load of the liner.
- Pd = Design pressure (pressure resulting from design basis accident plus safety margin).
- Td = Load due to thermal expansion, resulting when the liner is exposed to the design temperature.
- DBE = Shear stresses in the liner derived from applying the effect of the design basis earthquake.

Table 5.2-13 (Cont'd)

LOAD COMBINATIONS FOR LINER PLATE

OBE =	Shear stresses in the liner derived from applying the effect of the operating basis earthquake. The anticipated number of cycles, 150 cycles of OBE, is an assumed number of cycles of this type of earthquake for a 60 year span.
ΔT =	Load due to thermal expansion resulting when the liner is exposed to the differential temperature between operating and seasonal refueling temperatures (35°F). 600 cycles are assumed on the basis of 10 such variations per year, on a 60 year span. The design limit for operating temperature variations is 4000 cycles.
P_{\min} =	Minimum pressure resulting during operation of the containment.
T_{\min} =	Load due to thermal expansion resulting when the liner is exposed to the minimum pressure.
S_y =	Yield strength of the material.
S_m =	The smaller of 1/3 ultimate strength or of 2/3 yield strength.
S_u =	Ultimate strength of anchor material.
ΔP_o =	Differential Pressure between operating pressure and atmospheric pressure ($\Delta P_o = 6.7$ psi). The anticipated number of cycles, 150 cycles, are assumed on the basis of ΔP_o 2.5 refueling cycles per year on a 60 year span. The design limit for operating pressure cycles is 1000 cycles.

Table 5.2-14

LOAD COMBINATION FOR PENETRATIONS

Area of Analysis See Fig. 5.2-20	Stress Allowables (Per ASME III Nomenclature)	Load Combinations	Category
1	$P_m < 0.9 S_y$	M_p or T_p or Jax	Design
	$P_m + P_b < 0.9 S_y$		
	Pconcrete bearing		
	< 24 copsi		
	$P_m + P_b + Q < 3S_m$	$P_d + T_d + R_o$	Normal
	$P_m < .9 S_y$		
	$P_m + P_b < .9 S_y$ (Note 1)	M_p or T_p or Jax	Design (1)
2	(P_m+P_b) design (1) +		
	$(P_m+P_b+Q)P_b+T_d < 3S_m$	$P_d + T_d + \text{Design}(1)$	Design (2)
Note (1)	Applies to the penetration sleeve extension and attachment plate only.		
(2)	For the pipe portion refer to Section 3.7.3.1.		

Where:

M_p = Yielding moment = Required bending moment to produce stresses equal to the yield strength of the pipe material

T_p = Yielding torque = Required torsional moment to produce stresses equal to the yield strength of the material

Jax = Axial Jet Force = Load equal to the piping pressure times inside area of the pipe, acting in the axial direction of the piping

P_d = Piping Design Pressure

T_d = Piping Design Temperature

R_o = Piping reaction due to normal operating conditions

Table 5.2-14 (Cont'd)

LOAD COMBINATION FOR PENETRATIONS

S_y =	Yield strength of the material	
S_m =	The smaller of 1/3 ultimate strength or of 2/3 yield strength.	
S_u =	Ultimate strength of anchor material	

Table 5.2-15

INTERNAL MISSILE SUMMARY

<u>Missile Source</u>	<u>Missile (See Notes)</u>	<u>Actual Weight (lb)</u>	<u>Actual Impact Area (in.²)</u>	<u>Speed Torque or Pressure</u>	<u>Targets</u>	<u>Type KE (ft-lb)</u>	<u>Type KE/A (ft-lb/in.²)</u>
Fuel Pool Purification Pump	Type 1B (1 x 1 x 2)	0.58	1.0	Diam of coupling 6" at 1,750 rpm spider coupling	Fuel Pool Cooling HX and Pump	19.015	19.015
Fuel Pool Cooling Pump	Type 1B (1 x 1 x 2)	0.58	1.0	Diam of coupling 6" at 1,750 rpm spider coupling	Fuel Pool Cooling HX and Pump	19.015	19.015
Component Cooling Pump	Type 1B (1 x 1 x 2)	0.58	1.0	Diam of coupling 6" at 1,750 rpm spider coupling	Component Cooling HX and Pump	19.015	19.015
Quench Spray Pump	Type 1A (3/8" diam x 5" lg)	0.159	0.11	DBC is 6" at 3,600 rpm	Quench Spray Pump	22.05	200.54
Refueling Water Recirc. Pump	Type 1B (1 x 1 x 2)	0.58	1.0	Diam of coupling 6" at 1,750 rpm spider coupling	Quench Spray Pump	19.015	19.015
Electrical Aux. Feed Pump	Type 1A (3/8" diam x 5" lg)	0.159	0.11	DBC is 6" at 3,560 rpm	Aux. Feed Pumps	22.05	200.54
Turbine Aux. Feed Pump	Type 2 (1.5 x 1.5 x 1.5)	0.978	2.25	Diam of wheel 19" at 4,200 rpm	Aux. Feed Pumps	1852	823.1
Plant Bottle Storage	Type 3	2.0	0.5	Bottle pressure 2,600 psi	Refueling Water Tank	3828	7656

Table 5.2-15 (CONT'D)

INTERNAL MISSILE SUMMARY

<u>Missile Source</u>	<u>Missile (See Notes)</u>	<u>Actual Weight (lb)</u>	<u>Actual Impact Area (in.²)</u>	<u>Speed Torque or Pressure</u>	<u>Targets</u>	<u>Type KE (ft-lb)</u>	<u>Type KE/A (ft-lb/in.²)</u>
Propane Bottles	Type 3	2.0	0.5	Bottle pressure 264.7 psi	Refueling Water Tank	Less than 3828	Less than 7656
CO ₂ Storage Bottle	Type 3	2.0	0.5	Bottle pressure 305 psi	Fuel Pool Cooling HX and Pump	Less than 3828	Less than 7656
Aux. Bldg. Supply Fan	Type 5	87.0	15.0	Diam of fan 30" at 1,390 rpm	Leak Detection Exhaust Fans	54240	3616
Purge Exhaust Fan	Type 5	87.0	15.0	Diam of fan 40" at 1,220 rpm	Leak Detection Exhaust Fans	54240	3616

- Notes: Types of Possible Missiles
1. Pump motor coupling (A) bolt (B) gear tooth
 2. Turbine blade less energy required to penetrate casing
 3. Bottle blowout plug or valve regulator
 4. Compressor head bolt and nut 100% stored strain energy transformed into K.E.
 5. Fan blade less energy required to penetrate the fan housing. 40" diam fan at 1,220 rpm gives greatest K.E. generated by the fans. 40" diam fan at 1,390 rpm gives greatest penetration ability. Number indicates K.E. after penetration.

Table 5.2-16
PIPE BREAK LOCATIONS

Location	Main Steam Lines			Feedwater Lines		
	<u>32-SHP-56</u>	<u>32-SHP-57</u>	<u>32-SHP-58</u>	<u>16-WFPD-22</u>	<u>16-WFPD-23</u>	<u>16-WFPD-24</u>
Terminal Points	3 195 ⁽¹⁾	200 276 ⁽¹⁾	1 75 ⁽¹⁾	1 220 ⁽¹⁾	5 ⁽²⁾ 115 ⁽¹⁾	5 165
Point of Maximum Primary + Secondary Stress	200 ⁽¹⁾ 204 ⁽²⁾	274 ⁽¹⁾ 270 ⁽²⁾	73 ⁽¹⁾ 71 ⁽²⁾	15 25	110 ⁽¹⁾	12
Point Where P + S > .8 (S _a + S _n)	None	None	None	None	5 ⁽²⁾ , 10 ⁽²⁾ 110 ⁽¹⁾ 105 ⁽¹⁾ 75	None
Total Points	4	4	4	4	6	3
Total Areas	3 ⁽¹⁾	3 ⁽¹⁾	3 ⁽¹⁾	4 ⁽¹⁾	3 ⁽¹⁾⁽²⁾	3

Where:

P = Primary Stress
S = Secondary Stress
S_n, S_a are defined in ASME III NC3611

- Note:
- (1) Because of the proximity of two points, the area between the two points is considered one break area.
 - (2) Because the highest stress point is in the area of the terminal end point, the next highest stress point has been considered a pipe break point.

Table 5.2-17

CONTAINMENT STRUCTURES, SYSTEMS AND COMPONENTS
NOT DESIGNATED AS SEISMIC CATEGORY I

1. Elevator
2. Primary drain transfer tank (DG-TK-1)
3. Primary drain transfer tank eductor (DG-EJ-1)
4. Primary drain transfer pumps (2) (DG-P-1A and 1B)
5. Incore instrumentation room sump pump (DA-P-5)
6. Reactor containment sump pumps (DA-P-4A and 4B)
7. Pressurizer relief tank (RC-TK-2)
8. Reactor head storage stand
9. Upper and lower internals storage stands
10. Upper internal lift rig
11. Corrosion control tank (NS-TK-2)
12. Neutron shield surge tank (NS-TK-1)
13. Ventilation system fans, cooling coils, and filters
14. Ventilation ductwork
15. Piping (NNS)
16. Reactor coolant pump standpipes beyond orifice
17. Lighting

Table 5.2-18

REDUCTION FACTORS FOR LOAD CAPACITIES OF FLEXURAL
AND COMPRESSION MEMBERS

<u>Type of Member</u>	<u>Load Capacity Reduction Factor</u>
Tension and Flexure	0.90
Diagonal Tension, Bond, and Anchorage	0.85
Compression with Spiral Reinforcement	0.75
Compression with Tied Reinforcement	0.70

Table 5.2-19

SHEAR FORCES AT THE BASE OF THE WALL UNDER
FACTORED OBE AND DBE

Load Limit	Calculated Shear Forces		
	Base	El. 695'-1"	El. 711'-0"
1.25 OBE	34.62 kips/ft	34.0 kips/ft	32.0 kips/ft
OBE	43.3 kips/ft	42.6 kips/ft	40.0 kips/ft

Table 5.2-20

DIAGONAL REINFORCING STEEL STRESSES AND
CONCRETE SHEAR STRESSES UNDER FACTORED OBE AND DBE
LOADING CONDITIONS

OBE Load Condition
(DL + 1.25 P + 1.0 (T + TL) + 1.25 OBE)

Containment Structure Material Analyzed	Calculated Stresses		
	Base	El. 695'-1"	El. 711'-0"
Rebar	N/A ⁽¹⁾	22.50 ksi ⁽²⁾	46.6 ksi ⁽²⁾
Concrete	N/A ⁽¹⁾	40.0 psi	40.0 psi

DBE Load Condition
(DL + OP + 1.0 (T + TL) + 1.0 DBE)

Containment Structure Material Analyzed	Calculated Stresses		
	Base	El. 695'-1"	El. 711'-10"
Rebar	N/A ⁽¹⁾	11.8 ksi ⁽²⁾	30.0 ksi ⁽²⁾
Concrete	N/A ⁽¹⁾	60.0 psi	60.0 psi

- NOTE 1. At the base of the containment structure, the concrete is uncracked for a depth of 18 inches from the outside face of the wall due to the mat-wall boundary bending force that occurs under internal pressure. This uncracked concrete encases the diagonal reinforcing. The total available resistance at the base due to cracked ($V_c = 40$ psi or 60 psi) concrete and uncracked ($V_c = 110$ psi per ACI-63) concrete is 41.1 K/ft for the factored OBE and 49.5 K/ft for the DBE. Both of these values exceed the actual seismic shear forces for the factored OBE and DBE.
2. In the calculation, it was assumed that diagonals at 8 foot horizontal spacing were available to resist hoop, vertical, and shear membrane forces.

Table 5.2-21

MODIFICATION TO REINFORCING BARS (SIZES NO. 14 AND 18)
 GUARANTEED MINIMUM YIELD POINT CHEMICAL AND
 PHYSICAL REQUIREMENTS⁽¹⁾

<u>Chemical Requirements</u>	<u>Optimum Values (Percent by Weight)</u>
Carbon	0.35 maximum
Manganese	1.25 maximum
Silicon	0.15 to 0.25
Phosphorus	0.05 maximum
Sulfur	0.05 maximum
<u>Physical Requirement</u>	<u>Optimum Values</u>
Tensile strength, psi	70,000 minimum
Yield strength, psi	50,000 minimum
Elongation in 2 inches, percent	16 minimum
Elongation in 8 inches, percent	13 minimum

Note:

1. See Section 5.2.3.3

Table 5.3-1

CONTAINMENT ISOLATION ARRANGEMENTS

Service	Penet. No.	Penet. Class	Nominal Line Size	Isolation Valves Provided (16)		Isolation Valve Position				Fluid Contained	Auto Actuation Signal (1)	1971 GDC Or Exception Met (2)	Valve Closure Time (Sec) (5)		Valve Type (3)		Power Source (4)		FSAR Fig. No.
				Inside	Outside	Normal	Shutdown (13)	DBA (14)	Failure (15)				Inside	Outside	Inside	Outside	Inside	Outside	
Reactor Coolant System Charging	15	A	3	Check	Auto-Trip	Open	Open	Closed	AS-IS	Liquid	SIS	55-(4)	Check	10	Check	MOV (Gate)	Check	MCC-E5	9.1-1
Component Cooling to Reactor Coolant Pumps (17)	58,17,18	A	6	Auto-Trip	Auto-Trip	Open	Open	Closed	Closed	Liquid	CIB	56-(4)	20	20	TV (Globe)	TV (Globe)	CA	CA	9.4-1
Component Cooling to Shroud Coolers (17)	16	A	6	Auto-Trip	Auto-Trip	Open	Open	Closed	Closed	Liquid	CIB	56-(4)	20	20	TV (Globe)	TV (Globe)	CA	CA	9.4-1
Seal Injection Water to Reactor Coolant Pump	35,36,37	A	2	Check	Rem-Man	Open	Open	Open	AS-IS	Liquid	None	FSAR 5.3.3.6	Check	10	Check (Spring Loaded)	MOV (Gate)	Check	MCC-E3	9.1-1
Primary Grade Water to Pressurizer Relief Tank	45	A	3	Check	Auto-Trip	Closed	Closed	Closed	Closed	Liquid	CIA	56-(4)	Check	8	Check (Spring Loaded)	TV (Diaphragm)	Check	CA	None
Nitrogen Supply to Pressurizer Relief Tank	49	A	3/4	Check	Auto-Trip	Closed	Closed	Closed	Closed	Gas	CIA	56-(4)	Check	5	Check (Spring Loaded)	TV (Globe)	Check	CA	None
Main Condenser Ejector Vent	89	D	6	Check	Auto-Trip	Closed	Closed	Closed	Closed	Gas	CIB	56-(4)	Check	20	Check (Weight Loaded)	TV (Globe)	Check	CA	None
Nitrogen Supply to Safety Injection Accumulators	53	A	1	Auto-Trip	Auto-Trip	Closed	Closed	Closed	Closed	Gas	CIA	56-(4)	5	5	TV (Globe)	TV (Globe)	CA	CA	6.3-1
Component Cooling from Reactor Coolant Pump B & C Thermal Barriers (17)	8	A	3	Auto-Trip	Auto-Trip	Open	Open	Closed	Closed	Liquid	CIB	56-(4)	8	8	TV (Globe)	TV (Globe)	CA	CA	9.4-1
Reactor Coolant Pump - Seal Water Return (17)	19	A	3	Auto-Trip Check Valve	Auto-Trip	Open	Open	Closed	AS-IS	Liquid	CIA	55-(4)	10	10	MOV (Gate) Check (Spring Loaded)	MOV (Gate)	MCC-E5	MCC-E6	9.1-1
Component Cooling from Reactor Coolant Pump B & C Motors (17)	25	A	6	Auto-Trip	Auto-Trip	Open	Open	Closed	Closed	Liquid	CIB	56-(4)	8	8	TV (Globe)	TV (Globe)	CA	CA	9.4-1
Reactor Coolant System Letdown (17)	28	A	2	3 Auto-Trip	Auto-Trip	Open	Sometimes Closed	Closed	Closed	Liquid	CIA	55-(4)	5	5	TV (Globe)	TV (Globe)	CA	CA	9.1-1
			2	Rem-Man		Closed	Sometimes Open	Closed	AS-IS	Liquid	None	FSAR 1A.55							
Primary Drain Transfer Pump No. 1 Discharge (17)	29	A	2	Auto-Trip	Auto-Trip	Open	Open	Closed	Closed	Liquid	CIA	56-(4)	5	5	TV (Globe)	TV (Globe)	CA	CA	None
Reactor Containment Sump Pump Discharge (17)	38	A	2	Auto-Trip	Auto-Trip	Sometimes Open	Sometimes Open	Closed	Closed	Liquid	CIA	56-(4)	5	5	TV (Globe)	TV (Globe)	CA	CA	None
Component Cooling from Reactor Coolant Pump A Thermal Barrier (17)	26	A	2	Auto-Trip	Auto-Trip	Open	Open	Closed	Closed	Liquid	CIB	56-(4)	5	5	TV (Globe)	TV (Globe)	CA	CA	9.4-1
Component Cooling from Reactor Coolant Pump A Motor (17)	27	A	4	Auto-Trip	Auto-Trip	Open	Open	Closed	Closed	Liquid	CIB	56-(4)	8	8	TV (Globe)	TV (Globe)	CA	CA	9.4-1

Table 5.3-1 (CONT'D)

CONTAINMENT ISOLATION ARRANGEMENTS

Service	Penet. No.	Penet. Class	Nominal Line Size	Isolation Valves Provided (16)		Isolation Valve Position				Fluid Contained	Auto Actuation Signal (1)	1971 GDC Or Exception Met (2)	Valve Closure Time (Sec) (5)		Valve Type (3)		Power Source (4)		FSAR Fig. No.
				Inside	Outside	Normal	Shutdown (13)	DBA (14)	Failure (15)				Inside	Outside	Inside	Outside	Inside	Outside	
Air Activity Monitor Return to Containment	43	A	1	Auto-Trip	Auto-Trip	Open	Open	Closed	Closed	Gas	CIA ⁽⁹⁾	56-(4)	5	5	TV (Globe)	TV (Globe)	SOV	SOV	None
Primary Vent Header	48	A	1 1/2	Auto-Trip	Auto-Trip	Open	Open	Closed	Closed	Gas	CIA	56-(4)	5	5	TV (Globe)	TV (Globe)	CA	CA	None
Safety Injection Accumulators Sample (17)	55-1	A	3/8	Auto-Trip	Auto-Trip	Open	Open	Closed	Closed	Liquid	CIA	56-(4)	5	5	TV (Globe)	TV (Globe)	CA	CA	None
Pressurizer Relief Tank Gas Sample	55-4	A	3/8	Auto-Trip	Auto-Trip	Open	Open	Closed	Closed	Gas	CIA	55-(4)	5	5	TV (Globe)	TV (Globe)	CA	CA	None
Pressurizer Liquid Space Sample (17)	56-1	A	3/8	Auto-Trip	Auto-Trip	Open	Open	Closed	Closed	Liquid	CIA	55-(4)	5	5	TV (Globe)	TV (Globe)	CA	CA	None
Reactor Coolant Cold Leg Samples (17)	56-2	A	3/8	Auto-Trip	Auto-Trip	Closed	Closed	Closed	Closed	Liquid	CIA(9)	55-(4)	5	5	TV (Globe)	TV (Globe)	SOV	SOV	None
Reactor Coolant Hot Leg Samples (17)	56-3	A	3/8	Auto-Trip	Auto-Trip	Closed	Closed	Closed	Closed	Liquid	CIA(9)	55-(4)	5	5	TV (Globe)	TV (Globe)	SOV	SOV	None
Pressurizer Vapor Space Sample (17)	105-2	A	3/8	Auto-Trip	Auto-Trip	Open	Open	Closed	Closed	Gas	CIA	55-(4)	5	5	TV (Globe)	TV (Globe)	CA	CA	None
Component Cooling Water from Residual Heat Exchangers and Residual Heat Removal Pump Seal Coolers (17)	2,4	D	18	Rem-Man	Manual	Closed	Sometimes Open	Closed	AS-IS	Liquid	None	56-(1) FSAR 1A.56	60	Manual	MOV (Butterfly)	Manual (Butterfly)	A: MCC-E5 Manual B: MCC-E6	9.4-1	
Component Cooling Water to Residual Heat Exchangers and Residual Heat Removal Pump Seal Coolers (17)	1,5	D	18	Rem-Man	Manual	Closed	Sometimes Open	Closed	AS-IS	Liquid	None	56-(1) FSAR 1A.56	60	Manual	MOV (Butterfly)	Manual (Butterfly)	A: MCC-E5 Manual B: MCC-E6	9.4-1	
Air Recirculation Cooling Water - In (17)	14	A	8	Auto-Trip	Auto-Trip	Open	Open	Closed/ Open	Closed	Liquid	CIB	56-(4)	20	20	TV (Globe)	TV (Globe)	CA	CA	None
Air Recirculation Cooling Water - out (17)	11	A	8	Auto-Trip	2 Auto-Trip	Open	Open	Closed/ Open	Closed	Liquid	CIB	56-(4)	20	20	TV (Globe)	TV (Globe)	CA	CA	None
Steam Generator Blowdown	39,40,41	B	3	Sealed System	Auto-Trip	Open	Open	Closed	Closed	Liquid	CIA	57	Sealed System	8	Sealed System	TV (Globe)	Sealed System	CA	10.3-6
Steam Generator Blowdown Samples	56-4, 105-1, 97-4	B	3/8	Sealed System	Auto-Trip	Open	Open	Closed	Closed	Liquid	CIA	57	Sealed System	5	Sealed System	TV (Globe)	Sealed System	CA	None
Feedwater	76,77,78	B	16	Sealed System	Auto-Trip	Open	Open	Closed	AS-IS	Liquid	FWI	57	Sealed System	10	Sealed System	TV (Gate)	Sealed System	MCC-E5	10.3-5
			3	Sealed System	Check	Closed	Sometimes Open	Open	AS-IS	Liquid	None	FSAR 5.3.3.9	Sealed System	Check	Sealed System	Check	Sealed System	Check	10.3-5
Residual Heat Removal Inlet and Outlet Sample (17)	97-1, 97-2	A	3/8	Auto-Trip	Auto-Trip	Open	Open	Closed	Closed	97-1 Liquid 97-2 Liquid	CIA	56-(4)	5	5	TV (Globe)	TV (Globe)	CA	CA	None

Table 5.3-1 (CONT'D)

CONTAINMENT ISOLATION ARRANGEMENTS

Service	Penet. No.	Penet. Class	Nominal Line Size	Isolation Valves Provided (16)		Isolation Valve Position				Fluid Contained	Auto Actuation Signal (1)	1971 GDC Or Exception Met (2)	Valve Closure Time (Sec) (5)		Valve Type (3)		Power Source (4)		FSAR Fig. No.
				Inside	Outside	Normal	Shutdown (13)	DBA (14)	Failure (15)				Inside	Outside	Inside	Outside	Inside	Outside	
Component Cooling Water from Shroud Coolers (17)	9	A	6	Auto-Trip	Auto-Trip	Open	Open	Closed	Closed	Liquid	CIB	56-(4)	20	20	TV (Globe)	TV (Globe)	CA	CA	9.4-1
River Water to Recirculation Spray Heat Exchangers	79,80,81,82	C	14	Sealed System	Rem-Man	Open	Open	Open	AS-IS	Liquid	None	57	Sealed System	60	Sealed System	MOV (Butterfly)	Sealed System	A&C: MCC-E5 B&D: MCC-E6	9.9-1
River Water from Recirculation Spray Heat Exchangers (17)	83,84,85,86	C	14	Sealed System	Rem-Man Manual	Open Open	Open Open	Open Open	AS-IS AS-IS	Liquid Liquid	None None	57 FSAR 5.3.3.10	Sealed System	60 Manual	Sealed System	MOV (Butterfly) Ball	Sealed System	A&C: MCC-E5 B&D: MCC-E6	9.9-1
High Head Safety Injection to Hot Legs	7,33	C	3	Check	Rem-Man	Closed	Closed	Closed	AS-IS	Liquid	None	FSAR 5.3.3.1	Check	10	Check	MOV (Gate)	Check	869A: MCC-E5 869B: MCC-E6	6.3-8
Low Head Safety Injection	60,61,62	C	6	Check	Rem-Man	Closed (60,62) Open (61)	Closed (60,62) Open (61)	Closed/ Open (60,62) Open (61) Note 18	AS-IS AS-IS	Liquid	None	FSAR 5.3.3.1	Check	10	Check	MOV (Venturi Gate)	Check	890A: MCC-E5 890 B&C: MCC-E6	6.3-8
Quench Spray Pump - Discharge	63,64	C	10	Check	Auto-Trip	Closed	Closed	Open	AS-IS	Liquid	CIB	FSAR 5.3.3.5	Check	62 (to open)	Check (Weight Loaded)	MOV (Gate)	Check	A: MCC-E5 B: MCC-E6	6.4-1A
Outside Recirculation Spray Pump - Discharge	70,71	C	10	Check	Auto-Trip	Open	Open	Open	AS-IS	Liquid	CIB	FSAR 5.3.3.5	Check	60 (to open)	Check (Weight Loaded)	MOV (Gate)	Check	A: MCC-E5 B: MCC-E6	6.4-1B
Boron Injection (High Head Safety Injection to Cold Legs)	113	C	3	Check	2 Auto-Trip	Closed	Closed	Open	AS-IS	Liquid	SIS	FSAR 5.3.3.1	Check	867C: 10 (to open) 867D: 15 (to open)	Check	2 MOV (Gate)	Check	867C: MCC-E5 867D: MCC-E6	6.3-8
High Head Safety Injection to Cold Legs	96	C	3	Check	Rem-Man	Closed	Closed	Closed/ Open	AS-IS	Liquid	None	FSAR 5.3.3.1	Check	10	Check	MOV (Gate)	Check	MCC-E5	6.3-8
Outside Recirculation Spray Pump Suction from Containment Sump	66,67	C	12	None	Rem-Man	Open	Open	Open	AS-IS	Liquid	CIB	FSAR 5.3.3.2	None	60	None	MOV (Gate)	None	A: MCC-E5 B: MCC-E6	6.4-1B
Low Head Safety Injection Pump Suction from Containment Sump	68,69	C	12	None	Rem-Man	Closed	Closed	Open (6)	AS-IS	Liquid	None	FSAR 5.3.3.2	None	120	None	MOV (Gate)	None	860A: MCC-E5 860B: MCC-E6	6.3-8
Containment to Air Activity Monitor	44	A	1	None	2 Auto-Trip in Series	Open	Open	Closed	Closed	Gas	CIA	56-(4) FSAR 1.3.3.11	None	5	None	TV (Globe)	None	CA	None

Table 5.3-1 (CONT'D)

CONTAINMENT ISOLATION ARRANGEMENTS

Service	Penet. No.	Penet. Class	Nominal Line Size	Isolation Valves Provided (16)		Isolation Valve Position				Fluid Contained	Auto Actuation Signal (1)	1971 GDC Or Exception Met (2)	Valve Closure Time (Sec) (5)		Valve Type (3)		Power Source (4)		FSAR Fig. No.
				Inside	Outside	Normal	Shutdown (13)	DBA (14)	Failure (15)				Inside	Outside	Inside	Outside	Inside	Outside	
Containment Vacuum Pump/Hydrogen Recombiner Suction	92,93	A	2	None	2 Auto-Trip in Series; 2 Manual	One Open/ One Closed	One Open/ One Closed	Closed/ Open	Closed	Gas	CIA(9)	FSAR 5.3.3.8	None	5	None	2-TV (Globe)	None	CA,SOV	None
						Closed	Closed	Closed	AS-IS	Gas	None	FSAR 5.3.3.8	None	Manual	None	2 Manual (Ball)	None	Manual	6.5-1
Reactor Coolant System Fill	46	D	2	Check	Rem-Man	Closed	Sometimes Open	Closed	Closed	Liquid	None	56-(2) FSAR 1A.56	Check	5	Check (Spring Loaded)	TV (Globe)	Check	CA	9.1-1
Safety Injection Accumulator Make-up	20	A	1	Check	Manual	Closed	Sometimes Open	Closed	AS-IS	Liquid	None	56-(2)	Check	Manual	Check (Spring Loaded)	Manual (Globe)	Check	Manual	6.3-1
Residual Heat Removal to Refueling Water Storage Tank	24	D	6	2 Manual	Manual	Closed	Closed	Closed	AS-IS	Air	None	56-(1)	Manual	Manual	Manual (Gate) (Ball)	Manual (Gate)	Manual	Manual	9.3-1
Fuel Transfer Tube	65	D	20	Blind Flange	Manual	Closed	Closed	Closed	AS-IS	Liquid	None	FSAR 5.3.3.3 9.12.2.2	Blind Flange	Manual	Blind Flange	Manual (Gate)	Blind Flange	Manual	5.3-1
Safety Injection Accumulator Test Line	106	D	3/4	Auto-Trip	Auto-Trip	Closed	Closed	Closed	Inside: AS-IS Outside : Closed AS-IS	Air	CIA	56-(4)	10	5	MOV (Globe)	TV (Globe)	MCC-E6	CA	6.3-1
Containment Purge Exhaust	90	D	42	Auto-Trip (7)	Auto-Trip (7)	Closed	Closed	Closed	AS-IS	Gas	Valves trip shut on containment high-high radiation during normal operation or refuel- ing (7)	56-(1)	8	8	MOV (Butter- fly)	MOV (Butter- fly)	MCC-E12	MCC-E11	5.4-1
Containment Purge Supply	91	D	42	Auto-Trip (7)	Auto-Trip (7)	Closed	Closed	Closed	AS-IS	Gas	Valves trip shut on containment high-high radiation during normal operation or refuel- ing (7)	56-(1)	11	8	MOV (Butter- fly)	MOV (Butter- fly)	MCC-E12	MCC-E11	5.4-1
Containment Vacuum Ejector Suction	94	D	8	Manual	Manual	Closed	Closed	Closed	AS-IS	Gas	None	56-(1)	Manual	Manual	Manual (Butter- fly)	Manual (Butter- fly)	Manual	Manual	None

Table 5.3-1 (CONT'D)

CONTAINMENT ISOLATION ARRANGEMENTS

Service	Penet. No.	Penet. Class	Nominal Line Size	Isolation Valves Provided (16)		Isolation Valve Position				Fluid Contained	Auto Actuation Signal (1)	1971 GDC Or Exception Met (2)	Valve Closure Time (Sec) (5)		Valve Type (3)		Power Source (4)		FSAR Fig. No.
				Inside	Outside	Normal	Shutdown (13)	DBA (14)	Failure (15)				Inside	Outside	Inside	Outside	Inside	Outside	
Refueling Cavity to Purification Pump Suction	104	D	6	Manual	Manual	Closed	Closed	Closed	AS-IS	Air	None	56-(1)	Manual	Manual	Manual (Ball)	Manual (Ball)	Manual	Manual	9.5-1
Purification Pump Discharge to Refueling Cavity	103	D	6	Manual	Manual	Closed	Closed	Closed	AS-IS	Air	None	56-(1)	Manual	Manual	Manual (Ball)	Manual (Ball)	Manual	Manual	9.5-1
Compressed Air to Containment	42	D	2	Check	Manual	Closed	Open/ Closed	Closed	AS-IS	Gas	None	56-(2)	Check	Manual	Check (Spring Loaded)	Manual (Gate)	Check	Manual	9.8-1
Instrument Air	47	D	2	Check	Manual/ Auto trip	Closed/ Open	Closed/ Open	Closed/ Open	AS-IS/ Closed	Gas	None/ CIB	56-(2)	Check	Manual/ < 60	Check (Spring Loaded)	Manual (Gate)/ TV (Butterfly)	Check	Manual/ CA/SOV PNL-DC2 Train B PNL-DC3 Train A	9.8-1
Pressurizer Dead Weight Calibrator	110	D	1/8	None	2 Manual in series	Closed	Closed/ Open	Closed	AS-IS	Gas	None	FSAR 5.3.3.4	None	Manual	None	Manual (Globe)	None	Manual	None
Containment Leakage Monitoring - Open Taps and Containment Wide Range Pressure Monitoring	55-2, 57-1, 97-3, 57-2	A	3/8(8)	None	Manual Valve (11) and capped pipe	Closed	Closed	Closed	AS-IS	Gas	None	FSAR 5.3.3.4	None	Manual	None	Manual (Globe)	None	Manual	None
Hydrogen Recombiner Discharge to Containment	87,88	A	2	None	2 Manual in series	Closed	Closed	Closed/ Open	AS-IS	Gas	None	FSAR 5.3.3.8	None	Manual	None	Manual (Ball)	None	Manual	6.5-1
Main Steam (17)	73,74,75	B	32	Sealed system	Auto-Trip	Open	Closed/ Open	Closed	Closed ⁽¹²⁾	Steam	SLI	57	Sealed System	5	Sealed System	TV (Nonreturn)	Sealed System	CA	10.3-1
Main Steam to Turbine Drive for Steam Generator Auxiliary Feed Pump (17)	73,74,75	B	3	Sealed System	Rem-Man	Open	Open/ Closed	Open	AS-IS	Steam	None	57	Sealed System	30	Sealed System	MOV (Gate)	Sealed System	MCC-E6	10.3-1
Main Steam Safety Valves (17)	73,74,75	B	6	Sealed System	Safety Valves	Closed	Closed	Closed	Closed	Steam	None	57	Sealed System	--	Sealed System	Safety Valves	Sealed System	--	10.3-1
Main Steam Atmospheric Dump Valves (17)	73,74,75	B	6	Sealed System	Pressure Control Valves	Closed	Open/ Closed	Closed/ Open	Closed	Steam	None	57	Sealed System	--	Sealed System	Control Valve (Globe)	Sealed System	CA	10.3-1
Main Steam Line Drains (17)	73,74,75	B	1 1/2	Sealed System	Auto-Trip	Open	Open	Closed	Closed	Liquid	SLI	57	Sealed System	5	Sealed System	TV (Globe)	Sealed System	CA	10.3-1
Main Steam Residual Heat Release Valve (17)	73,74,75	B	4	Sealed System	Residual Heat Release Valve	Closed	Closed/ Open	Closed/ Open	Closed	Steam	None	57	Sealed System	--	Sealed System	Control Valve (Globe)	Sealed System	CA	10.3-1
Main Steam Isolation Bypass (17)	73,74,75	B	2	Sealed System	Rem-Man	Closed	Closed	Closed	AS-IS	Steam	SLI	57	Sealed System	--	Sealed System	MOV	Sealed System	MCC E06	10.3-1
Containment Hydrogen Monitoring System	95,109	C	3/8	SOV	SOV	Closed	Closed	Open	Closed	Gas	None	56-(4)	--	--	SOV (Globe)	SOV (Globe)	PNL-DC2 Train B PNL-DC3 Train A	PNL-DC2 Train B PNL-DC3 Train A	6.5-4

Table 5.3-1 (CONT'D)

CONTAINMENT ISOLATION ARRANGEMENTS

Service	Penet. No.	Penet. Class	Nominal Line Size	Isolation Valves Provided (16)		Isolation Valve Position				Fluid Contained	Auto Actuation Signal (1)	1971 GDC Or Exception Met (2)	Valve Closure Time (Sec) (5)		Valve Type (3)		Power Source (4)		FSAR Fig. No.
				Inside	Outside	Normal	Shutdown (13)	DBA (14)	Failure (15)				Inside	Outside	Inside	Outside	Inside	Outside	
Personnel Air Lock (PH-P-1)	None	D	1 1/2	2 Manual in Series	2 Manual in Series	Admin. Closed	Closed	Closed	AS-IS	Air	None	56	Manual	Manual	Ball	Ball	Manual	Manual	5.2-23
Emergency Air Lock (PH-P-2)	None	D	2	Manual	Manual	Admin. Closed	Closed	Closed	AS-IS	Air	None	56	Manual	Manual	Ball	Ball	Manual	Manual	5.2-23
Reactor Vessel Level Instrumentation System (RVLIS)	95,109	D	1/4	None	(10)	--	--	--	--	Liquid	None	FSAR 5.3.3.7	--	--	None	--	--	--	None
Fire Protection - Containment Hose Reel Stations	13	D	4	Check	Auto-Trip	Closed	Closed	Closed	Closed	Liquid	CIA	56	Check	15	Check (Weight Loaded)	TV (Globe)	Check	CA	None
Fire Protection - Containment Cable Penetrations	31	D	4	Check	Auto-Trip	Closed	Closed	Closed	Closed	Liquid	CIA	56	Check	15	Check (Weight Loaded)	TV (Globe)	Check	CA	None
Fire Protection - Containment RHR Platform	32	D	3	Check	Auto-Trip	Closed	Closed	Closed	Closed	Liquid	CIA	56	Check	15	Check (Weight Loaded)	TV (Globe)	Check	CA	None

- Notes: (1) Sections 7.3 and Tables 7.3-1 and 7.3-2 describe actuation signals.
- (2) 1971 General Design Criterion met is listed with appropriate subparagraph in parentheses. Exceptions to General Design Criterion arrangements are noted by explanatory Updated FSAR section numbers.
- (3) MOV = Electric Motor Operated Valve
TV = Trip Valve
- (4) For electric motor-operated valves (MOVs) the appropriate motor control center is listed. Those which have an even number receive power from Emergency Generator 2. Those which have an odd number receive power from Emergency Generator 1 (Section 8.5). For air-operated trip valves (TVs), the power supply is designated CA for compressed air (Section 9.8). For solenoid operated trip valves (TVs), the power supply is designated SOV for solenoid operator. Secondary modes of operation are provided for MOVs in the form of manually operated (handwheels) overrides; for TVs, CA pressure can be bled off or solenoid operator can be de-energized to trip valves to fail-safe positions. Manual and check valves have no specific secondary modes of operation.
- (5) Closure times listed for air-operated trip valves are calculated assuming no plug weight or stem friction. These values are not included to represent limits of operation, rather they are reflective of the original purchase specification. These do not necessarily represent values relied upon in DBA analyses. Refer to LRM Table 5.1-1 for such information.
- (6) Opened to shift low head safety injection from refueling water storage tank to containment sump (Section 6.3).
- (7) Containment purge isolation valves are normally shut, remote manually operated, administratively controlled valves, meeting General Design Criterion 56-(1). During refueling operations, these valves are opened and may be configured to trip shut automatically on receipt of a containment high-high radiation signal (Section 11.3).
- (8) Each of the containment leakage monitoring and containment wide range pressure monitoring lines have a one-eighth inch orifice on the inside of containment located adjacent to the three-eighth inch penetration, consistent with the requirement of Safety Guide 11 (Section 1.3). Penetrations 57-3 and 57-4 have been capped inside and outside of containment.
- (9) CIA can be overridden for sample inlet valves (Section 9.6.1.1).
- (10) Hydraulic isolation device.
- (11) Penetrations 55-2, 57-1, 57-2, and 97-3 are isolated outside of containment downstream of the containment pressure instrumentation by a sealed manual valve and pipe cap.
- (12) Main Steam Isolation Valve fails closed on loss of air and fails AS-IS on loss of 125 VDC.
- (13) Shutdown position in Modes 3 and 4.
- (14) DBA position is for LBLOCA.
- (15) Failure position is for power failure.
- (16) Relief valves listed in [Licensing Requirements Manual](#) Table 3.6.1-1.
- (17) Relief valve between isolation valves.
- (18) Based on single failure considerations, MOV-1SI-890C (Penetration No. 62) may be required to be closed to isolate LHSI flow to the cold legs during transfer to post-DBA simultaneous cold leg and hot leg recirculation.

Table 5.6-1

CLARIFICATION OF ASME BOILER AND PRESSURE VESSEL CODE,
SECTION III PARAGRAPHS

<u>Paragraph Number</u>	<u>Clarification</u>
N-511	All welding material was tested per paragraphs N-511.4 or N-511.5.
N-522	Deposited weld metal was subjected to the drop weight tear test as developed by the U.S. Naval Research Laboratory (NRL Report 6300) or the ASTM E 208 Drop Weight Test, depending upon the thickness of the weld joints.
N-526	When it was physically impossible to remove backing strips, they are not removed. Only the radiographed portions of the welded joints were prepared as specified in the ASME Boiler and Pressure Vessel Code, Section IX, paragraph 333 (a).
N-527	If the proximity and size of welds are such that, using established techniques and procedures, the resultant distortion was not sufficiently restrained to retain the required configuration, controlled peening was performed.
N-611	Compliance with paragraph N-611.2(c) was confined to monitoring compliance with subject paragraphs only.

Removed in Accordance with RIS 2015-17

FIGURE 5.1-1
MACHINE LOCATION REACTOR
CONTAINMENT SH. 1
PLAN EL. 767'-10"
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

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FIGURE 5.1-2
MACHINE LOCATION REACTOR
CONTAINMENT - SH. 2
PLAN ELEVATION 738'-10"

BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

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UFSAR FIGURE 5.1-3
MACHINE LOCATION REACTOR
CONTAINMENT SH.3 EL.718'-6"
BEAVER VALLEY POWER STATION UNIT NO.1
UPDATED FINAL SAFETY ANALYSIS REPORT

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FIGURE 5.1-4

MAC. LOC. REACTOR CONTAINMENT SH. 4
PLAN EL. 692'-11"

BEAVER VALLEY POWER STATION UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT

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UFSAR FIGURE 5.1-5

FENOC FIRST ENERGY NUCLEAR OPERATING COMPANY		BEAVER VALLEY POWER STATION UNIT 1	
MAC. LOC. REACTOR CONTAINMENT SH. 5		L.I. SECTION 1-1, 6-6, 22-22 AND 24-24	
8700-AM-1E		AA	

Removed in Accordance with RIS 2015-17

FIGURE 5.1-6
MACHINE LOCATION REACTOR
CONTAINMENT SH. 6 SECTIONS
2-2, 5-5 AND 23-23

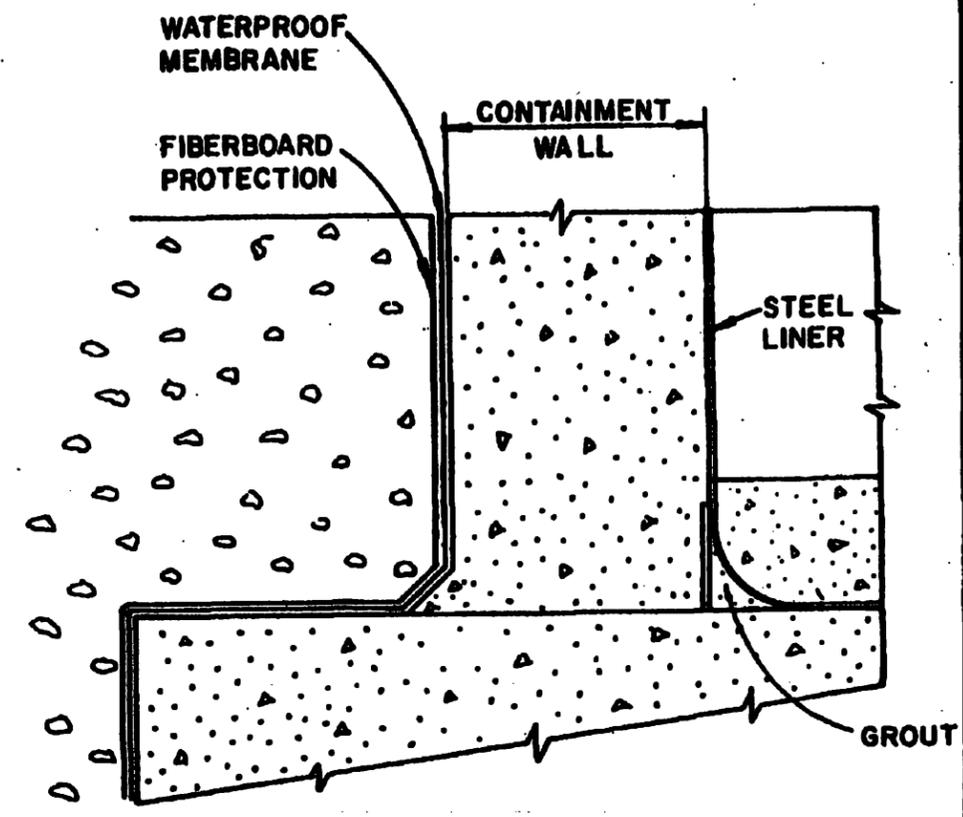
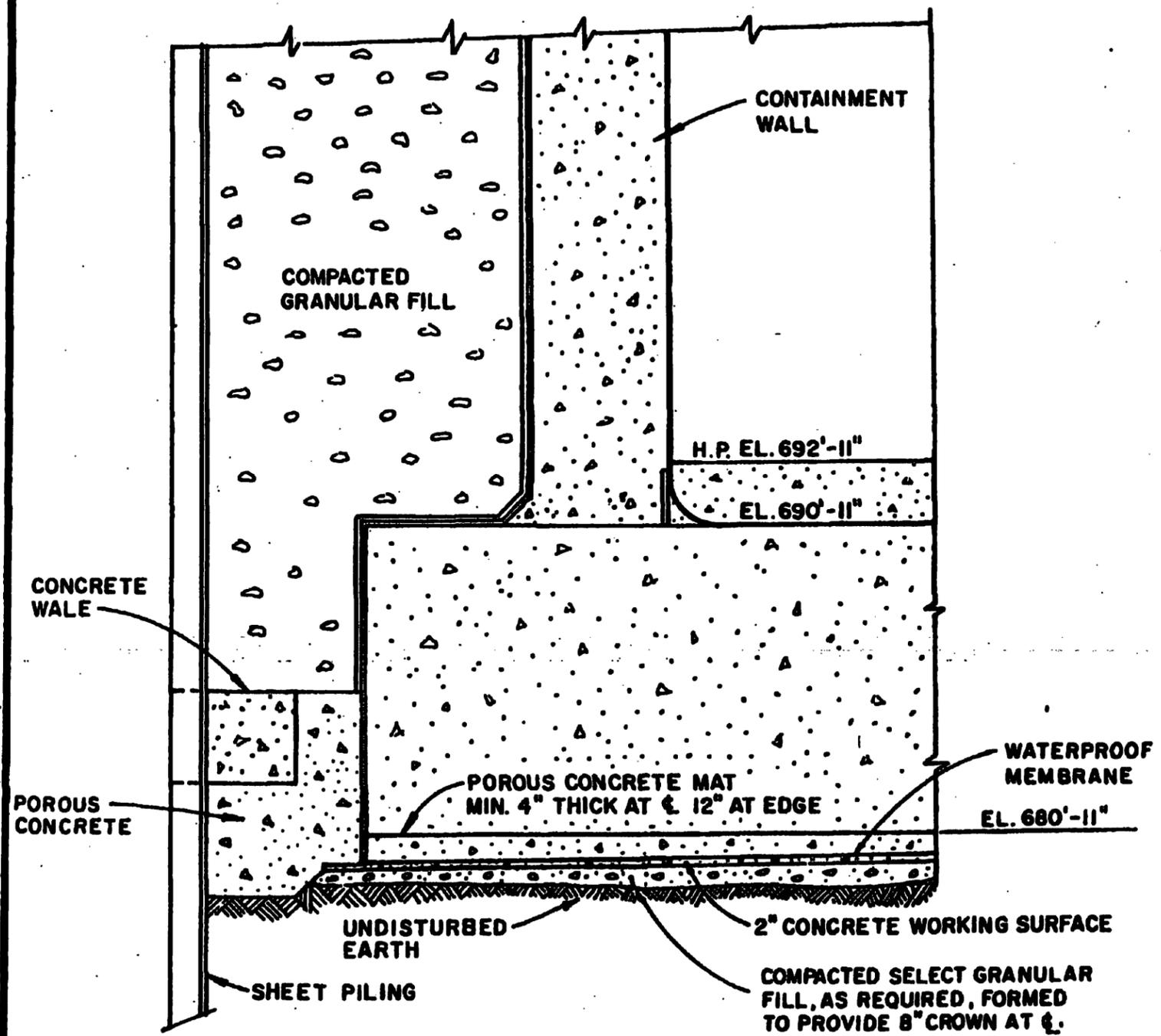
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

UFSAR FIGURE 5.1-7

FENOC FIRST ENERGY NUCLEAR OPERATING COMPANY		BEAVER VALLEY POWER STATION UNIT 1	
MACH. LOC. REACTOR CONTAINMENT SH. 7		SECTIONS 3-3 AND 4-4	
DATE: 01/20/00	BY: J. J. [unreadable]	SCALE: 1/8" = 1'-0"	REV: AA
APP: [unreadable]	CHK: [unreadable]	DATE: 01/20/00	8700-RM-10 14

8 9 1



DETAIL

- NOTES:
1. NORMAL GROUND WATER EL. 665'±
 2. ORDINARY HIGH WATER EL. 675'±
 3. STANDARD PROJECT FLOOD EL. 705'
 4. PROBABLE MAXIMUM FLOOD EL. 730'

FIGURE 5-2-1
REACTOR CONTAINMENT WATERPROOFING
 BEAVER VALLEY POWER STATION UNIT NO. 1
 UPDATED FINAL SAFETY ANALYSIS REPORT

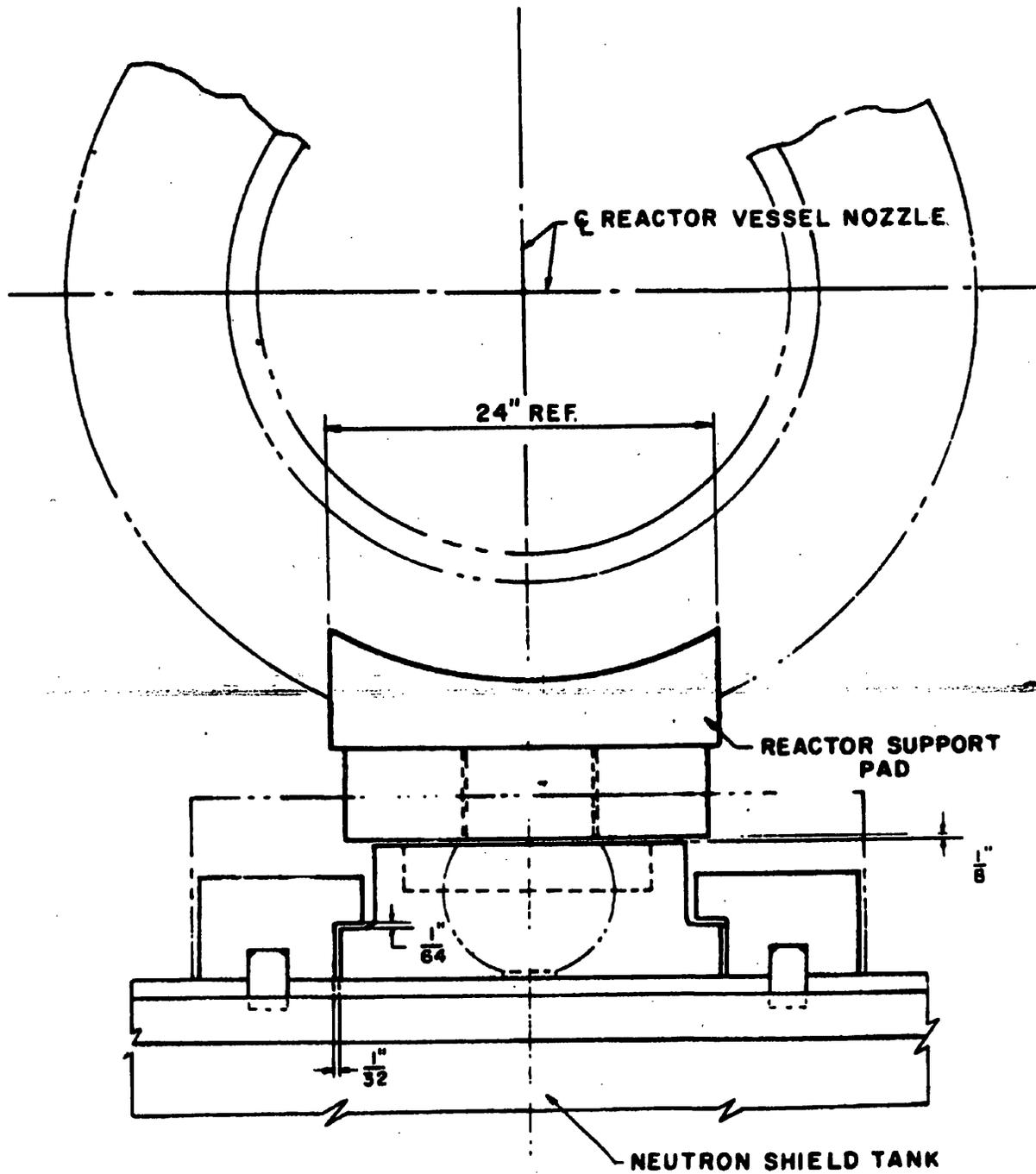


FIGURE 5-2-2
REACTOR VESSEL SUPPORT
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

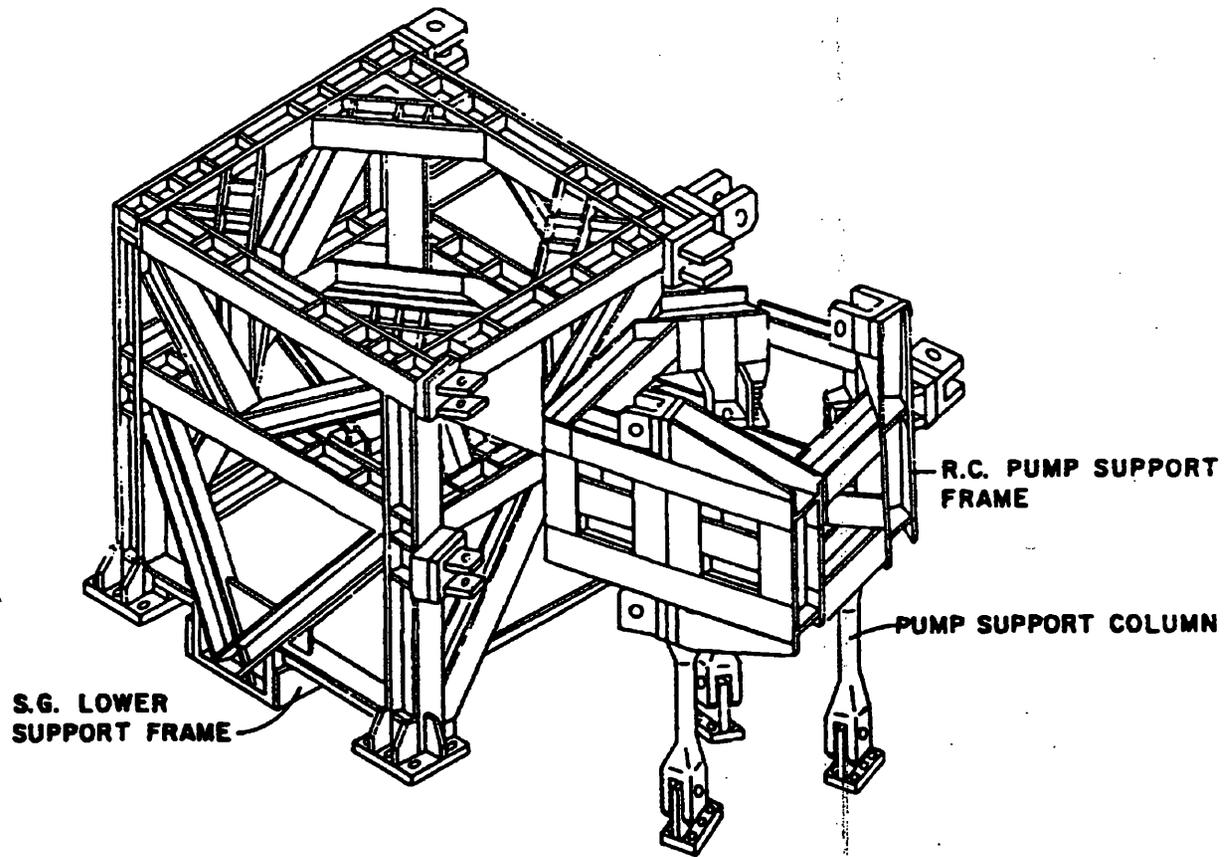


FIG. 5.2-3
STEAM GENERATOR AND REACTOR
COOLANT PUMP LOWER SUPPORTS
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

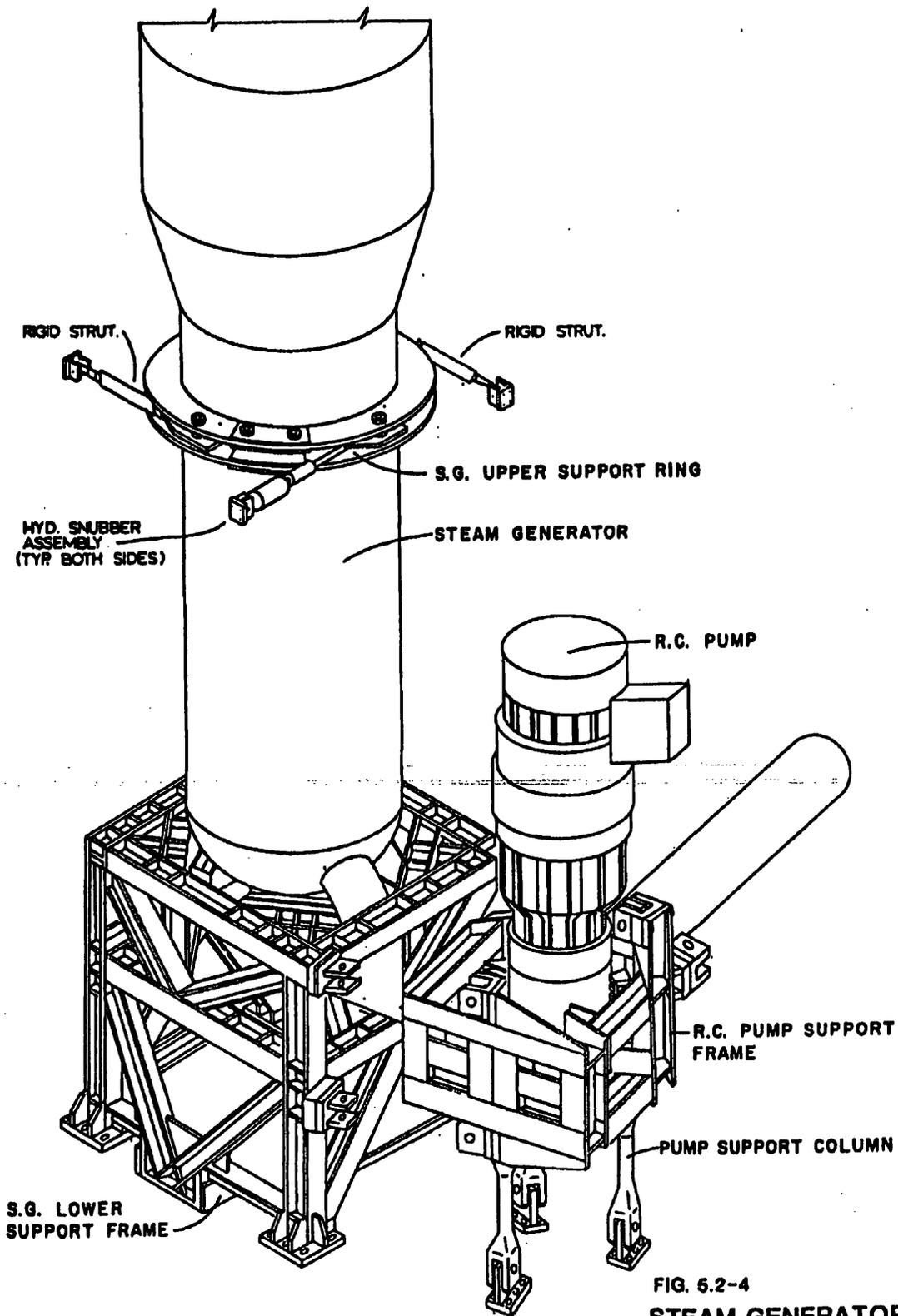


FIG. 6.2-4
STEAM GENERATOR AND
REACTOR COOLANT PUMP
BEAVER VALLEY POWER STATION UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT

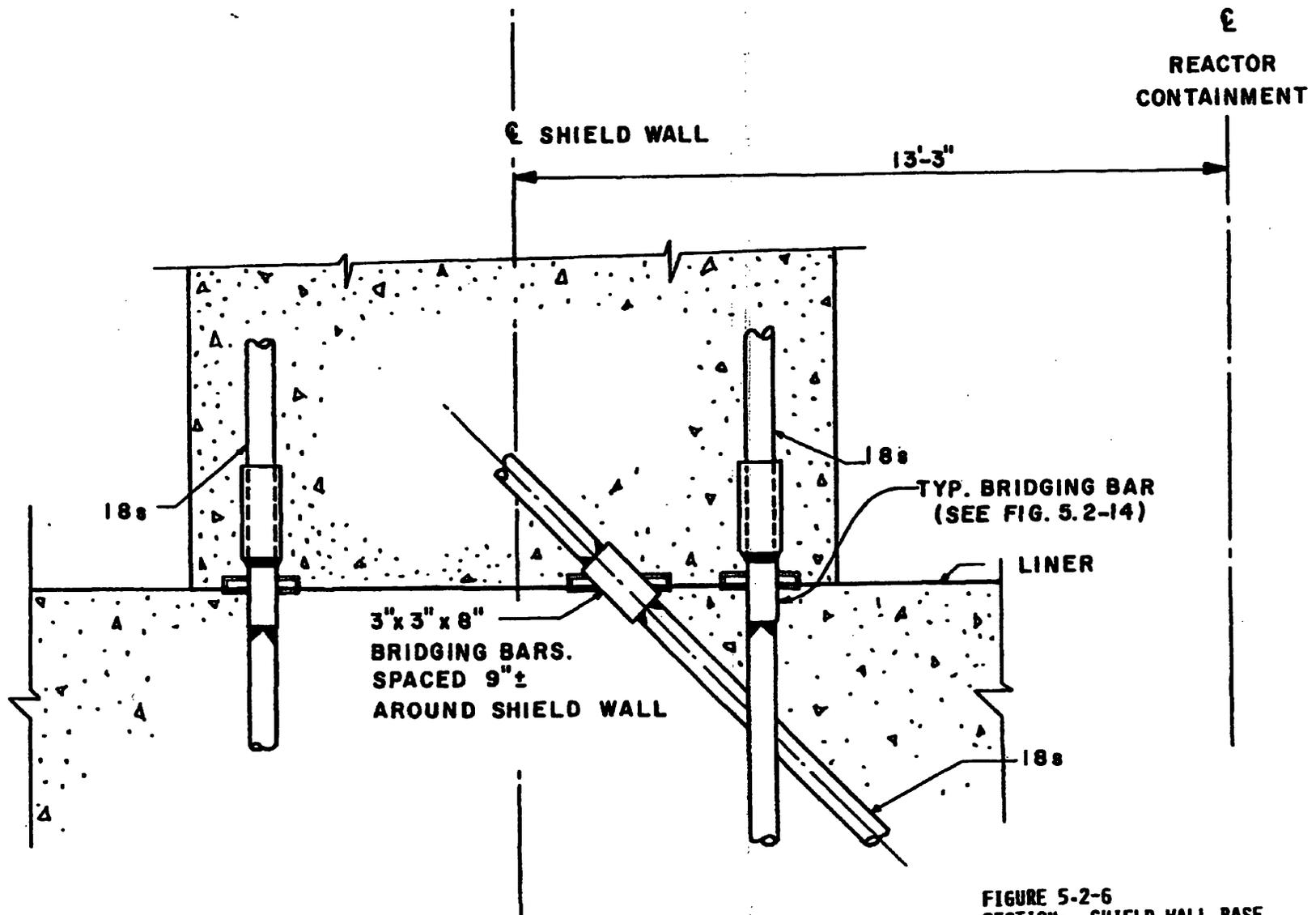


FIGURE 5-2-6
SECTION: SHIELD WALL BASE
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

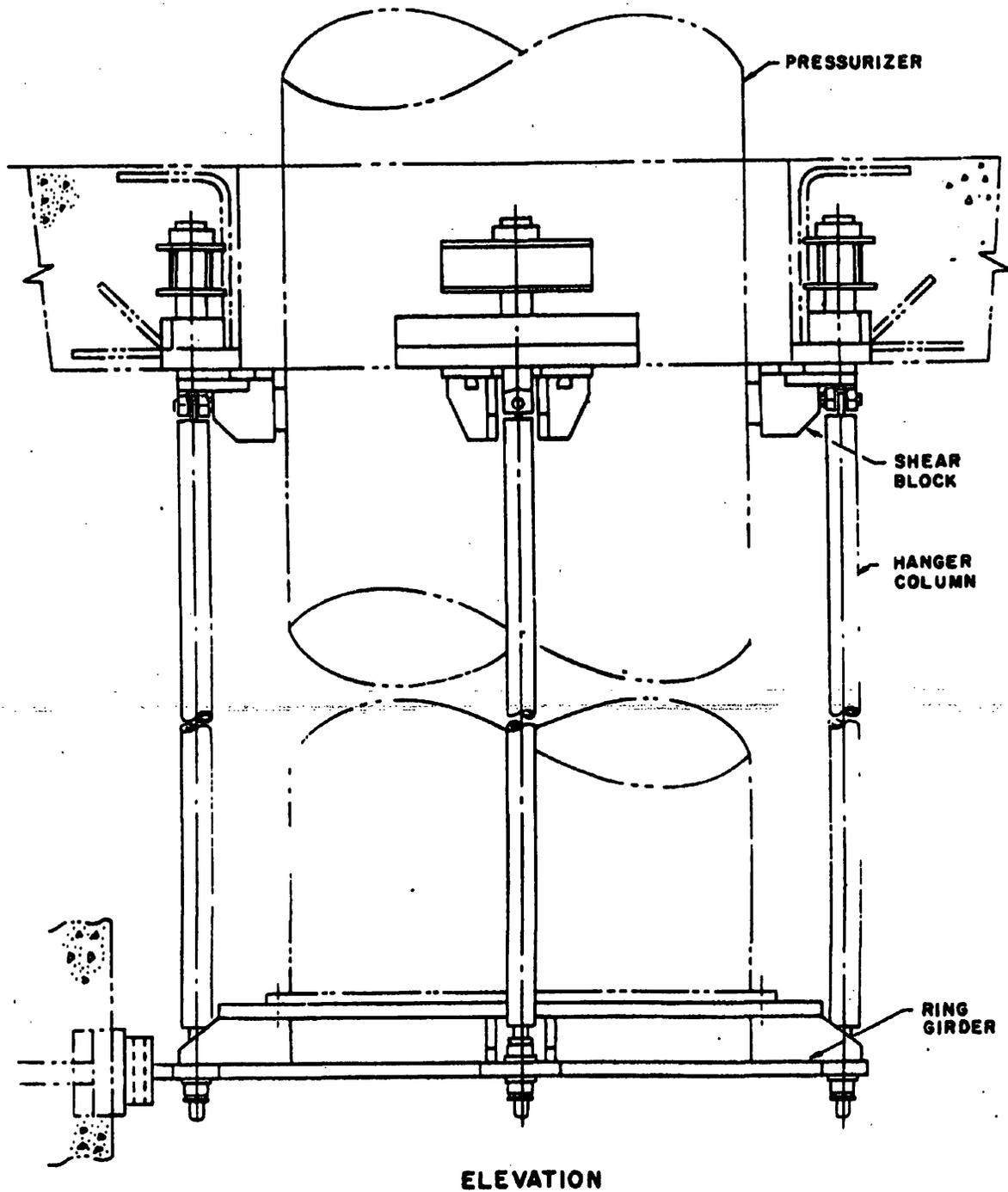


FIGURE 5-2-7
PRESSURIZER SUPPORT
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

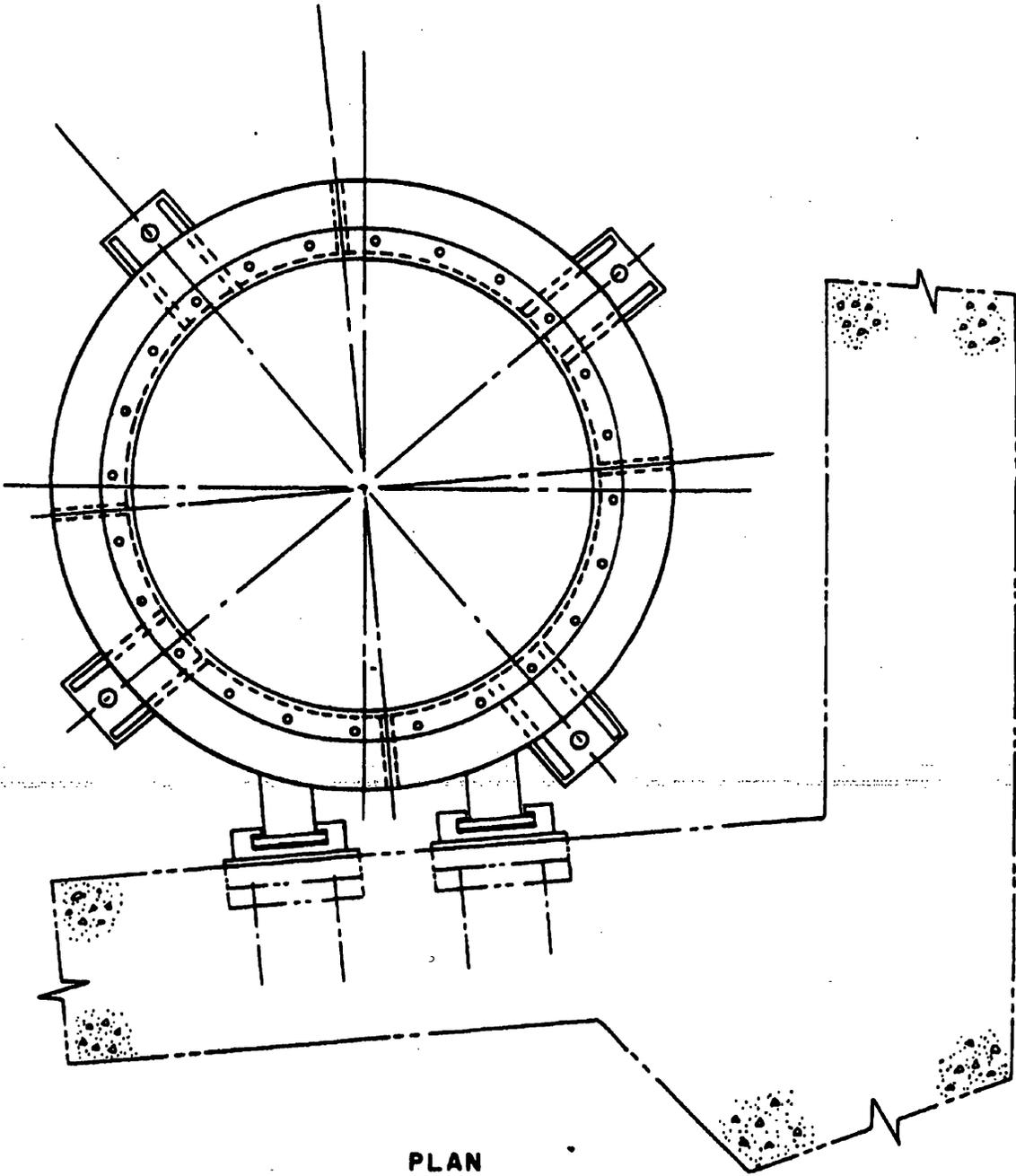


FIGURE 5-2-8
PRESSURIZER SUPPORT
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

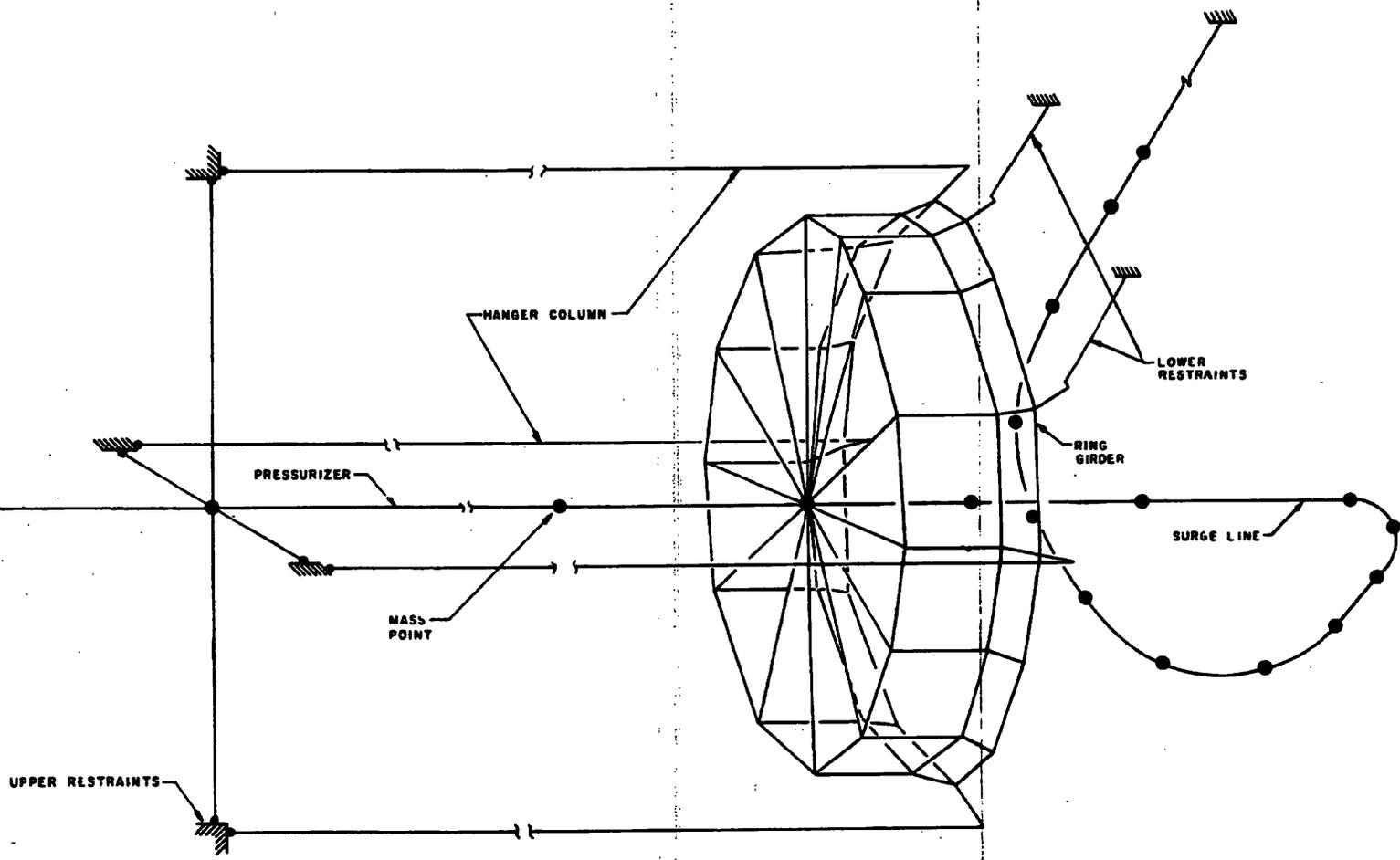


FIGURE 5-2-9
DYNAMIC MODEL FOR PRESSURIZER SUPPORTS
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

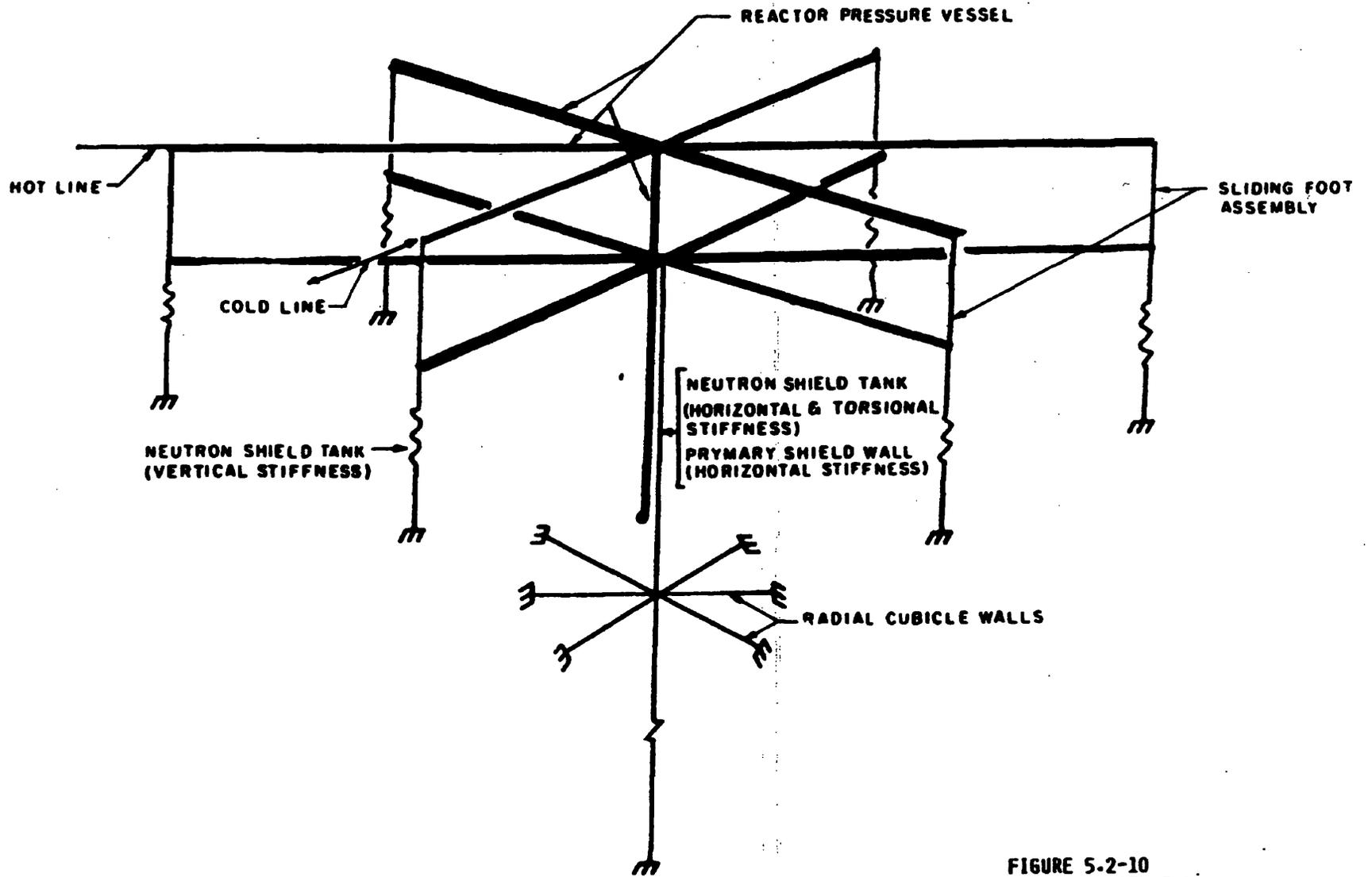
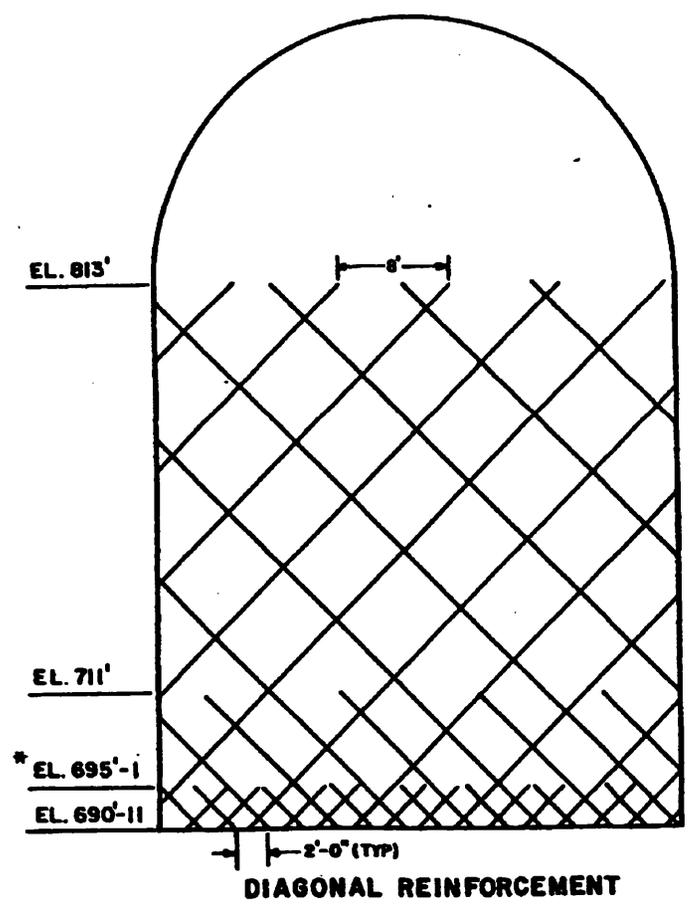


FIGURE 5-2-10
DYNAMIC MODEL FOR
REACTOR VESSEL SUPPORTS
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT



*
 AVERAGE OF 4 CUT-OFF-POINTS
 (692'-1", 694'-1", 696'-1", 698'-1")

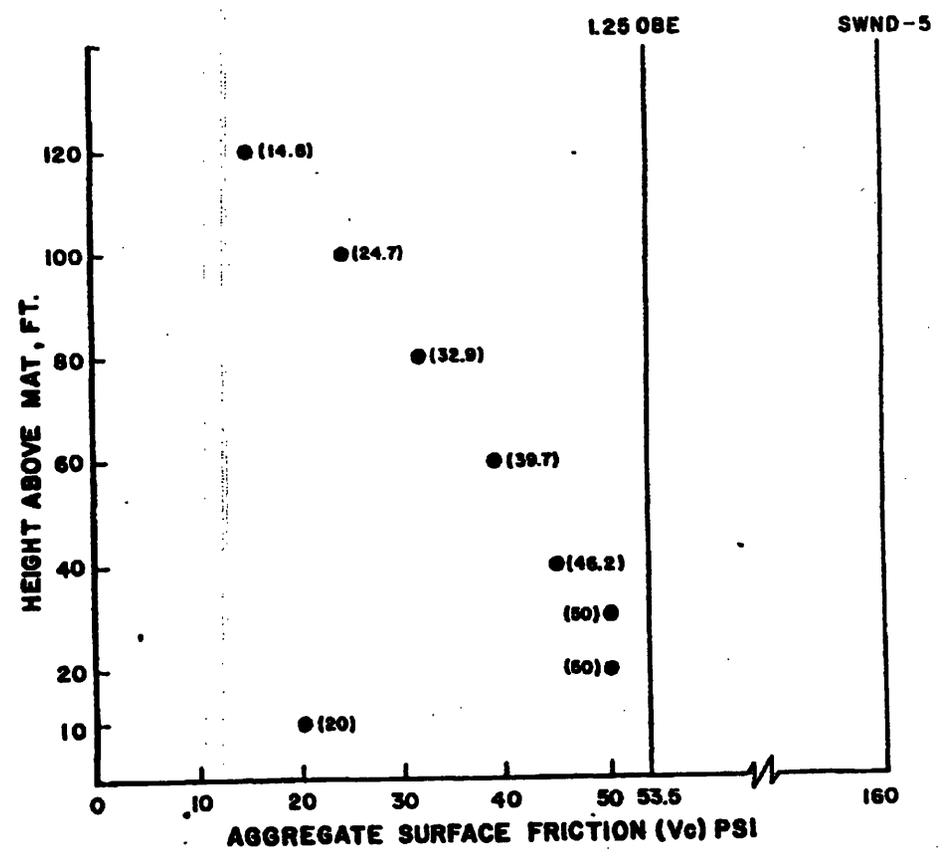


FIGURE 5-2-11
 SEISMIC SHEAR ALLOWANCE FOR AGGREGATE
 SURFACE FRICTION, CONTAINMENT STRUTURE
 BEAVER VALLEY POWER STATION UNIT NO. 1
 UPDATED FINAL SAFETY ANALYSIS REPORT

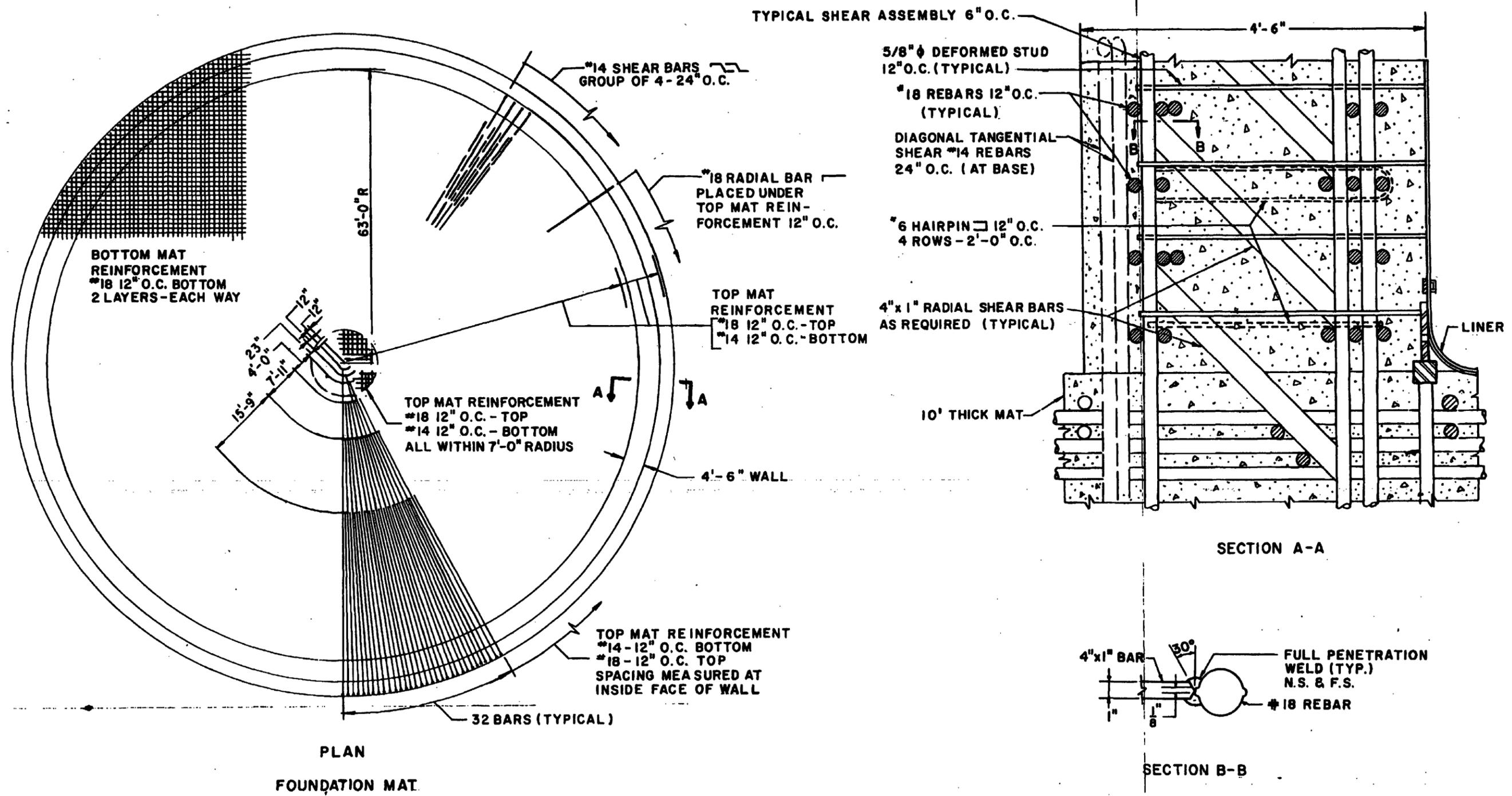


FIGURE 5.2-12
FOUNDATION MAT AND FIXED BASE
 BEAVER VALLEY POWER STATION UNIT NO. 1
 UPDATED FINAL SAFETY ANALYSIS REPORT

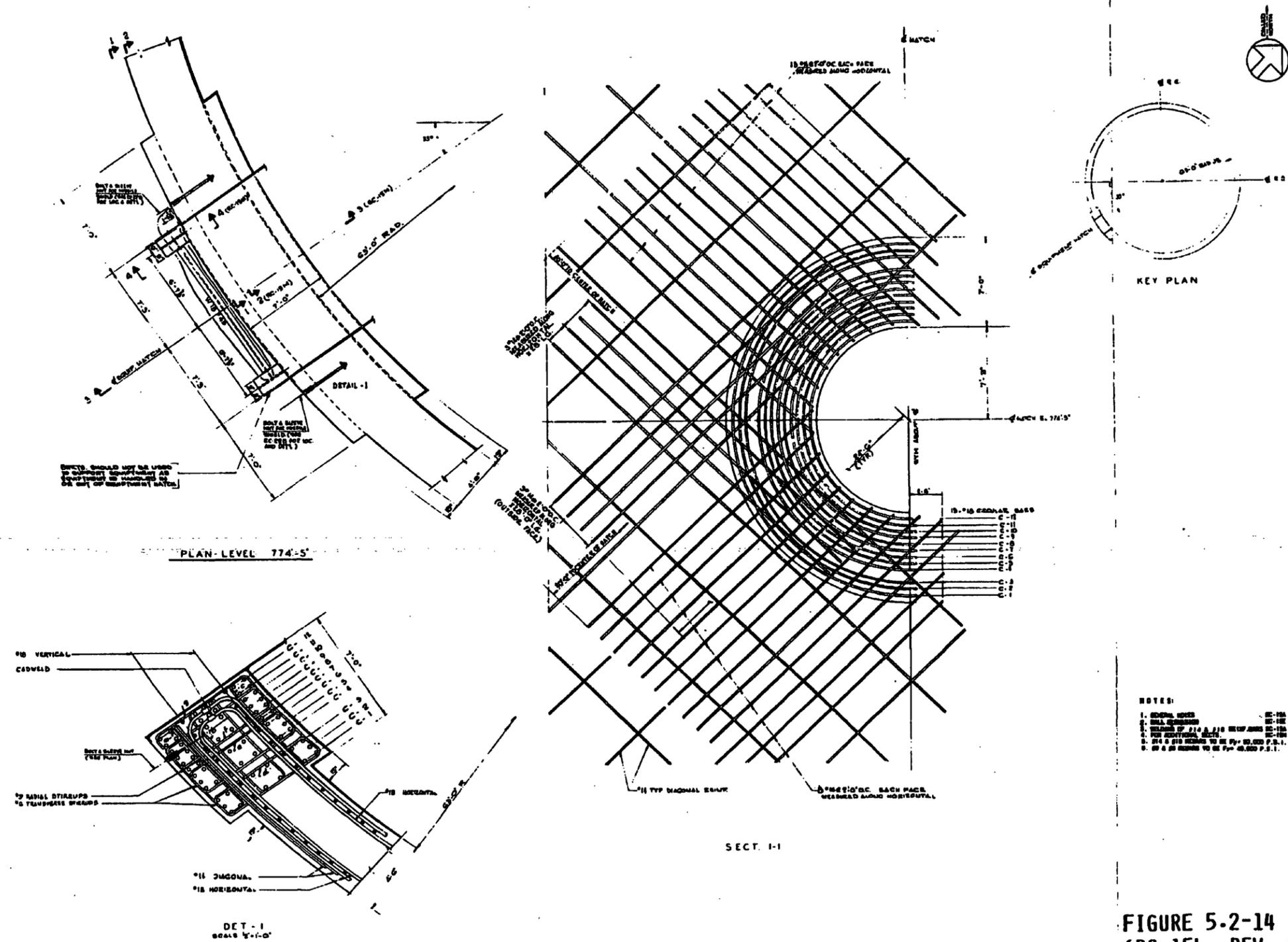


FIGURE 5-2-14
 (RC-15L, REV. 5)
 EQUIPMENT HATCH
 REINFORCING DETAILS - SH 1
 BEAVER VALLEY POWER STATION UNIT NO. 1
 UPDATED FINAL SAFETY ANALYSIS REPORT

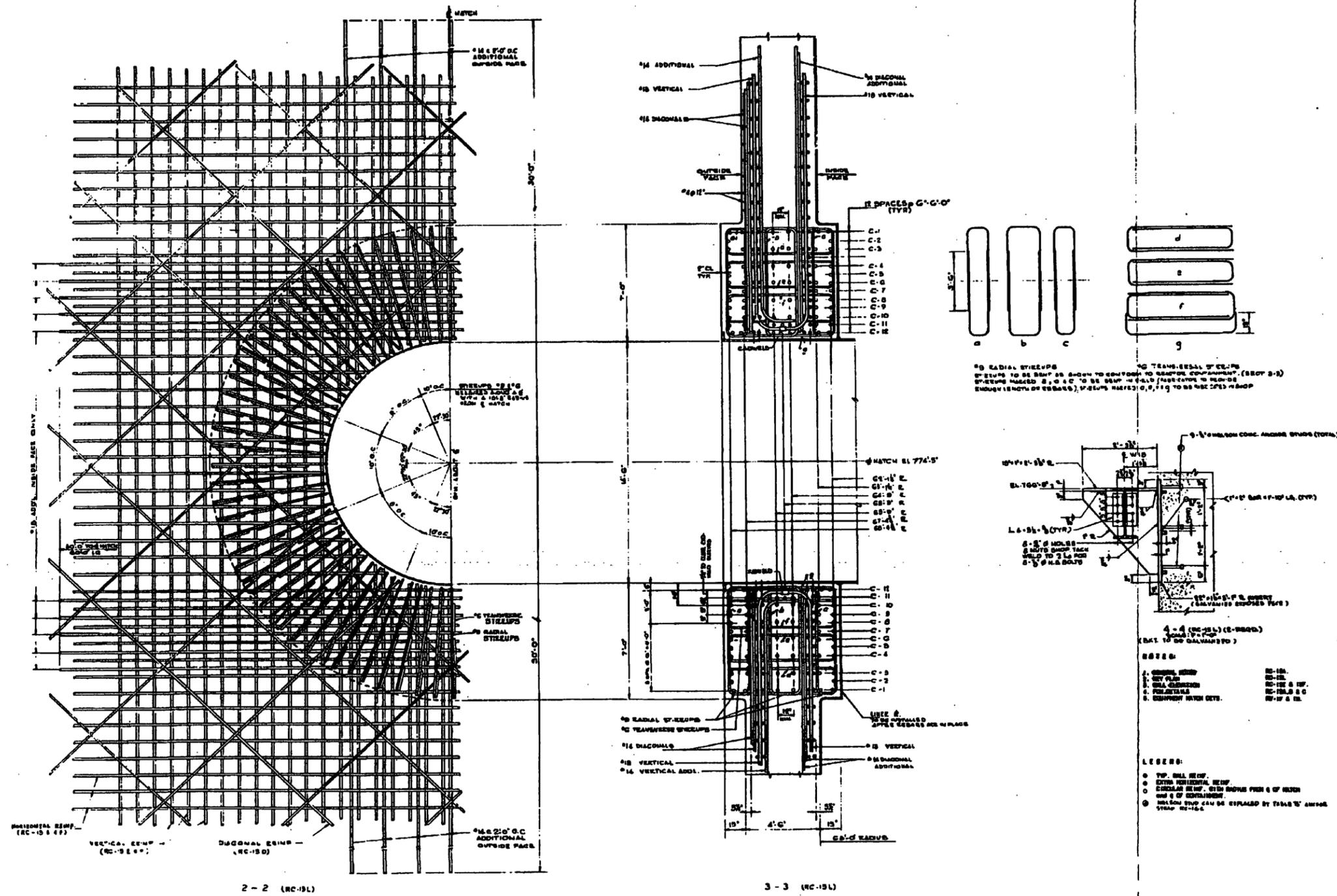


FIGURE 5-2-15
 (RC-15M, REV. 5)
 EQUIPMENT HATCH
 REINFORCING DETAILS - SH 2
 BEAVER VALLEY POWER STATION UNIT NO. 1
 UPDATED FINAL SAFETY ANALYSIS REPORT

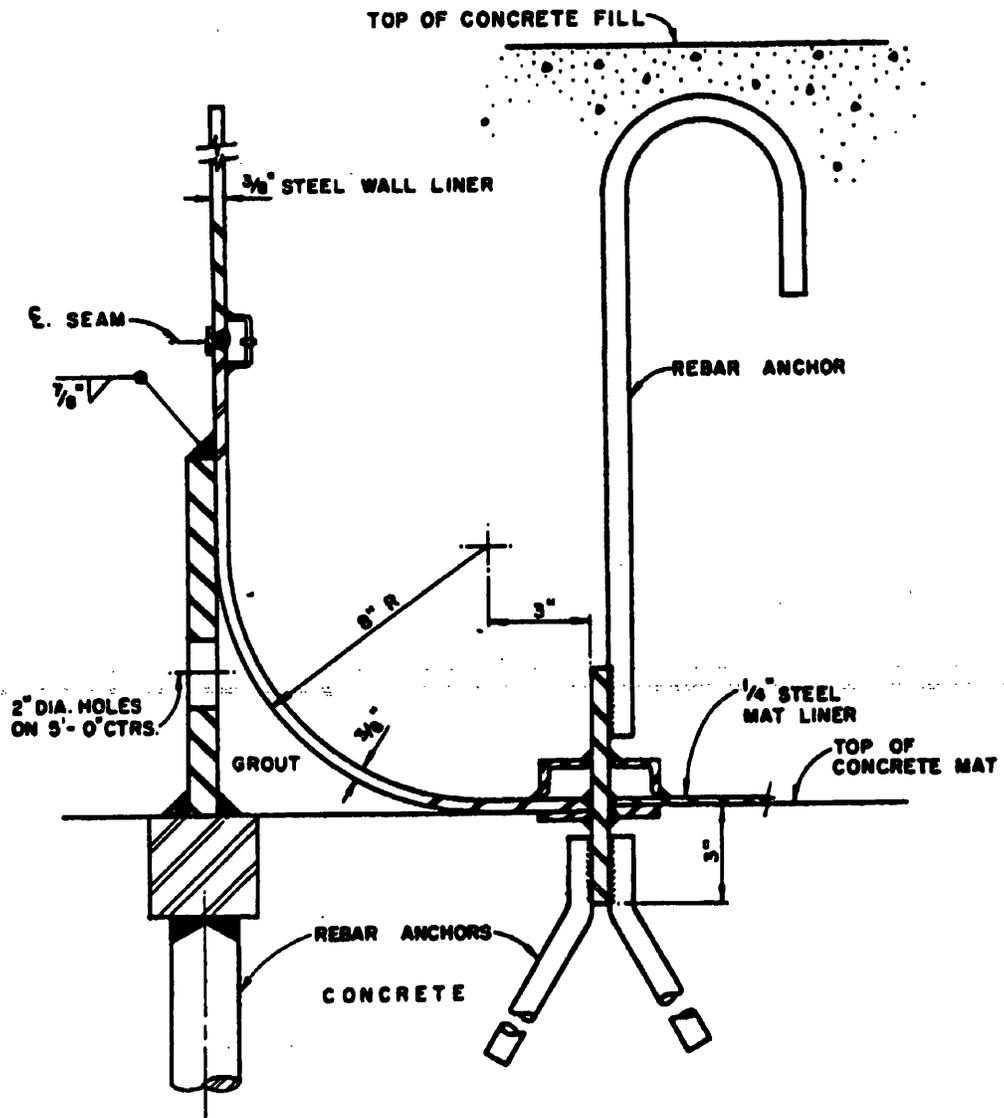


FIGURE 5.2-16
WALL AND MAT JOINT
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

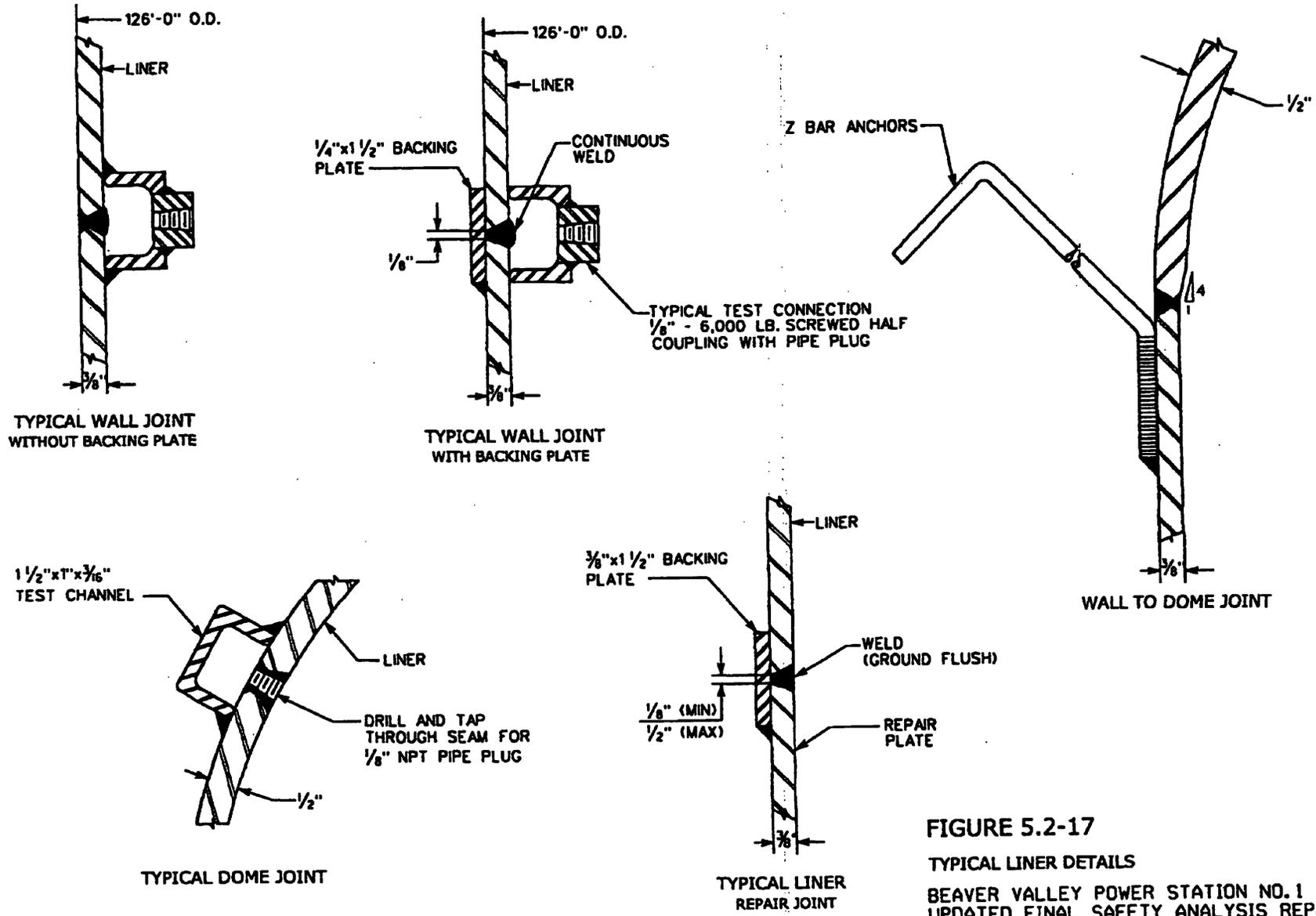
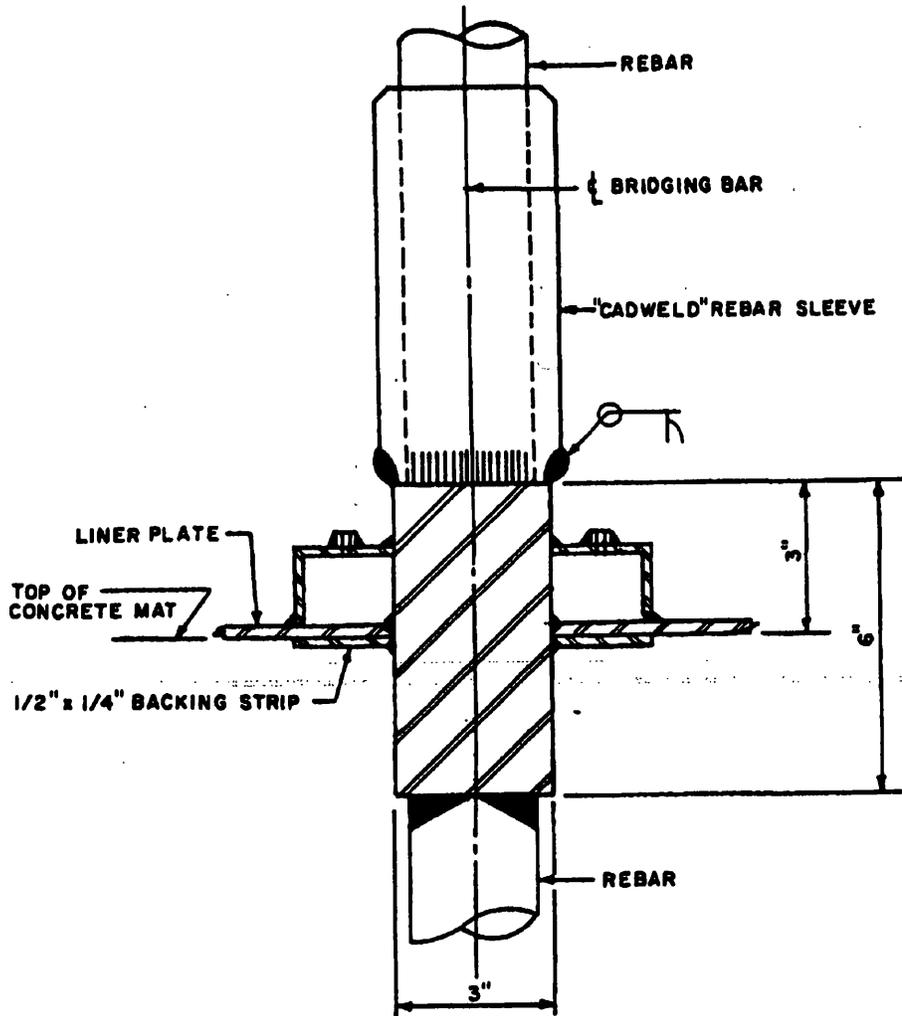
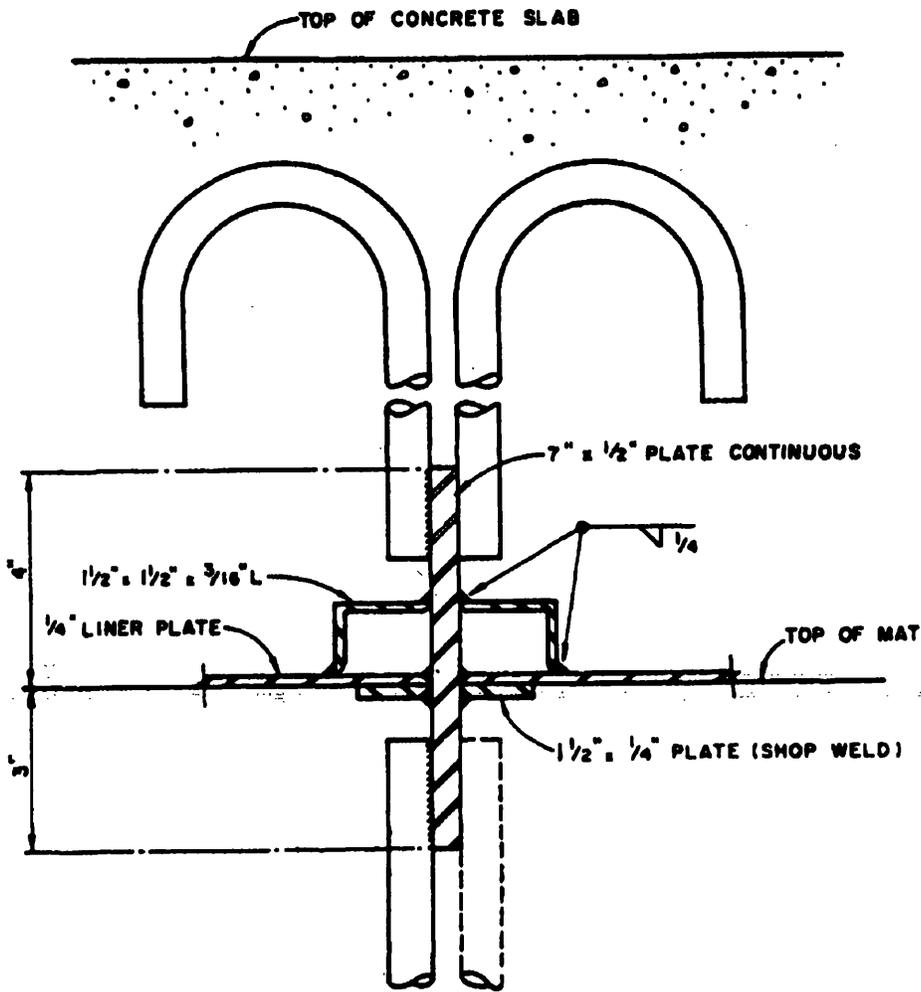


FIGURE 5.2-17
 TYPICAL LINER DETAILS
 BEAVER VALLEY POWER STATION NO. 1
 UPDATED FINAL SAFETY ANALYSIS REPORT



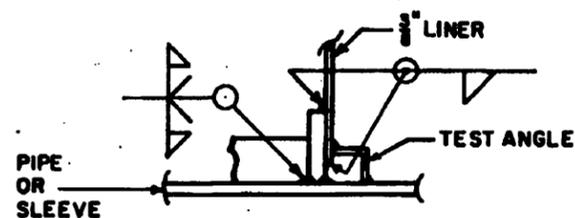
TYPICAL SECTION THROUGH BRIDGING BAR USED TO PROVIDE MAIN REINFORCING STEEL CONTINUITY THROUGH MAT LINER

FIGURE 5-2-18
SECTION-TYPICAL BRIDGING BAR - SH 1
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

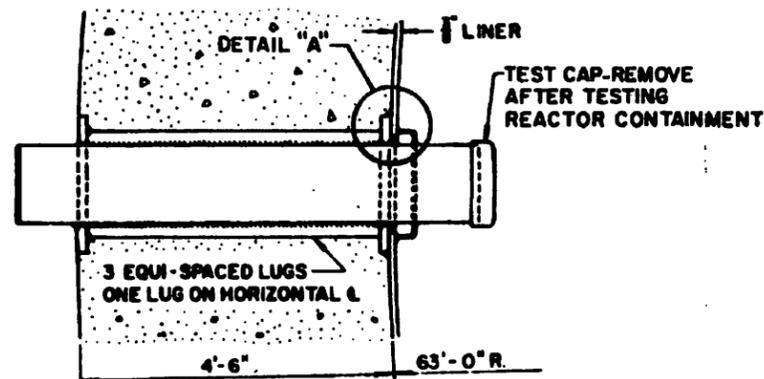


TYPICAL SECTION BRIDGING BAR USED TO ANCHOR
CONCRETE SLAB TO CONTAINMENT MAT THROUGH MAT LINER

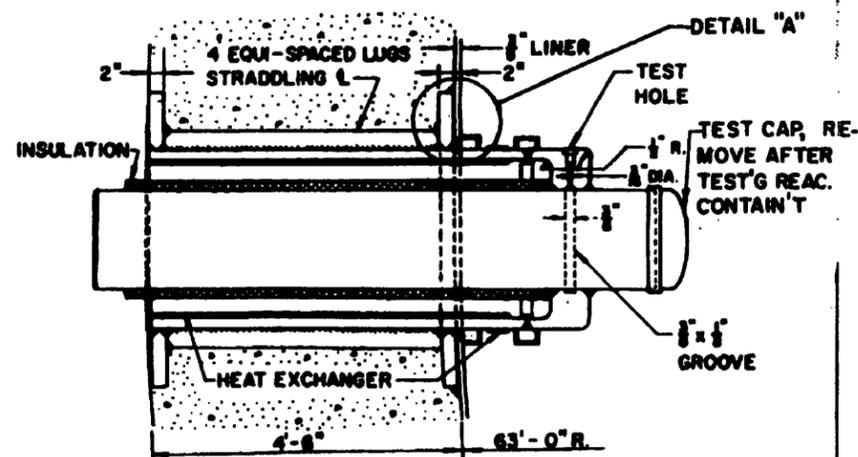
FIGURE 5-2-19
SECTION-TYPICAL BRIDGING BAR - SH 2
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT



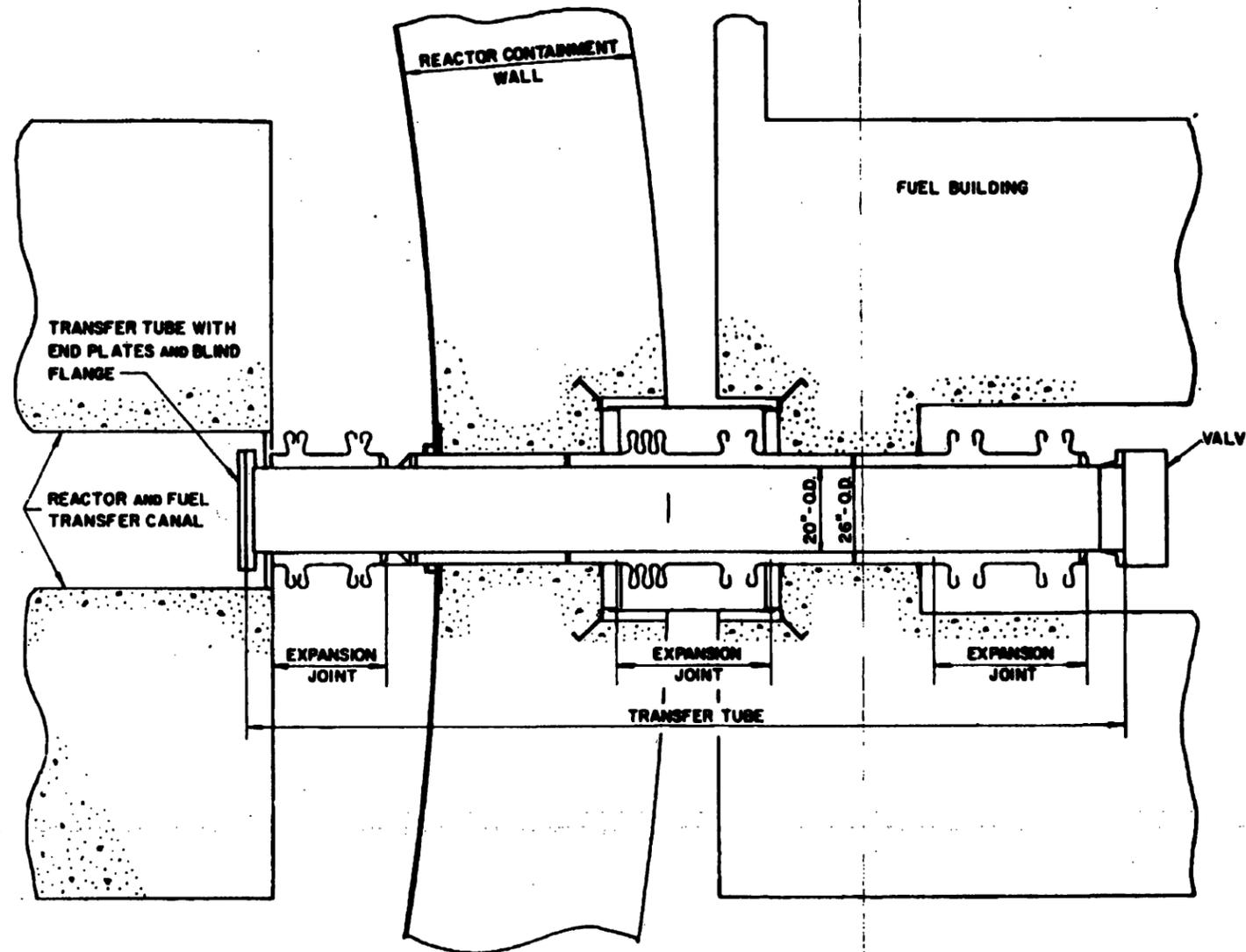
DETAIL "A"
TYPICAL FOR COLD AND
HOT PIPE PENETRATION



TYPICAL COLD PIPE PENETRATIONS



TYPICAL HOT PIPE PENETRATION



PLAN - FUEL TRANSFER PENETRATION
PENETRATION No. 65

MATERIAL NOTES

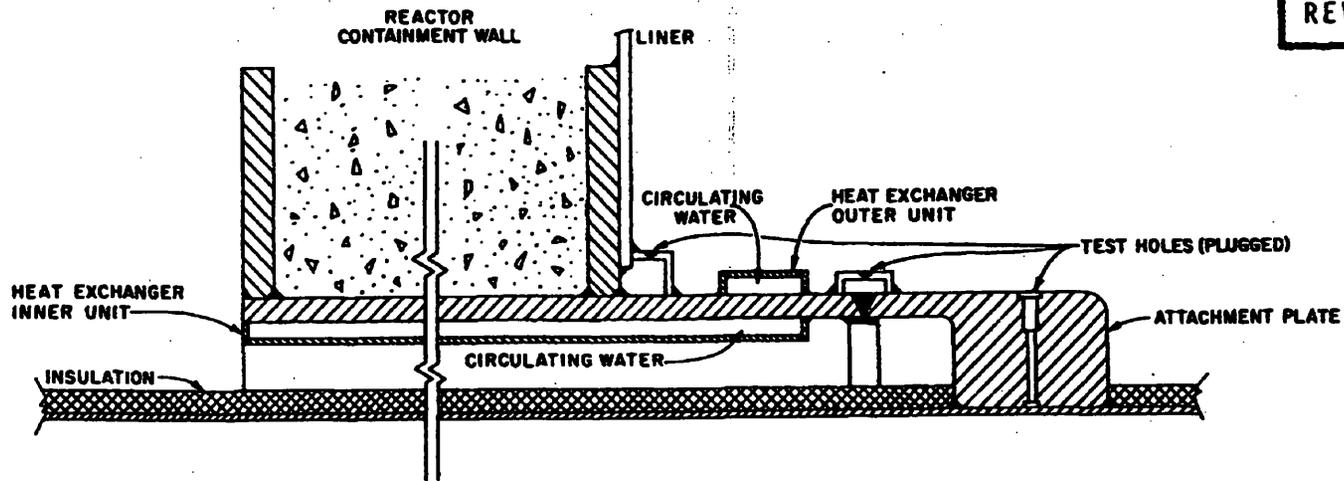
- | | |
|---------------------------|----------------------------|
| CARBON STEEL PLATES | - ASTM - A516 - GR60 |
| | - FINE GRAINED, NORMALIZED |
| CARBON STEEL FORGINGS | - ASTM - A350 - GR LF2 |
| CARBON STEEL PIPE | - ASTM - A333 - GR3 |
| CARBON STEEL PIPE SLEEVES | - ASTM - A333 - GR3 |
| STAINLESS STEEL FORGINGS | - ASTM - A182 - F304 |
| STAINLESS STEEL PIPE | - ASTM - A312 - TYP 304 |
| STAINLESS STEEL TUBING | - ASTM - A269 - TYP 304 |
| BOLTS | - ASTM - A193 - B7 |
| NUTS | - ASTM - A194 - 2H |

WELDING ELECTRODES

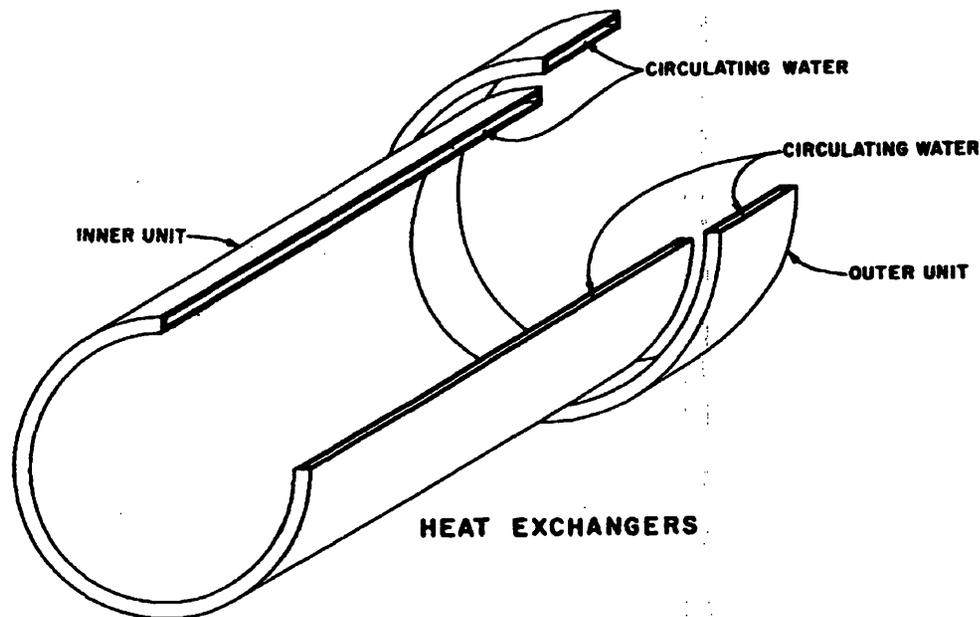
- | | |
|------------------------------------|------------------|
| CARBON STEEL TO CARBON STEEL | - ASTM - E 7018* |
| STAINLESS STEEL TO STAINLESS STEEL | - ASTM - E 308 |
| CARBON STEEL TO STAINLESS STEEL | - ASTM - E 309 |

*OR APPROVED EQUAL.

FIGURE 5-2-21
TYPICAL PIPING PENETRATIONS - SH 1
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

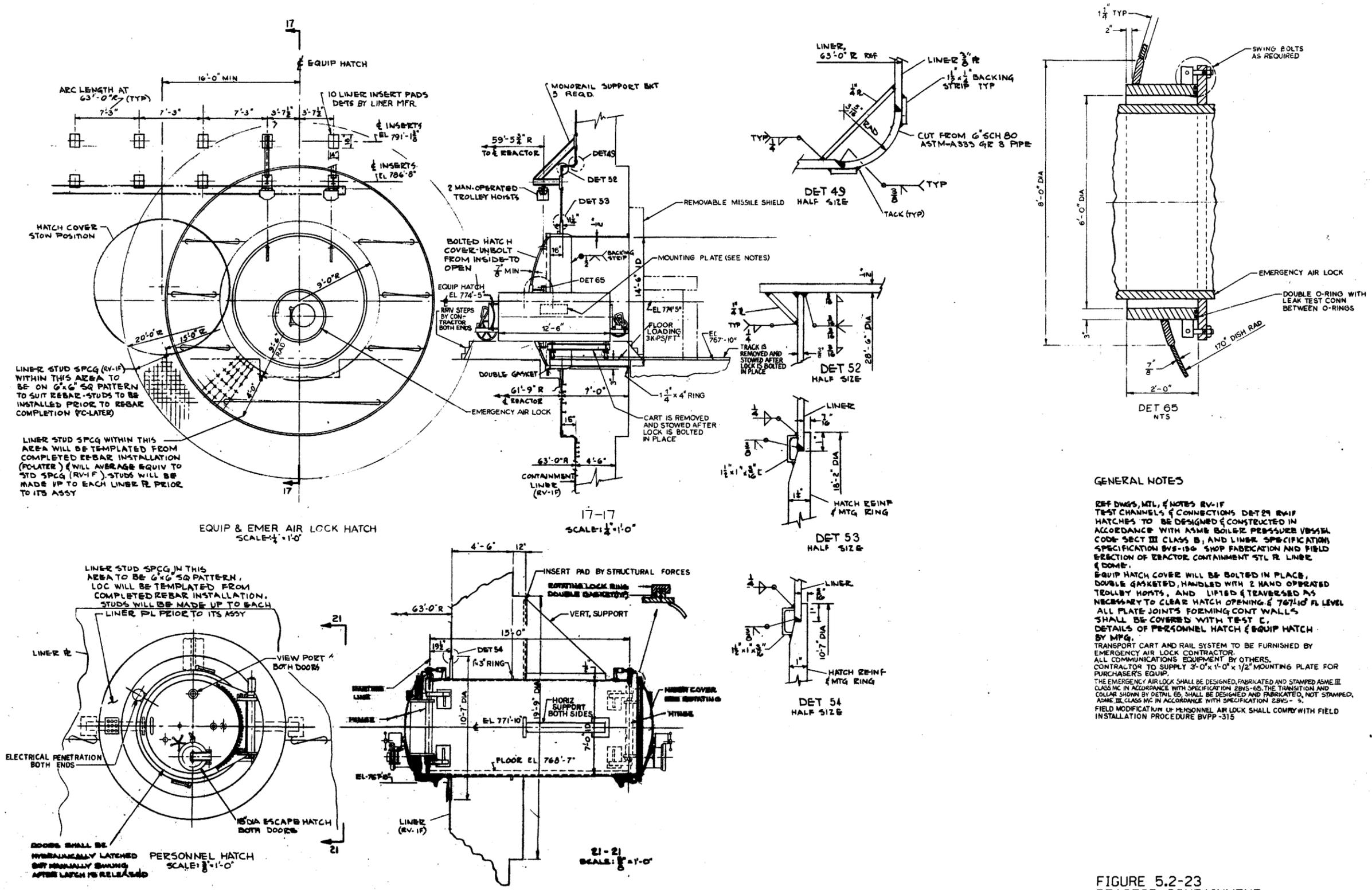


HOT PIPE PENETRATION



HEAT EXCHANGERS

**FIGURE 5-2-22
TYPICAL PIPING PENETRATIONS - SH 2
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT**



GENERAL NOTES

REF DWGS, MIL, & NOTES RV-1F TEST CHANNELS & CONNECTIONS DET 29 HALF HATCHES TO BE DESIGNED & CONSTRUCTED IN ACCORDANCE WITH ASME BOILER PRESSURE VESSEL CODE SECT III CLASS B, AND LINER SPECIFICATION SPECIFICATION BVS-150 SHOP FABRICATION AND FIELD ERECTION OF REACTOR CONTAINMENT STL R LINER & DOME.

EQUIP HATCH COVER WILL BE BOLTED IN PLACE, DOUBLE GASKETED, HANDLED WITH 2 HAND OPERATED TROLLEY HOISTS, AND LIFTED & TRAVERSED AS NECESSARY TO CLEAR HATCH OPENING & 76710 FL LEVEL ALL PLATE JOINTS FORMING CONT WALLS SHALL BE COVERED WITH TEST C.

DETAILS OF PERSONNEL HATCH & EQUIP HATCH BY MFG.

TRANSPORT CART AND RAIL SYSTEM TO BE FURNISHED BY EMERGENCY AIR LOCK CONTRACTOR.

ALL COMMUNICATIONS EQUIPMENT BY OTHERS.

CONTRACTOR TO SUPPLY 3'-0" x 1'-0" x 1/2" MOUNTING PLATE FOR PURCHASER'S EQUIP.

THE EMERGENCY AIR LOCK SHALL BE DESIGNED, FABRICATED AND STAMPED ASME III CLASS IC IN ACCORDANCE WITH SPECIFICATION BVS-150. THE TRANSITION AND COLLAR SHOWN BY DETAIL 65, SHALL BE DESIGNED AND FABRICATED, NOT STAMPED, ASME III CLASS IC IN ACCORDANCE WITH SPECIFICATION BVS-150.

FIELD MODIFICATION OF PERSONNEL AIR LOCK SHALL COMPLY WITH FIELD INSTALLATION PROCEDURE BVPP-315

FIGURE 5.2-23
REACTOR CONTAINMENT
LINER DETAILS SH.4
 BEAVER VALLEY POWER STATION UNIT NO.1
 UPDATED FINAL SAFETY ANALYSIS REPORT

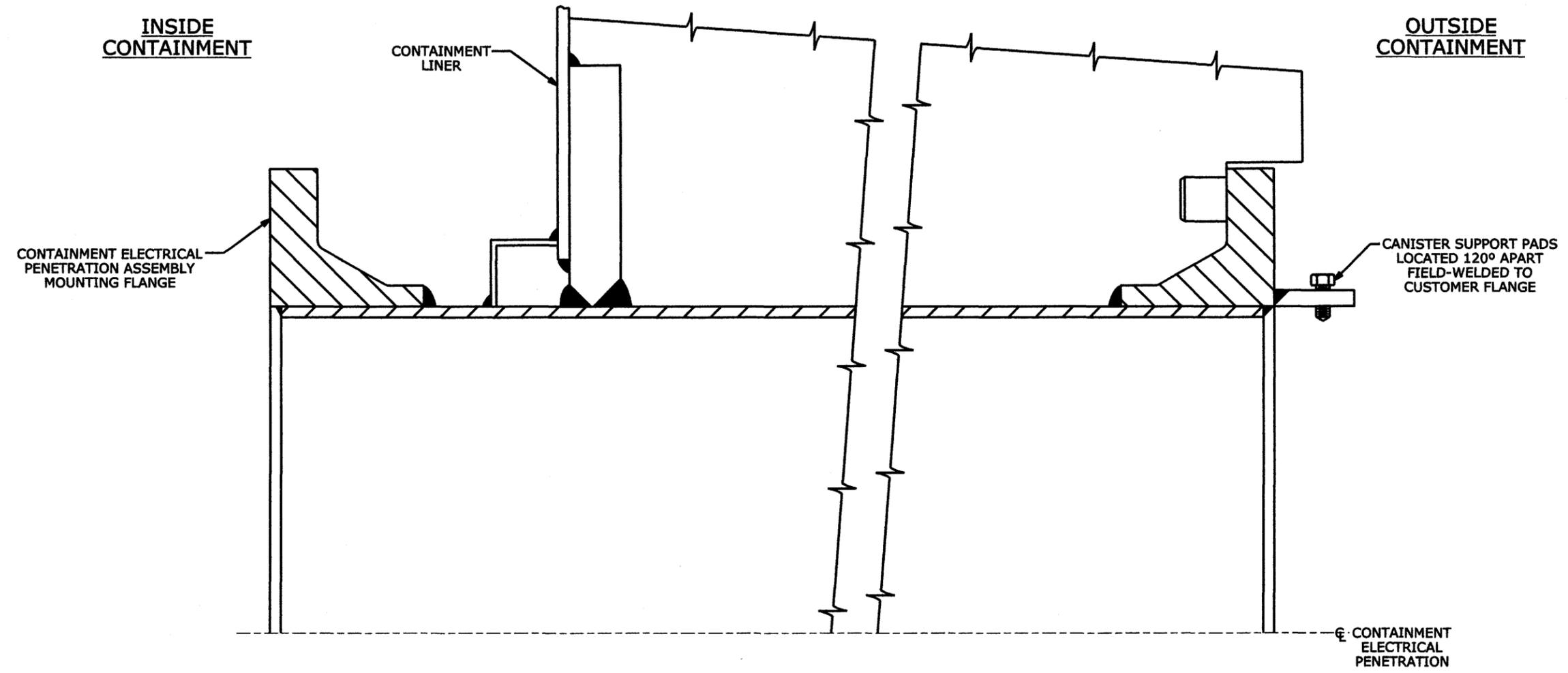


FIGURE 5.2-24
TYPICAL ELECTRICAL PENETRATION
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

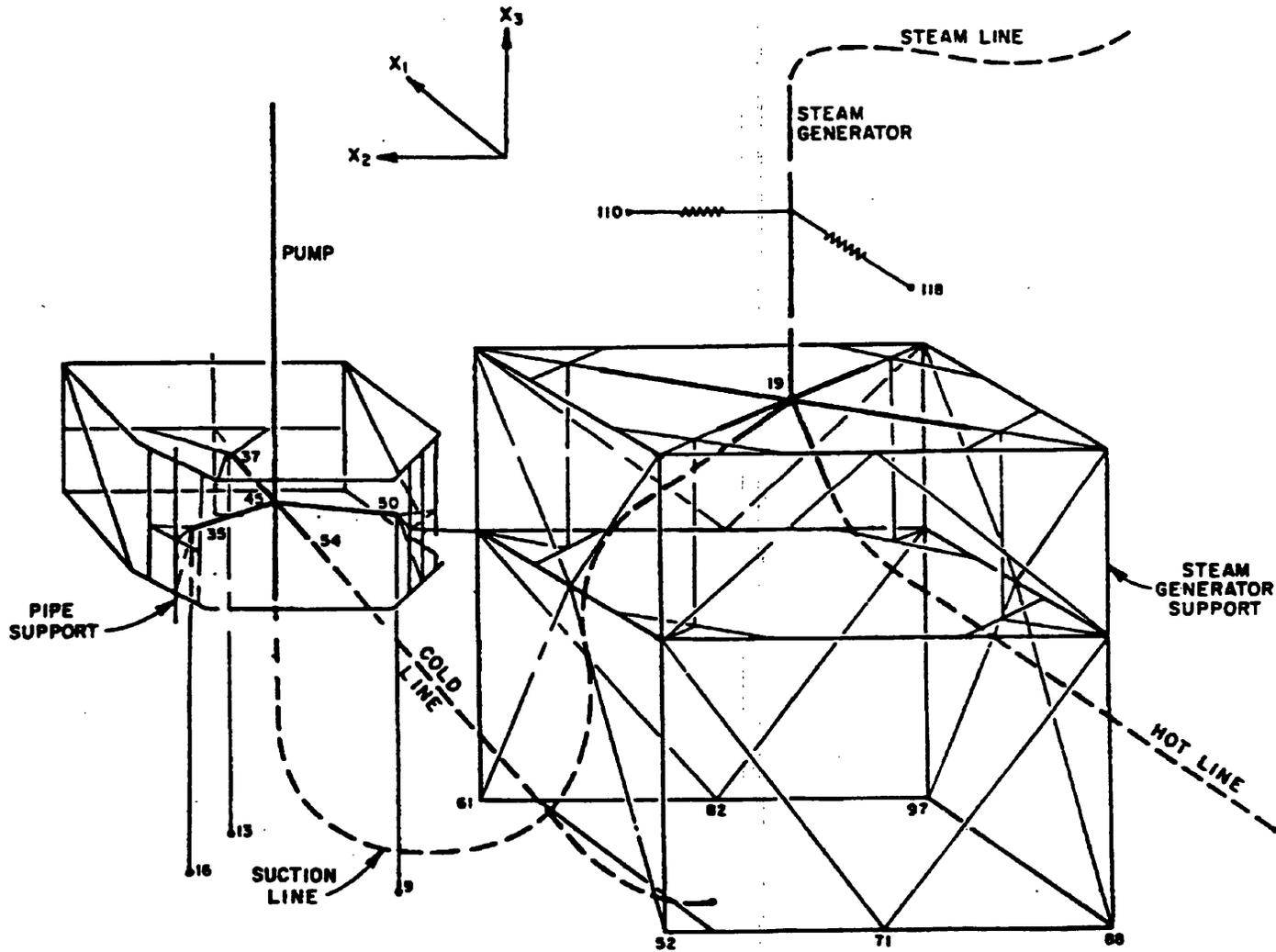


FIG. 5.2-26
 STRESS MODEL FOR STEAM GENERATOR
 AND PUMP SUPPORTS
 BEAVER VALLEY POWER STATION UNIT NO. 1
 UPDATED FINAL SAFETY ANALYSIS REPORT

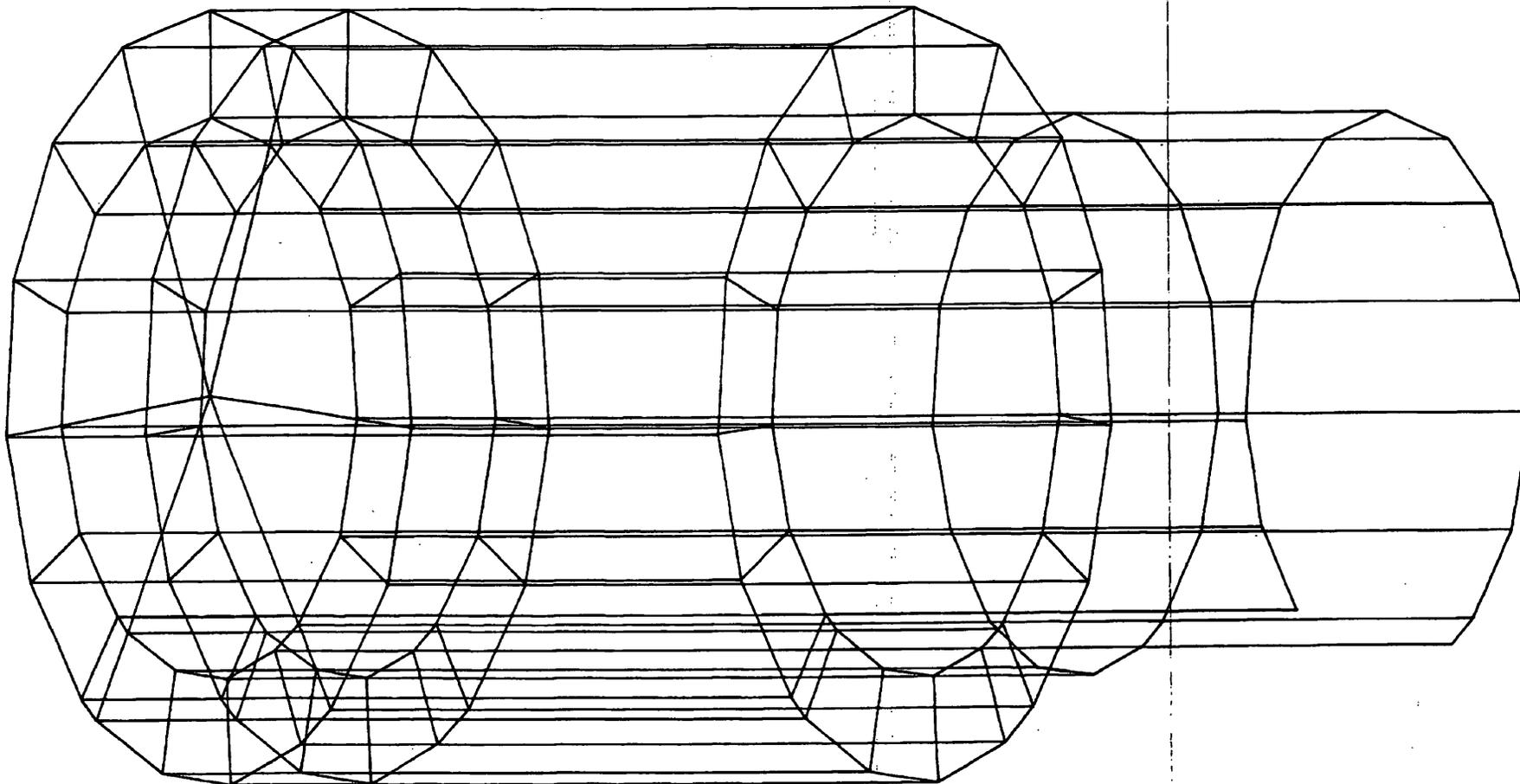


FIGURE 5-2-27
STRESS MODEL FOR REACTOR VESSEL
SUPPORTS (NEUTRON SHIELD TANK)
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

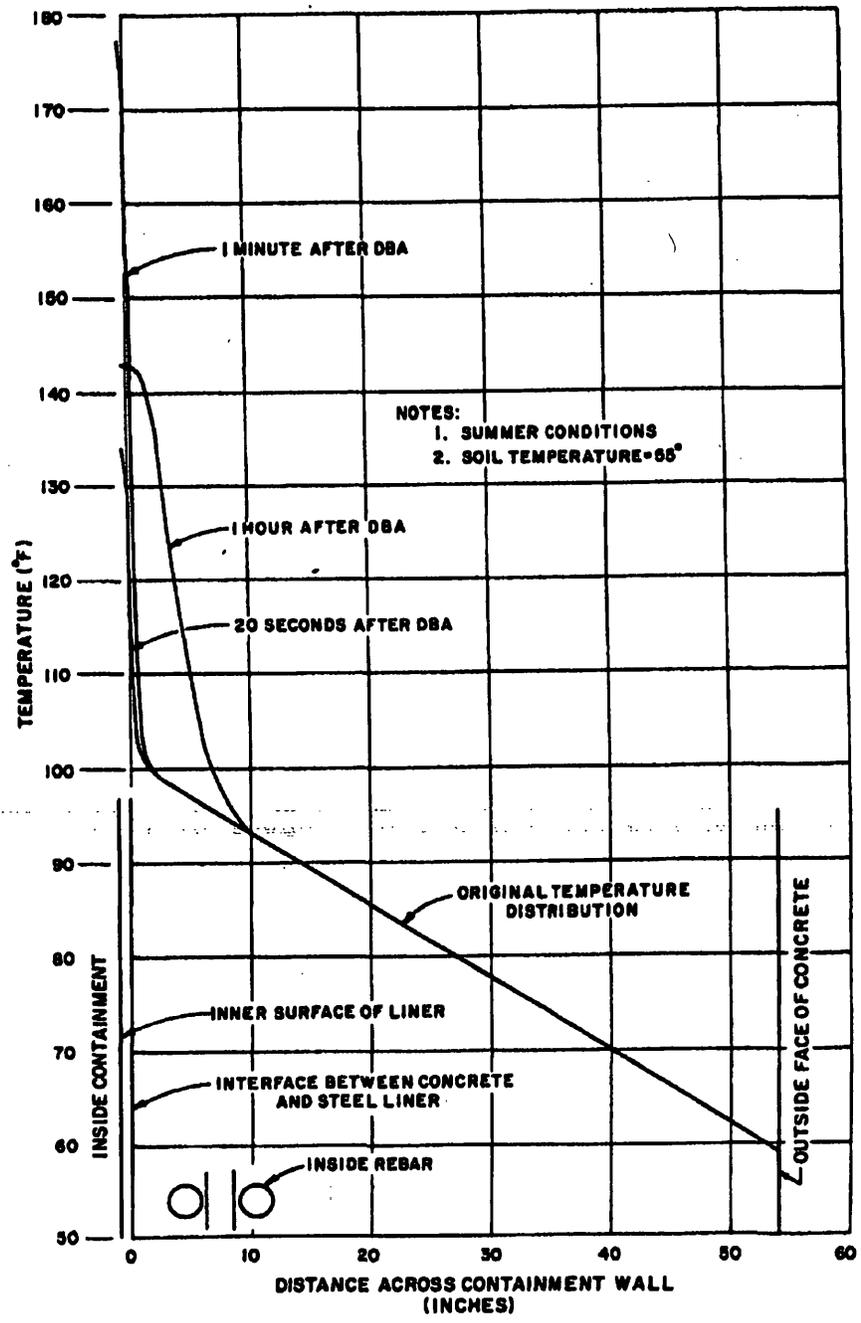


FIGURE 5-2-28
TRANSIENT TEMPERATURE PROFILES IN
CONTAINMENT WALL BELOW GRADE
FOLLOWING HOT LEG DER
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

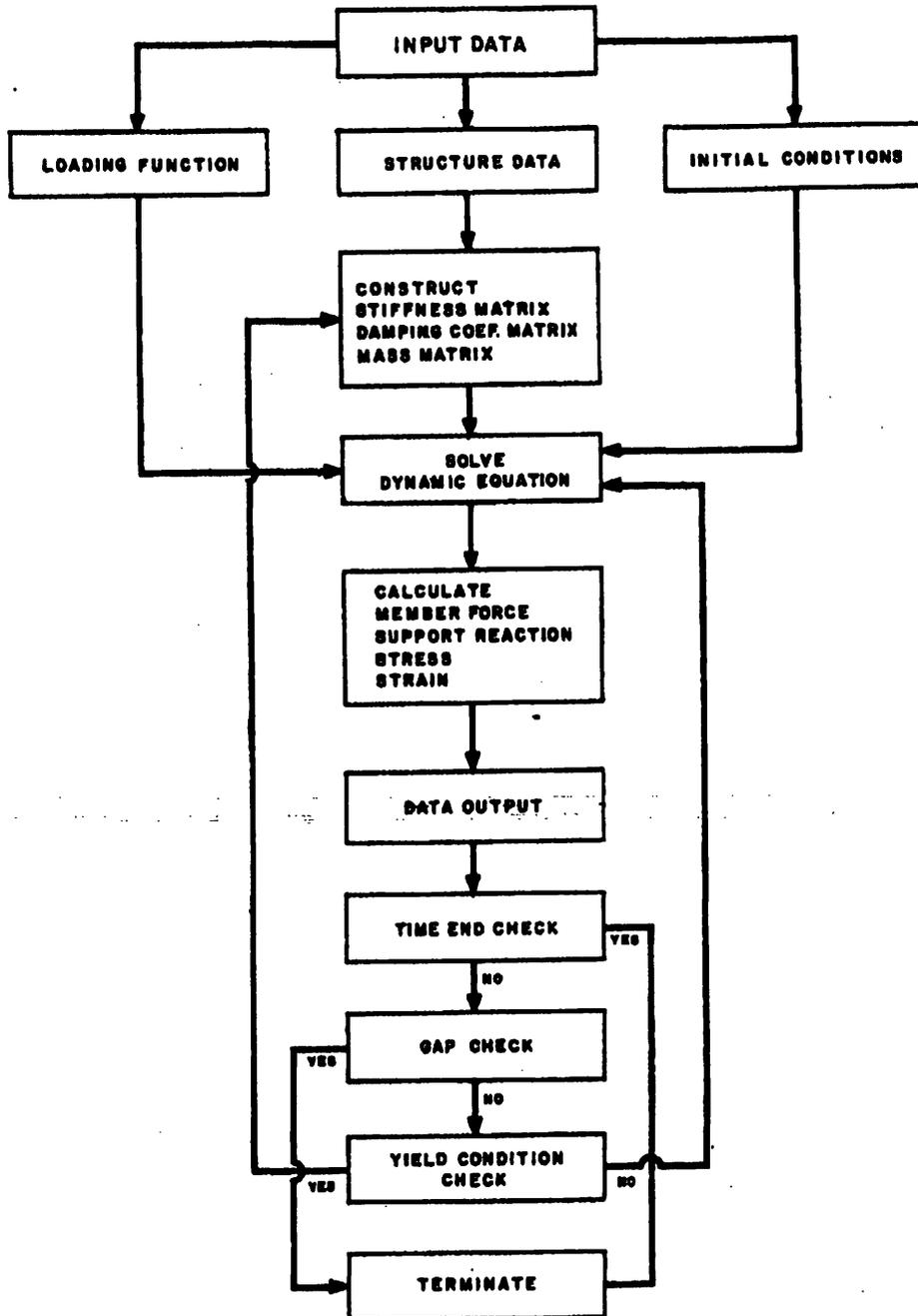


FIGURE 5-2-45
 FLOW CHART LIMITA II
 BEAVER VALLEY POWER STATION UNIT NO. 1
 UPDATED FINAL SAFETY ANALYSIS REPORT

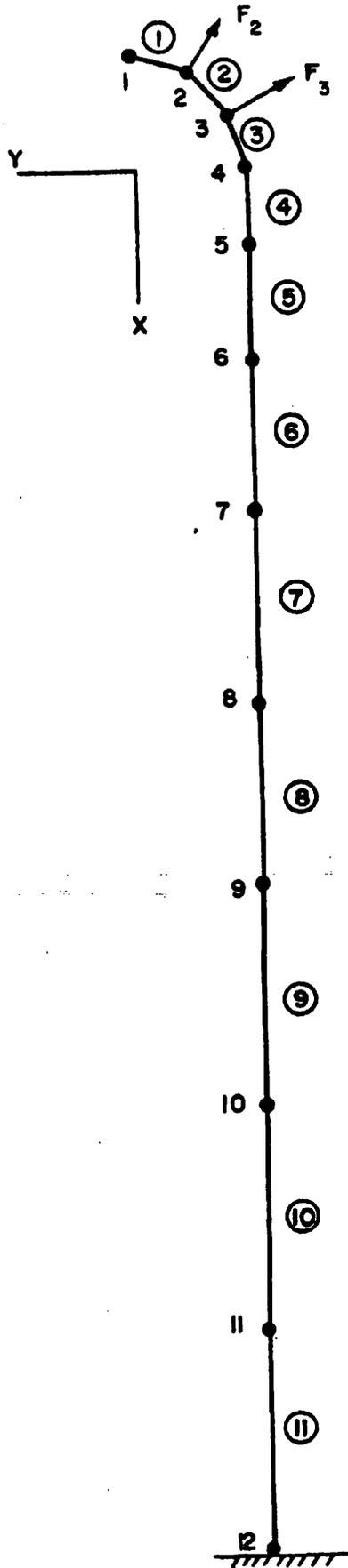


FIGURE 5.2-46
MATHEMATICAL MODEL
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

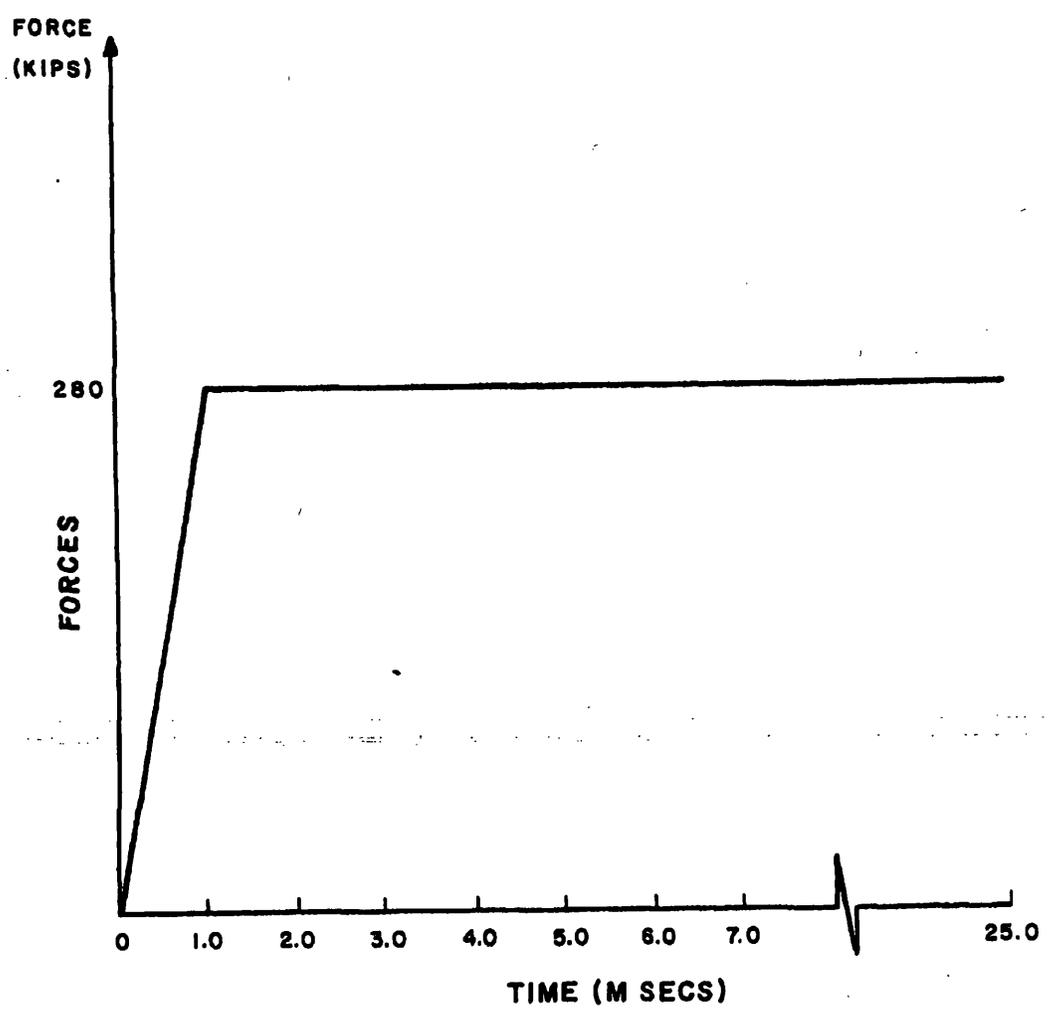


FIGURE 5.2-47
FORCING FUNCTION
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

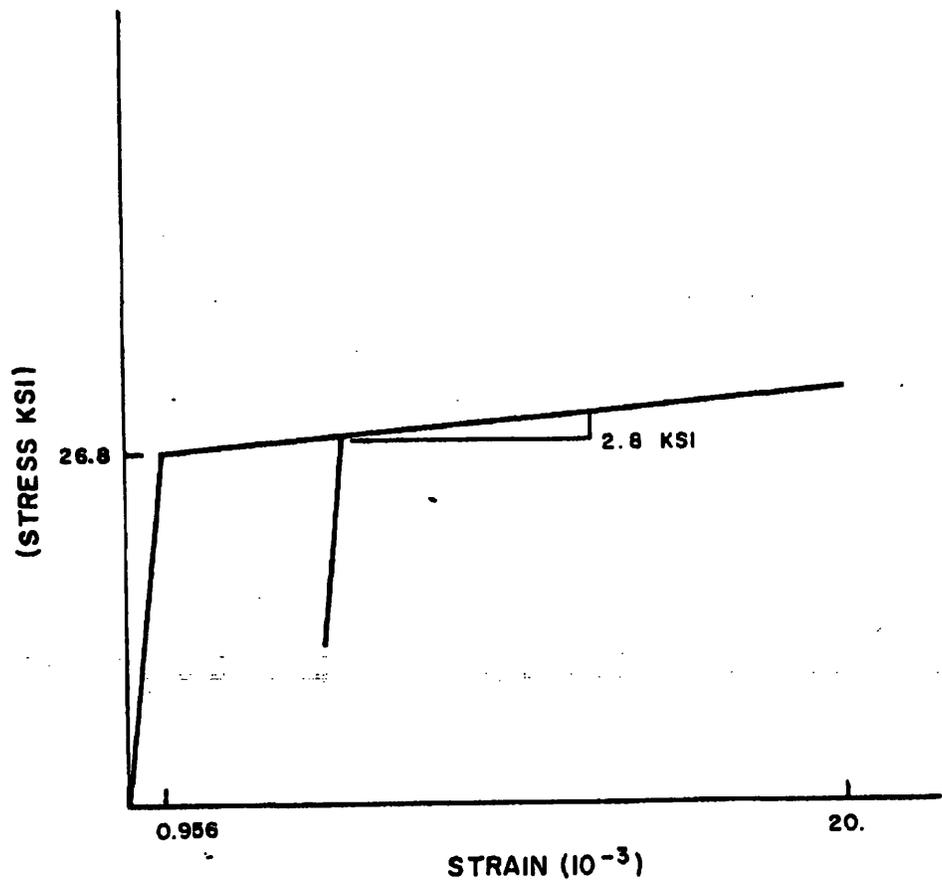
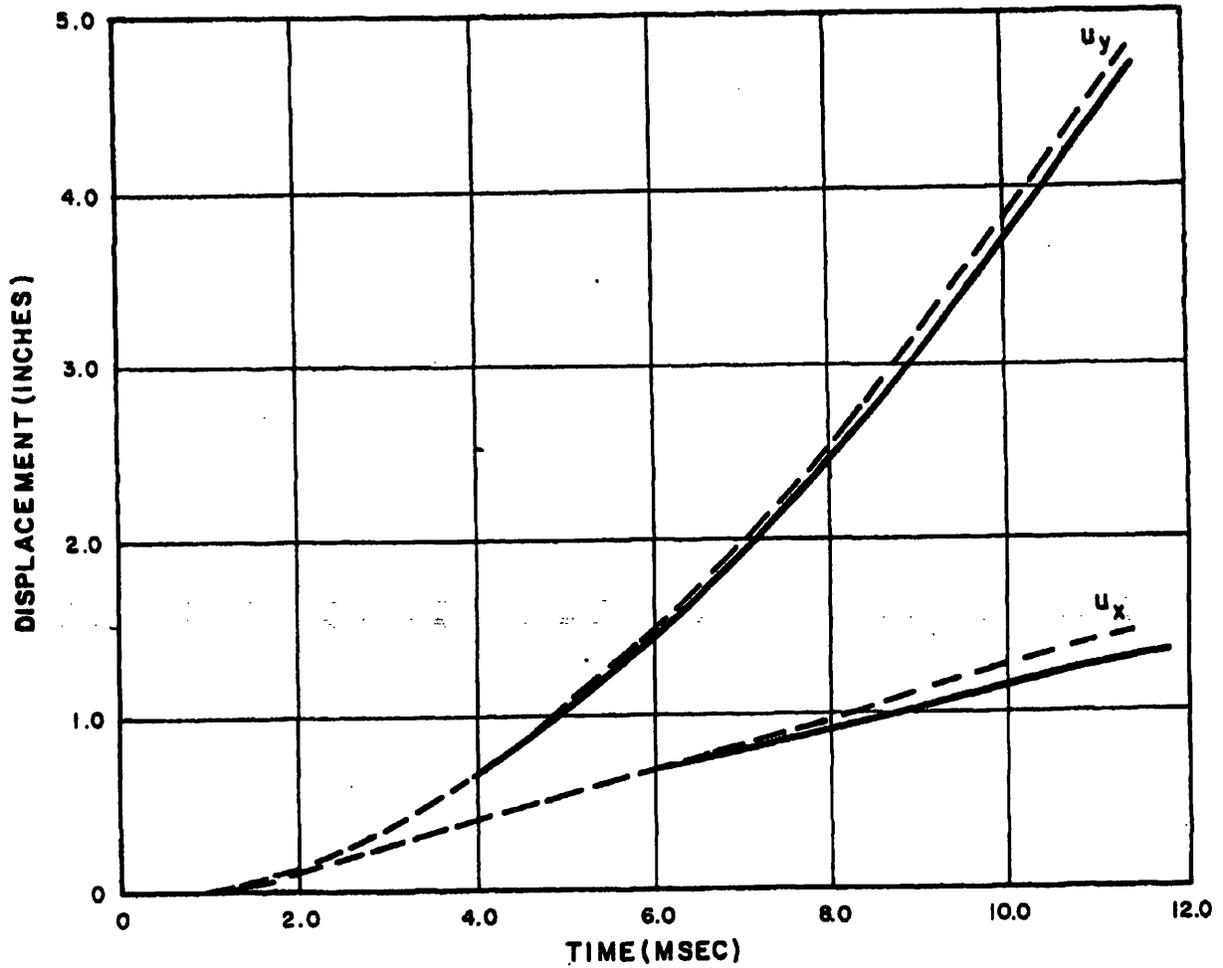


FIGURE 5-2-48
STRESS - STRAIN RELATION
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT



LEGEND:
- - - ANSYS
— LIMITA II

FIGURE 5.2-49
DISPLACEMENT RESPONSES AT JOINT I
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

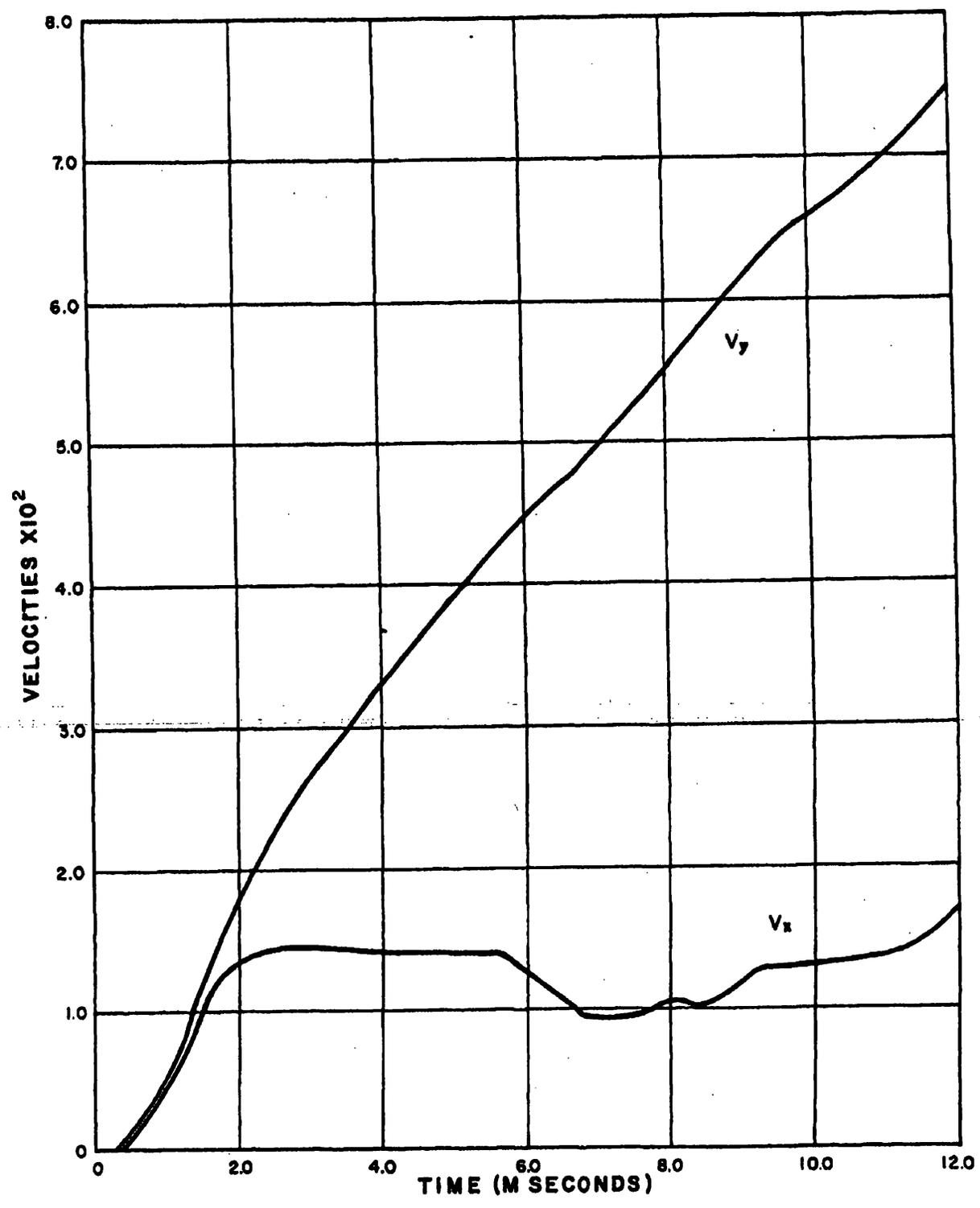


FIGURE 5-2-50
VELOCITY RESPONSES AT JOINT I
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

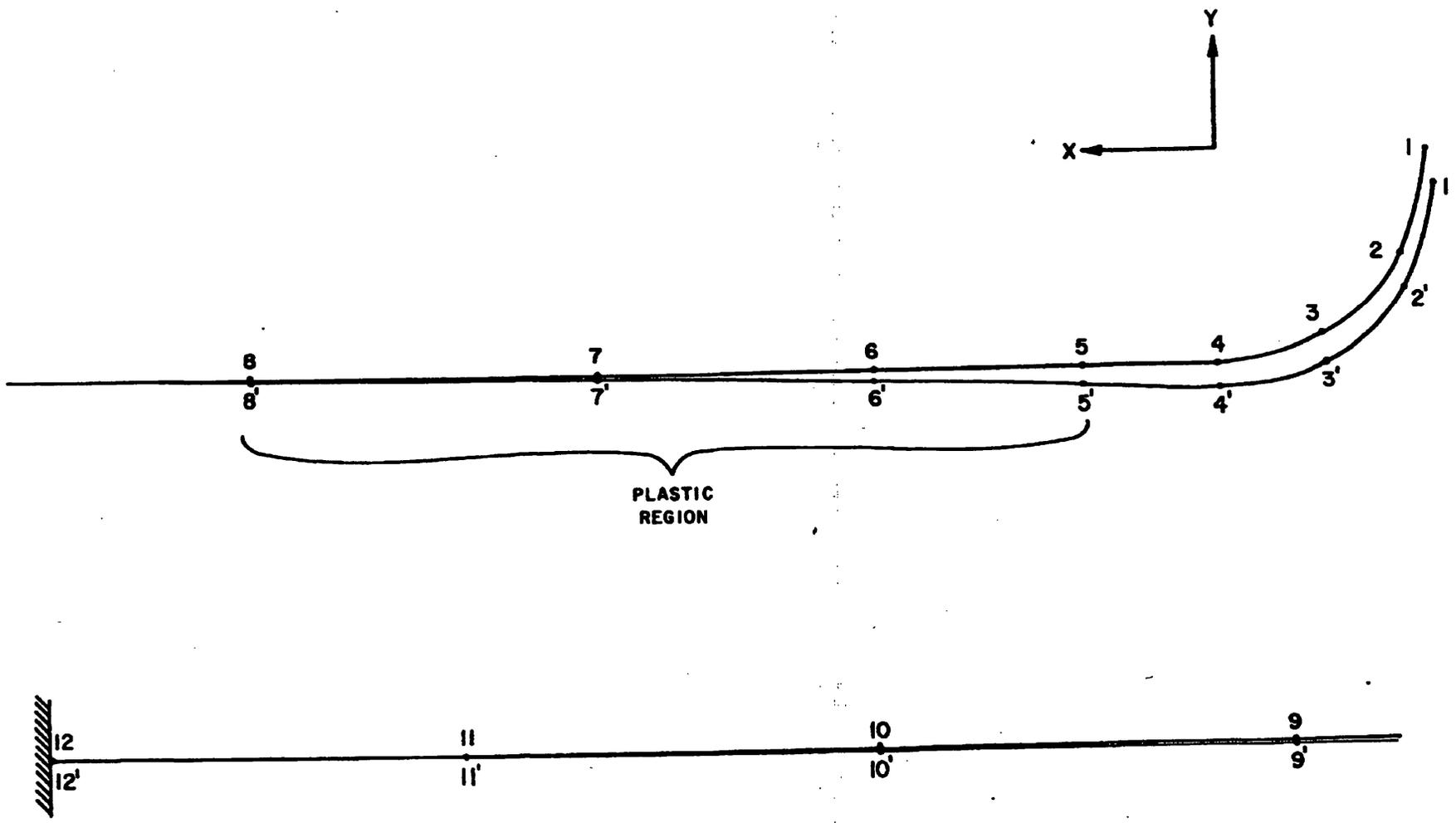


FIGURE 5-2-51
PIPE DEFORMATION AT TIME
0.013200 SECONDS
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

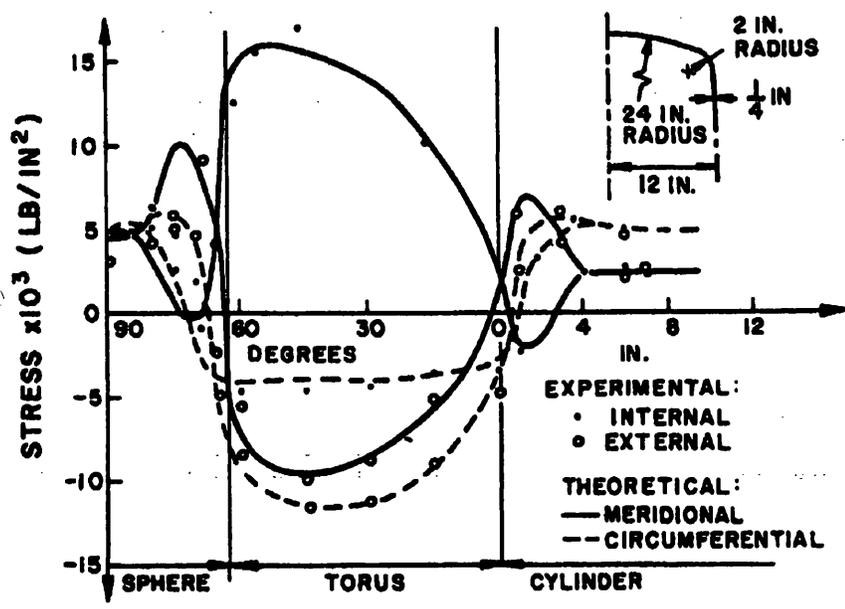


FIGURE 5-2-52
STRESS DISTRIBUTION FOR TORISPHERICAL
VESSEL, 100 LBF/IN²
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

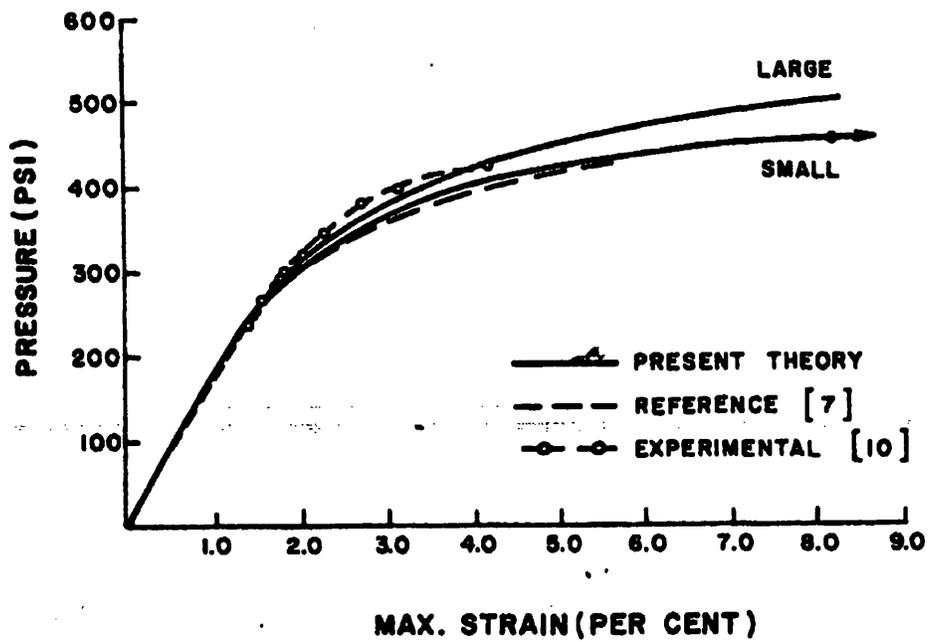


FIGURE 5-2-53
PRESSURE - MAXIMUM STRAIN IN TORISPHERE
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

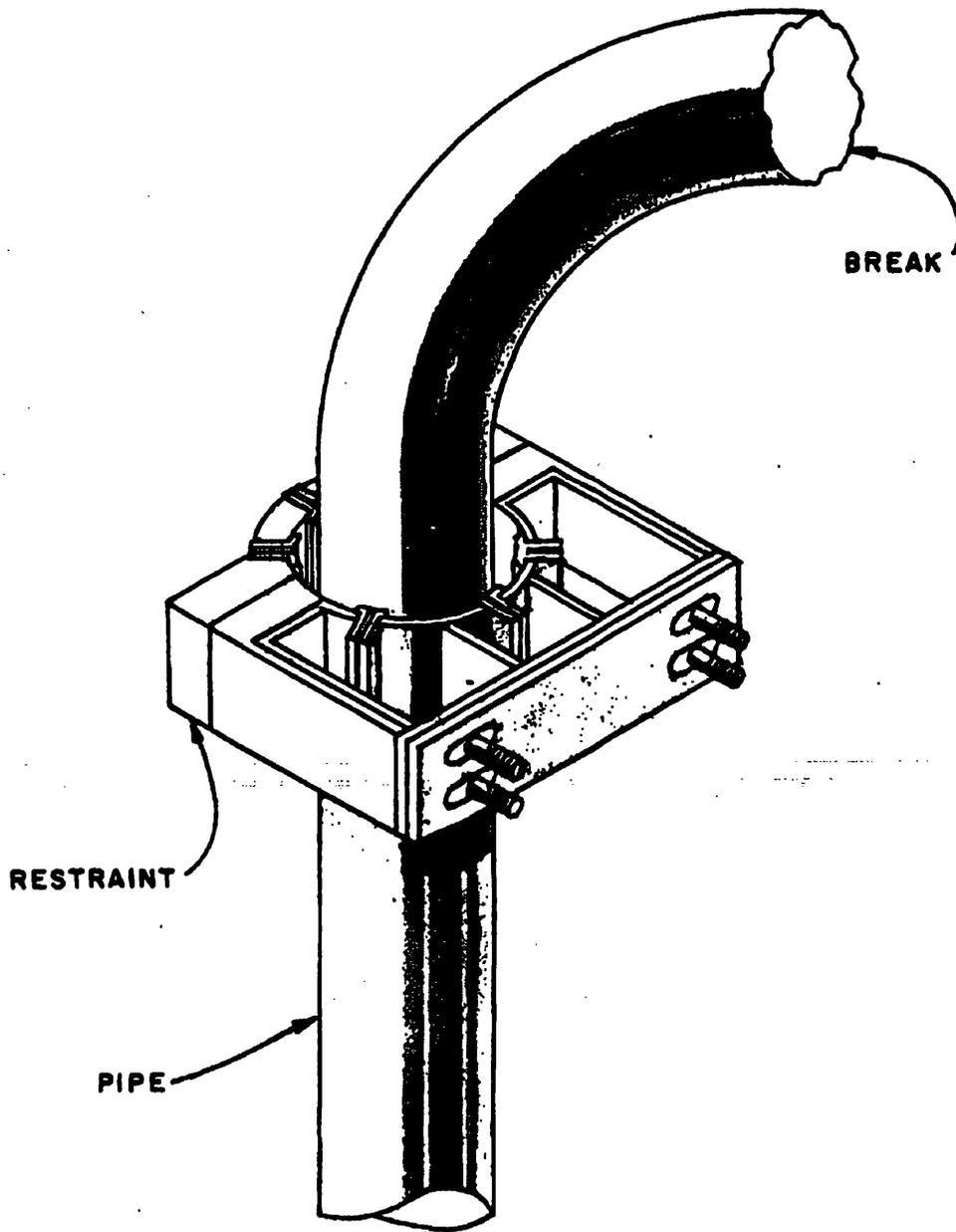


FIGURE 5-2-54
PIPE - RESTRAINT SYSTEM
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

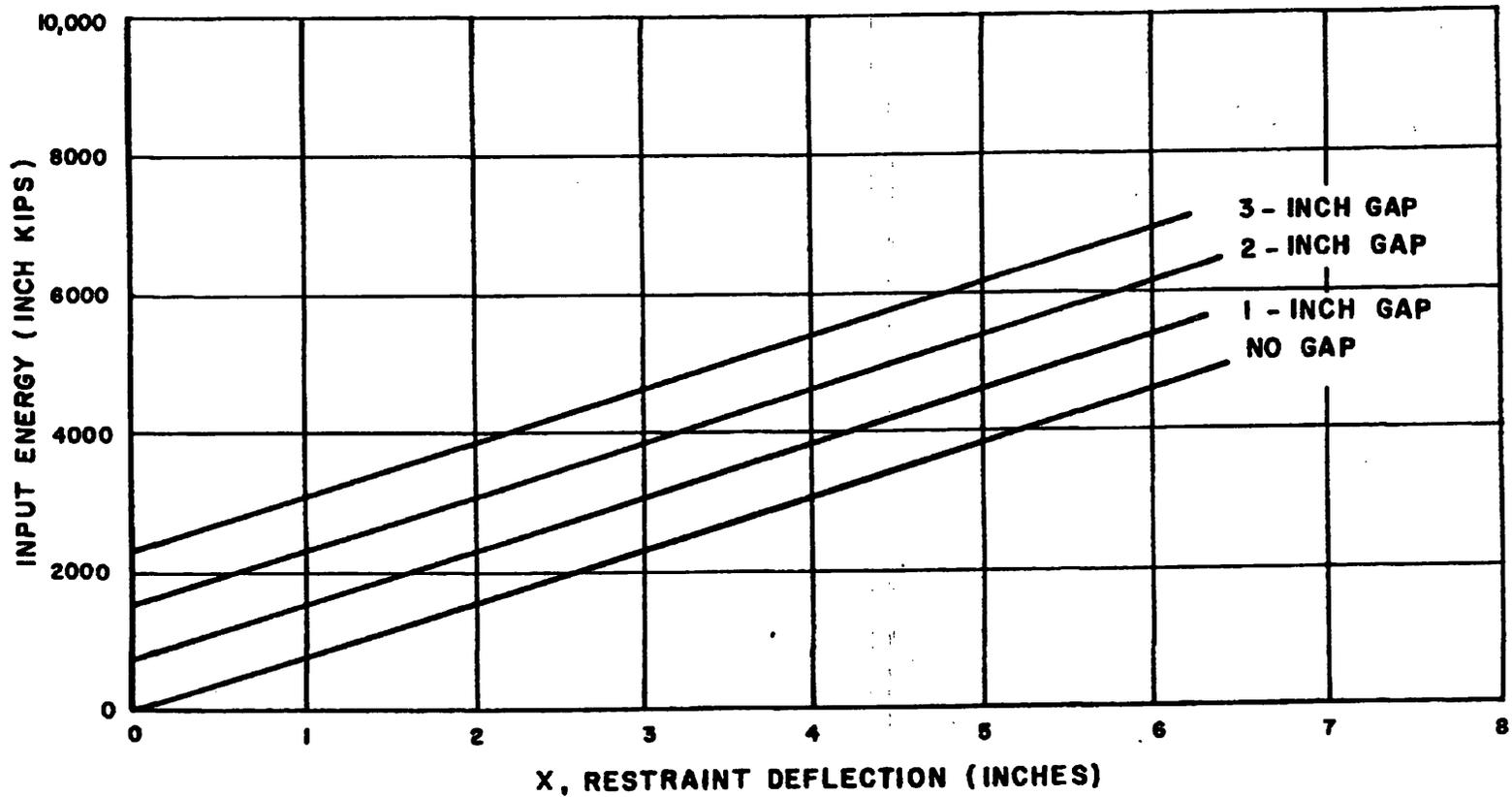


FIGURE 5-2-55
ENERGY FROM BLOWDOWN THRUST
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

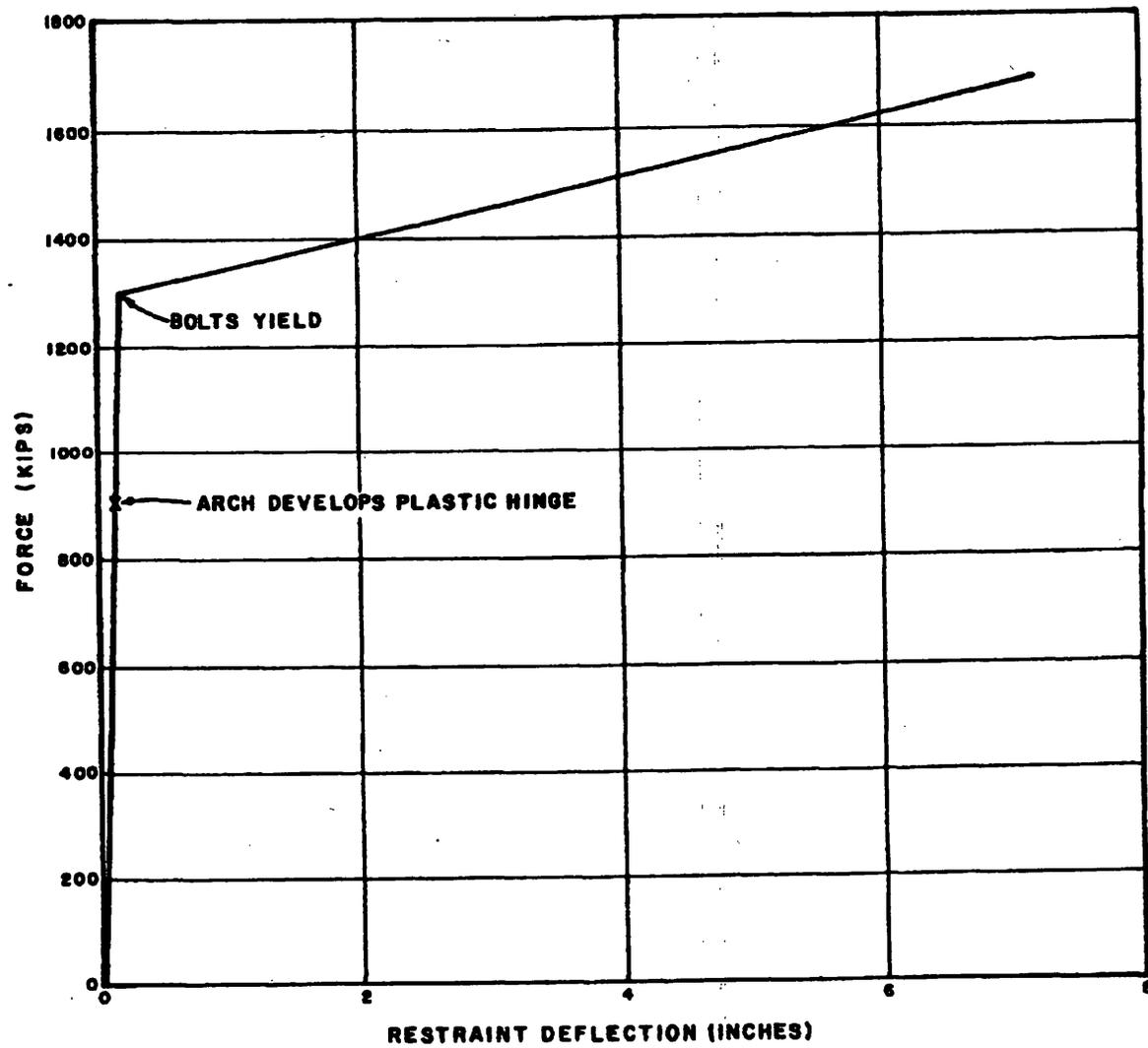


FIGURE 5-2-57
RESTRAINT REACTION RADIAL OUTWARD LOAD
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

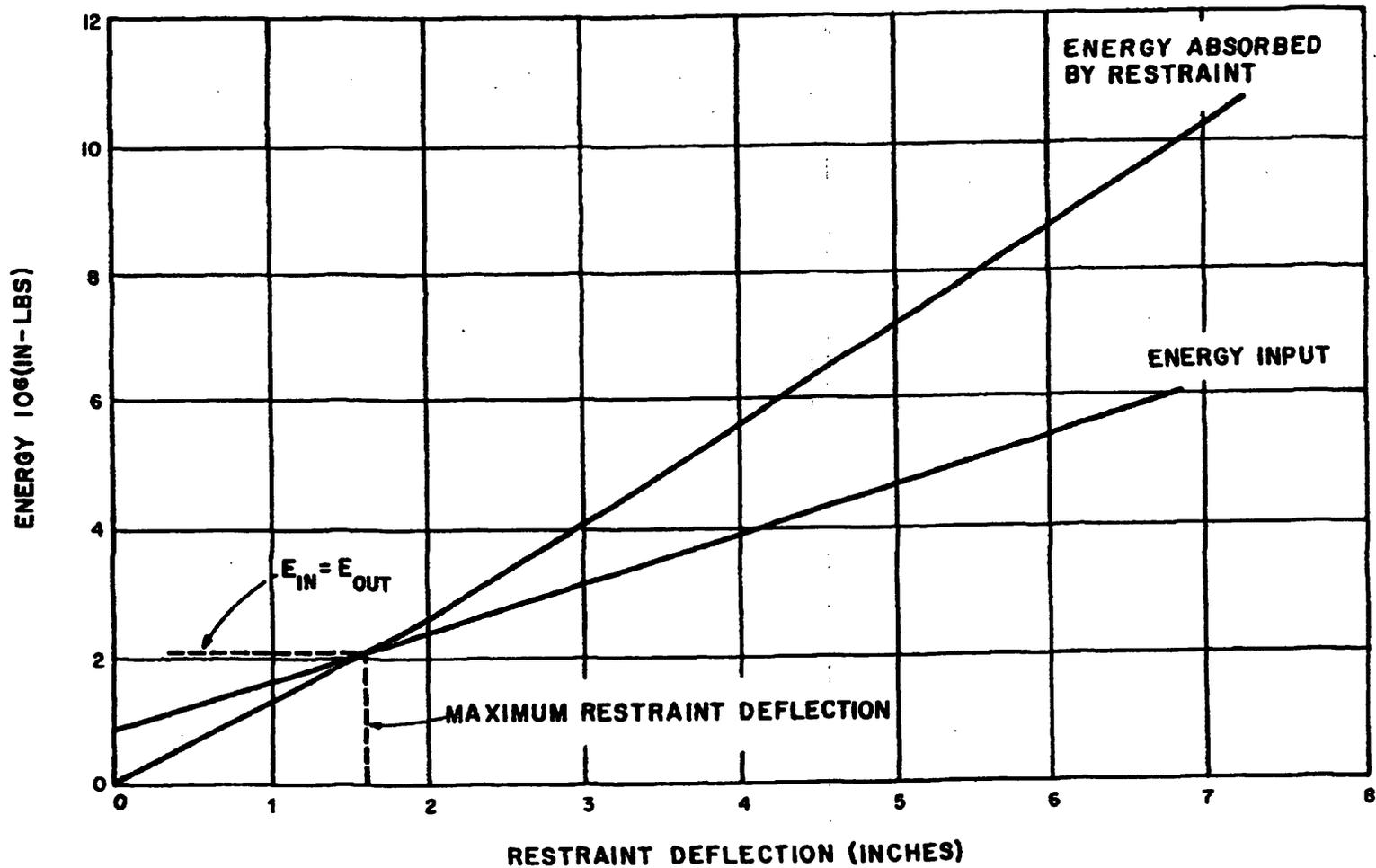
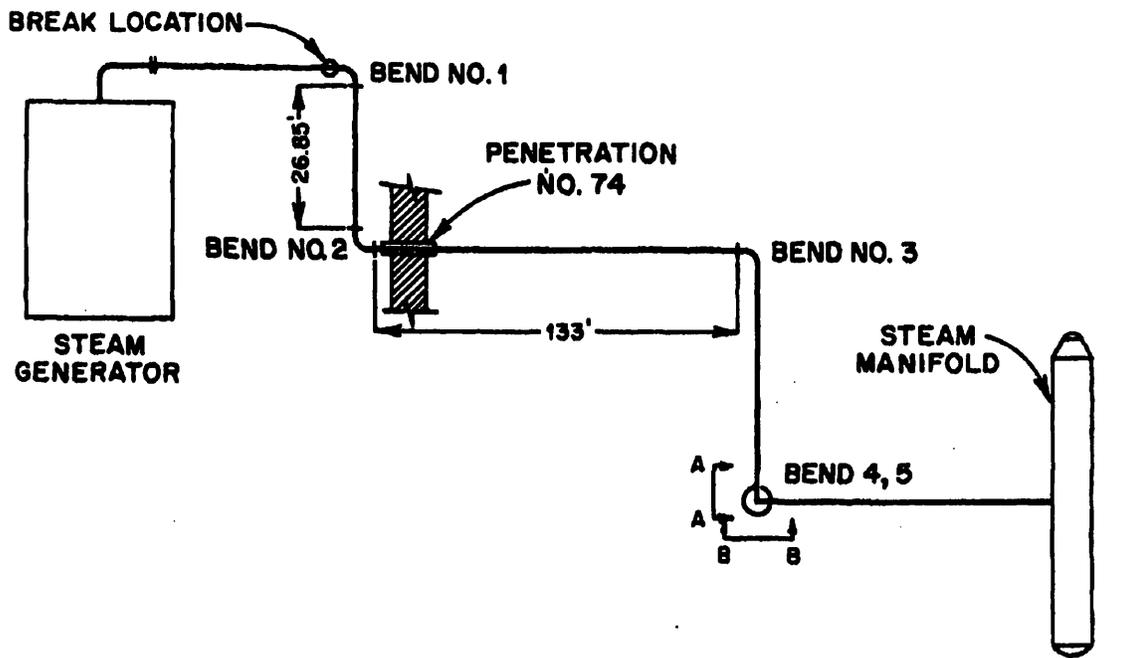
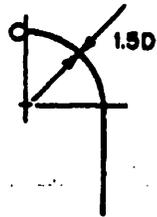


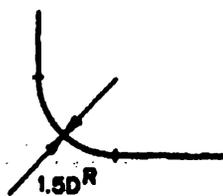
FIGURE 5-2-58
RESULTS OF ENERGY ABSORPTION METHOD
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT



BEND NO. 1



BEND NO. 2



BEND NO. 3

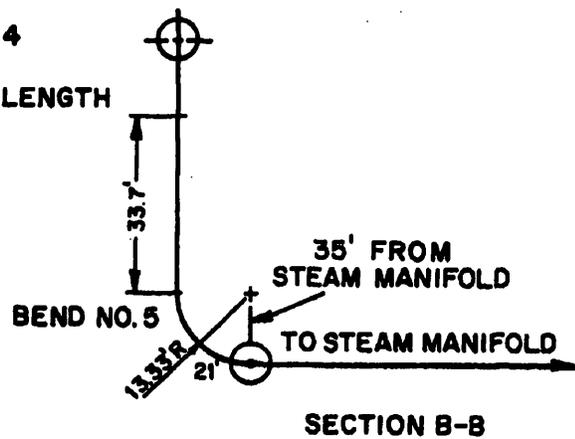
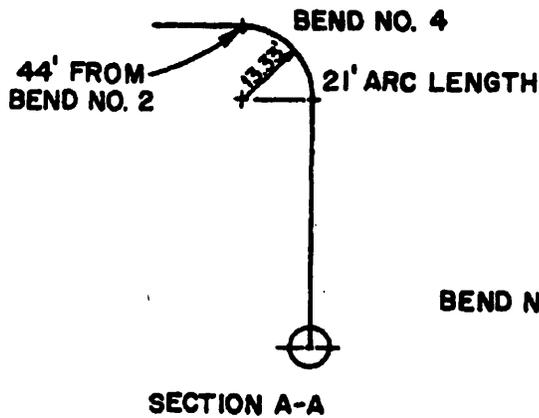
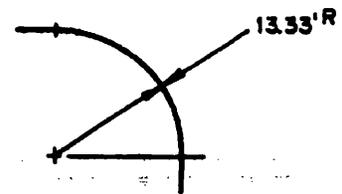


FIGURE 5-2-59
SCHEMATIC DRAWING OF STEAM LINE BREAK
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

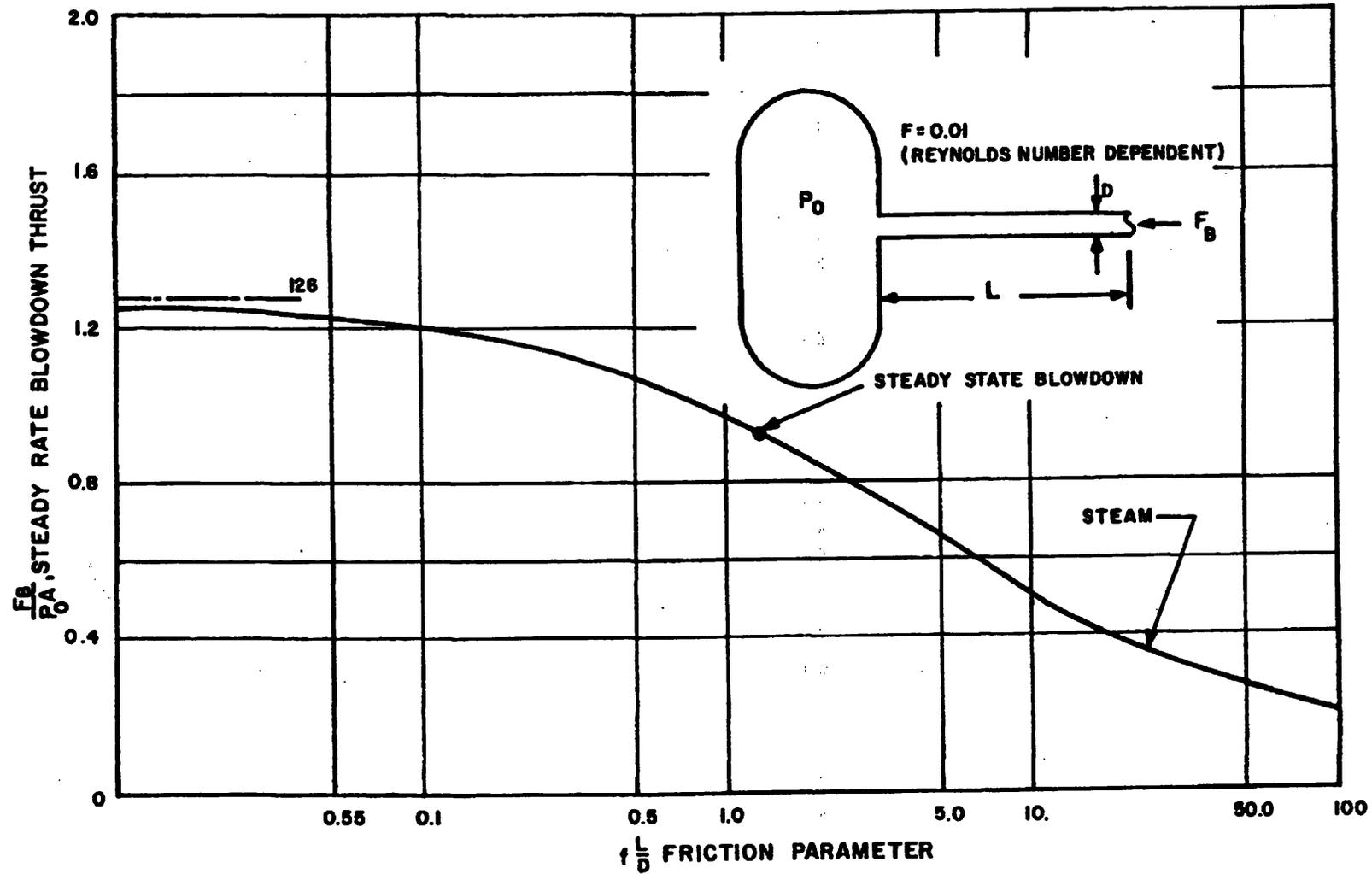
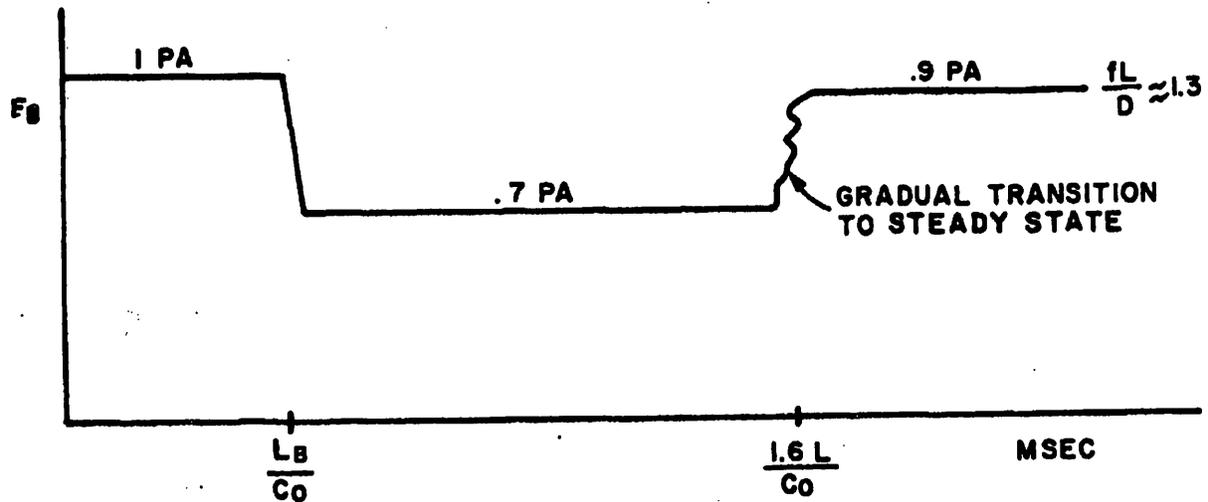


FIGURE 5-2-60
 THE STEADY STATE BLOWDOWN FORCES
 BEAVER VALLEY POWER STATION UNIT NO. 1
 UPDATED FINAL SAFETY ANALYSIS REPORT



L_B = LENGTH OF ELBOW

$$\frac{L_B}{C_0} = \frac{1.5 \times 2.5 \times \pi}{2 (1000)} = 3.68 \text{ MSEC}$$

L = 326 FT. FROM BREAK LOCATION TO THE STEAM MANIFOLD

$$\frac{1.6L}{C_0} \approx 326 \text{ MSEC}$$

$$\text{FRICTION PARAMETER } \frac{fL}{D} = \frac{.01 \times 326}{2.5} = 1.3$$

(FOR THIS CALCULATION ELBOW LOSS WAS NOT CONSIDERED)

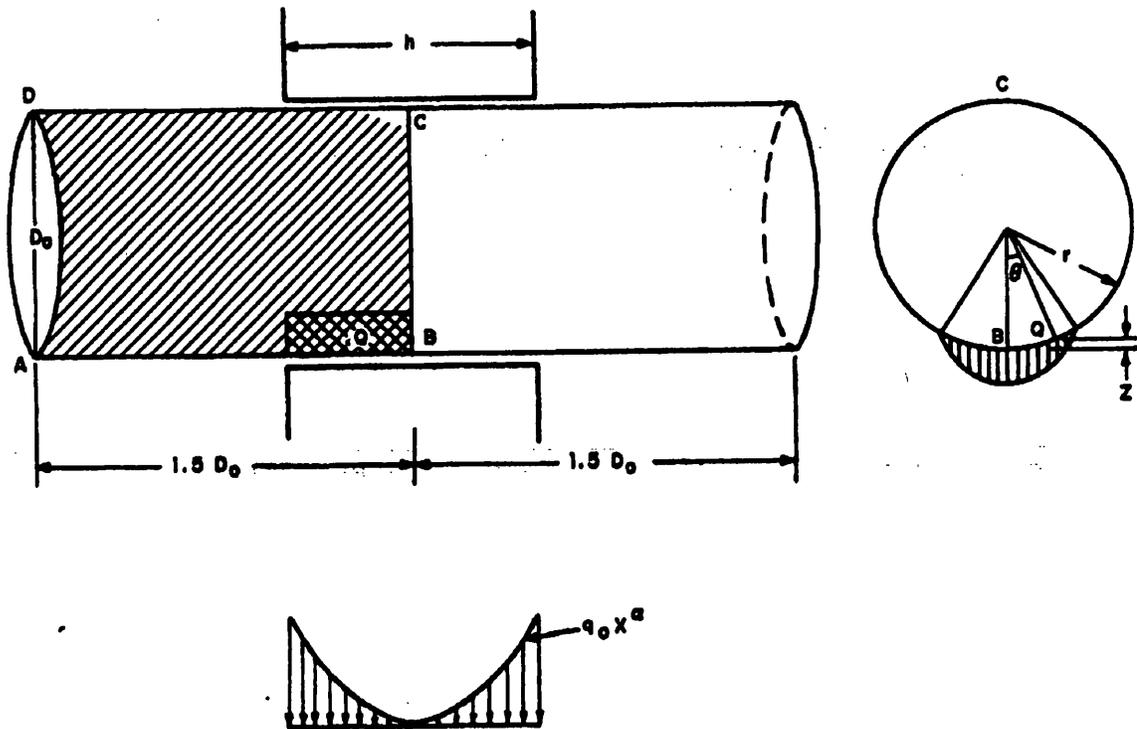


FIGURE 5-2-62
GEOMETRY FOR PIPE INDENTATION
CALCULATION
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

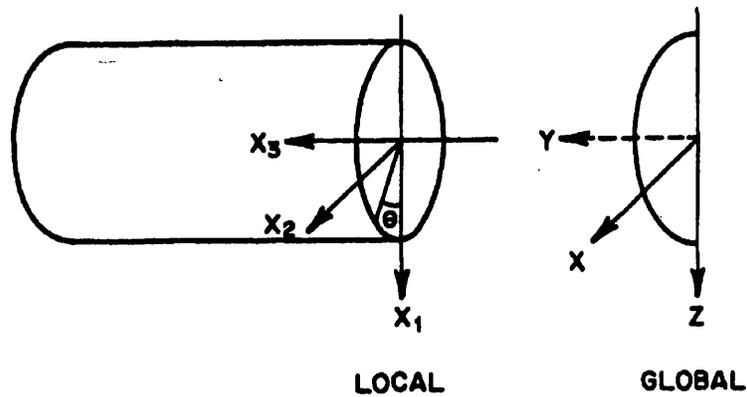
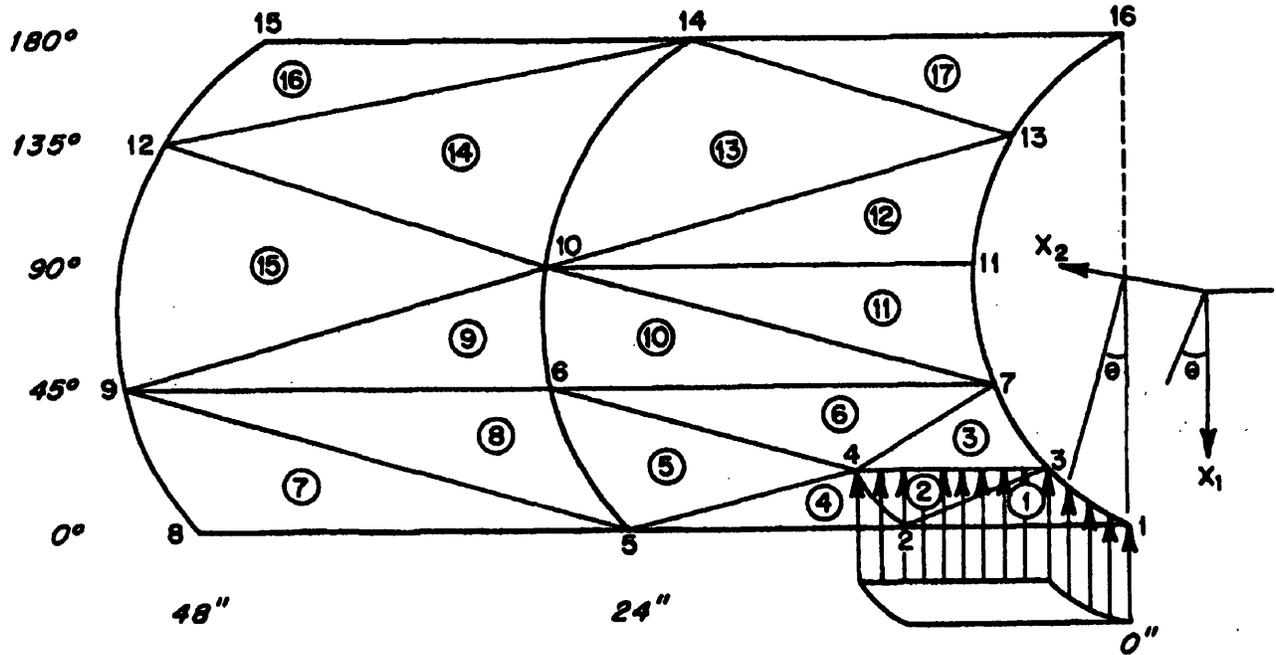


FIGURE 5-2-63
 MATHEMATICAL MODEL FOR PIPE
 INDENTATION
 BEAVER VALLEY POWER STATION UNIT NO. 1
 UPDATED FINAL SAFETY ANALYSIS REPORT

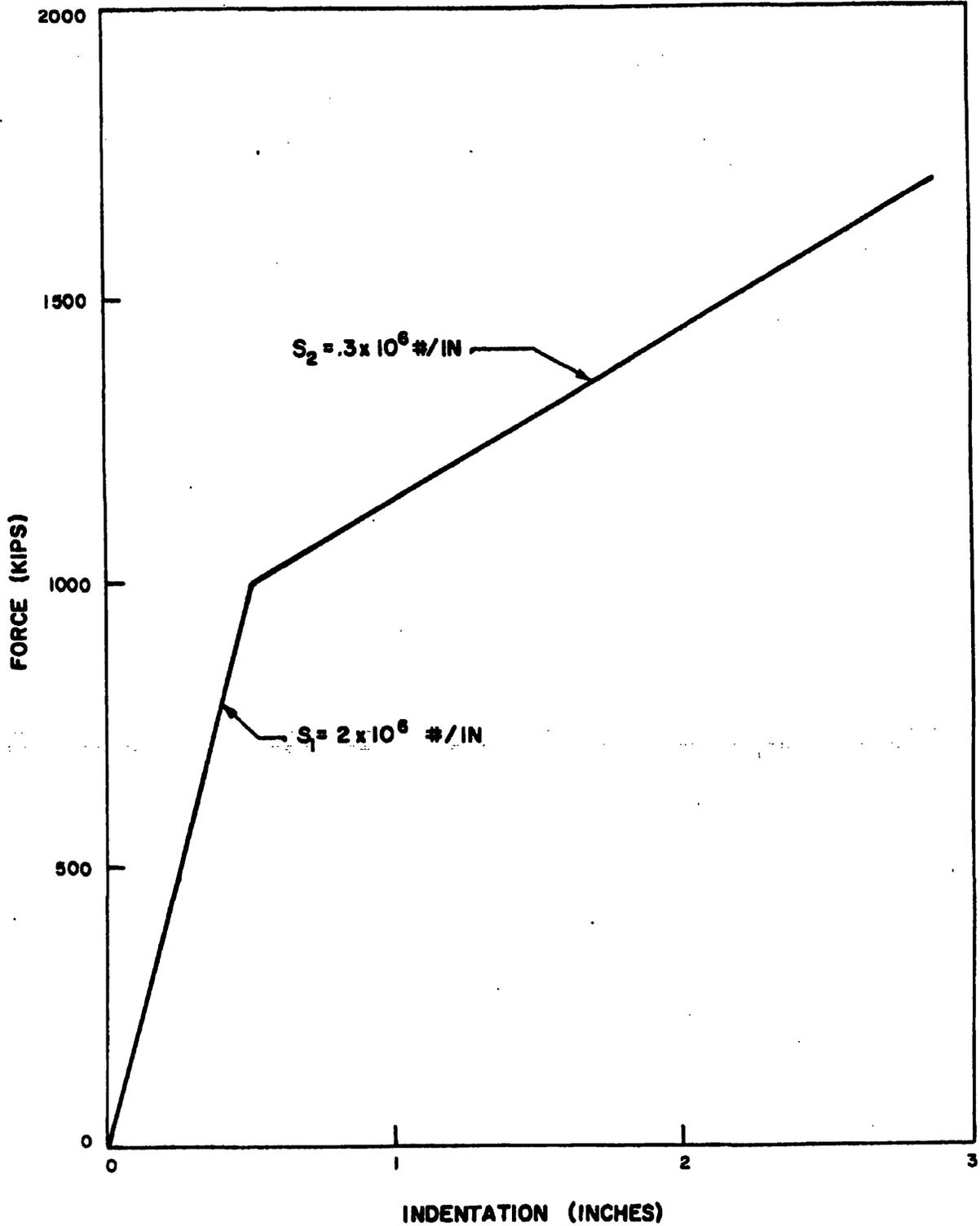


FIGURE 5-2-64
INDENTATION STIFFNESS OF
32 INCH PIPE
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

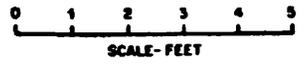
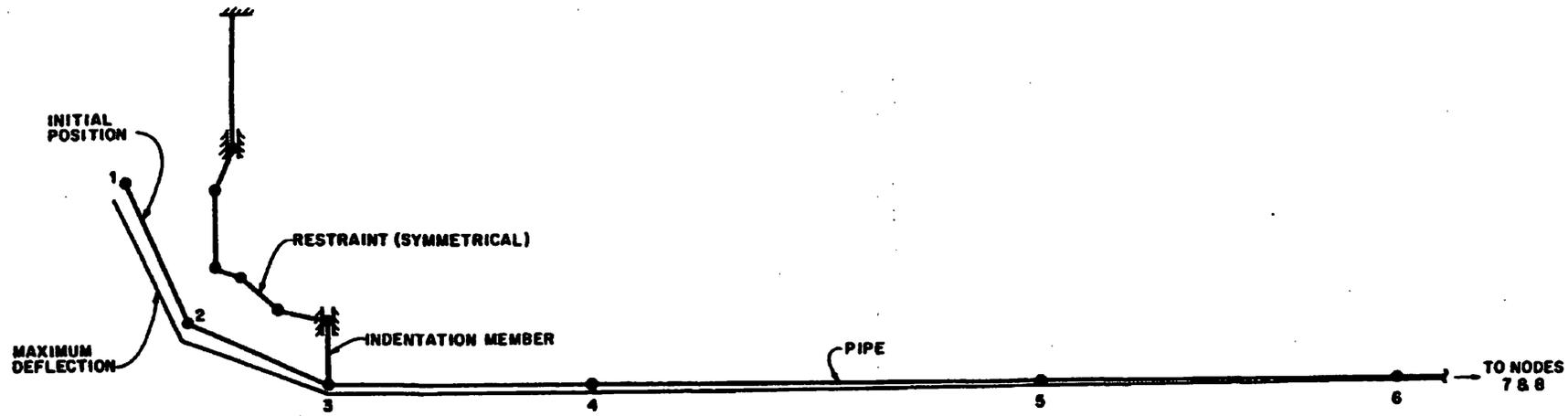


FIGURE 5-2-65
MATHEMATICAL MODEL OF
PIPE AND RESTRAINT
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

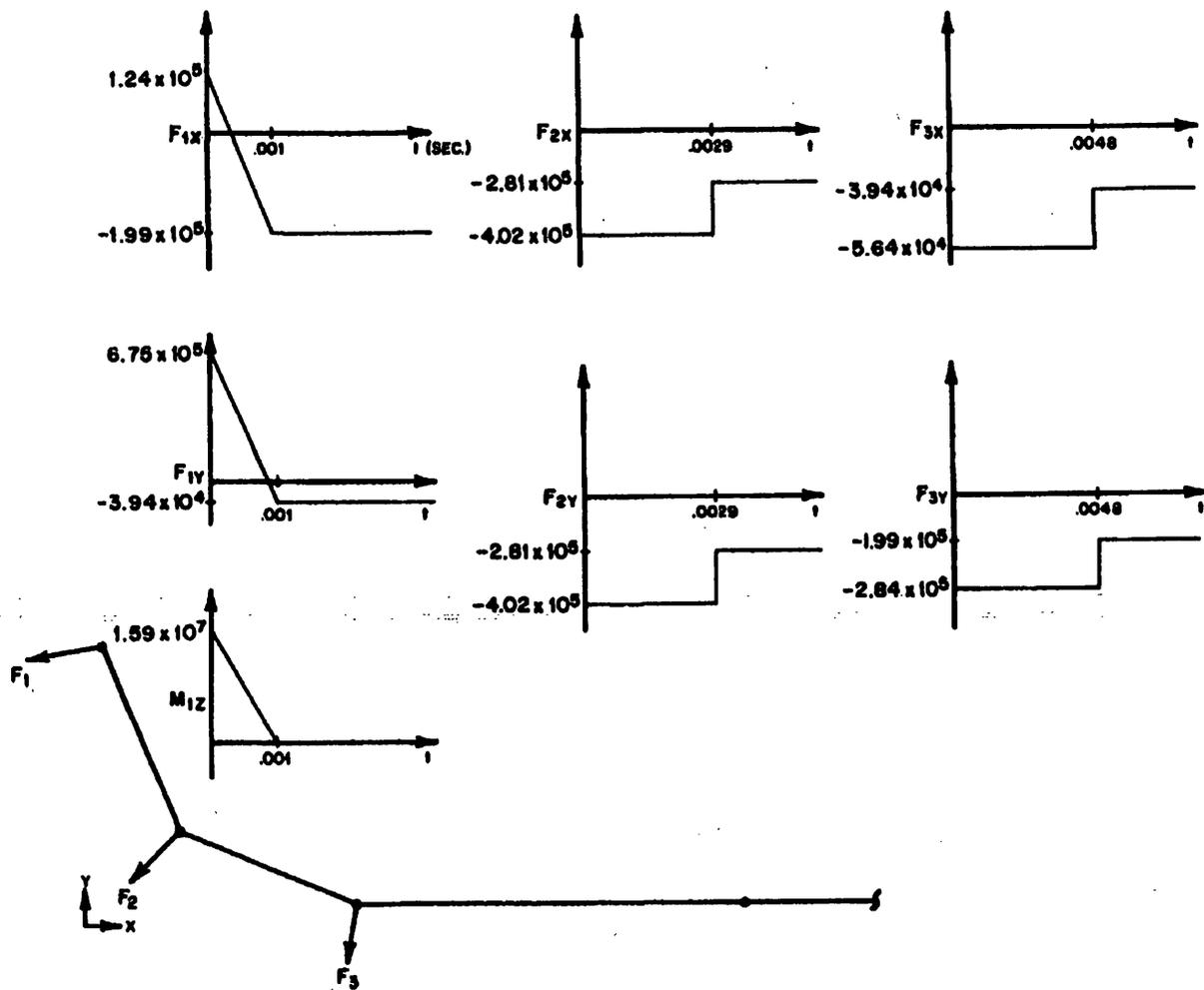


FIGURE 5-2-66
APPLICATION OF BLOWDOWN THRUST
TO MATHEMATICAL MODEL
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

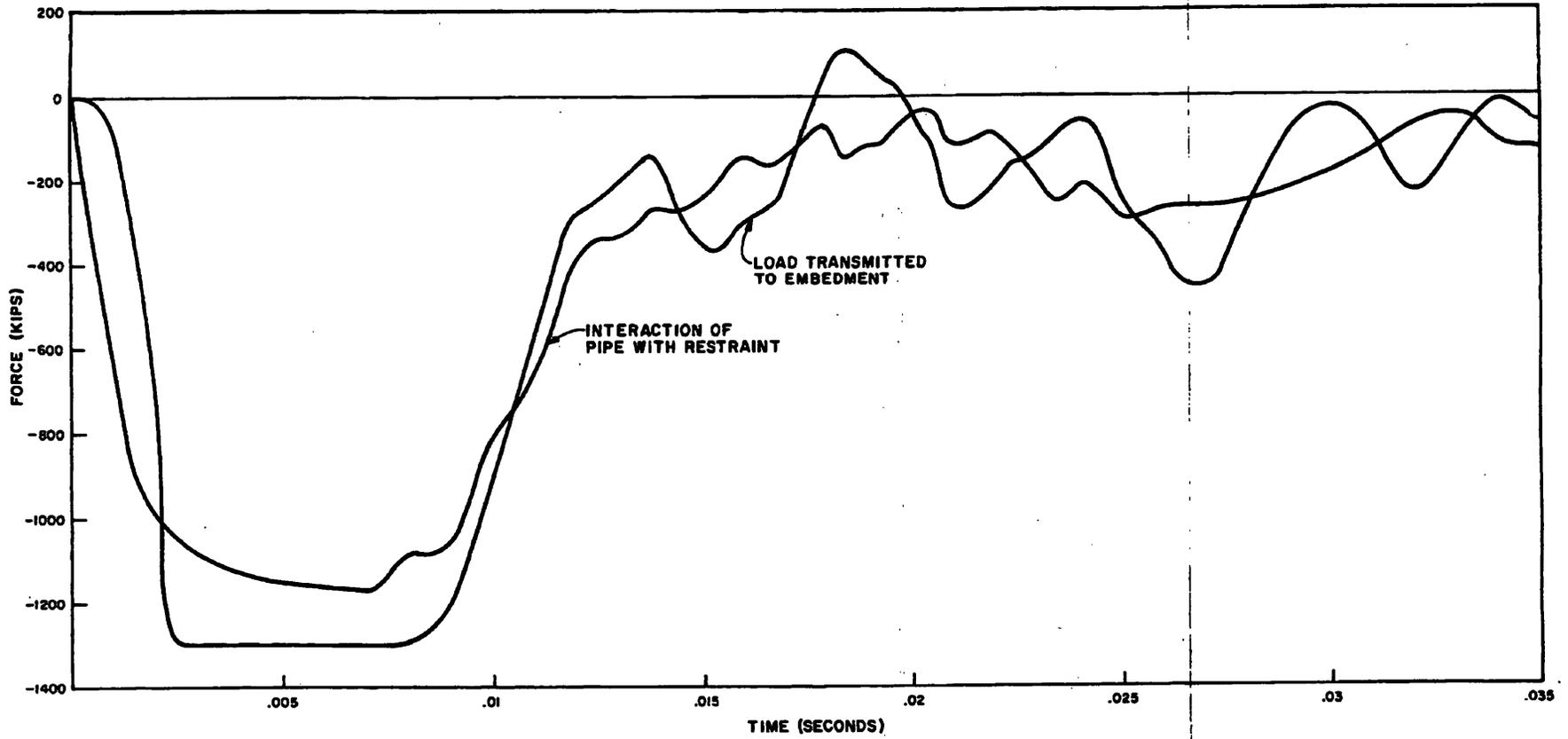


FIGURE 5-2-67
DYNAMIC RESPONSE OF PIPE
RESTRAINT SYSTEM
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

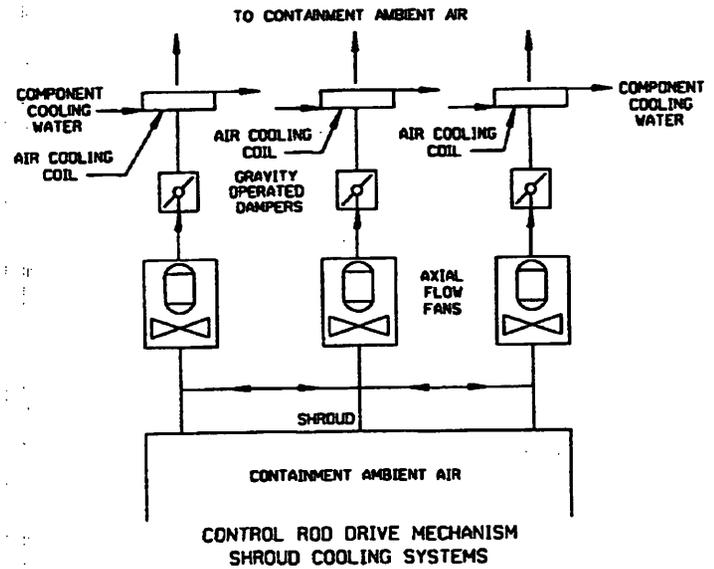
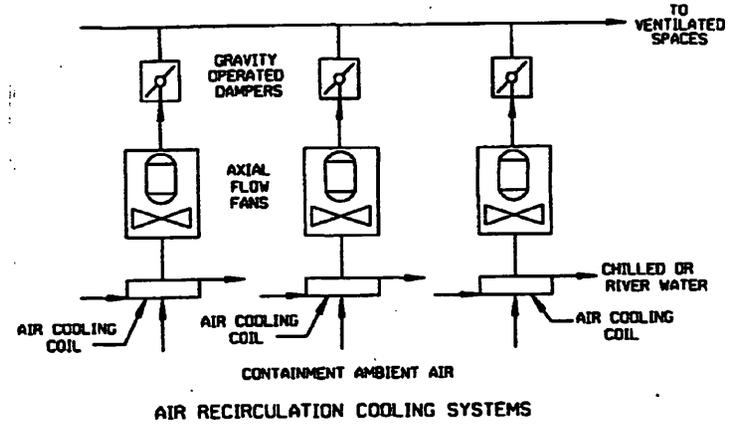
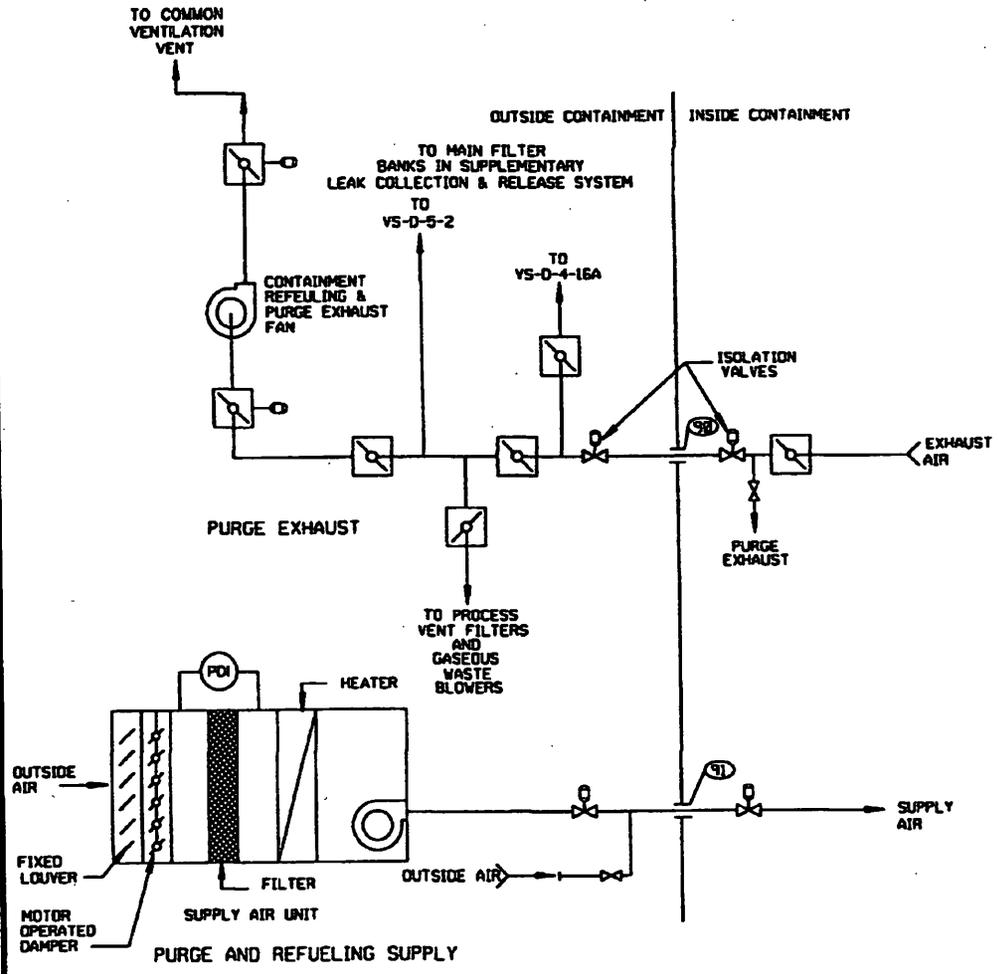


FIGURE 5.4-1
VENTILATION SYSTEMS CONTAINMENT
 BEAVER VALLEY POWER STATION UNIT NO. 1
 UPDATED FINAL SAFETY ANALYSIS REPORT