1. 3-ft thick fill slab.
2. 3-ft thick crane wall.
3. 4-ft to 6-ft thick refueling canal.
4. 2-ft thick operating floor slab.
5. Primary shield wall.

The method of design, stress analysis, critical stresses and locations are as follows:

1. 3-ft thick fill slab - The controlling loads on the 3-ft slab are the reactions are from the primary equipment supports due to various postulated pipe breaks. The slab was designed as a series of radial beams running under the equipment supports and spanning between the reactor support wall and the crane wall. Stresses in reinforcing were limited to 0.9 fy. Maximum stresses occur immediately below the primary equipment supports.

2. 3-ft thick crane wall - The crane wall was designed for a 7 psi differential pressure occurring immediately after a primary pipe break and prior to pressure equalization.

   Although the stress levels associated with this pressure differential were sufficiently low to establish that the concrete could resist the pressure loading, sufficient reinforcing was provided to resist all membrane forces without any contribution from the concrete. Stresses were limited to 0.9 fy. The membrane hoop stress was 33 ksi and the axial vertical rebar stress was 14.3 ksi.

A two dimensional finite element analysis was performed to determine the effect of the jet forces associated with the pipe break on the crane wall.

The jet force associated with a pipe break has been based on the static force PA where P is the primary system operating pressure and A is the cross sectional area of the coolant pipe. The analysis indicated that in local areas (at the application of the force) yielding of the crane wall rebar will occur. The load was assumed to act at the mid-height of the wall, thus causing maximum bending moment. The ability of the wall to support the dead load of the crane was checked, considering the yielded area indicated by the computer analysis as unable to carry load. A beam 12-ft long and 5-ft deep (the underside of the operating floor to the top of the potential yield portion of the crane wall) was found to provide more than twice the ultimate capacity required. This analysis was very conservative for three reasons:

1. A jet force load at this location would cause little yielding since it is not located at mid span.

2. The haunch at the underside of the operating floor was not considered.

3. The membrane effect of the circular crane wall was not taken into account.

Further stability of the crane wall was demonstrated by determining the ultimate failure load by means of a yield line analysis. This analysis indicated that the structure has the capacity, through strain energy of structural response, to resist the uniform jet force load of 1500 kips or 975 kips with the 7 psi pressure differential without failure.

The containment internal concrete is essentially rigid; (fundamental frequency 18.6 cps) therefore, seismic loads were calculated using the maximum ground acceleration (0.15g).

The crane wall was initially considered as a cantilever beam with a frequency of approximately 13 cps and the base shear was determined by the response spectrum approach. The base shear was distributed to the individual nodes by the formula:

$$Fx = \frac{W_x h_x V}{\Sigma \; W h}$$

Where

V	=	base shear
W_x	=	weight of node under consideration
h_x	=	distance from base to section under consideration.

3 Wh = Summation of the product of weights and heights of all nodes

The moment at the base was determined and the uplift calculated by considering a circular ring of thickness equal to the area of steel per in. This maximum uplift, which occurs at one point at the base of the structure stresses the rebar to 5.2 ksi.

The crane wall was also designed to resist steam and feed water pipe break reactions of 340 kips and 200 kips where supports are connected to the wall. The extra steel provided for pipe break loads is available in the form of steel buttresses to resist pressure, jet force, and seismic loads; however, it was not considered in the analysis.

3. <u>4-ft to 6-ft thick refueling canal</u> - The refueling canal was designed for the 7 psi pressure differential. The wall resists the pressure by spanning vertically between the refueling floor and the operating floor. Stresses were limited to 0.9 fy.

An analysis was performed to check the effects of the jet force load the cross section was found to be sufficient to provide stability. A yield line analysis was performed and provided the basis for the above.

The seismic load was determined by the same procedure used for the crane wall. The average load in kips/ft was distributed over the wall and the vertical span was conservatively assumed to carry the entire load. The resulting bending moment was found to be well within the capacity of the wall.

4. <u>2-ft thick operating floor slab</u> - Because of the many openings in the floor for equipment, the floor was designed as a series of beams. Principal loadings were D.L. + 500 psf live load and 7 psi upward pressure differential + D.L. The first loading (D.L. + 500 psf live load) was designed in accordance with Part IV-B of ACI 318. Stresses for the pressure differential case were limited to 0.9 fy.

   The operating floor was investigated. There appears to be very little area of the operating floor, which could be reached by the expanding jet of water from a break in the reactor coolant system. The jet will be greatly dispersed in the distance between the primary coolant piping and the underside of the operating floor. The only area of the floor, which could be struck by a jet spans between areas of the floor heavily reinforced as beams. The span cross section consists of a T-beam with the 2-ft thick floor acting as the flange and the 7-ft high biological shielding wall as the web. This section can resist the jet force load within 0.9 fy stress limit on the rebar.

5. Primary Shield Wall - This was designed for two loading conditions due to a split in the reactor. The stress in the reinforcing was limited to the tensile strength of the bars. The first load considered was a 1-ft wide longitudinal split along the length of the reactor. The vessel is assumed accelerated through a 6-in. distance against the support wall by the jet force caused by a 2200 psi pressure acting through a 26.4-ft long by 1-ft wide longitudinal vessel rupture, which results in an impact load of 650 k/ft. This load is imposed by considering an impact factor of two. The maximum rebar stress is 69.5 ksi. The second load considered a pressure buildup of 1000 psi inside the pit due to release of reactor contents. This produces a rebar stress of 86 ksi. The rebar used is ASTM A 432 with specified yield of 60 ksi and ultimate tensile strength of 90 ksi.

   To protect the containment base liner, an average of 2-ft of concrete above the containment liner plus a 1-in. liner plate embedded on top of the concrete was provided at the bottom of the containment reactor cavity pit. Below the containment liner plate is 4.5-ft of structural concrete poured on rock.

   Temperature differential conditions as a result of a LOCA are considered to be of such short duration that the effects were not used in the design of interior structures for stress analysis. A sketch of the design conditions is given in Figure 5.1-24.

   During normal operations, the only significant transient temperature gradients occur during startup. The minimum containment internal temperature is limited to 50°F. The maximum operating containment internal temperature is 130°F. Forced movement of containment air is used to limit the concrete temperature surrounding the reactor vessel. This forced air movement of the containment air as well as normal convection and radiation is expected to limit the concrete temperature differentials in the range of 5°F to 10°F. To demonstrate the large margin available in the concrete crane wall and the primary shield wall, a conservative assumption of a 30°F temperature gradient has been evaluated. The evaluation included the gradient effect through the crane wall, the 6-ft thick portion of the primary shield wall below the reactor coolant pipe nozzle, the 5-ft thick portion of the primary shield wall where the nozzles penetrate the wall, and the 4-ft thick wall above the shield wall.

   The maximum rebar stress was found to be 4500 psi and occurs in the vertical rebar in the crane wall. The maximum compressive concrete stress was found to be 226 psi and occurs in the hoop direction in the 5-ft portion of the primary shield wall. These stresses are approximately 20-percent of the allowable working stress values and will have no significant effect on the design adequacy of the structures analyzed.

5.1.3.8 Pressure Stresses

5.1.3.8.1 Accident Pressure

Pressure effects on the containment structure may be divided into two types:
(1) membrane stresses and (2) discontinuity stresses.

1. For membrane stress analysis, the dome and cylinder are treated as thin-walled shell structures. (The thickness to radius ratio for the dome is 1/20 and the cylinder 1/15. These ratios are smaller than the 1/10 criterion for thin-walled shell analysis.[8] Membrane forces are resisted by steel reinforcing.

2. Discontinuity stresses occur at the juncture of the cylinder and the mat and the juncture of the cylinder and dome. Discontinuity effects are determined as follows:

   a. The radial growth of the shell is computed based on membrane stress in the reinforcing and liner.

   b. The flexural rigidity of the meridional wall section is determined based on a cracked section analysis in accordance with conventional reinforced concrete design techniques.

   c. Moments and shears are calculated based on having consistent deformation for the two elements at the point of discontinuity.

Discontinuity effects at the spring line are very slight due to the small difference in radial growth between the dome and cylinder. Since the circumferential reinforcing in the dome and cylinder vary, stresses and therefore deformations are essentially equal.

The mat is considered as offering complete fixity; no credit is taken for the liner at the base in resisting moments since at the point of maximum shear the bond between the liner and concrete is insufficient to transmit complementary beam shear. A slip surface between the concrete and liner is formed and the liner is subjected to membrane forces only.

The 9-ft thick mat is subjected to the following due to pressure inside the containment building:

1. Uplift at the juncture with the wall.

2. Moment and shear due to discontinuity effects with the wall.

3. Downward pressure loading due to internal pressure.

The 9-ft mat is designed to accommodate the flexural effects of these loads. At the crane wall, the mat is founded on the unyielding rock and further pressure loads are transmitted through bearing directly into the rock.

Resistance to these loads is based on a cracked concrete section. No credit is taken for the liner for the same reasons given for the wall.

Discontinuity shears in both the cylinder and mat are resisted by either bent bars or stirrups.

In the outer portions of the base mat, the slab is raised off of the rigid foundation under accident loadings; thus no frictional resistance can be offered by the rigid foundation. Where the uplift is overcome, the only load of any consequence, which must be resisted by the mat is the radial tension. The restraint, which is imposed by the rigid foundation on the bottom portion of the base mat, effectively eliminates all radial tension in the mat. However, for conservatism this restraint has been neglected in the analysis of the mat for radial tension. The hoop and radial reinforcing supplied as temperature reinforcing is more than adequate for this purpose.

5.1.3.8.2 Soil Pressure

Portions of the containment structure are subjected to the effects of backfill bearing against the containment wall. The effects on the structure are:

1. Shear and overturning effects due to seismic response and interaction between the soil and structure.

2. Discontinuity effects caused by the soil restraining deformation of the structure under accident pressures.

To determine the shear and overturning effects two limiting cases were investigated. The first was the case where the structure and soil move out of phase. It was assumed that the structure was subjected to the passive pressure of the soil with the mass of soil, within the shear failure envelope, accelerated against the structure with ground acceleration. In the second case the soil and structure move in phase. For this case it was assumed that the structure was subjected to the active pressure of the soil with the mass of soil, within the shear failure envelope, accelerated with the structure at ground acceleration.

These loads were then treated as external loads on the structure. See Section 3.1.5 of the Containment Design Report for additional information.

To determine the discontinuity effects caused by soil restraint, the structure was analyzed for the passive pressure case. The restraint of the deformation of the structure due to the soil was calculated. Vertical and circumferential bending moments due to this restraint were then determined. Reinforcing bar stresses were calculated and found to be minor. This analysis was then verified by a finite element analysis.

In this analysis, full contribution of the backfill was assumed. During the course of construction it became necessary to build a retaining wall in a substantial area of the backfill, to facilitate construction. The retaining wall extends over 50-ft in plan and includes all of the high fill points assumed in the analysis and design. It can therefore be concluded that the analysis was conservative in that the backfill effects on the completed structure would be only a fraction of that assumed in the original design.

5.1.3.9 Thermal Stresses

Temperature effects on the containment structure may be divided into two separate considerations: one effect is due to a thermal gradient through the wall, the other is caused by the rapid temperature rise of the liner under accident conditions. The reinforced-concrete wall restrains the liner from growing, resulting in compression in the liner and additional tension in the reinforcing.

1. Calculation of gradient stresses is based on method of analysis outlined in ACI 505-54, "Specification for the Design and Construction of Reinforced Concrete Chimneys."[9] The gradient used is linear with 120°F on the inside and 0°F exterior concrete temperature (-5°F ambient). The maximum operating temperature of the containment is 130°F. The effect of elevated operating temperature (up to 150°F) on the structural elements was evaluated in 1987 and was found to be acceptable.

   The ACI method assumes a cracked section in which the concrete carries no tension. The neutral surface (surface at which no thermal stress exists) is determined. Stresses in the liner and reinforcing are calculated based on the assumption that there is no distortion of the wall; i.e., variation of strain through the wall thickness is linear.

2. To determine the effects due to rapid rise in liner temperature, there are two basic assumptions made. The first is that the effects are internal in nature; i.e., the compressive force in the liner is balanced by a tensile force in the reinforcing. The second is that there is no distortion of the wall.

   Because temperature effects are internal in nature and do not affect the overall tensile load carrying capability of the structure, local yielding of reinforcing under accident conditions is acceptable.

   The temperature gradient through the wall is essentially linear on both the insulated and uninsulated portions and is a function of the operating temperature internally and the average ambient temperature externally. Accident temperatures mainly affect the liner, rather than the concrete and reinforcing bars, due to the insulating properties of the concrete. By the time the temperature of the concrete adjacent to the liner begins to rise significantly, the internal pressure and temperature in the containment shell due to maximum thermal gradient will not influence the capacity of the structure to resist the other forces. Temperature effects induce stresses in the structure, which are internal in nature; tension outside and compression in the inside of the shell such that the resultant force is zero. Loading combinations concurrent with these temperature effects may cause local stresses in the outside horizontal and vertical bars to reach yield; however, as local yielding is reached, any further load is transferred to the unyielded elements. At the full yield condition, the magnitude of final load resisted across a horizontal and vertical section remains identical to that which would be carried if the temperature effects were not considered. Thus, the overall carrying capacity of the structure and the factor of safety of the structural elements are not affected.

5.1.3.10 <u>Analysis of Openings</u>

The methods followed in design of large openings are described in Section 3.4 of the Containment Design Report (CDR). Included are descriptions of the safety factors used in design. Sample calculations are provided, listing all the criteria and analyzing the effects of all pertinent factors, such as cracking. Also addressed in the CDR is how the existence of biaxial tension in concrete (cracking) has been taken care of in the design, and how the normal and shear stresses due to axial load, two-directional bending, two-directional shear, and torsion are combined. Additionally, the criteria for the design of the thickened part of the wall around the openings is stated.

The methods used to check the design of the thickened stiff part of the shell around large openings and its effect on the shell, torsional stresses, and shrinkage considerations are also addressed in Section 3.4 of the Containment Design Report. This section also describes how deformations and forces are handled around the large openings and in the transition zones into the main portion of the structure.

In the cylindrical section of the containment, where there are large openings for access hatchways and penetrations, the reinforcing bars (hoop, vertical and diagonal) are continued without interruption around the openings.

No bar terminates at any openings as illustrated around the penetration in Figure 5.1-1. Also additional bars have been furnished locally to take the stresses developed around large openings. Concrete is locally thickened at the equipment access hatchway area to accommodate all the reinforcing bars required in this area.

A finite element analysis is performed on the large openings. Representation of the structure is by rectangular elements; each element consists of ten layers of orthotropic, elastic material to represent the reinforcement, concrete and the liner. About 1000 degrees of freedom are considered in the model. This analysis is used as a check on the adequacy of the large openings. Results appear in the Containment Design Report.

A finite element analysis of the equipment hatch area indicated local liner plastic deformations during the pressure test. For the order of magnitude and location of these stresses, see Section 3.4 of the Containment Design Report. These deformations have no influence on the structure during the pressure test due to the ductility of the studs and liner plate.

The limiting elastic liner deformations during test pressure will be from tensile stresses. During an accident loading they will be from compressive stresses. Therefore, a relationship between the pressure and accident loads cannot be determined directly. However, the test pressure demonstrates the ductile behavior of the liner.

Since the containment is not subject to accident temperatures during the testing, no direct correlation between test and accident conditions can be made in evaluating thermal stresses at large openings.

The liner is stressed beyond the yield point in very local areas adjacent to the transition from the thickened equipment hatch boss to the cylinder wall. The maximum stress is equal to 39.28 ksi for the 1.5P loading condition. The strain corresponding to this stress (0.17-percent) is below the limits (0.5-percent) stated in Section 2.2.4 of the Containment Design Report. The average liner stress in the cylinder for the 1.5P load combination is approximately -15 ksi in the vertical direction and -2.0 ksi in the horizontal direction.

The maximum rebar stress associated with the 1.5P load combination is approximately 66 ksi in the 4'-6" portion of the containment wall cylinder.

For a complete discussion of liner stresses, see the Containment Liner Stress Analysis Report. For a detailed discussion of liner stresses in the equipment hatch area and further justification of the stresses noted above, see Section 3.4.4 of the Containment Design Report.

All reinforcing is continuous around penetrations. Steps have been taken to ensure that no local crushing of concrete will occur. From Reference 16, it has been determined that in order to prevent local crushing of the concrete, a minimum bend diameter of 31 times the bar diameter is required when the reinforcing is stressed to yield. The angle of bend in the rebar determines the force that will be transmitted to the concrete in the event the bar tries to straighten out due to tension. For this reason most bars are bent at 10 degrees except at large penetrations including the equipment hatch, personnel lock, main steam and feedwater, and air purge penetrations, where the deviation of the bar from its centerline is too large to permit a 10 degree bend. In these cases the bars have been bent at 30 degrees but a tie-back system is used, which prevents a buildup of forces. To prevent this buildup, (in all cases except the equipment hatch penetration), the line of force makes an angle of one-half of the angle of bend, from a horizontal line from the vertical bars and from a vertical line for the horizontal bars and is tangent to the outside of the penetration.

At the personnel and equipment hatches a large void will be carried since, due to the large offset of the bars from their centerline, it will take the bars longer to return to their centerline after passing the penetration. To prevent any cracking and spalling of concrete and to add lost strength to the cross-section, these voids have been filled with added rebar, which achieves bond by means of mechanical anchorage.

The same precautions mentioned above have been taken with the seismic bars. See Figure 5.1-25.

For penetrations between 9-in. and 18-in. in diameter, all the reinforcing bars including primary and secondary vertical bars and diagonal bars have been grouped around the penetrations. Due to the continuity of the bars and the relatively small opening size, no special provisions need be made to resist normal, shear, and bending stresses. The penetrations are keyed into the concrete, thus creating an edge loading, which will put torsion into the wall. The loads are small and the rebar will feel little effects from this torsional loading.

For penetrations greater than 18-in. up to 48-in. in diameter, the bars are continuous. Due to the large angle of bend of these bars, a tie-back system is used, which offers additional resisting strength to shear, bending, and torsional stresses.

5.1.3.11 Seismic and Wind Design

The design of the containment, which is a Class I structure (see Section 1.11), is based on a "response spectrum" approach in the analysis of the dynamic loads imparted by earthquake. The seismic design takes into account the acceleration response spectrum curves as developed by G. Housner. Seismic accelerations have been computed as outlined in TID-7024[10] and Portland Cement Association Publication.[11]

The following damping factors have been used:

	Component	Percent Critical Damping
1.	Containment structure	2.0
2.	Concrete support structure of reactor vessel	2.0
3.	Steel assemblies:	

	a.	Bolted or riveted	2.5
	b.	Welded	1.0
4.	Vital piping systems		0.5
5.	Concrete structures above ground:		
	a.	Shear wall	5.0
	b.	Rigid frame	5.0

As indicated in Section 5.1.2.2, ground accelerations used for design purposes are 0.1g applied horizontally and 0.05g applied vertically. The natural period of vibration is computed by the Rayleigh method; in this method, the containment structure is analyzed as a simple cantilever intimately associated with the rock base and with broad base sections of adequate strength to assure full and continued elastic response during seismic motions. Further, both bending and shear deformations are considered.

The structure is divided into sections of equal length and loaded laterally by dead weight of the section and any equipment and live load occurring at the section. Deflections caused by shear and moments are then determined, and the end deflection is given the value $\phi' = 1.0$ with corresponding values determined for other sections. The natural period of vibration for the structure is then determined by setting potential energy equal to kinetic energy and solving for the period.

$$T = 2\Pi \left[\frac{Y_0 \sum \varnothing^2 \, dm}{g \sum \varnothing \, dm} \right]^{½}$$

where

Y_0 = maximum actual deflection

$$\phi = \frac{\text{deflection of section under consideration}}{\text{maximum actual deflection}}$$

g = acceleration due to gravity

dm = weight of section under consideration

T = period in sec.

Based on an uncracked concrete section, the period is determined to be 0.241 sec. A more realistic calculation for a cracked section, using reinforcing steel and liner as the resisting elements, yields a period 0.936 sec.

Using the derived period and entering the acceleration spectral curves, Figures 1.11-1 and 1.11-2 of Section 1.11, and applying a 2-percent critical damping, a spectral acceleration for the containment was selected. This value was derived to determine the base shear. The distribution of base shear is a triangular loading assumption.

This assumption yields a load distribution pattern with zero loading at the base to a maximum loading at the spring line of the dome. Above this line, the loading decreases due to a change in section and consequently change in weight. This load distribution allows the determination of shears and moments at any critical section through the containment from which the appropriate unit stresses are obtained.

Seismic shears are resisted by diagonal reinforcing except in the upper areas of the dome. No credit is taken for the reinforcing in compression.

From 30 degrees above the springline, where the seismic shears are small, the shears are carried by dome reinforcing steel lying in the plane of principal tension

A finite element analysis was performed on the basemat using loads determined for the three basic loading conditions specified in the Containment Design Report. Maximum hoop moment caused by lack of symmetry of the seismic loading was found to be 454 in.–kips/in. This compares with a capacity of 690-in.-kips/in. for the in-place hoop reinforcing.

Tornado loads have not been considered in the design of the Indian Point Unit 2 Containment Building; however, similarity in design of Indian Point Unit 3 (where such loads are considered) indicates that seismic reinforcement bars provide a more than adequate mechanism to withstand the torsional effect of Tornado loads.

The torsional effect results from wind striking the containment building at an angle α from the normal, as shown in Figure 5.1-26. The torsional force is due to the component of the wind tangential to the surface of the containment building and is equal to:

$$F_t = AC_D\ (q)\ (\sin \alpha)$$

Where

A = surface area of the containment

C_D = 0.5 from A.S.C.E. Paper 3269 - "Transactions of the A.S.C.E.," Vol. 126 Part II 1961, p. 1165 (coefficient of drag)

q = 0.002558 V^2 (wind pressure)

α = 45 degrees

This assumption is conservative in that the actual tangential force would be the result of skin friction and the effects would be negligible.

This component of torsional force is computed from a direct wind loading as based on A.S.C.E. Paper 3269.

Torsional shear is a maximum at the juncture of the walls and base slab and varies to zero at the top of the dome.

The torsional effect can be converted to a shear per lineal foot around the circumference of the containment by distributing the shear over the circumference of the seismic reinforcing.

The seismic bars provide a more than adequate mechanism to withstand this torsional effect. The maximum stress in the bars under this loading is 17 ksi. See Figure 5.1-26.

5.1.3.12 Cathodic Protection

A complete survey and tests to determine the need for cathodic protection on Indian Point Unit 2 was made by the A. V. Smith Engineering Company of Narberth, Pennsylvania. Electrical resistivity measurements and a visual inspection of the area away from the river, where the turbine generator building, reactor building, primary auxiliary building and associated facilities are located indicated that the environment is mostly rock with areas of dry sandy clay. The electrical resistivity of the soil ranged from 3,500 to 30,000 ohm-cm with the majority of the readings being above 10,000 ohm-cm. On this basis, it was determined that cathodic protection was not required on underground facilities in areas away from the river or the containment building liner, although a protective coating on pipes was recommended to eliminate any random localized corrosion attack. An analysis of Hudson River water data, obtained from the Con Edison plant chemist, showed the electrical resistivity of the water to vary over an extremely wide range due to salt intrusion from the ocean. The range of resistivity has been from 59 to 10,000 ohm-cm with a large number of reading in the 300 ohm–cm area. This value was considered to be extremely corrosive and the following structures in the area near the river were placed under cathodic protection:

1. Circulating water lines.
2. Service water lines.
3. Bearing piles.
4. Sheet piling (earth and water side) and wing wall anchorage system.
5. Metallic structures inside intake structure (traveling screens, bar racks, circulating water pump suction, service water pump suction).

5.1.3.13 Containment - Shear Crack

The arrangement of reinforcing bars in the containment shell is such that a reinforcing bar crosses any potential crack plane. Any cracks resulting from diagonal tension caused by shearing forces will be carried by reinforcing bars, which span across the crack. Thus all shears will be carried by the reinforcing bars and none by the concrete.

The reinforcing bars are almost all continuous throughout the containment structure; however, where a bar terminates this is accomplished by means of a 180 degree hooked bar. In no case are bars simply terminated without providing means for additional anchorage.

Throughout the cylinder, the meridional reinforcing is continuous. Beyond the springline, the bars extend radially toward the center of dome. As the bars reach a 6-in. spacing, which is one-half the required spacing, alternate bars have been dropped off by means of reinforcing splice plates. The splice piece consists of a plate with two Cadweld sleeves welded on the incoming side and one sleeve welded on the outgoing side. Thus, the number of bars present is halved and the spacing is increased to the required 12-in.

This is repeated to the top of the dome where a three layered grid pattern has been used to maintain the continuity of the rebars. The bars in the grid pattern have been Cadwelded to the same type reinforcing splice plates described above, but the Cadweld is beveled to obtain the desired direction of the grid.

At the base in the area of high discontinuity stresses, additional No. 18S bars have been provided. At the point where they were no longer needed, they have been Cadwelded to a No. 11 bar, which is terminated with a 180 degree hook.

All seismic bars have been terminated in a 180 degree hook. In no case was a No. 18S bar terminated in this way since the minimum 180 degree hook could not be provided in a 4-ft 6-in. thick wall.

Radial shear reinforcing stirrups were terminated by hooking around vertical bars.

5.1.4 Containment Penetrations

5.1.4.1 General

In general, a penetration consists of a sleeve embedded in the concrete and welded to the containment liner. The weld to the liner is shrouded by a continuously pressurized channel, which is used to demonstrate the integrity of the penetration-to-liner weld joint. The pipe, electrical conductor cartridge, duct or equipment access hatch passes through the embedded sleeve and the ends of the resulting annulus are closed off, either by welded end plates, bolted flanges or a combination of these. (See Figures 5.1-27 through 5.1-31.)

Differential expansion between a sleeve and one or more hot pipes passing through it is accommodated by using a bellows type expansion joint between the outer end of the sleeve and the outer end plate, as shown on Figure 5.1-30. Pressurizing connections are provided to continuously demonstrate the integrity of the penetration assemblies.

5.1.4.2 Types of Penetrations

5.1.4.2.1 Electrical Penetrations

The electrical penetration system consists of 60 electrical penetrations including the following: 48 Crouse-Hinds, 1 Westinghouse, 10 Conax and 1 spare sleeve (below flood-up level.)

The Crouse-Hinds and Westinghouse types are identical in design (see Fig 5.1-27). This is because Westinghouse took over the Crouse-Hinds manufacturing facility and design after the original plant penetrations were purchased. The design of this type of electrical penetration utilizes a single canister that is sealed at both ends by a combination of metal and ceramic seals. Epoxy layers on both ends provide a physical support for the conductors within the penetration canister. All of the Westinghouse and Crouse-Hinds penetrations are welded to the sleeve inside containment. The entire canister assembly is constantly pressurized by the weld channel pressurization system and monitored for any leakage.

The Conax penetrations are of a modular design consisting of a stainless steel header and 18 independently mounted conductor feedthrough modules (Figure 5.1-28 and 5.1-29), which can be individually removed and relocated. The header plate and the individual feedthrough modules are the pressure-retaining boundary. This type penetration does not have a sealed canister. The conductor modules are threaded into the header plate and the header plate itself is welded to the sleeve, which goes though the containment wall. Leakage monitoring of the

Conax penetrations is accomplished by interconnecting ports machined in the header plate to each conductor feedthrough module. A small hole is provided on each conductor feedthrough module stainless steel tubular housing allowing the feedthrough module to be pressurized when the header plate's parts are pressurized. Metal compression fittings (swaging type) are used for mounting the conductor feedthrough modules to the header in a double seal manner. The individual conductors passing through the feedthrough module are surrounded by polysulfone and are sealed (swaged) at each end of the feedthrough housing. The length of the housing (feedthrough tube) is roughly 2-ft longer than the sleeve within, which the penetration is installed. Six of the Conax penetrations are welded to their sleeve outside containment and four are welded to their sleeve inside containment to accommodate differences in the sleeves into which they are welded.

Weld channel rings are used to create a double weld seal between the header plate and the containment sleeve. All of the weld joints necessary maintain containment integrity are monitored for leaks with the weld channel pressurization system.

If a minor leak should develop at any of the plant's electrical penetrations, a release from inside containment to outside should not occur since each penetration is double sealed and pressurized to maintain a positive pressure (between 49 and 55 psi), which is higher than anticipated containment accident pressures.

5.1.4.2.2 Piping Penetrations

Double barrier piping penetrations are provided for all piping passing through the containment. The pipe is centered in the embedded sleeve, which is welded to the liner. End plates are welded to the pipe at both ends of the sleeve. Several pipes may pass through the same embedded sleeve to minimize the number of penetrations required. In this case, each pipe is welded to both end plates. A connection to the penetration sleeve is provided to allow continuous pressurization of the compartment formed between the piping and the embedded sleeve. . These penetrations are listed as "Hot" in Table 5.2-1. In the case of piping carrying hot fluid, the pipe is insulated and cooling is provided to reduce the concrete temperature adjoining the embedded sleeve. Local areas are allowed to have increased temperatures not to exceed 250ºF. Cooling is provided for hot penetrations through the use of air-to-air heat exchangers. These are made in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII, by welding together one flat sheet and one embossed sheet of 10 gauge carbon steel material, the embossment forming coolant passages. The unit is rolled into the form of a cylinder with an outside diameter slightly smaller than the respective inside diameter of the penetration sleeve. The exchanger is placed inside the sleeve and outside the pipe insulation, with the inlet and outlet coolant connections penetrating the sleeve between the outside concrete wall surface and the bellows expansion joint.

The coolant to be used is ambient air fed by a rotary blower, which is backed up with a full sized spare. The isolation features and criteria for piping penetrations are given in Chapter 6. Figure 5.1-30 shows typical hot and cold pipe penetrations.

A total of 107 pipes pass through 53 penetration sleeves, 23 of which are considered thermally hot. In addition, two spare penetration sleeves (capped and pressurized) are available for the possible future addition of piping.

All piping penetrations are designed for normal loads within the stress limits of the ASME Code, Section VIII.

All piping penetrations except main steam and feedwater are designed as anchors for the pipes passing through them and will transmit piping loads to the reinforced concrete wall. The anchorage strength exceeds the maximum combined forces imposed by the effects on the piping penetration of dead load, loads induced from a loss of coolant accident, thermal expansion of the pipe, penetration air pressure, and earthquake loads. The piping penetrations are designed to transmit the above combined loadings to the concrete structure without exceeding the yield strength of penetration steel.

In addition, each piping penetration is designed to withstand, within emergency load criteria, the effect of the rupture of a pipe passing through that penetration at or near the penetration.

The main steam and feedwater penetrations are designed so that the pipes themselves are effectively enclosed for blowdown just inside and just outside the wall. These anchors are designed to prevent a main steam or feedwater pipe rupture from causing a breach of containment at the penetrations. The anchors are designed to 90-percent of yield strength.

All piping penetrating the containment is designed to meet the requirements of USAS B31.1 (1955) Power Piping Code.

Pipes that penetrate the containment building wall and that are subject to machinery-originated vibratory loadings, such as from the reactor coolant pumps, have their supports spaced in such a manner that the natural frequency of the piping system immediately adjacent to the penetrations is greater than the dominant frequencies of the pump. Pipeline vibration was checked during preliminary plant operation and where necessary, vibration dampers were fitted. This checking and fitting effectively eliminates vibrating loads as a design consideration.

5.1.4.2.3 Equipment and Personnel Access Hatches

An equipment hatch has been provided. It is fabricated from welded steel and furnished with a double-gasketed flange and a bolted, dished door. The hatch barrel is embedded in the containment wall and welded to the liner. Provision is made to continuously pressurize the space between the double gaskets of the door flanges, and the weld seam channels at the liner joint, hatch flanges and dished door. Pressure is relieved from the double gasket spaces prior to opening the joints. The personnel hatch is a double door, mechanically latched, welded steel assembly. A quick-acting type equalizing valve connects the personnel hatch with the interior of the containment vessel for the purposes of equalizing pressure in the two systems when entering or leaving the containment. Two spring-loaded check valves in series are installed to allow pressure relief inside the air locks to the containment interior. The 16-ft diameter equipment hatch opening and the 8-ft 6-in. diameter personnel hatch are the only openings, which require special design consideration. The personnel hatch doors are interlocked to prevent both being opened simultaneously and to ensure that one door is completely closed before the opposite door can be opened.

Remote indicating lights and annunciators situated in the control room indicate the door operational status. An emergency lighting and communications system operating from an external emergency supply is provided in the lock interior. Emergency access to either the inner door, from the containment interior, or the outer door, from outside, is possible by the use special door unlatching tools. The design is in accordance with Section VIII of the ASME Code.

During refueling, the equipment hatch may be replaced by a temporary closure plate, which provides penetrations for temporary services. This enables maintenance activities to be performed while maintaining adequate containment integrity during fuel assembly movement. This plate is seismically designed and can withstand 3 psi differential pressure. During non-outage periods, this temporary closure plate may be stored within the Containment.

During refueling, a roll-up door may be substituted for the temporary closure plate. The design basis Fuel Handling Accident does not credit accident mitigation via Containment isolation subsequent to a Fuel Handling Accident. However, the roll-up door serves as a mechanism that will support rapid closure of Containment in the event a radiation release occurs during fuel handling. Because the roll-up door is not considered airtight, it may only be used under specific circumstances: either the reactor cavity water level is at least 14 feet above the reactor vessel flange, or the core configuration must consist of at least 72 unirradiated fuel assemblies (reload core) with the Reactor Coolant System elevation >66 feet (i.e., **not** in reduced inventory). Under these circumstances, if RHR cooling should be lost completely, the operators will have approximately 30 minutes or more to restore cooling prior to the onset of boiling in the reactor vessel.

5.1.4.2.4 Special Penetrations

1. Fuel Transfer Penetration - A fuel transfer penetration is provided for fuel movement between the refueling transfer canal in the reactor containment and the spent fuel pit. The penetration consists of a 20-in. stainless steel pipe installed inside a 24-in. pipe. The inner pipe acts as the transfer tube and is fitted with a pressurized double-gasketed blind flange in the refueling canal and a standard gate valve in the spent fuel pit. This arrangement prevents leakage through the transfer tube in the event of an accident. The outer pipe is welded to the containment liner and provision is made by use of a special seal ring for pressurizing all welds essential to the integrity of the penetration during plant operation. Bellows expansion joints are provided on the pipes to compensate for any differential movement between the two pipes or other structures. Figure 5.1-31 shows a sketch of the fuel transfer tube.

2. Containment Supply and Exhaust Purge Ducts - The ventilation system purge ducts are each equipped with two quick-acting, tight-sealing valves (one inside and one outside of the containment) to be used for isolation purposes. The valves are manually opened for containment purging, but are automatically closed upon receipt of a safety injection signal or high-containment radiation signal. The space between the valves is pressurized above design pressure while the valves are normally closed during plant operation. See Section 5.3, Containment Ventilation System, and Section 6.4, Containment Air Recirculation Cooling System.

   Seismic Class I debris screens inside the primary containment protect the primary containment isolation valves in the containment purge and pressure relief exhaust ducts from debris that may inhibit their correct operation. The screens are stainless-steel wire mesh and are mounted over the exhaust ducts.

   Two solenoid-controlled, pneumatically operated butterfly valves are provided for each purge penetration, one on each side of the containment building wall. Two penetrations, one supply and one exhaust, are required. Valves are spring-loaded to fail closed.

The space between the valves is pressurized from the pressurization system through an electrically operated three-way solenoid valve. This pressure is maintained only when valves are closed and must be relieved before butterfly valves can be opened. Failure to release this pressure will prevent the inside containment valves from opening. By procedure the outside containment valves are opened after the inside containment valves are open.

Failure of any of the valves to open will prevent the containment building purge supply fan from running. Tripping of the containment building purge supply fan will automatically close the inside containment butterfly valves. By procedure the outside containment butterfly valves must then be closed. When these valves are closed the space between the valves is automatically pressurized. Failure of any valves to close will prevent the adjacent space from being pressurized and will sound the loss of pressurization alarm. Loss of pressure for either zone will be displayed by individual indicating lights at the main control board.

The valve control solenoids for the inside containment isolation valves FCV-1170 and FCV-1172 and pressurization solenoids are controlled from a single control switch on the fan room control panel. The valve control solenoids for outside containment isolation valves FCV-1171 and FCV-1173 are controlled from a switch in the control room. The cycle is initiated by setting the fan room control switch to the "open" position. This will energize the pressurization alarm.

When the pressure between the valves has been relieved, the valve control solenoids for the inside containment isolation valves are energized and these two valves are opened. If for any reason, either of the two inside containment isolation valves fail to open within a given time after the cycle is initiated, both of these valves will close and pressure will be restored. The circuit is interlocked to prevent inadvertent opening of the valves during a safety injection condition.

Once the inside containment purge valves have been opened, the operator has a predetermined time to place the control switch for the outside containment purge valves to the "open" position and once opened to start the purge supply fan. Failure to do so will cause the inside containment purge valves to close.

Position indicating lights for each of the four valves are provided on the fan room control panel and the main control board.

3. <u>Sump Penetrations</u> - The piping penetration in the containment sump area is not of the typical sleeve-to-liner design. In this case, the pipe is welded directly to the base liner. The weld to the liner is shrouded by a test channel, which is used to demonstrate the integrity of the liner.

5.1.4.3 Design of Containment Penetrations

5.1.4.3.1 Criteria

The liner is basically not a load-carrying member. Because it is subjected to strains imposed by the reinforced concrete, the liner has been reinforced at each penetration in accordance with the ASME Code Section VIII. The weldments of liner to penetration sleeve are of sufficient strength to accommodate stress concentrations and adhere strictly to ASME Code Section VIII requirements for both type and strength. The penetration sleeves and plates are designed to accommodate all loads imposed on them under operating conditions (thermal effects and internal penetrations and test pressures) and accident conditions (loads resulting from all strains, internal pressures, and seismic movements). All reinforcing bars except stirrups and facing bars that are not counted on to carry any load are continuous around the openings.

Liner stress is imposed on the cylindrical penetration as a circular uniform load acting around the circumference of the penetration. The liner plate is locally thickened at the penetrations to take care of additional stresses.

5.1.4.3.2 Materials

The materials for penetrations, including the personnel and equipment access hatches, together with the mechanical and electrical penetrations, are carbon steel, conforming with the requirements of the ASME Nuclear Vessels Code and exhibiting ductility and welding characteristics compatible with the main liner material. As required by the Nuclear Vessels Code, the penetration materials meet the necessary Charpy V-notch impact values at a temperature 30°F below lowest service metal temperature, which is 50°F within containment and -5°F outside the containment.

The stainless steel bellows of the hot penetration expansion joints were protected from damage in transit and during construction by sheet metal covers fastened in place at the fabricator's shop.

1. Piping Penetrations: Materials

Piping Penetration Material	Specification
Penetration Sleeve - 12-in. dia. and under	ASTM-A333, Gr. 1
Over 12-in. dia.	ASTM-A201, Gr. B
(see exception below)	normalized to A300 CL. 1, Firebox
- 22-in. dia. containment	ASTM-A53, Gr. B sump suction
- Rolled shapes	ASTM-A36, A131, Gr. C

2. Electrical Penetrations: Materials

The penetration sleeves to accommodate the electrical penetration assembly cartridges are schedule 80 carbon steel in accordance with ASTM-A333, Gr. 1, except where otherwise noted. The electrical cartridges have been secured to the penetration sleeve so that all possible leak paths between the cartridge and sleeve will be blocked by a pressurized zone.

3. Access Penetrations: Materials

The equipment and personnel access hatch material is as follows:

Item	Material Specification
Equipment hatch insert	ASTM A516, Gr. 60 normalized to ASTM A300, CL. 1, Firebox
Equipment hatch flanges	ASTM A516, Gr. 60 normalized to ASTM A300, CL. 1, Firebox
Equipment hatch head	ASTM A516, Gr. 60 normalized to ASTM A300, CL. 1, Firebox
Personnel hatch	ASTM A516, Gr. 60 normalized to ASTM A300, CL. 1, Firebox

5.1.4.4 Leak Testing of Penetration Assemblies

A preoperational proof test was applied to each penetration by pressurizing the necessary areas to 54 psig. This pressure was maintained for a sufficient time to allow soap bubble and Freon sniff tests of all welds and mating surfaces. Any leaks found were repaired and retested; this procedure was repeated until no leaks existed.

5.1.4.5 Construction

The qualification of welding procedures and welders has been in accordance with Section IX, "Welding Qualifications," of the ASME Boiler and Pressure Vessel Code. The repair of defective welds has been in accordance with Paragraph UW-38 of Section VIII, "Unfired Pressure Vessels."

5.1.4.6 Testability of Penetrations and Weld Seams

All penetrations, the personnel air lock, and the equipment hatches are designed with double seals, which will be normally pressurized at or above the containment design pressure. Individual testing at 115-percent of containment design pressure is also possible.

The containment ventilation purge ducts are equipped with double isolation valves and the space between the valves is permanently piped into the penetration pressurization system. The space can be pressurized to 115-percent of design pressure when the isolation valves are closed. The purge valves fail in the closed position upon loss of power (electric or air).

All welded joints in the liner have steel channels welded over them on the inside of the vessel. During construction, the channel welds were tested by means of pressurizing sections with Freon gas and checking for leaks by means of a Freon sniffer. These welds were also then continuously pressurized at 50 psig.

5.1.4.7 Accessibility Criteria

Limited access to the containment through personnel air locks is possible with the reactor at power or with the primary system at design pressure and temperature at hot shutdown. After shutdown, the containment vessel is purged to reduce the concentration of radioactive gases and airborne particulates. This purge system has been designed to reduce the radioactivity level to doses defined by 10 CFR Part 20 for a 40-hr occupational work week within 2 to 6 hr after plant shutdown. Since negligible fuel defects are expected for this reactor, much less than the 1-percent fuel rod defects used for design and purging of the containment is normally accomplished in less than 2 hr. To ensure removal of particulate matter and radioactive gases, the purge air is passed through a high efficiency and charcoal filters before being released to the atmosphere through the purge vent. The primary reactor shield has been designed so that access to the primary equipment is limited by the activity of the primary system equipment and not the reactor.

5.1.4.8 Penetration Design – Computations

The penetration sleeves and end plates are designed to accommodate all loads imposed on them. The sleeve and end plate loads include the effects of internal pressure; concentrated loads imposed by the sleeve anchors on the concrete as the anchors strain in conjunction with wall movement under both operating and accident conditions; thermal effects due to both gradient and thermal reactions of the particular item passing through the sleeve; shear, bending, and compression due to accident end pressures; and shear and bending due to seismic movements of the particular item passing through the penetration. The sleeve and expansion joint are designed to remain within ASME Code Section VIII stress limitations with small strains under all or any combinations of loadings mentioned above.

For design computations of penetrations and the shell adjacent to them, see Figures 5.1-32 and 5.1-33. In Section 5.1.4.8.1, the formula for radial deformation of a hole in a plate subjected to biaxial stresses is determined by performing an integration of the tangential strains around the periphery of the hole.

In Section 5.1.4.8.2, the relationship between the deflection determined from above to the final plate and penetration sleeve deformations is developed and the formulas for stress in the liner and the stress in the penetration sleeve are developed.

Section 5.1.4.8.3 shows a summary of the liner and penetration stresses and states the assumptions made in the analysis.

In addition, thermal loads have been investigated for their effect on the shell adjacent to the penetration sleeve and found to be insignificant (38 psi bearing stress on the concrete is the maximum stress on the concrete shell).

5.1.4.8.1 Radial Deformation of a Hole in a Plate

From Reference 17, page 81

$$\sigma o = S - 2 S \cos 2\theta + [S' - 2S' \cos(2\theta - \pi)]$$

where

S = Horizontal stress in liner
S' = Vertical stress in liner

$$\delta D = \frac{1}{E}\int_{o}^{\pi} \left(S - 2S\cos 2\theta + \left[S' - 2S'\cos\left(2\theta - \pi\right)\right]\right) r\sin\theta d\theta$$

$$\delta D = \frac{r}{E}\left[\int_{o}^{\pi} S\sin\theta d\theta - 2S\int_{o}^{\pi}\cos 2\theta\sin\theta d\theta + S'\int_{o}^{\pi}\sin\theta d\theta - 2S'\int_{o}^{\pi}\cos\left(2\theta - \pi\right)\sin\theta d\theta\right]$$

$$\int\cos\left(2\theta - \pi\right)\sin\theta d\theta = -\int\cos 2\theta\sin\theta d\theta$$

$$= -\int\left(1 - 2\sin^2\theta\right)\left(\sin\theta\right)d\theta$$

$$= -\int\left(\sin\theta - 2\sin^3\theta\right)d\theta$$

$$= -\left[\left(-\cos\theta\right) - 2\frac{\sin^2\theta\cos\theta}{3} + \frac{2}{3}\int\sin\theta d\theta)\right]$$

$$= -\left(-\cos\theta + 2/3\sin^2\theta\cos\theta + \frac{4}{3}\cos\theta\right)$$

therefore

$$\int\cos(2\theta - \pi)\sin\theta d\theta = \frac{-\cos\theta}{3} - \frac{2}{3}\sin^2\theta\cos\theta$$

$$\delta = \frac{r}{E}\left[-S\cos\theta - 2S\left(\frac{\cos\theta}{3} + \frac{2}{3}\sin^2\theta\cos\theta\right) - S'\cos\theta - 2S'\left(\frac{-\cos\theta}{3} - \frac{2}{3}\sin^2\theta\cos\theta\right)\right]_{o}^{\pi}$$

$$\delta = \frac{r}{E}\left[\left(S + \frac{2}{3}S + S' - \frac{2}{3}S'\right) - \left(-S - \frac{2}{3}S - S' + \frac{2}{3}S'\right)\right]$$

$$\delta = \frac{r}{E}\left[2S + \frac{4}{3}S + 2S' - \frac{4}{3}S'\right]$$

$$\delta = \frac{r}{E}\left[\ \frac{10}{3}\ S + \frac{2}{3}\ S'\right]$$

$\delta = \frac{2}{3}\frac{r}{E}\ [\ 5S + S']$ (for stresses in the same direction)

$\delta = \frac{2}{3}\frac{r}{E}\ [\ 5S - S']$ (for stresses in the opposite direction)

5.1.4.8.2 Plate and Sleeve Deformation

$$\Delta_{UN} = \Delta_{Pl\,Res.} + \Delta_{Sleeve}$$

$$\Delta_{UN} = \frac{S_1}{E}(1-\nu)R + \frac{S_l(t_{pl})R^2\lambda}{2Et_{sleeve}}^{*}$$

$$\Delta_{UN} = \frac{S_1}{E}\left[R(1-\nu) + \frac{t_{pl}\;R^2\lambda}{2t_{sleeve}}\right]$$

$$S_1 = \frac{\Delta_{UN}E}{R\left[(1-\nu) + \frac{t_{pl}\,R\lambda}{2t_{sleeve}}\right]}$$

$$S_{sleeve} = \frac{S_1(t_{pl})\,R\lambda}{2t_{sleeve}}$$

$$S_{sleeve} = \frac{\Delta_{UN}E\;t_{p1}\;R\lambda}{R\left[(1-\nu) + \frac{t_{p1}\,R\lambda}{2t_{sleeve}}\right]2t_{sleeve}}$$

where:

$$\lambda = \left[\frac{3(1-\nu^2)}{R^2t^2_{sleeve}}\right]^{\frac{1}{4}}$$

*S_1 = Stress in Liner
t_{pl} = plate thickness, in.

R = radius, in.
ν = poisson ratio
E = modulus of elasticity

5.1.4.8.3 Summary

Penetration	Stress in Sleeve (ksi)	Stress in Liner (ksi)
Air purge	-23.8	-19.5
Main steam	-33.4	-27.94
Typical mech. Penetration	-31.0	-31.1
Electrical penetration		
A) C and T[1]	-22.5	-29.5
B) T and T[1]	+18.2	+19.7
Fuel transfer		
A) C and T[1]	-25.7	-25.6
B) T and T[1]	+20.8	+16.6

[*Note – 1. First letter represents direction of vertical liner stress; second letter represents direction of horizontal liner stress, C signifies compression and T signifies tension.*]

A) Ignores effects of insulation in the vertical direction.

B) Considers effects of insulation.

Conservative Assumptions

1. The weld pressurization channel stiffens the area.

2. The liner alone was designed for stress concentration effects while the cracked concrete was ignored.

3. The unrestrained growth is based on maximum growth from a stress concentration consideration.

4. The main steam and mechanical penetrations have been considered in a noninsulated zone when they are just inside the insulated zone. The compression in the hoop direction will be greatly reduced or perhaps go into tension, thus reducing the stresses.

5. The allowable stress in the sleeve is 56,700 psi except for the stainless steel fuel transfer penetration, which is 49,500 psi. These values come from Table N-421 and Figure N-414 of the ASME Nuclear Vessel Code Section III.

5.1.5 Primary System Supports

In 1989, the NRC approved changes to the design bases with respect to dynamic effects of postulated primary loop pipe ruptures, as discussed in Section 4.1.2.4.

In 2000, an analysis (Reference 19) of the reactor coolant loop and its component supports, which incorporates the NRC approved changes, was performed to reflect the replacement steam generator and removal of sixteen of the original twenty-four steam generator support frame hydraulic snubbers. The analysis also reflected the de-activation of the original horizontal and vertical pipe rupture restraints, located on the cross over legs at the steam generator end. Based on this revised analysis, it was concluded that the Unit 2 reactor coolant system can withstand the combination of blowdown and seismic loads within acceptable stress limits. By reducing the number of snubber and de-activating the rupture restraints the extent of maintenance, inspection and testing requirements is reduced and the reliability of the Reactor Coolant System is enhanced by reducing the possibility of equipment malfunction. In 2003, the reactor coolant loop and its component supports were re-analzed due to a power uprate. This latest analysis does not consider the coincident combination of blowdown and seismic loads.

The original design basis is described in the following paragraphs.

The primary system supports, steam generator, reactor coolant pump, pressurizer, and reactor vessel were designed to withstand pipe break or seismic acceleration based on the following:

1. The break is either a circumferential or longitudinal pipe rupture of area equivalent to the pipe cross section occurring anywhere in the system piping. The longitudinal rupture occurs at any point 360 degrees around the pipe. The support system is designed to withstand the steady thrust equivalent to the product of system operating pressure and pipe rupture area without exceeding yield stress in the support members. The stress limits on the vessels and piping are tabulated in Section 1.11. The component supports prevent rupture of reactor coolant piping in the remaining intact loops which could result from an assumed rupture in any one loop, thereby ensuring that the path for safety injection flow to the core is available. Additionally, the supports are designed to prevent secondary piping rupture as a result of rupture in the primary loop and vice versa.

2. The nuclear steam supply system and its support system are designed such that the nuclear steam supply system is capable of continued safe operation for the combination of normal loads and the design earthquake loading. The equipment and supports operate within normal design limits for the design earthquake. The system and its supports are also designed to withstand the maximum potential earthquake without loss of function. The seismic response curves for both the design and maximum potential earthquake and the stress limits are presented in Section 1.11. Component loads are obtained from the curve using the appropriate period and damping.

3. The primary system supports were not originally designed to resist combined seismic and accident loads. They were designed as statically uncoupled component supports.

   A complete reactor coolant system loop, including the steam generator and the reactor coolant pump supports, has been analyzed for combined dead, seismic, and blowdown loads. Stresses were determined by means of the three-dimensional frame computer program, STRUDL. The dead load assumed is the flooded weight of the component. The seismic load considered is 0.6g horizontal acceleration times the flooded mass of the component at the center of gravity of the component acting in the N-S, E-W, NW-SE and SW-NE directions analyzed separately. The horizontal earthquake component acting on the steam generator is assumed to be carried by the upper steam generator. The vertical component of earthquake is assumed equal to 0.4g acting simultaneously with the horizontal load at the center of gravity of both the pump and steam generator. The system was analyzed for each separate accident or pipe rupture resulting in a jet load equal to 1500 kips as shown in Figure 5.1-34.

The combined dead plus seismic plus accident maximum resultant member axial stress and axial plus bending stress (in parentheses) for the steam generator and pump supports is shown in Figures 5.1-35 through 5.1-42 (stresses are expressed in ksi.) The section views of the support shown can be identified by the isometric views of the pump and steam generator supports shown in Figures 5.1-43 and 5.1-44. Negative values indicate compression and positive values indicate tensile stress. Since response of the primary systems is elastic, deformations are very small and were not considered as design parameters required to verify the design adequacy of the supports.

It should be noted the stresses shown are not for a particular combined blowdown or seismic load case but rather the worst combination for a given member; hence, the values shown are upper limits for each member and could not in fact actually occur in the combination shown. It should also be noted that the primary support structures are designed as trusses rather than frames hence the bending stresses indicated are secondary in nature.

5.1.5.1 Steam Generator

The steam generators are supported within a caged structural system, consisting of four connected trusses, all welded together, fabricated of carbon steel members, with provisions for limited movement of the structure in a horizontal direction to accommodate piping expansion with a system of "Lubrite" plates, hydraulic snubbers, guides, and stops. The "Lubrite" plates, hydraulic snubbers, guides, and stops were originally designed as a rigid support to resist the action of seismic and pipe break loads. In 2000, the number of hydraulic snubbers supporting the steam generator frame in the direction of the hot leg, has been reduced from the original six down to two per steam generator. The two remaining snubbers are located at the upper support point of the frame at Elevation 92'-0". The analysis of the reactor coolant loop and of the steam generator support structure accounts for the replacement steam generator and for the reduced number of hydraulic snubbers. The following are loading conditions that the structure was originally designed to resist:

1. Vertical dead weight of pipe and vessel, flooded = 1,000 kips

2. Seismic loads:

    a. Horizontal load of 474 kips acting at the centroid of the steam generator vessel, located near the top of the support structure, which is directly transferred to the hydraulic snubbers, guides, and stops, and in turn to the bottom of the 2-ft thick concrete operating floor slab at elevation 93-ft.

    b. Vertical load of 320 kips transferred as axial load to the base plates and anchor bolts at elevation 46-ft.

3. Primary system - longitudinal pipe rupture:

   a. Reaction at the nozzle of the steam generator from the pipe between the reactor and the steam generator elbow, produces a force of 1090 kips in any direction and an overturning moment or torsional moment of (1090 kips x 4.25-ft) 4632-ft-kips. Overturning and torsional moments are resisted by the support system at elevation 46-ft and horizontal forces are distributed, through the truss action, to elevations 46-ft and 93-ft.

   b. Reactions at the nozzle of the steam generator from the pipe between the steam generator elbow and reactor coolant pump elbow, produces a force of 850 kips in any direction and a torsional moment or overturning moment of (850 kips x 5.0-ft) 4250-ft-kips. Overturning and torsional moments are resisted by the support system at elevation 46-ft, and horizontal forces are distributed, through the truss action, to elevations 46-ft and 93-ft.

4 Primary system - circumferential break:

   a. Reactions at the nozzle of the steam generator from the pipe between the reactor and steam generator produces a horizontal force of 1490 kips. This force is transferred through the vessel support to the two vertical trusses of the structural system, which in turn, transmits it as horizontal reactions at the slabs at elevations 46-ft and 93-ft. The moment produced by this force is (1490 kips x 2-ft) 2980-ft-kips and is less than the dead load resisting moment (500 kips x 10-ft) 5000-ft-kips, and the vertical forces at elevation 46-ft are all compressive, no uplift.

   b. Reactions at the nozzle of the steam generator from a pipe between the steam generator and the reactor coolant pump produces a horizontal force of 1700 kips plus an overturning moment of (1700 kips x 4.25-ft) 7225-ft-kips, or a vertical force of 1700 kips and an overturning moment of (1700 kips x 5.33-ft) 9061-ft-kips. The horizontal force and moments are transferred to the structural system and the reactions are resisted at the slabs at elevations 46-ft and 93-ft, or the vertical force and moment are resisted at elevation 46 ft.

5. Secondary system - longitudinal rupture in steam pipe:
   Reactions at the nozzle of the steam generator from the steam pipe longitudinal rupture at the top of the vessel produce:

   a. Horizontal force of 600 kips and a torsional moment of 2400-ft-kips. Horizontal force is transferred through the vessel to the structural support system, which in turn transmits it as horizontal reactions to the slabs at elevations 46-ft and 93-ft. The torsional moment is transferred through the vessel to the structural system, which in turn, transmits it to the base at elevation 46-ft.

   b. Vertical upward or downward force of 600 kips and an overturning moment of 2400-ft-kips. Upward forces are overcome by the operating weight of the steam generator. Downward force is added to the operating weight and transferred to the base at elevation 46-ft. Overturning moment is transferred through the vessel supports to the structural system, which in turn, transmits it as vertical reactions at the base, elevation 46-ft.

6. Secondary system - circumferential break in steam pipe:

   Reaction at the nozzle of the steam generator from the steam pipe guillotine break at the top of the vessel produces a horizontal force of 600 kips. This force is transferred through the vessel to the structural system, which in turn transmits it as horizontal reactions of 1085 kips at elevation 93-ft and 485 kips at elevation 46-ft.

7. Secondary system - feedwater pipe breaks:

   The reactions from circumferential and longitudinal pipe breaks in the feedwater system are resisted in a manner similar to steam pipe breaks listed under preceding sections (5) and (6), but are much smaller in magnitude. Maximum longitudinal 1600-ft-kips, maximum circumferential 200 kips.

5.1.5.2 Reactor Coolant Pump

The reactor coolant pump is supported on a three-legged structural system consisting of three connected trusses fabricated of carbon steel members, structural sections and pipe, supported from elevation 48-ft-6-in. Provisions for limited movement of the structure in any horizontal direction to accommodate piping expansion is accomplished with a sliding "Lubrite" base plate arrangement and a system of tie rods and anchor bolts that restrain the structure from movement beyond the calculated limits. To improve the ability of the reactor coolant pumps to meet combined LOCA and seismic loads, two of the reactor coolant pump holddown bolts have been replaced with higher strength ASTM A540 steel bolts.

The following are loading conditions that the structure was originally designed to resist:

1. Vertical dead weight of pipe and pump flooded = 206 kips.

2. Seismic:

   a. Horizontal load of approximately 117 kips acting at the centroid of the pump assembly, which is transferred by the structural system and piping to the tie rods and base of the supporting structure at elevation 48-ft-6-in. This load includes the seismic effect of the support self-weight.

   b. Vertical seismic load of approximately 78 kips transferred directly as axial load to the base plates and anchor bolts. This load includes the seismic effect of the support self-weight.

3. Primary system - longitudinal rupture:

   a. Reaction at the nozzle of the pump from a pipe break in the pipe between the steam generator elbow and pump elbow produces a torsional moment of 3825-ft-kips, together with a horizontal force of 850 kips or an overturning moment of 3825-ft-kips, together with a vertical up or down force of 850 kips.

Torsional forces are resisted by the structural stability of the primary piping connected to the pump.

Reactions from horizontal forces are resisted by the tie rods connected to the steam generator and reactor support structures. Forces caused by an overturning moment are resolved into horizontal and vertical components, which are resisted by tension in the anchor bolts, axial load on the foundations, and tension in the tie rods.

b. Reaction at the nozzle of the pump from a pipe break in the pipe between the pump and the reactor, produces a torsional moment of 6880-ft-kips, together with a horizontal force of 1165 kips, or an overturning moment of 6880-ft-kips, together with a vertical up or down force of 1165 kips.

Torsional forces are resisted by the structural stability of the primary piping connected to the pump. Reactions from the horizontal forces are resisted by the tie rods connected to the walls.

Forces caused by an overturning moment are resolved into horizontal and vertical components, which are resisted by:

(1) Tension in the anchor bolts.
(2) Axial load on the foundations.
(3) Tension in the tie rods.

4. Primary system - circumferential break:

a. Reactions at the nozzle of the pump from a pipe break in the pipe between the steam generator and pump produces a horizontal force on the structure of 1700 kips. This force is resisted directly by the bumper located against the elbow of the pipe. Components of the force are then transferred to the base of the structure and the tie rods connecting the pump support to the steam generator support system.

b. Reactions at the nozzle of the pump from a pipe break in the pipe between the pump and the reactor produces a torsional moment of 3240-ft-kips and a horizontal force of 1340 kips on the structure.

Torsional forces are resisted by the structural stability of the remaining primary piping connected to the pump.

Reactions from the horizontal forces are resisted by tie rods connected to the walls.

5.1.5.3 Pressurizer

Pressurizer is supported on a free-standing structural system, consisting of six connected trusses fabricated of carbon steel members, all welded together and secured at the base by anchor bolts at elevation 46-ft.

The following are loading conditions that the structure has been designed to resist:

1. Vertical dead weight of pipe and vessel flooded is 360 kips. The self-weight of the support is 21 kips.

2. Seismic:

   a. Horizontal seismic load of 174 kips acting at the centroid of the pressurizer vessel, which coincides in elevation with the slab at elevation 95-ft, is directly transferred through the concrete embedded guides to the slab. This load excludes the seismic effect of the support self-weight.

   b. Vertical seismic load of 123 kips transferred through the structural system as axial forces to the base plates and anchor bolts at elevation 46-ft. This load excludes the seismic effect of the support self-weight.

3. Longitudinal pipe rupture

   a. Reaction at the surge pipe nozzle of the pressurizer produces either a torsional moment of 734-ft-kips and a horizontal force of 234 kips or an overturning moment of 734-ft-kips and a horizontal or vertical force of 234 kips.

      These moments and forces are resisted by the structural system and transferred to the base at elevation 46-ft.

4. Circumferential pipe break:

   a. Reaction at the surge pipe nozzle of the pressurizer produces a horizontal force of 234 kips and an overturning moment of 734-ft-kips.

      These moments and forces are resisted by the structural system and transferred to the base at elevation 46-ft.

5.1.5.4 Reactor Vessel Support Girder

The reactor vessel is supported on four cooling plates that are fastened to the top flange of a circular box section ring girder, fabricated of carbon steel plates. The bottom flange of the girder is in continuous contact with a nonyielding concrete foundation.

In addition to the reactor vessel weight and piping reactions of the girder has been designed to support the conditions of loading for pipe break and seismic forces as outlined in Figure 5.1-45.

5.1.5.5 Reactor Vessel Rupture

The reactor pressure vessel is enclosed by a 6-ft thick circular reinforced concrete shield wall that is designed to sustain the internal pressure and provide missile protection for the containment liner in the highly unlikely failure of the reactor vessel due to a longitudinal split. All stresses will be maintained within specified minimum ultimate rebar tensile stress.

In the event of a circumferential reactor break, the 0.25-in. basemat liner plate at the bottom of the containment reactor cavity pit directly under the reactor vessel is protected by 2-ft of concrete with a 1-in. steel liner plate embedded on top of the concrete. Directly below the reactor cavity pit containment basemat liner plate, 4.5-ft of concrete is poured on rock. Refer to Figures 5.1-46 through 5.1-51.

As discussed in Section 5.1.3.7, in the event of reactor vessel failure, a pressure build up of 1000 psi and rebar stresses of 86 ksi (assuming all concrete is cracked) inside the pit due to release of reactor contents is assumed. Since the integrity of the wall is not jeopardized, the integrity of the vessel support that is supported on the wall will not be jeopardized. Deflection of the shield wall will not cause large stresses in the vessel support since a lubricated surface is provided on the shoes, allowing the vessel support to slide.

5.1.5.6 Circumferential Cracking

The worst circumferential crack location from the standpoint of downward missiles is just below the reactor coolant system piping nozzles. As the following calculations show, this missile will not violate the containment structure and liner integrity.

As a consequence of this circumferential crack, the downward missile represented by bottom vessel head has the following characteristics at the time of impact on the cavity floor:

1. Weight: 381,000 lb
2. Cross sectional area of crater: 63-ft^2
3. Downward velocity: 213-ft/sec
4. Concrete crushing strength: 4000 psi

The depth of penetration has been calculated by using the Petri formula for penetration into an infinite, thick concrete slab, as reported in Nav. Docket P-51:

$$D = K\frac{W}{A} \log_{10}\left(1 + \frac{V^2}{215,000}\right)$$

where:

D = depth of penetration, ft
K = penetration coefficient for 4000 psi concrete
W = missile weight, lb
A = missile area, ft^2
V = missile velocity, ft/sec

The following parameters have been used:

$K = 2.8 \times 10^{-3}$
W = 381,000 lb
A = 63-ft^2
V = 213 ft/sec

The result is a depth of penetration of 1.4-ft.

As mentioned above, the 0.25-in. basemat liner is covered by 2-ft of concrete with a 1-in. steel plate on top. As it can be readily seen, even neglecting the 1-in. steel plate in the penetration calculations, the containment liner will not be reached.

5.1.5.7 Longitudinal Splitting

The cavity wall is designed to withstand the forces and internal pressurization associated with a longitudinal split without gross damage. See Section 5.1.3.7 for a discussion of the analysis of this assumed accident condition.

5.1.6 Containment Structure Design Evaluation

5.1.6.1 Reliance On Interconnected Systems

The containment leakage limiting boundary is provided in the form of a single, carbon steel liner on the vessel having double barrier weld channels and penetrations. Each system whose piping penetrates this boundary is designed to maintain isolation of the containment from the outside environment. Provision is made to continuously pressurize penetrations and weld channels and to monitor leakage from this pressurization.

5.1.6.2 System Integrity and Safety Factors

Pipe Rupture - Penetration Integrity - The penetrations for the blowdown and sample lines are designed so that the penetrations are stronger than the piping system and so that the vapor barrier will not be breached due to a hypothesized pipe rupture. The pipe rupture loads for the main steam and feedwater lines are resisted by the supports located away from their penetrations and do not affect the integrity of the penetrations for these lines.

Major Component Support Structures - The support structures for the major components are designed to resist all thrust forces, moments and torques associated with either a reactor coolant system or main steam pipe break. All primary structural steel elements are designed for stresses not exceeding yield stress due to these forces.

Containment Structure Components Analyses - The details of radial, longitudinal, and horizontal shear analyses for the containment reinforced concrete are given in Section 5.1.3.

5.1.6.3 Performance Capability Margin

The containment structure is designed based upon limiting load factors, which are used as the ratio by which accident and earthquake loads are multiplied for design purposes to ensure that the load/deformation behavior of the structure is one of elastic, low strain behavior. This approach places minimum emphasis on fixed gravity loads and maximum emphasis on accident and earthquake loads. Because of the refinement of the analysis and the restrictions on construction procedures, the load factors primarily provide for a safety margin on the load assumptions. Load combinations and load factors used in the design, which provide an estimate of the margin with respect to all loads, are tabulated in Section 5.1.2.

5.1.7 Liner Insulation

Insulation is provided on approximately the first 43-ft of the containment liner to limit the temperature rise in the liner under accident conditions to 80°F above ambient and thereby avoid excessive liner compressive stress during the accident. The first 18-ft (elev. 46-ft to 64-ft, except in the piping penetration area in the southeast quadrant where the insulation rose only 16-ft to the 62-ft elevation) was installed as part of the original containment design. In 1973 an additional 25-ft (elev. 64-ft to 89-ft) was added. The first 18-ft (elev. 64-ft to 82-ft) covers the entire circumference of the liner while the upper 7-ft (elev. 82-ft to 89-ft) only covers part of the circumferential area in the north and south-southwest quadrants where the main steam and feedwater lines extend up along the crane wall. The insulation panels are attached to the steel containment liner by means of 3/16-in. diameter stainless steel studs welded to the liner on the basis of six per panel. The insulation panels are protected by stainless steel jacketing on the exposed faces and sealed at the joints. Details of the insulation installation are given in Table 5.1-2.

The insulation has been designed to meet the following operational requirements:

1. Normal operating temperature of 120°F. (The maximum normal operating temperature of the containment was changed from 120°F to 130°F by Amendment 149 to the Facility Operating License DPR-26 for IP-2 dated March 27, 1990. Evaluations performed show the insulation material used on the containment liner is adequate for use at the higher operating temperature.)

2. Under accident conditions, the rise in liner temperature not to exceed 80°F. The analyses performed to support the Stretch Power Uprate (SPU) also performed analyses of the containment liner under the most limiting conditions for liner stress and showed a temperature rise well under allowed 80°F.

3. Insulation panels to be rated nonburning in accordance with ASTM procedure D–1692.

4. To be removable by sections for inspection of the containment liner.

5.1.8 Minimum Operating Conditions (For Containment Integrity)

Containment integrity internal pressure limitations and leakage rate requirements are established in the facility Technical Specifications.

5.1.9 Containment Structure-Inspection And Testing

5.1.9.1 Initial Containment Leakage Rate Testing

Criterion: Containment shall be designed so that integrated leakage rate testing can be conducted at the peak pressure calculated to result from the design basis accident after completion and installation of all penetrations and the leakage rate shall be measured over a sufficient period of time to verify its conformance with required performance. (GDC 54)

After completion of the containment structure and installation of all penetrations and weld channels, an initial integrated leakage rate test was conducted at the containment design pressure (47 psig), maintained for a minimum of 24 hr, verifying that the leakage rate is no greater than 0.1-percent by weight of the containment volume per day at design basis accident conditions. This leakage rate test was performed using the absolute method. In addition, a reduced pressure integrated leakage rate test was conducted at a pressure not less than 50 percent of the containment design pressure and maintained for a minimum of 24 hr.

5.1.9.2 Periodic Containment Leakage Rate Testing

Criterion: The containment shall be designed so that an integrated leakage rate can be periodically determined by test during plant lifetime. (GDC 55)

The containment is tested in accordance with 10 CFR 50 Appendix J as discussed in section 5.1.12.

A leak rate test at the containment design pressure using the same method as the initial leak rate test can be performed at any time during the operational life of the plant, provided the plant is not in operation and precautions are taken to protect instruments and equipment from damage.

5.1.9.3 Provisions for Testing of Penetrations

Criterion: Provisions shall be made to the extent practical for periodically testing penetrations, which have resilient seals or expansion bellows to permit leak tightness to be demonstrated at the peak pressure calculated to result from occurrence of the design basis accident. (GDC 56)

Penetrations are designed with double seals, which are continuously pressurized above accident pressure. The large access openings such as the equipment hatch and personnel air locks are equipped with double gasketed doors and flanges with the space between the gaskets connected to the pressurization system. The system uses a supply of clean, dry, compressed air that will place the penetrations under an internal pressure above the peak calculated accident pressure.

A permanently piped monitoring system is provided to continuously measure leakage from all penetrations.

Leakage from the monitoring system is checked by continuous measurement of the integrated makeup air flow. In the event excessive leakage is discovered, each penetration can then be checked separately at any time.

5.1.9.4 Provisions for Testing of Isolation Valves

Criterion: Capability shall be provided to the extent practical for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits. (GDC 57)

Capability is provided to the extent practical for testing the functional operability of valves and associated apparatus during periods of reactor shutdown.

Initiation of containment isolation employs coincidence circuit, which allow checking of the operability and calibration of one channel at a time. Removal or bypass of one signal channel place that circuit in the half-tripped mode.

Hydrostatic tests of isolation valves in series are performed by first testing the upstream valve with the second valve open, then opening the upstream valve and closing the second valve, so that each valve will have an independent test.

The main steam and feedwater barriers and isolation valves in systems that connect to the reactor coolant system are hydrostatically tested to measure leakage.

Valves in the residual heat removal system are not considered to be isolating valves in the usual sense inasmuch as the system would be in operation under accident conditions.

Field and operational inspection and testing have been divided into three phases:

1. Construction tests; those taking place during erection of the containment building liner.
2. Preoperational tests; those taking place after the containment structure was erected and all penetrations were complete and installed.
3. Postoperational tests; monitoring during reactor operation.

5.1.10 Construction Tests

During erection of the liner, the following inspection and tests were performed.

5.1.10.1 Bottom Liner Plates

All liner plate welds are tested for leaktightness by vacuum box. The box is evacuated to at least a 5 psi pressure differential with the atmospheric pressure.

After completion of a successful leak test, the welds were covered by channels. A strength test was performed by applying 54 psig air pressure to the channels in the zone for a period of 15 min.

The zone of channel covered welds was pressurized to 47 psig with a 20-percent by weight of freon-air mixture. The entire run of the channel-to-plate welds was then traversed with a halogen leak detector.

The sensitivity of the leak detector was 1 x 10^{-9} standard cc per second. The sniffer was held approximately 0.5-in. from the weld and traversed at a rate of about 0.5-in./sec. The detection of any amount of halogen indicated a leak requiring weld repairs and retesting.

After the halogen test was completed, all liner welds not accessible for radiography were pressurized with air to 47 psig and soap-tested. Any leaks indicated by bubbles were repaired and retested. Where leaks occurred, welds were removed by arc gouging, grinding, chipping, and/or machining before rewelding. In addition, the zone of channels was held at the 47 psig air pressure for a period of at least 2 hr. The drop in pressure did not exceed the equivalent of a leakage of 0.05-percent of the containment building volume per day. Compensation for change in ambient air temperature was made.

5.1.10.2 Vertical Cylindrical Walls and Dome

For the liner, a complete radiograph was made of the first 10-ft of full penetration weld made by each welder or welding operation. A minimum of a 12-in. film "spot" radiograph was made every 50-ft of weld thereafter on the side walls and dome, except where backup plates were used. The radiograph films were reviewed by United Engineers and Constructors. When a spot radiograph showed defects that required repair, two adjacent spots were radiographed. If defects requiring repair were shown in either of these, all of the welding performed by the responsible operator or welder was 100-percent radiographed to determine the area of defect.

The performance and acceptance standards for all radiography were ASME Section VIII, Paragraph UW51.

The liner plate-to-plate welds were tested for leaktightness by vacuum box techniques. After successful completion of the spot radiography and vacuum box tests and subsequent repair of all defects, the channels were welded in place over all seam welds in a predetermined zone. A strength test was performed on the liner plate weld and the channel weld by pressurizing the channel with air at 54 psig for 15 min. In addition, each zone of channel covered weld was leak-tested using the freon-air mixture at 47 psig.

In locations where radiography was not possible, such as the lower courses of shell plates where backup plates were used, and where liner bottom welds and floor plate welds were made to angles and tees, the liner fabricator welded on a 2-in. long overrun coupon. The overrun coupon was chipped off, marked for location and given to United Engineers and Constructors for testing. These welds were also vacuum box tested.

Welded studs were visually inspected, and at least one at the beginning of each day's work and another at approximately mid-day were bend-tested to 45 degrees for each welder. Studs failing visual or bend-testing were removed.

While the liner is not a pressure vessel, industry experience has shown that leaks in pressure vessels normally occur at joints. For this reason, and following current liner fabrication practice, there was no radiographic or other nondestructive examination of liner plate.

5.1.10.3 Penetrations

Strength and leak tests of individual penetration internals and closures and sleeve weld channels were performed in a similar manner to the above and all leaks repaired and the penetration or weld channel retested until no further leaks were found. See Figures 5.1-53 through 5.1-56 for the areas of the containment and liner, which were instrumented for the strength test.

5.1.11 Preoperational Tests

All penetrations and the welds joining these penetrations to the containment liner and the liner seam welds were designed to provide a double barrier, which can be continuously pressurized at a pressure higher than the design pressure of the containment. This blocks all of these potential sources of leakage with a pressurized zone and at the same time provides a means of monitoring the leakage status of the containment, which is more sensitive to changes in the leakage characteristics of these potential leakage sources.

After the containment building was complete with liner, concrete structures, and all electrical and piping penetrations, equipment hatch and personnel locks were in place, the following tests were performed.

5.1.11.1 Strength Test

A pressure test was made on the completed building using air at 54 psig. This pressure was maintained on the building for a period of at least 1 hr. During this test, measurements and observations were made to verify the adequacy of the structural design. For a description of observations, cracks, strain gauges, etc., refer to Reference 18.

5.1.11.2 Integrated Leakage Rate Test: (Type A)

The integrated leakage rate tests were performed on the containment building at 47 psig using the absolute method. This leakage test was performed with the double penetration and weld channel zones open to the containment atmosphere. The leakage rate demonstrated by this test was equal to or less than 0.1-percent of the containment free volume per day at design basis accident conditions. After it was assured that there were no defects remaining from construction, a sensitive leak rate test was conducted.

5.1.11.3 Sensitive Leak Rate Test: (Type B)

The sensitive leak rate test included only the volume of the weld channels and double penetrations. This test was considered more sensitive than the integrated leakage rate test, as the instrumentation used permitted a direct measurement of leakage from the pressurized zones. The sensitive leak rate test was conducted with the penetrations and weld channels at 50 psig and with the containment building at atmospheric pressure. The leak rate for the double penetrations and weld channel zones was equal to or less than 0.2-percent of the containment free volume per day.

5.1.11.4 Containment Isolation Valve Test: (Type C)

These tests were conducted to detect leaks through certain containment isolation valves.

5.1.12 Postoperational Tests

Containment testing is conducted in accordance with the Technical Specifications and 10 CFR Appendix J, including integrated leakage rate tests at the containment design pressure, In 1997, the Technical Specifications were amended to allow the use of 10 CFR 50 Appendix J, Option B (as modified by approved exemptions) and NRC Regulatory Guide 1.163 dated September 1995 for integrated leakage rate tests, air lock tests, and containment isolation valve operability tests.

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TABLE 5.1-1
Flooded Weights - Containment Building

Item		Flooded Operating Weight, lbs
Pressurizer -1		346,000
Steam generators - 4		3,746,000
Reactor - 1		
(a)	Vessel	868,000
(b)	Internals	420,000
(c)	Piping	1,000,000
Reactor pumps - 4		824,000
Accumulator tanks - 4		529,000
175-ton polar crane - 1		650,000
Ventilation fans - 5		656,000
Reactor coolant drain tank - 1		20,000
Pressure relief tank - 1		100,000
Other miscellaneous equipment		100,000
	Total	9,259,000

TABLE 5.1-2
Containment Liner Insulation Properties

1. Elevation 46-ft to 64-ft liner insulation:

- 1-1/4-in. polyvinylchloride insulation, Vinylcel, as manufactured by Johns-Mansville and 1-1/2 in. Pittsburgh Corning Foamglass Insulation.
- 0.019-in. thick stainless steel jacket (exposed side) except for areas using Pittsburgh Corning Foamglass Insulation in which a jacket thickness of 0.024" is used.
- Insulation adhesive is Johns-Manville Dutch Brand FN12 or an approved equal.

2. Elevation 64-ft to 89-ft liner insulation:[1]

- 1.5-in. thick FOAMGLAS® with density of 8.5 to 9 lb/ft^3, as manufactured by Pittsburgh Corning Corporation. This insulation has a thermal conductivity of 0.5 - 0.525 BTU-in/hr-ft^2-°F and a specific heat (Cp) of 0.18 BTU/lb-°F.
- 1/16-in. commercial grade pure asbestos paper backing adjacent to the liner plate on the unexposed face.
- The adhesive bonding the FOAMGLAS® to the asbestos paper is Cadoprene No. 434 and bonding the stainless steel jacket to the FOAMGLAS® is Cadoseal No. 700 by Epolux Manufacturing Corporation.

Note:
[1] Insulation from Elevation 82-ft to 89-ft only covers part of the circumferential area in the north and south-southwest quadrants.

5.1 FIGURES

	Test Locations
Figure 5.1-55	Containment Strain Gauge Test Locations
Figure 5.1-56	Containment Proof Test Gross Deformation Measurements

5.2 CONTAINMENT ISOLATION SYSTEM

5.2.1 Design Basis

Each system whose piping penetrates the containment leakage limiting boundary is designed to maintain or establish isolation of the containment from the outside environment under the following postulated conditions:

1. Any accident for which isolation is required (severely faulted conditions) with

2. A coincident independent single failure or malfunction (expected fault condition) occurring in any active system component within the isolated bounds.

Piping penetrating the containment is designed for pressures at least equal to the containment design pressure. Containment isolation valves are provided as necessary in lines penetrating the containment to ensure that no unrestricted release of radioactivity can occur. Such releases might be due to rupture of a line within the containment concurrent with a loss-of-coolant accident, or due to rupture of a line outside the containment that connects to a source of radioactive fluid within the containment.

In general, isolation of a line outside the containment protects against rupture of the line inside concurrent with a loss-of-coolant accident, or closes off a line, which communicates with the containment atmosphere in the event of a loss-of-coolant accident.

Isolation of a line inside the containment prevents flow from the reactor coolant system or any other large source of radioactive fluid in the event that a piping rupture outside the containment occurs. A piping rupture outside the containment at the same time as a loss-of-coolant accident is not considered credible, as the penetrating lines are seismic Class I design up to and including the second isolation barrier and are assumed to be an extension of containment.

The isolation valve arrangement provides two barriers between the reactor coolant system or containment atmosphere and the environment.

System design is such that failure of one valve to close will not prevent isolation, and no manual operation is required for immediate isolation. Automatic isolation is initiated by a containment ventilation isolation signal, a Phase A isolation signal ("T" signal), a Phase B isolation signal ("P" signal), or manually. See Section 5.2.4 or Chapter 7.0 for further details.

The containment isolation valves have been examined to ensure that they are capable of withstanding the maximum potential seismic loads.

To ensure their adequacy in this respect:

1. Valves are located in a manner to reduce the accelerations on the valves. Valves suspended on piping spans are reviewed for adequacy for the loads to which the span would be subjected. Valves are mounted in the position recommended by the manufacturer.

2. Valve yokes are reviewed for adequacy and strengthened as required for the response of the valve operator to seismic loads.

3. Where valves are required to operate during seismic loading, the operator forces are reviewed to ensure that system function is preserved. Seismic forces on the operating parts of the valve are small compared to the other forces present.

4. Control wires and piping to the valve operators are designed and installed to ensure that the flexure of the line does not endanger the control system. Appendages to the valve, such as position indicators and operators, are checked for structural adequacy.

Isolation valves are provided as necessary for all fluid system lines penetrating the containment to ensure at least two barriers for redundancy against leakage of radioactive fluids to the environment in the event of a loss-of-coolant accident. These barriers, in the form of isolation valves or closed systems, are defined on an individual line basis. In addition to satisfying containment isolation criteria, the valving is designed to facilitate normal operation and maintenance of the systems and to ensure reliable operation of other engineered safeguards systems.

With respect to numbers and locations of isolation valves, the criteria applied are generally those outlined by the seven classes described in Section 5.2.2 below.

5.2.2 System Design

The seven classes listed below are general categories into which lines penetrating containment may be classified. The seal water referred to in the listing of categories is provided by the isolation valve seal water system described in Section 6.5. The following notes apply to these classifications.

1. The "not-missile-protected" designation refers to lines that are not protected throughout their length inside containment against missiles generated as the result of a loss of coolant accident. These lines, therefore, are not assumed invulnerable to rupture as a result of a loss of coolant.

2. In order to qualify for containment isolation, valves inside the containment must be located behind the missile barrier for protection against loss of function following an accident.

3. Manual isolation valves that are locked closed or otherwise closed and under administrative control during power operation qualify as automatic trip valves.

4. A check valve qualifies as an automatic trip valve in certain incoming lines not requiring seal water injection.

5. The double disc type of gate valve is used to isolate certain lines. When sealed by water injection, this valve provides two barriers against leakage of radioactive liquids or containment atmosphere.

6. In lines isolated by globe valves and provided with seal water injection, the valves are generally installed so that the seal water wets the stem packing.

7. Loss of seal water through those isolation valves closed only by a containment isolation phase B signal is prevented by solenoid operated valves in the seal water injection lines. Excessive loss of seal water through motor operated isolation valves that could fail to close in response to a containment isolation signal is limited by flow restrictive orifices installed in the seal water lines. A water seal at the failed valve is ensured by proper slope of the protected line, or a loop seal, or by additional valves on the side of the isolation valves away from the containment.

8. Isolated lines between the containment and the second outside isolation valve are designed to the same seismic criteria as the containment vessel and are considered to be an extension of containment.

5.2.2.1 Class 1, Outgoing Lines, Reactor Coolant System

Outgoing lines connected to the reactor coolant system that are normally or intermittently open during reactor operation are provided with at least two automatic trip valves in series located outside the containment. Automatic seal water injection is provided for lines in this classification.

An exception to the general classification is the residual heat removal loop's reactor coolant system suction line, which has two barriers established by normally closed valves located outside containment.

5.2.2.2 Class 2, Outgoing Lines

Outgoing lines not connected to the reactor coolant system that are normally or intermittently open during operation and not missile protected or that can otherwise communicate with the containment atmosphere following an accident are provided at a minimum with two automatic trip valves in series or a single automatic double-disc gate valve outside containment. Automatic seal water injection is provided for lines in this classification. Most of these lines are not vital to plant operation following an accident.

5.2.2.3 Class 3, Incoming Lines

Incoming lines connected to open systems (i.e., systems that are in some way connected to the containment environment) outside containment, and not missile protected or that can otherwise communicate with the containment atmosphere following an accident are provided with one of the following arrangements outside containment:

1. Two automatic trip valves in series, with automatic seal water injection. This arrangement is provided for lines that are not necessary to plant operation after an accident.

2. Two manual isolation valves in series, with manual seal water injection. This arrangement is provided for lines that remain in service for a time, or are used periodically, subsequent to an accident.

Incoming lines connected to closed systems outside containment, and not missile protected or that can otherwise communicate with the containment atmosphere, are provided either with two isolation valves in series outside containment with seal water injection, or at a minimum, with one check valve or normally closed isolation valve located either inside or outside containment. The closed piping system outside containment provides the necessary isolation redundancy for lines that contain only one isolation valve.

Exception is the containment spray headers, for which valving is based on safeguards requirements.

5.2.2.4 Class 4, Missile Protected Lines

Incoming and outgoing lines that penetrate the containment and that are normally or intermittently open during reactor operation and are connected to closed systems inside the containment and protected from missiles throughout their length, are provided with at least one manual isolation valve located outside the containment. Seal water injection is not required for this class of penetration. An exception is the residual heat exchanger cooling water lines for which design is based on safeguards requirements.

5.2.2.5 Class 5, Normally Closed Lines Penetrating the Containment

Lines that penetrate the containment and that can be opened to the containment atmosphere but that are normally closed during reactor operation are provided with two isolation valves in series or one isolation valve and one blind flange.

5.2.2.6 Class 6, Special Service Lines

There are a number of special groups of penetrating lines and containment access openings. These are discussed below.

Each ventilation purge duct penetration is provided with two tight-closing butterfly valves, which are normally closed during reactor power operation and are actuated to the closed position automatically upon a containment isolation or a containment high radiation signal. One valve is located inside and one valve is located outside the containment at each penetration. The space between valves is pressurized by air from the weld channel and penetration pressurization system whenever they are closed. Blind flanges can also be used for containment isolation in place of automatic purge isolation valves, provided they meet the same design criteria as the isolation valves.

The containment pressure relief line is similarly protected. However, because the line is periodically opened during reactor power operation, three tight closing butterfly valves in series are provided, one inside and two outside the containment. These valves also are actuated to the closed position upon a containment isolation or containment high radiation signal. The two intravalve spaces are pressurized by air from the weld channel and penetration pressurization system whenever they are closed.

The equipment access closure is a bolted, gasketed closure that is air sealed during reactor operation. The personnel air locks consist of two doors in series with mechanical interlocks to ensure that one door is closed at all times. Each air lock door and the equipment closure are provided with double gaskets to permit pressurization between the gaskets by the weld channel and penetration pressurization system. (See Section 6.6.)

The fuel transfer tube penetration inside the containment is designed to present a missile-protected and pressurized double barrier between the containment atmosphere and the atmosphere outside the containment. The penetration closure is treated in a manner similar to the equipment access hatch. A positive pressure is maintained between the double gaskets of the tube cover flange to establish the double barrier between the containment atmosphere and the inside of the fuel transfer tube. The interior of the fuel transfer tube is not pressurized. Seal water injection is not required for this penetration.

The following lines would be subjected to pressure in excess of the isolation valve seal water system pressure (~52 psig) in the event of an accident, due to operation of the safety injection system recirculation pumps:

1. Residual heat removal loop inlet line.
2. Residual heat removal loop outlet line.
3. Bypass line from residual heat exchanger outlet to safety injection pumps suction.
4. Residual heat removal pumps mini-flow line.
5. Residual heat removal loop sample line.
6. Recirculation pump discharge sample line.

These lines are isolated by double disk gate valves or double valves, which can be sealed by nitrogen gas from the high pressure nitrogen supply of the isolation valve seal water system. A self-contained pressure regulator operates to maintain the nitrogen injection pressure slightly higher than the maximum expected line pressure. These valves are closed or intermittently operated during reactor operation, and the nitrogen gas injection is manually initiated.

Lines, which are capable of communicating with the containment atmosphere (normally filled with air or vapor) include:

1. Steam jet air ejector return line to containment.
2. Containment radiation monitor inlet and outlet lines.

In an accident condition the space between the two containment isolation valves in each line are sealed by pressurizing with air from the weld channel and penetration pressurization system. The air is introduced into each space at approximately 2 psi above the containment design pressure through a separate line from the weld channel and penetration pressurization system. Parallel, redundant, fail-open valves in each injection line open on the appropriate containment isolation signal to provide a reliable supply of pressurizing air. A flow-limiting orifice in each injection line prevents excessive air consumption if one of these valves spuriously fails to open, or if one of the containment isolation valves fails to respond to the "trip" signal.

5.2.2.7 Class 7, Steam and Feedwater Lines

These lines and the shell side of the steam generator are considered basically as an extension of the containment boundary and as such must not be damaged as a consequence of reactor coolant system damage. This requires that the steam generator shell, feed and steam lines within the containment are to be classified and designed for the reactor coolant system missile-protected category. The reverse is also true in that a steam break is not to cause damage to the reactor coolant system.

5.2.3 Isolation Valves And Instrumentation Diagrams

Figures 5.2-1 through 5.2-28 show all valves in lines leading to the atmosphere or to closed systems on both sides of the containment barrier, valve actuation and preferential failure modes, the application of "trip" (containment isolation) signals, relative location of the valves with respect to missile barriers, and the boundaries of seismic Class I designed lines. The item numbers in these figures align with the item numbers in Table 5.2-1. Figure 5.2-29 defines the nomenclature and symbols used.

5.2.4 Valve Parameters Tabulation

A summary of the fluid systems lines penetrating containment and the valves and closed systems employed for containment isolation is presented in Table 5.2-1. Each valve is described as to type, operator, position indication and open or closed status during normal operation, shutdown and accident conditions. Information is also presented on valve preferential failure mode, automatic trip by the containment isolation signal, and the fluid carried by the line.

Containment isolation valves are provided with actuation and control equipment appropriate to the valve type. For example, air-operated globe and diaphragm (Saunders Patent) valves are generally equipped with air diaphragm operators, with fail-safe operation provided by the control devices in the instrument air supply to the valve. Motor-operated gate valves are capable of being supplied from reliable onsite emergency power as well as their normal power source. Manual and check valves, of course, do not require actuation or control systems.

The containment isolation system is brought into service by one of three conditions: phase A isolation signal, phase B isolation signal, or containment ventilation isolation signal.

The automatically tripped isolation valves are actuated to the closed position by any of these isolation signals. The first of these signals is derived in conjunction with safety injection actuation, and trips the majority of the automatic isolation valves. These are valves in the so-called "nonessential" [*Note - "Nonessential" process lines are defined as those, which do not increase the potential for damage to in-containment equipment when isolated. "Essential" process lines are those providing cooling water and seal water flow for the reactor coolant pumps. These services should not be interrupted unless absolutely necessary while the reactor coolant pumps are operating.*] process lines penetrating the containment. This is defined as "phase A" isolation and the trip valves are designated by the letter "T" in the isolation diagrams, Figures 5.2-1 through 5.2-29. This signal also initiates automatic seal water injection (See Section 6.5). The second, or "phase B", containment isolation signal is derived upon actuation of the containment spray system, and trips the automatic isolation valves in the so-called "essential"* process lines penetrating the containment. These trip valves are designated by the letter "P" in the isolation diagrams. Containment ventilation isolation represents closing of the three ventilation lines to the containment and will be automatically activated by high containment radioactivity, a phase A isolation signal, or automatic containment spray (and associated phase B) actuation; see Section 5.3 for further information on the containment, heating, cooling and ventilation system.

A manual containment isolation signal can be generated from the control room for either phase A or phase B isolation. These signals perform the same functions as the automatically derived signals. The containment ventilation isolation signal can be manually activated by a manual safety injection signal, a manual phase A containment isolation signal, or a manual containment spray signal.

Nonautomatic isolation valves, i.e., remote stop valves and manual valves, are used in lines, which must remain in service, at least for a time, following an accident. These are closed manually if and when the lines are taken out of service.

Standard closing times available with commercial valve models are adequate for the sizes of containment isolation valves used. Valves equipped with air-diaphragm operators generally close in approximately 2 sec. The typical closing time available for large motor-operated gate valves is 10 sec. Closing times of greater than 10 seconds are permitted on a case by case basis if properly justified by an individual valve evaluation.

The large butterfly valves used to isolate the containment ventilation purge ducts are equipped with air piston operators and spring returns capable of closing the valves. The butterfly valves used to isolate the 10-in. pressure relief line are equipped with air piston operators each with a separate accumulator air supply on each valve capable of closing the valves. These valves all fail to the closed position on loss of control signal or instrument air. Allowable closure time for these valves is less than or equal to 3 seconds.

5.2.5 Valve Operability

All containment isolation valves, actuators, and controls are located so as to be protected against missiles that could be generated as the result of a loss-of-coolant accident. Only valves so protected are considered to qualify as containment isolation valves.

Only isolation valves located inside containment are subject to the high-pressure, high-temperature, steam-laden atmosphere resulting from an accident. Operability of these valves in the accident environment is ensured by proper design, construction, and installation, as reflected by the following considerations:

1. All components in the valve installation, including valve bodies, trim and moving parts, actuators, instrument air and control and power wiring, are constructed of materials sufficiently temperature resistant to be unaffected by the accident environment. Special attention is given to electrical insulation, air operator diaphragms and stem packing material.

2. In addition to normal pressures, the valves are designed to with-stand maximum pressure differentials in the reverse direction imposed by the accident conditions.

This criterion is particularly applicable to the butterfly-type isolation valves used in the containment purge lines. Valve actuators are installed on these butterfly valves and travel is limited to a maximum of 60 degrees to ensure that the valves will be able to close against the maximum calculated design-basis accident pressure of 47 psig. An adjustable position setting on the actuators allows the valves to be opened to a full 90-degree position when containment integrity is not required.

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 1 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Testing/ Sealing Method[1]	Used After Accid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
1	Pressurizer relief tank to gas analyzer RCS	5.2-1	549	Globe	Air	Yes	Op/Cl	Op/Cl	Closed	FC	T	Water (A)	No	G	Hot	
			548	Globe	Air	Yes	Op/Cl	Op/Cl	Closed	FC	T	Water (A)				
2.	Pressurizer relief tank N_2 supply tank RCS	5.2-1	518	Check	-	No	Closed	Closed	Closed	-	-	-	Yes/No	G	Cold	
			3418	Globe	Sole.	No	Open	Open	Op/Cl	FC	-	-				
			3419	Globe	Sole.	No	Open	Open	Op/Cl	FC	-	-				
			4136	Dia.	Manual	No	Closed/OI	Closed/OI	Op/Cl	-	-	-				
3.	Pressurizer relief tank makeup - RCS	5.2-1	552	Dia	Air	Yes	Closed	Closed	Closed	FC	T	Water (A)	No	W	Cold	
			519	Dia	Air	Yes	Closed	Closed	Closed	FC	T	Water (A)				
4.	Residual heat removal return - ACS/SIS	5.2-2	$741A_{1A}$	Check	-	No	Closed	Open	Open	-	-	RHR	Yes	W	Hot	
			744_{1A}	DDV	Motor	Yes	Open	Open	Op/CL	FAI	-	Nit (M)				May be closed depending on accident condition
5.	Resid. Heat removal loop to - S.I.pumps - ACS/SIS	5.2-2	888A	DDV	Motor	Yes	Closed/OI	Closed	Op/Cl	FAI	-	Nit (M)	Yes	W	Hot**	
			888B	DDV	Motor	Yes	Closed/OI	Closed	Op/Cl	FAI	-	Nit (M)				
	To sampling system ACS/SS	5.2-2	958	Globe	Motor	No	Closed/OI	Closed	Closed	FAI	-	Nit (M)	Yes	W	Hot	May be used during shutdown and after accident
			959	Globe	Motor	No	Closed/OI	Closed	Closed	FAI	-	Nit (M)	Yes	W		
			990D	Globe	Manual	No	LC/OI	LC	LC	-	-	Nit (M)	No	-	-	
	RHR pump mini-flow line	5.2-2	1870	Globe	Motor	Yes	Open	Open	Op/Cl	FAI	-	Nit (M)	Yes/No	W	Hot	
			743	Globe	Motor	Yes	Open	Open	Op/Cl	FAI	-	Nit (M)	Yes/No	W		

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 2 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Testing/ Sealing $Method_1$	Used After Accid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
6.	Residual heat removal loop-out - ACS	5.2-2	732_{1A}	DDV	Manual	No	LC/OI	Open	Closed	-	-	Nit (M)	No	W	Hot	
7.	Containment sump recirculation - ACS/SIS	5.2-2	885A 885B	DDV_{2A} DDV_{2A}	Motor Motor	Yes Yes	Closed/OI Closed/OI	Closed Closed	$Closed_2$ $Closed_2$	FAI FAI	- -	RHR RHR	No_2	W	Cold	2. Normally closed but may be opened after accident if normal recirculation path from recirculation pump not available 2A. The upstream disc (nearest containment) of 885A and the downstream disc (RHR Loop side) of 885B have a 3/16" hole to prevent pressure locking
8.	Letdown line - CVCS	5.2-3	201 202	Globe Globe	Air Air	Yes Yes	Open Open	Open Open	Closed Closed	FC FC	T T	Water (A) Water (A)	No	W	Hot	
9.	Charging line - CVCS	5.2-3	205 226 227	Gate Globe Globe	Motor Motor Motor	No No No	Open Open Closed/OI	Open Open Closed	Op/Cl Op/Cl Op/Cl	FAI FAI FAI	- - -	Water (M) Water (M) Water (M)	Yes* Yes* Yes*	W	Cold	* May be used depending on accident.
10.	Reactor coolant pump seal-water supply lines (4) - CVCS	5.2-4	250ABCD 4925, 4926, 4927, 4928	Globe Globe	Motor Motor	No No	Open Open	Open Open	Op/Cl Op/Cl	FAI FAI	- -	Water (M) Water (M)	Yes_3 Yes_3	W W	Cold Cold	3. Manual isolation if and when pumps are stopped.
11.	Reactor coolant pump seal water return - CVCS	5.2-4	222	DDV	Motor	Yes	Open	Open	Closed	FAI	P	Water (A)	No	W	Cold	
12.	Reactor coolant sample line - SS	5.2-5	956E 956F	Globe Globe	Motor Motor	Yes Yes	Op/Cl Op/Cl	Op/Cl Op/Cl	Closed Closed	FAI FAI	T T	Water (A) Water (A)	Yes_4	W	Hot	4. Used to take postaccident RCS samples
13.	Fuel transfer tube - FHS	5.2-5	A	Blind flange	-	No	Closed	-	-	-	-	Air_5	No	W	Cold	Flange is double gasketed in refuel-ing canal (missile protected). 5. Normally seal with air (WCPPS)

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 3 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Posit. Isolation Trip	Testing/ Sealing Method[1]	Used After Accid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
14.	Containment spray headers (2) - SIS	5.2-6	869A,B	DDV	Motor	No	Open	Op/Cl	Op/Cl	FAI	-	Water (M)	Yes	W	Cold	
			867A,B	Check	-	No	Closed	Closed	Op/Cl	-	-	-				
			878A	Globe	Manual	No	LC/OI	Closed	Closed	-	-	-				
15.	Safety injection headers (2) - SIS	5.2-7	850A	DDV	Motor	No	Open	Open	Op/Cl	FAI	-	Water(M)	Yes	W	Hot**	
			851A	DDV	Motor	Yes	Open	Open	Op/Cl	FAI	-	Water(M)	Yes	W	Hot**	
			851B	DDV	Motor	Yes	Open	Open	Op/Cl	FAI	-	Water(M)	Yes	W	Hot**	
			850B	DDV	Motor	No	Open	Open	Op/Cl	FAI	-	Water(M)	Yes	W	Hot**	
16.	Safety injection test line - SIS	5.2-7	859A	Globe	Manual	No.	LC/OI	Closed	Closed	-	-	Water (A)	No	W	Cold	
			859C	Globe	Manual	No	LC/OI	Closed	Closed	-	-	Water (A)				
17.	Accumulator/ OPS N_2 supply - SIS	5.2-8	4312	Check	-	No	Closed	Closed	Closed[6]	-	-	-	No	G	Cold	6. Could be opened depending on type of accident
			863	Globe	Air	Yes	Op/Cl	Op/Cl	Closed[6]	FC	T	-	No			
18.	Accumulator sample - SS	5.2-8	956G	Globe	Air	Yes	Op/Cl	Op/Cl	Closed	FC	T	Water (A)	No	W	Cold	Valves A and B opened intermittently to take sample
			956H	Globe	Air	Yes	Op/Cl	Op/Cl	Closed	FC	T	Water (A)				
19.	Primary system vent header and N_2 supply line - WDS	5.2-9	1786	Dia	Air	Yes	Open	Closed	Closed	FC	T	Water (A)	No	G	Hot	
			1787	Dia	Air	Yes	Open	Closed	Closed	FC	T	Water (A)				
			3416	Globe	Sole.	No	Open	Open	Op/Cl	FC	-	-	Yes/No	G	Hot	
			3417	Globe	Sole.	No	Open	Open	Op/Cl	FC	-	-				
			5459	Dia	Manual	No	Closed/OI	Closed	Op/Cl	-	-	-				
			1616	Check	-	No	Closed	Closed	Op/Cl	-	-	-				
20.	Reactor coolant drain tank to gas analyzer - WDS	5.2-9	1788	Dia	Air	Yes	Op/Cl	Op/Cl	Closed	FC	T	Water (A)	No	G	Hot	Valves opened intermittently
			1789	Dia	Air	Yes	Op/Cl	Op/Cl	Closed	FC	T	Water (A)				
21.	RCDT pump discharge – WDS	5.2-9	1702	Dia	Air	Yes	Open	Op/Cl	Closed	FC	T	Water (A)	No	W	Cold	Valves open intermittently
			1705	Dia	Air	Yes	Open	Op/Cl	Closed	FC	T	Water (A)				

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 4 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Testing/ Sealing Method[1]	Used After Accid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
22.	Reactor coolant pump cooling water in - ACS	5.2-10	797	DDV	Motor	Yes	Open	Op/Cl	Closed	FAI	P	Water (A)	No[7]	W	Cold	7. Could be used depending on the type of accident
23.	Reactor coolant pump water out (6”) - ACS	5.2-10	784	DDV	Motor	Yes	Open	Op/Cl	Closed	FAI	P	Water (A)	No[8]	W	Cold	8. Could be used depending on the type of accident
24.	Reactor coolant pump water out (3”) - ACS	5.2-10	FCV-625	DDV	Motor	Yes	Open	Op/Cl	Closed	FAI	P	Water (A)	No[9]	W	Cold	9. Could be used depending on the type of accident
25.	Resid. Heat exch. Cooling water in - ACS	5.2-11	CS	-	-	-	-	-	-	-	-	-	Yes	W	Hot	Residual heat exchanger and associated component cooling lines are a missile protected closed system
			CS	-	-	-	-	-	-	-	-	-	Yes	W	Hot	Component cooling system closed
26.	Resid. Heat exch. Cooling water return - ACS	5.2-11	$822A_{1B}$ $822B_{1B}$ CS	Gate Gate -	Motor Motor -	Yes Yes -	Closed Closed -	Open Open -	Open Open -	FAI FAI -	- - -	- - -	Yes Yes Yes	W W W	Cold Cold Cold	 Component cooling system closed
27.	Recir. Pump cooling water supply - ACS	5.2-12	$753H_{1B}$	Gate	Manual	No	Open	Open	Op/Cl	-	-	-	Yes	W	Cold	May be closed depending on accident condition
			CS	-	-	-	-	-	-	-	-	-	Yes	W	Cold	Component cooling system closed
28.	Recir. Pump cooling heater return - ACS	5.2-12	$753G_{1B}$	Gate	Manual	No	Open	Open	Op/Cl	-	-	-	Yes	W	Cold	May be closed depending on accident condition
			CS	-	-	-	-	-	-	-	-	-	Yes	W	Cold	Component cooling system closed
29.	Excess letdown eat exchanger cooling water in – ACS	5.2-13	791 798	Dia Dia	Air Air	Yes Yes	Open Open	Open Open	Closed Closed	FC FC	T T	Water (A) Water (A)	No	W	Cold	

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 5 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Testing/ Sealing Method[1]	Used After Accid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
30.	Excess letdown heat exchanger cooling water out - ACS	5.2-13	796 793	Globe Dia	Air Air	Yes Yes	Op/Cl Open	Op/Cl Open	Closed Closed	FC FC	T T	Water (A) Water (A)	No	W	Cold	
31.	Containment sump pump discharge – WDS	5.2-13	1728 1723	Dia Dia	Air Air	Yes Yes	Open Open	Open Open	Closed Closed	FC FC	T T	Water (A) Water (A)	No	W	Cold	
31a.	Sampling system return - WDS	5.2-13	5132 4399	Globe Globe	Motor Motor	Yes Yes	Closed Closed	Closed Closed	Cl/Op Cl/Op	FAI FAI	T T	Water (A) Water (A)	No/Yes[10] No/Yes[10]	W W	Cold Cold	10. Can be used to return highly radioactive water to containment after post-accident analysis of sampling system.
32.	Containment air sample in - rad. mon.	5.2-14	PCV-1234 PCV-1235	Dia Dia	Air Air	Yes Yes	Open Open	Open Open	Closed Closed	FC FC	T T	Air (A) Air (A)	No[11] No[11]	G G	Cold Cold	11. May be opened for air sampling following accident when the containment pressure is below 5 psig
33.	Containment air sample out - rad. mon.	5.2-14	PCV-1236 PCV-1237	Dia. Dia.	Air Air	Yes Yes	Open Open	Open Open	Closed Closed	FC FC	T T	Air (A) Air (A)	No[12] No[12]	G G	Cold Cold	12. May be opened for air sampling following accident when the containment pressure is below 5 psig
34.	Air ejector discharge to containment sec sys	5.2-14	PCV-1229 PCV-1230	Globe Globe	Air Air	Yes Yes	Closed Closed	Closed Closed	Closed Closed	FC FC	T T	Air (A) Air (A)	No No	G G	Cold Cold	
35.	Main steam headers[13]	-	CS	-	-	-	-	-	-	-	-	-	-	-	Hot	Steam generators 13. (four penetrations)

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 6 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Testing/ Sealing Method$_1$	Used After Acid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
36.	Main feedwater headers	-	CS	-	-	-	-	-	-	-	-	-	-	-	Hot	Steam generators (four penetrations)
37.	Steam generator blowdown/sample sec. sys.	5.2-15	PCV1214 PCV1215 PCV1216 PCV1217	Globe	Air	Yes	Open	Op/Cl	Closed	FC	T	Water (A)	No	W	Hot	*(four penetrations)
			PCV1214A PCV1215A PCV1216A PCV1217A	Globe	Air	Yes	Open	Op/Cl	Closed	FC	T	Water (A)	No	W	Hot	
38.	S.G. blowdown sample															System deleted
39.	Ventilation system water cooling water in – SWS_{14}	5.2-16	SWN-41	BV	Motor	No	Open	Open	Op/Cl	FAI	-	SWS	Yes	W	Cold	Fan cooler units - missile protected, closed system
			SWN-42	Relief	-	No	Closed	Closed	Closed	-	-	SWS	-	-	-	
			SWN-43	Gate(2) Globe(3)	Manual	No	LC/OI	Closed	Op/Cl	-	-	SWS	Yes	-	-	14. (five penetrations)
40.	Ventilation system motor cooling water out -	5.2-16	SWN-44	BV	Motor	No	Open	Open	Op/Cl	FAI	-	SWS	Yes	W	Cold	Fan cooler units - missile protected, closed system
	SWS_{15}	5–2–16	SWN-51	Globe	Motor	No	Open	Open	Op/Cl	FAI	-	SWS	Yes	W	Cold	15. (Five penetrations)
40a.	Ventilation system motor cooling water out - SWS_{16}	5–2–16	SWN-71	Globe	Motor	No	Open	Open	Op/Cl	FAI	-	SWS	Yes	W	Cold	16. Five penetrations
41.	Service air	5.2-17	SA-24	Dia	Manual	No	LC/OI	LC	LC	-	-	Water (A)	No	G	Cold	
			SA-24-1	Dia	Manual	No	LC/OI	LC	LC	-	-	Water (A)				
42.	Not assigned															

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 7 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Testing/ Sealing Method[1]	Used After Accid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
43.	Weld channel pressurization air supply (PPS)[17]	5.2-17	PCV1111-1[18] PCV1111-2[18] CS	Ball Ball -	Manual - -	No - -	Open	Open	Op/Cl	- -	- -	- -	Yes -	G -	Cold -	17. Two penetrations penetration press system
44.	Spare	5.2-17	580A 580B	Needle Needle	Manual Manual	No No	LC/OI LC/OI	LC LC	LC LC	- -	- -	- -	No	G	Cold	Penetration capped inside containment and outside containment downstream of valve 580B
45.	Auxiliary steam supply	5.2-18	UH-43	DDV	Manual	No	LC/OI	Closed[18]	LC	-	-	Water (A)	No	G	Hot	18. May be opened during shutdown for cont. heating
46.	Auxiliary steam supply condensate return	5.2-18	UH-44	DDV	Manual	No	LC/OI	Closed[19]	LC	-	-	Water (A)	No	W	Hot	19. May be opened during shutdown for cont. heating
47.	City water	5.2-18	MW-17 MW-17-1	Gate Gate	Manual Manual	No No	LC/OI LC/OI	Closed[20] Closed[20]	LC LC	- -	- -	Water (A) Water (A)	No	W	Cold	20. May be opened during shutdown for maintenance or fire protection purposes
48.	Purge supply duct in - vent. sys.	5.2-19	FCV-1170 FCV-1171	BV BV	Air Air	Yes Yes	Closed[21] Closed[21]	Open Open	Closed Closed	FC FC	CVI CVI	Air (A) Air (A)	No	G	Cold	21. May be open for safety related purging, or to facilitate safety related surveillance or maintenance.
49.	Purge exhaust duct out - vent. sys.	5.2-19	FCV-1172 FCV-1173	BV BV	Air Air	Yes Yes	Closed[22] Closed[22]	Open Open	Closed Closed	FC FC	CVI CVI	Air (A) Air (A)	No	G	Cold	22. May be open for safety related purging, or to facilitate safety related surveillance or maintenance.
50.	Containment pressure relief - vent	5.2-19	PCV-1190 PCV-1191 PCV-1192	BV BV BV	Air Air Air	Yes Yes Yes	Closed[23] Closed[23] Closed[23]	Closed Closed Closed	Closed Closed Closed	FC FC FC	CVI CVI CVI	Air (A) Air (A) Air (A)	No	G	Cold	23. Opened intermittently for pressure relief.
51.	Recirculation pump discharge sample line	5.2-20	990A 990B	Globe Globe	Motor Motor	Yes Yes	Closed Closed	Closed Closed	Op/Cl Op/Cl	FAI FAI	T T	Nit (M) Nit (M)	No/Yes[24]	W	Cold	24. Used periodically after accident to sample recirculation fluid.

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 8 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Testing/ Sealing Method[1]	Used After Accid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
52.	Pressurizer steam space sample line	5.2-20	956A	Globe	Air	Yes	Op/Cl	Op/Cl	Closed	FC	T	Water (A)	No/Yes[25]	W	Hot	25. Could be used for taking postaccident samples.
			956B	Globe	Air	Yes	OP/Cl	OP/Cl	Closed	FC	T	Water (A)	No/Yes[25]	W	Hot	
53.	Pressurizer liquid space sample line	5.2-20	956C	Globe	Air	Yes	Op/Cl	Op/Cl	Closed	FC	T	Water (A)	No/Yes[26]	W	Hot	26. Could be used for taking postaccident samples.
			956D	Globe	Air	Yes	Op/Cl	Op/Cl	Closed	FC	T	Water (A)	No/Yes[26]	W	Hot	
54.	Containment	5.2-21	1814A	Globe	Manual	No	LO	Open	Op/Cl	-	-	-	Yes	G	Cold	
55.	Pressure		1814B													
56.	Instrumentation		1814C													
			CS	-	-	-	-	-	-	-	-	-	-	-	-	
57.	Postaccident containment sampling system supply and return lines (7)	5.2-22	SOV-5018	Globe	Sole.	Yes	Closed	Closed	Both[26a]	FC	-	-	Yes	G	Cold	26a. Isolation valves are opened intermittently after an accident.
			SOV-5020	Globe	Sole.											
			SOV-5022	Globe	Sole.											
			SOV-5024	Globe	Sole.											
			SOV-5019	Globe	Sole.	Yes	Closed	Closed	Both[26a]	FC	-	-	Yes	G	Cold	
			SOV-5021	Globe	Sole.											
			SOV-5023	Globe	Sole.											
			SOV-5025	Globe	Sole.											
58.	Spare	5.2-23														
59.	Spare	5.2-24														
60.	Spare															
61.	Spare															
62.	Spare															
63.	Not assigned															

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 9 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Testing/ Sealing Method$_{1}$	Used After Accid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
64.	Instrument/air postaccident venting supply	5.2-25	IA-39	Check	-	No	Open	Open	Both	-	-	-	Yes/No$_{27}$	G	Cold	27. Could be used to resupply instrument air to containment post-accident.
			PCV1228	Dia	Air	Yes	Open	Open	Both	FC	T	-	Yes/No$_{27}$	G	Cold	
65.	Postaccident venting exhaust line	5.2-25	E-2	Dia	Air	No	Closed/OI	Closed	Both	FC	-	Air (A)	Yes/No$_{28}$	G	Cold	28. Could be used after accident if containment venting were deemed necessary.
			E-1	Dia	Air	No	Closed/OI	Closed	Both	FC	-	Air (A)	Yes/No$_{28}$	G	Cold	
			E-3	Dia	Air	No	Closed/OI	Closed	Both	FC	-	Air (A)	Yes/No$_{28}$	G	Cold	
			E-5	Dia	Air	No	Closed/OI	Closed	Both	FC	-	Air (A)	Yes/No$_{28}$	G	Cold	
66.	Deleted															
67.	Containment leak test air line$_{30}$	5.2-26	A	Blind Flange	-	No	Closed	Closed	Closed	-	-	-	No.	Gas	Cold	30. Two penetrations
			B	Blind Flange W/Test Conn.	-	No	Closed	Closed	Closed	-	-	-	No	Gas	Cold	
68.	Equipment access	-	CS	-	-	-	-	-	-	-	-	Air (A)	No	-	-	
69.	Personnel air lock (2)	5.2-27	85A, 95A$_{31}$	Ball	Interlk w/door	Yes	Closed	Closed	Closed	-	-	Air (A)	No	Gas	Cold	31. 85A & 95A may be open when 85B & 95B are closed. 85B & 95B may be open when 85A & 95A are closed.
			85B, 95B$_{31}$	Ball								Air (A)				
			85C, 95C	Spring check	-	No	Closed	Closed	Closed	-	-	-	No	Gas	Cold	
			85D, 95D	Spring check	-	No	Closed	Closed	Closed	-	-	-	No	Gas	Cold	
70.	Steam generator level, pressurizer level, and pressure pneumatic indication lines (4)	5.2-28	IIP-500	Globe	Manual	No	Closed/OI	Closed	Both	-	-	-	Yes$_{32}$	-	Cold	32. Depending on accident type
			IIP-501	Globe	Manual	No	Closed/OI	Closed	Both	-	-	-	Yes$_{32}$	-	Cold	
			IIP-502	Globe	Manual	No	Closed/OI	Closed	Both	-	-	-	Yes$_{32}$	-	Cold	
			IIP-503	Globe	Manual	No	Closed/OI	Closed	Both	-	-	-	Yes$_{32}$	-	Cold	
			IIP-504	Globe	Manual	No	Closed/OI	Closed	Both	-	-	-	Yes$_{32}$	-	Cold	
			IIP-505	Globe	Manual	No	Closed/OI	Closed	Both	-	-	-	Yes$_{32}$	-	Cold	
			IIP-506	Globe	Manual	No	Closed/OI	Closed	Both	-	-	-	Yes$_{32}$	-	Cold	
			IIP-507	Globe	Manual	No	Closed/OI	Closed	Both	-	-	-	Yes$_{32}$	-	Cold	

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 10 of 10)

Key:

FAI	- fail as is	RCS	- reactor coolant system	Hot – insulated & cooled penetration
FC	- fail closed	ACS	- auxiliary coolant system	Cold – standard piping or equipment penetration
FO	- fail open	WDS	- waste disposal system	Hot** - insulated non cooled penetration
LC	- locked closed	SIS	- safety injection system	IVSWS -isolation valve seal water system
LO	- locked open	SS	- sampling system	WCPPS -weld channel pressurization system
BV	- butterfly valve	CVCS	- chemical and volume control system	RHR -residual heat removal system
DDV	- double disk gate valve	Vent	- ventilation system	
Dia	- diaphragm valve	SWS	- service water system	
T	- containment isolation signal - phase A	FH	- fuel handling	
P	- containment isolation signal - phase B	PPS	- penetration pressurization system	
A	- automatic	CVI	- containment ventilation isolation signal	
M	- manual	CS	- closed system	
Op/Cl	- open/closed	Nit	- nitrogen	
OI	- may be opened intermittently to support plant operations			

Notes:

1. Sealing Methods and Type C Testing:
   - For valves sealed by IVSWS water (designated “Water (A)” or “Water (M)”), minimum Type C test pressure is 52 psig.
   - For valves sealed by IVSWS nitrogen (designated “Nit (M)”), minimum Type C test pressure is 47 psig.
   - For valves sealed by WCPPS (designated “Air (A)”), minimum Type C test pressure is 47 psig.
   - For valves sealed by RHR system fluid (designated “RHR”), minimum Type C test pressure is 52 psig (valves 741A,885A,885B).
   - For valves sealed by service water system (designated “SWS”), minimum Type C test pressure is 52 psig (valve series SWN-41, SWN-42, SWN-43, SWN-44, SWN-51, SWN-71). Either the “A” or “B” valve(s) may serve as the required containment isolation valve(s) for the SWN-41, SWN-44 and SWN-71 series. Designation of the “B” valve(s) in the SWN-44 series requires the codesignation of the SWN-51 valves associated with the penetration(s) as an additional required containment isolation valve(s) (see Figure 5.2-16).
   - For all other isolation valves not sealed by a system, gas (ie. Nitrogen or air) is the test medium at a minimum Type C pressure of 47 psig.

1A. These valves testable only when at cold shutdown (741A, 744, 732).

1B. These valves are excluded from Type C testing per License Amendment N0. 63, dated August 28, 1980 (822A, 822B, 753G, 753H, PCV-1111-1, PCV-1111-2).

5.2 FIGURES

5.3 CONTAINMENT HEATING, COOLING AND VENTILATION SYSTEM

5.3.1 Design Basis

5.3.1.1 Performance Objectives

The containment heating, cooling and ventilation systems are designed to accomplish the following:

1. Remove the normal heat loss from all equipment and piping in the reactor containment during plant operation and to maintain a normal ambient temperature of 130°F or less.

2. Provide sufficient air circulation and filtering of iodine throughout all containment areas to permit safe and continuous access to the reactor containment within two hours after reactor shutdown assuming defects exist in 1-percent of the fuel rods.

3. Provide for positive circulation of air across the refueling water surface to assure personnel access and safety during shutdown.

4. Provide containment heating, if required, to assure a minimum containment ambient temperature of 50°F before the reactor is taken above the cold shutdown condition.

5. Provide for purging of the containment vessel to the plant vent for dispersion to the environment. The rate of release does not permit offsite dose to exceed Offsite Dose Calculation Manual (ODCM).

6. Provide for depressurization of the containment vessel following an accident. The postaccident design and operating criteria are detailed in Section 6.4.

7. Provide for continuous pressure relief via an exhaust system.

In order to accomplish these objectives the following systems are provided:

1. Containment recirculation cooling system
2. Control rod drive mechanism cooling system
3. Containment purge and pressure relief system
4. Containment auxiliary charcoal filter system
5. Steam heating system

5.3.1.2 Design Characteristics-Sizing

The design characteristics of the equipment required in the containment for cooling, filtration and heating to handle the normal thermal and air cleaning loads during normal plant operation are presented in Table 5.3-1. In certain cases where engineered safeguards functions also are served by the equipment, component sizing is determined from the heavier duty specifications associated with the design basis accident detailed further in Section 6.4.

5.3.2 System Design

5.3.2.1 Piping and Instrumentation Diagram

The containment ventilation, purging, and recirculation cooling and filtration systems flow diagram is shown in Plant Drawing 9321-4022 [Formerly UFSAR Figure 5.3-1]. The containment ventilation systems and main plant vent are designed as Class I structures.

5.3.2.2 Containment Cooling and Ventilation System

Air recirculation cooling during normal operation is accomplished using air handling units discharged into a common header ductwork distribution system to ensure adequate flow of cooled air throughout the containment. The cooling coils in each air handling unit transfer up to 61.7 x 10^6 Btu/hr in the event of an accident when supplied with approximately 1600 gpm cooling water at 95°F inlet temperature and steam-air flowrate of 64,500 cfm.

Each air-handling unit consists of the following equipment arranged so that, during normal and accident operation, air flows through the unit in the following sequence: cooling coils, moisture separators (demisters), centrifugal fan with direct-drive motor, and distribution header. The fans and motors of these units are equipped with vibration sensors to detect abnormal operating conditions in the early stages of the disturbance. The normal air flow rate per air-handling unit is approximately 70,000 cfm. Section 6.4.2 provides additional information on the operation of this system.

The following additional systems supplement the main containment recirculation system:

1. Control rod drive cooling system consisting of fans and ductwork to circulate air through the control rod drive mechanism shroud and discharge it to the main containment volume. Four direct driven axial flow fans are provided for use. There are two power supplies for each fan.

2. Two unit steam heaters are located in containment to provide additional area heating as required. The containment purge supply is also provided with steam pre-heating to supplement containment heating as required.

5.3.2.3 Containment Purge System

The containment purge system is independent of the primary auxiliary building exhaust system, (except for the common exhaust fans) and includes provisions for both supply and exhaust air. The supply system includes roughing filters, heating coils, fan, supply penetration with two butterfly valves for bubble tight shutoff, and a purge supply distribution header inside containment. The exhaust system includes exhaust penetration with two butterfly valves identical to those above, exhaust ductwork, filter bank with roughing, HEPA and charcoal filters, fans and exhaust vent. The purge supply and exhaust flow rates are nominally 23,000 cfm and 25,000 cfm respectively. The quick closing purge isolation valves close upon receipt of an accident signal. Allowable closure times for these valves are specified in Section 5.2.4.

During power operation, the purge system is routinely not operated. Prior to purging the containment air, particulate and gas monitor indications of the closed containment activity levels will be used to guide routine releases from the containment. During power operation, the containment air particulate and gas monitor indications will help determine desirability of using either one or both of two auxiliary particulate and charcoal filter units installed in the containment primarily for preaccess cleanup.

When containment purging is in progress for access following reactor shutdown, releases from the plant vent are continuously monitored with a gas monitor, as described in Section 11.2.

5.3.2.4 Purge System Isolation Valves

The purge supply and exhaust duct butterfly valves, both inside and outside the containment, are normally closed during power operation. They may be opened for safety related reasons, i.e., pressure control or to facilitate safety related surveillance or maintenance. The opening angle is limited during operation so that the valves can close against a differential pressure (see Section 5.2.5). The spaces between the closed valves are pressurized with air by the weld channel and penetration pressurization system. The valves are designed for rapid automatic closing by the containment isolation signal (derived from any safety injection signal), upon a signal of high activity level within the containment in the event of a radioactivity release when the purge line is open, or upon a manually initiated signal. To ensure optimum sealing of the resilient valve seats, the two valves located outside containment are enclosed and a minimum ambient temperature is maintained.

5.3.2.5 Containment Pressure Relief Line

The normal pressure changes in the containment during reactor power operation, and during plant cooldown if the containment purge system is not operating, will be handled by the containment pressure relief line. This line is equipped with three quick-closing butterfly type isolation valves, one inside and two outside the containment. The valves will be automatically actuated to the closed position by the containment isolation signal, by a containment high radioactivity signal, or upon a manually initiated signal. Each of these air operated valves is equipped with an accumulator to assure each can close even if the air supply is lost. The two intra-valve spaces are pressurized with air by the Weld Channel and Penetration Pressurization System when the valves are closed. The pressure relief line discharges to the plant vent. The opening angle of the pressure relief valves is limited during operation so the valves can close against a differential pressure.

5.3.2.6 Containment Purge and Pressure Relief Isolation Reset

Opening of the purge and pressure relief isolation valves following an isolation signal requires deliberate operator action by resetting all isolation signals and depressing both Containment Ventilation Isolation reset push buttons. Further, in order to reset the Containment Ventilation Isolation signal for Train B, the control switches for the purge and pressure relief isolation valves in Train B must first be placed in the closed position. In addition, guard plates are placed over the reset buttons. These three features preclude the possibility of inadvertently opening these valves.

REFERENCES FOR SECTION 5.3

1. Deleted

TABLE 5.3-1 (Sheet 1 of 2)
Containment Cooling and Ventilation System - Principal Component Data Summary

System	Units Installed	Unit Capacity	Units Required for Normal Operation
Containment cooling and recirculation			
Demister	5	72,500 cfm	5_2
Cooling coils - normal	5	2.20 x 10^6 Btu/hr$_1$	5_2
Cooling coils - DBA	5	61.7 x 10^6 Btu/hr$_3$	
Fans	5	72,500 cfm	5_2
Fan pressure	-	7.21-in. H_2O (Note 7)	
Fan motors (440 V, 3-phase)	5	350 hp	5_2
Control rod drive mechanism cooling			
Fans, standard conditions	4	15,000 cfm	3
Fan pressure	-	5.5-in. H_2O	
Fan motors	4	30 hp	3
Reactor compartment cooling			
Part of containment recirculation system	-	12,000 cfm	
Refueling canal air sweep			
Part of containment recirculation system	-	17,500 cfm	
Purge supply	1	23,000 cfm$_6$	Optional
Fan pressure	-	2.5-in. H_20	
Fan motors	1	40 hp	
25 psig steam preheat coils	1 set	3 x 10^6 Btu/hr	Optional
Air filters, roughing	-	-	1

TABLE 5.3-1 (Sheet 2 of 2)
Containment Cooling and Ventilation System - Principal Component Data Summary

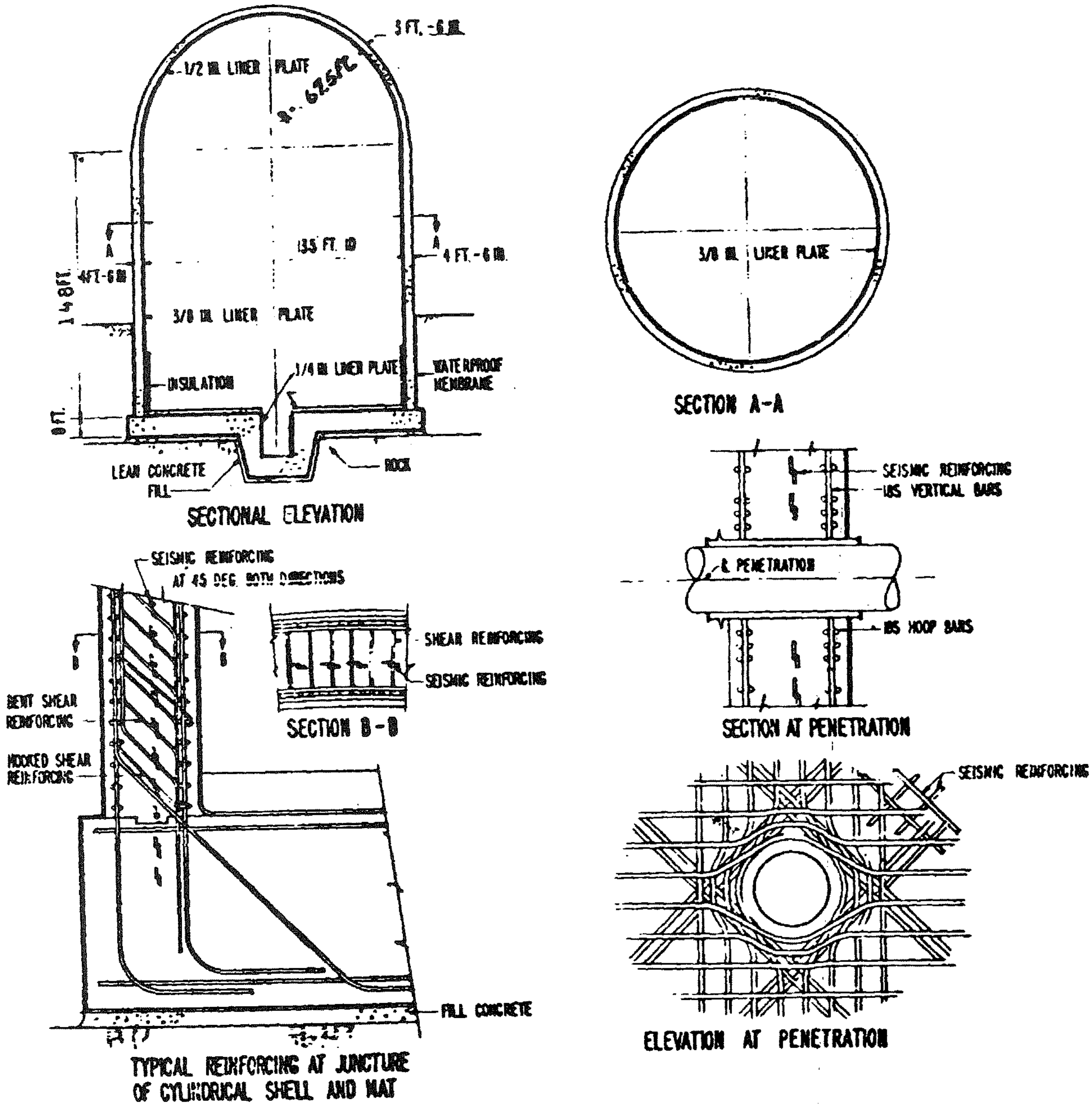
System	Units Installed	Unit Capacity	Units Required for Normal Operation

Purge exhaust			
Fans, standard conditions	2[4]	55,500 cfm[6]	Optional
Fan pressure	-	10.3-in. H_2O	Optional
Fan motors	2	125 hp	Optional
Plenums	1	-[5]	Optional
HEPA filters	-	-	Optional
HECA filters (charcoal adsorbers)	1	-	Optional
Containment auxiliary charcoal filter			
Fans, standard conditions	2	8,000 cfm	Optional
Fan pressure	-	5.0-in. H_20	Optional
Fan motors	2	10 hp	Optional
Filters and charcoal filters; roughing, HEPA	2	8,000 cfm	Optional
Steam heating			
Heaters, 25 psig steam	2	400,000 Btu/hr each	Optional

Notes:
1. This value reflects the increase in air side flow rate due to removal of the original plant HEPA Filters.
2. Depends on time of year and containment atmospheric temperature.
3. Based on minimum assumed performance at 271°F containment temp 95°F Service Water temp, 1600 gpm Service Water flow and 64,500 cfm air flow rate.
4. The two exhaust fans are used interchangeably or as backup for:
   1. Ventilation of primary auxiliary building.
   2. Containment building purge system.
5. Purge (25,000 cfm) and primary auxiliary building exhaust (55,500 cfm) are fed into a common plenum.
6. Purge supply (23,000 cfm) and purge exhaust (25,000 cfm) are the nominal, as built, flow rates for the purge system (± 10%).
7. At 72,500 cfm flow rate.

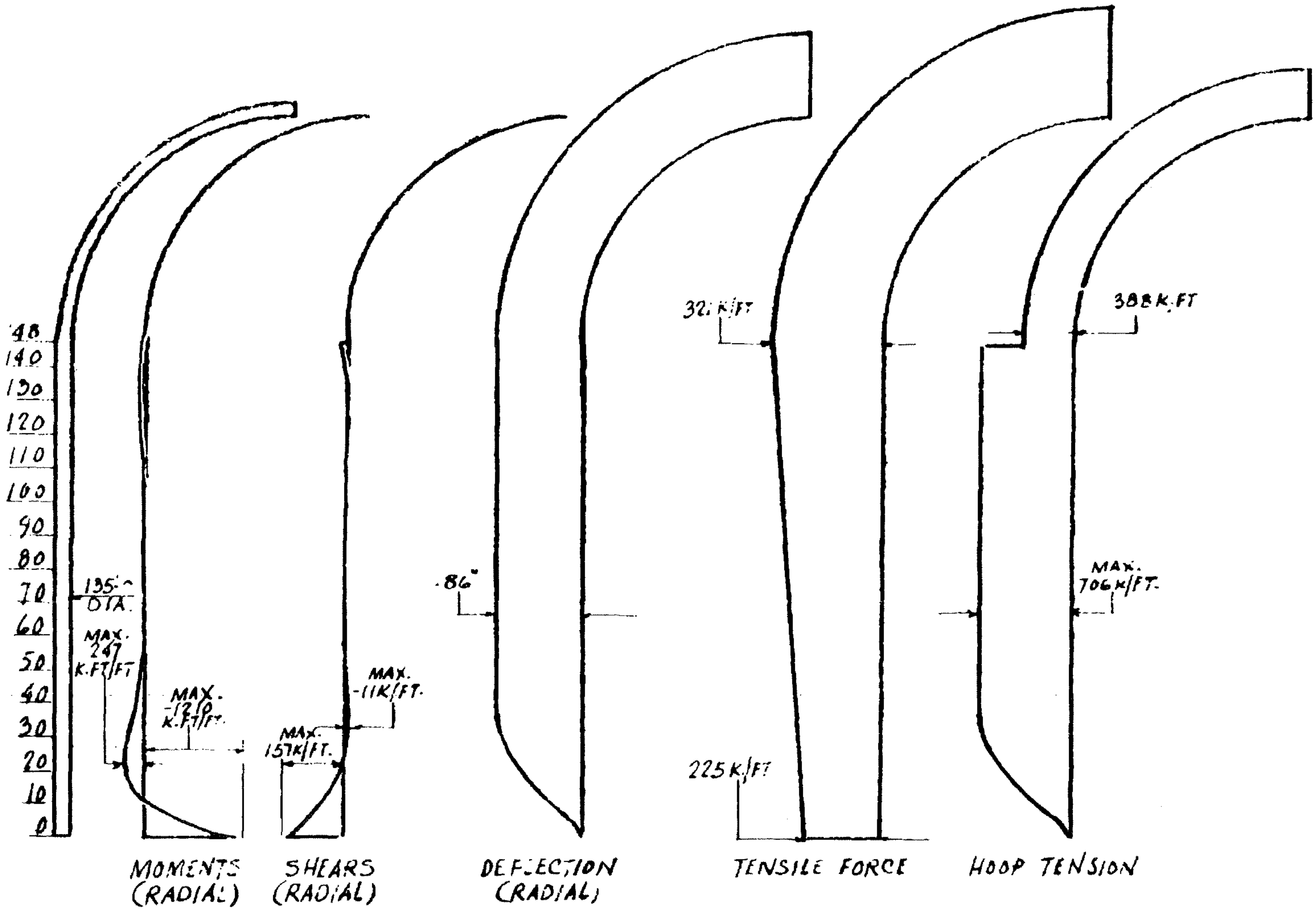
5.3 FIGURES

Figure No.	Title
Figure 5.3-1	Containment Cooling and Ventilation System [Replaced with Plant Drawing 9321-4022]



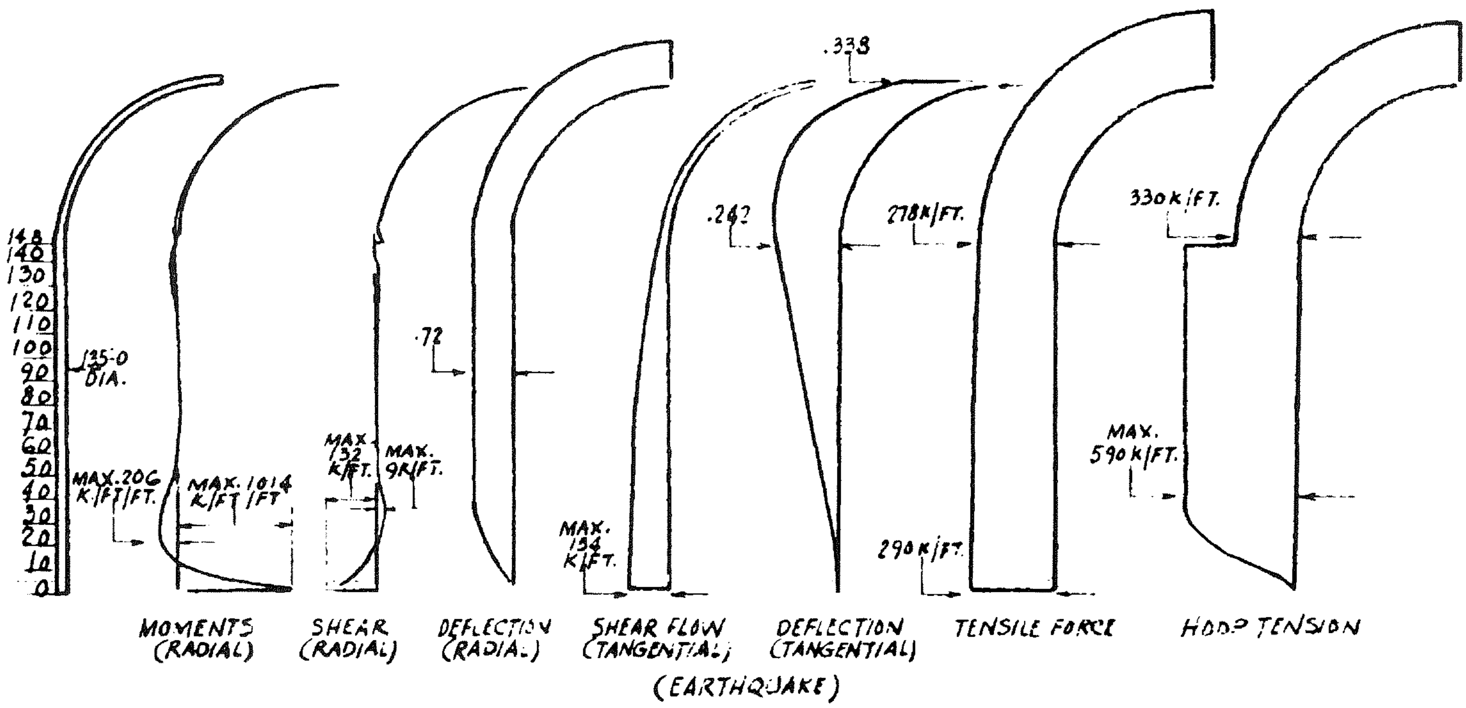
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-1 CONTAINMENT STRUCTURE	
MIC. No. 1999MC3746	REV. No. 17A

$C = 1.0D \pm 0.05D + 1.5P + 1.0(T+TL)$



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-11 CYLINDER AND DOME-LOAD CONDITION (A) - 1.5P	
MIC. No. 1999MC3750	REV. No. 17A

$$C = 1.0D \pm 0.05D + 1.25P + 1.0(T' + TL') + 1.25E$$



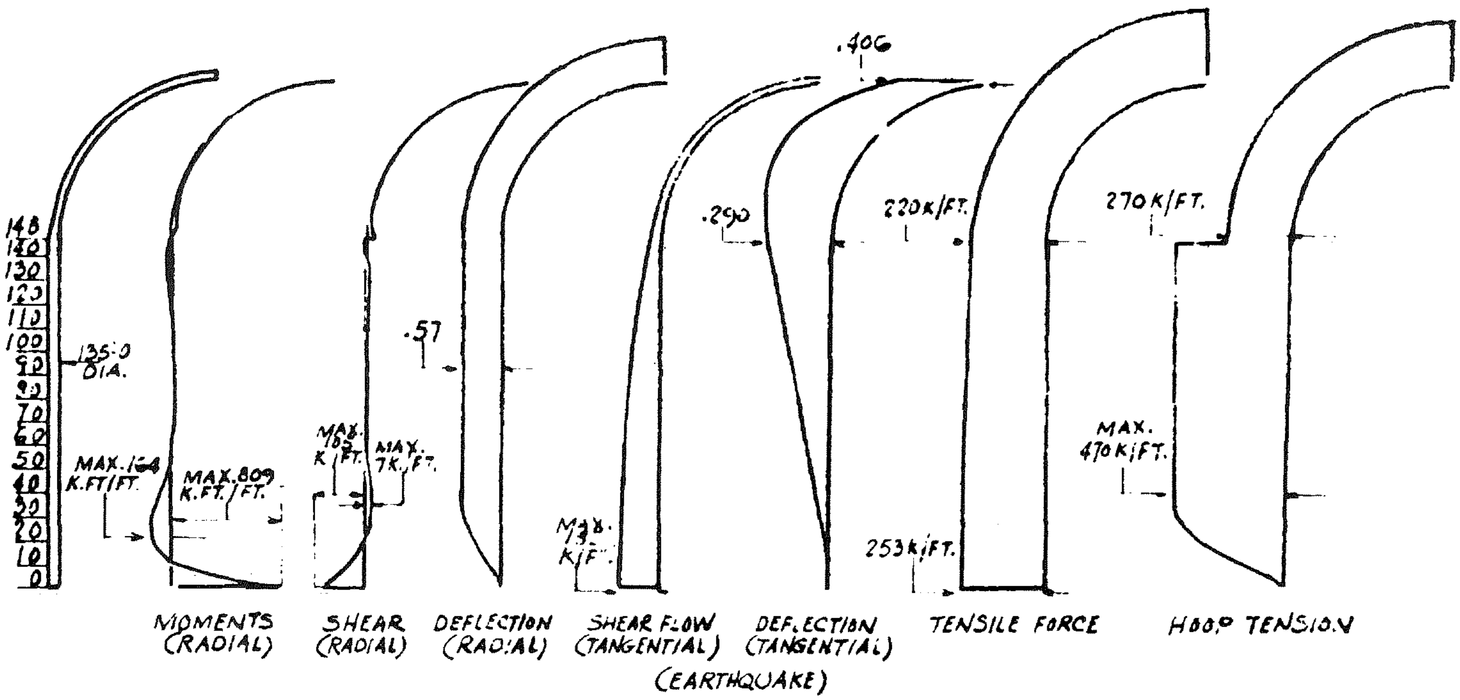
INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1-12

CYLINDER AND DOME-LOAD CONDITION (B) - 1.25P

MIC. No. 1999MC3751	REV. No. 17A

$C = 1.0D \pm 0.05D + 1.0P + 1.0(T'' + TL'') + 1.0E'$

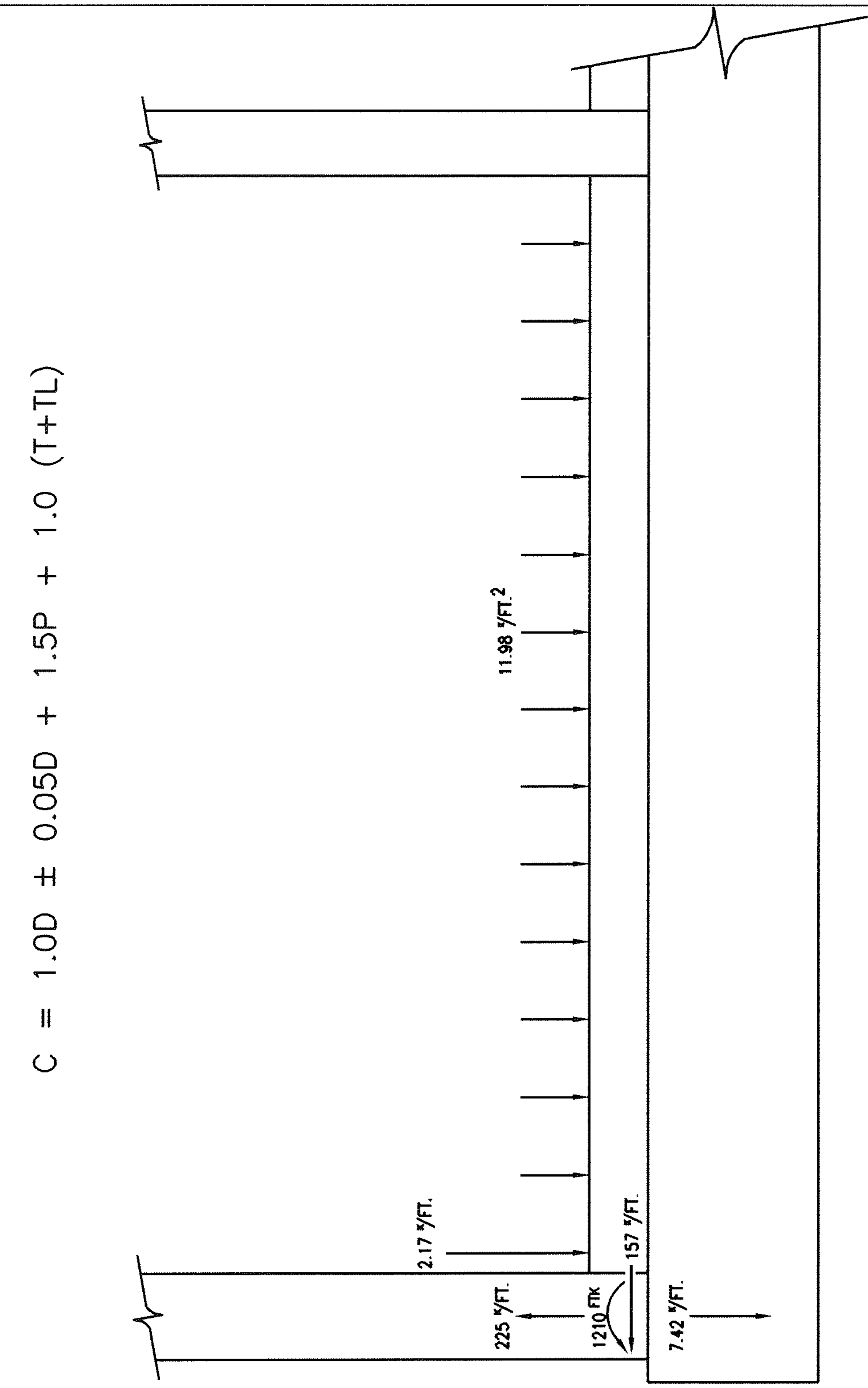


INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1–13

CYLINDER AND DOME–LOAD CONDITION (C) – 1.0P

MIC. No. 1999MC3752	REV. No. 17A

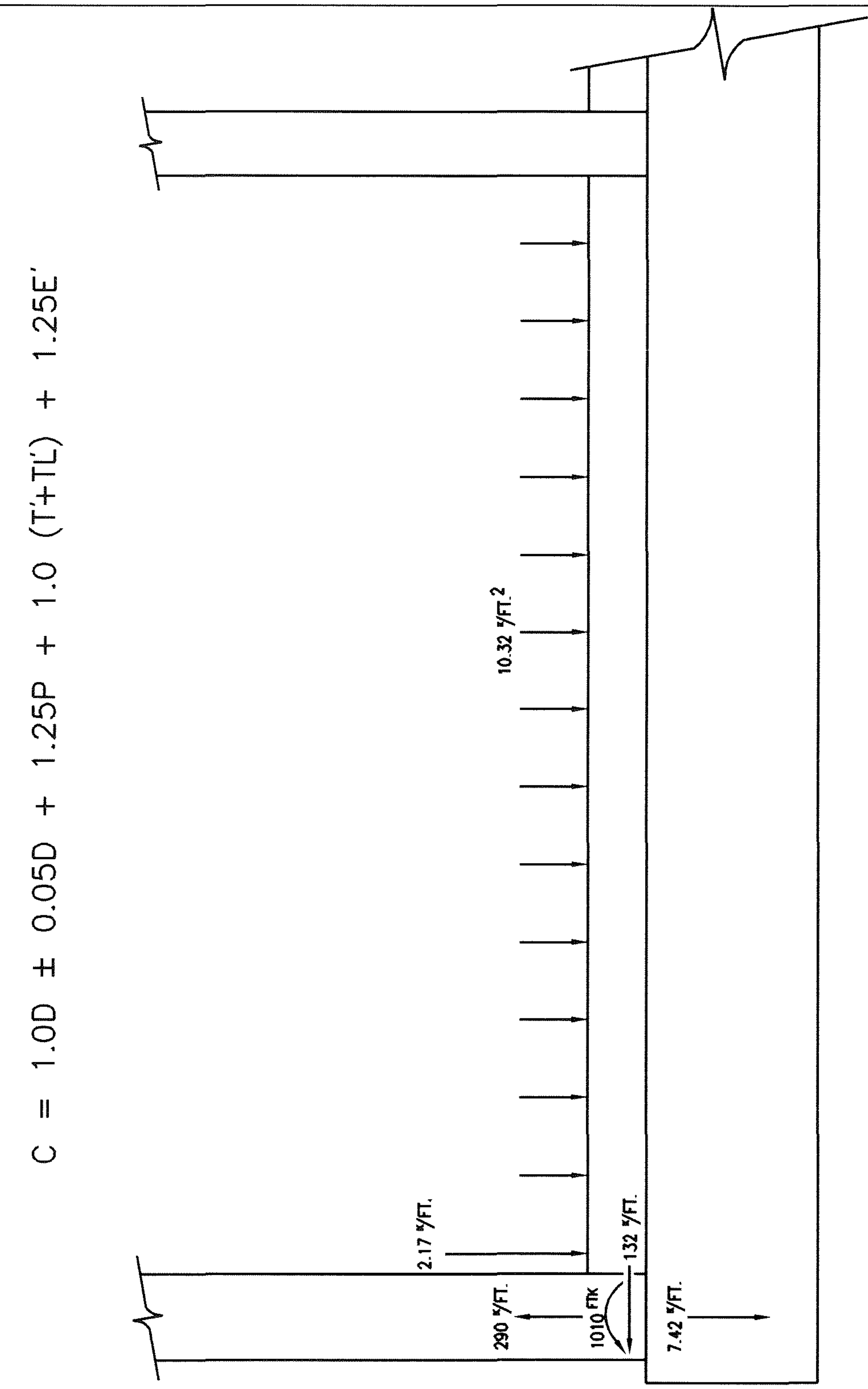


INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1-14

LOADING DIAGRAM IN MAT-LOAD CONDITION (A) - 1.5P

MIC. No. 1999MC3753	REV. No. 17A

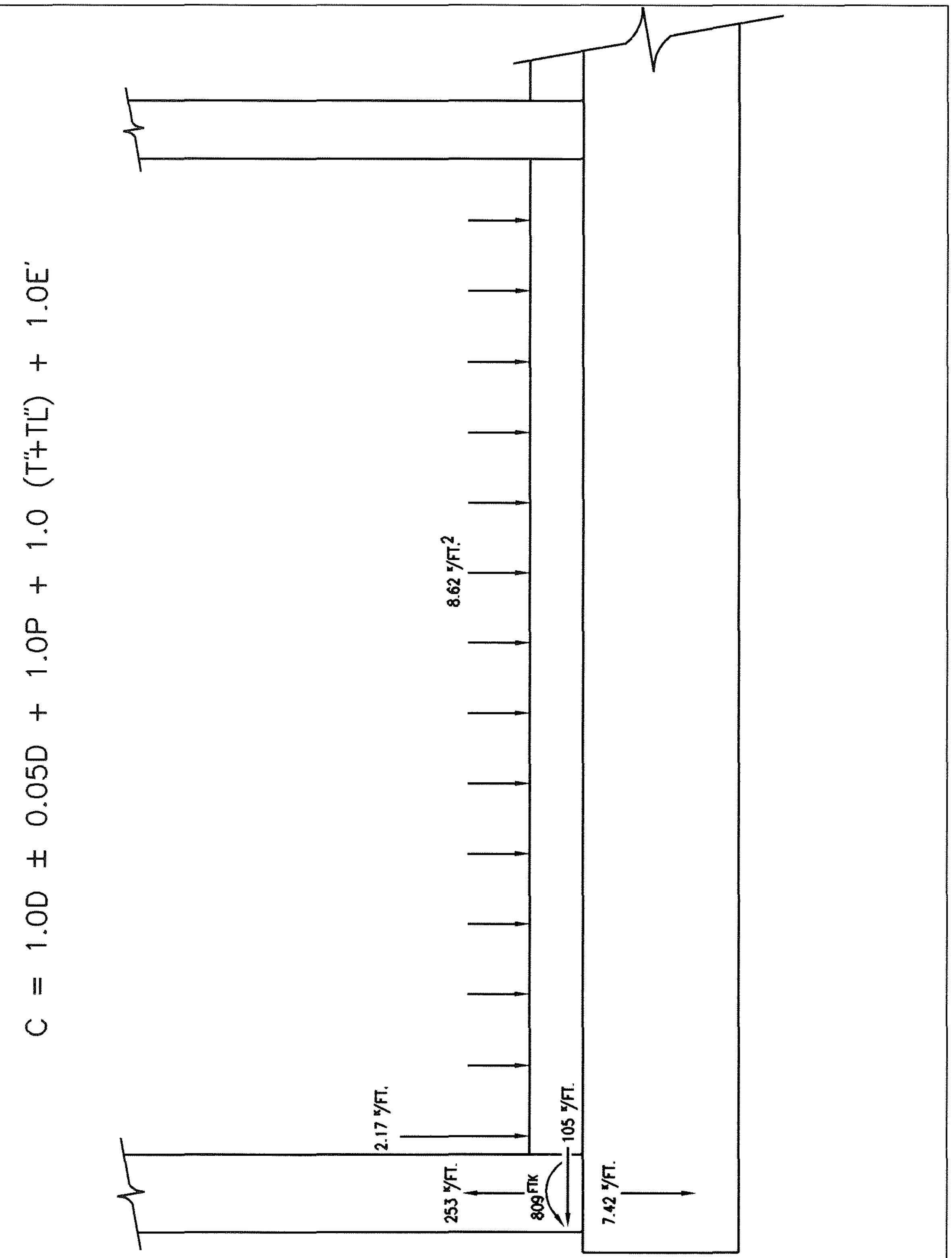


INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1-15

LOADING DIAGRAM IN MAT-LOAD CONDITION (B) - 1.25P

MIC. No. 1999MC3754	REV. No. 17A

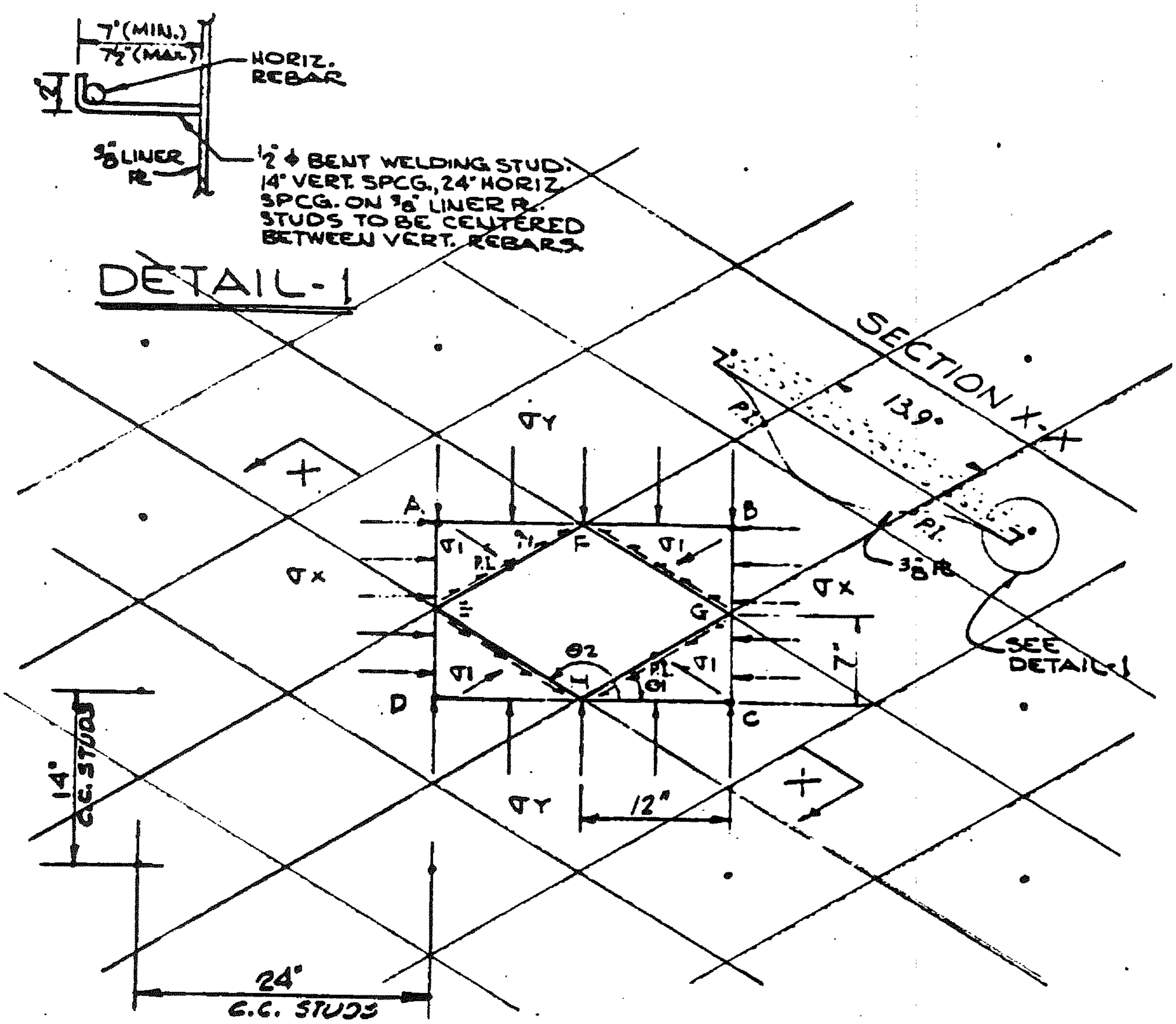


INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1–16

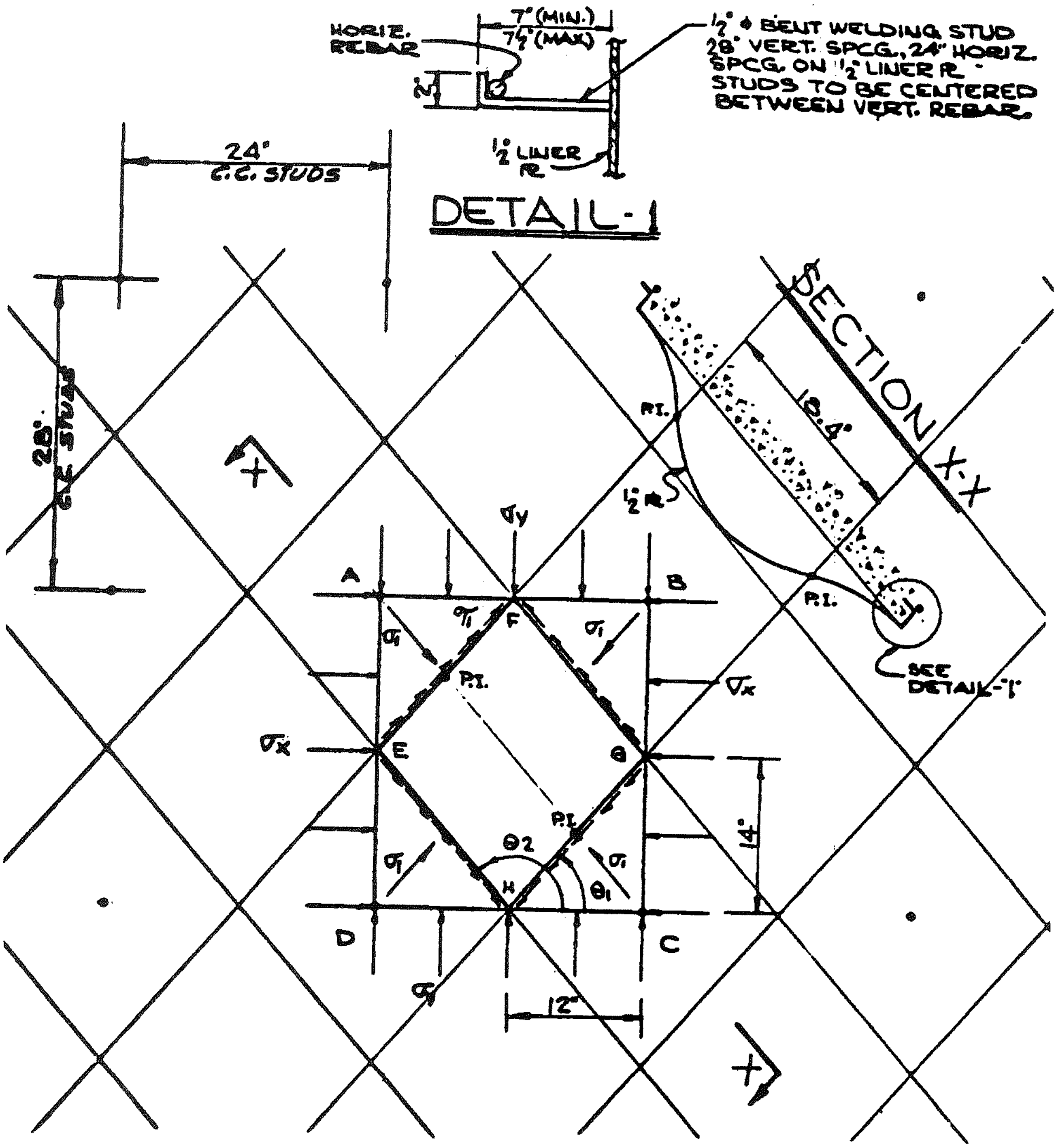
LOADING DIAGRAM IN MAT–LOAD CONDITION (C) – 1.0P

MIC. No. 1999MC3755	REV. No. 17A



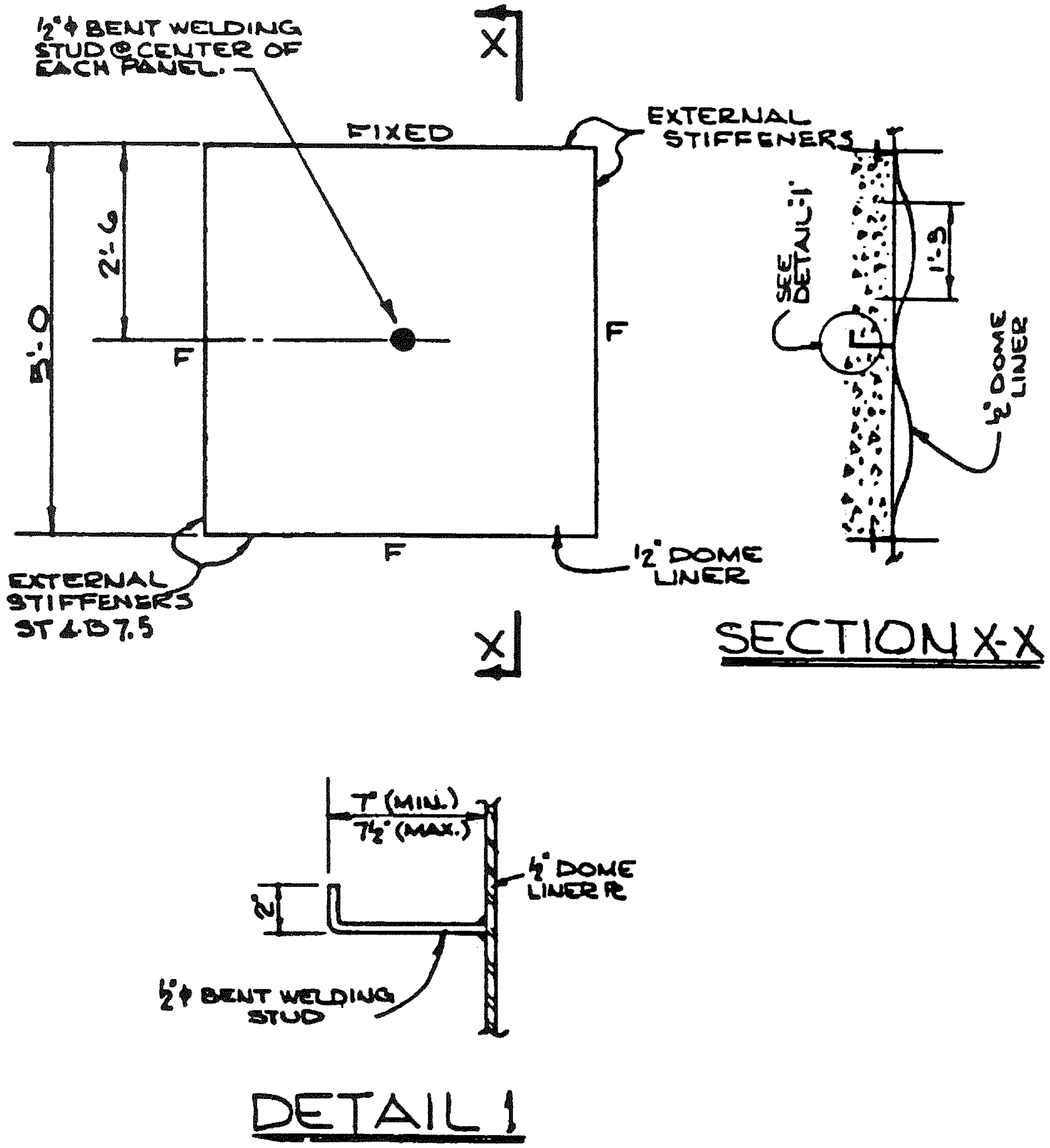
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-17 WELD STUD CONNECTION AT PANEL LOW POINT	
MIC. No. 1999MC3756	REV. No. 17A

OAGI0000215_0634

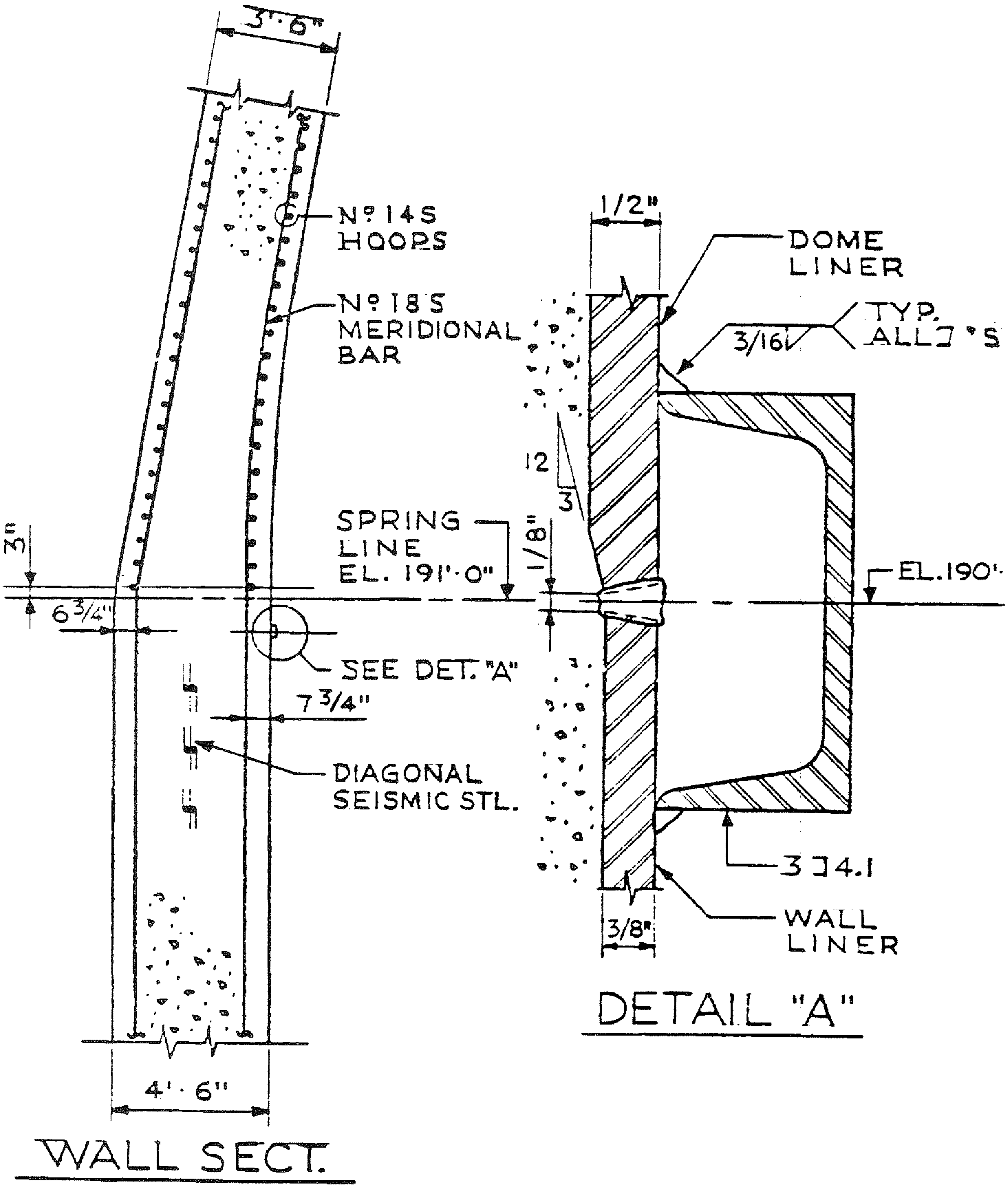


INDIAN POINT UNIT No. 2
UFSAR FIGURE 5.1–18
WELD STUD CONNECTION AT PANEL LOW POINT

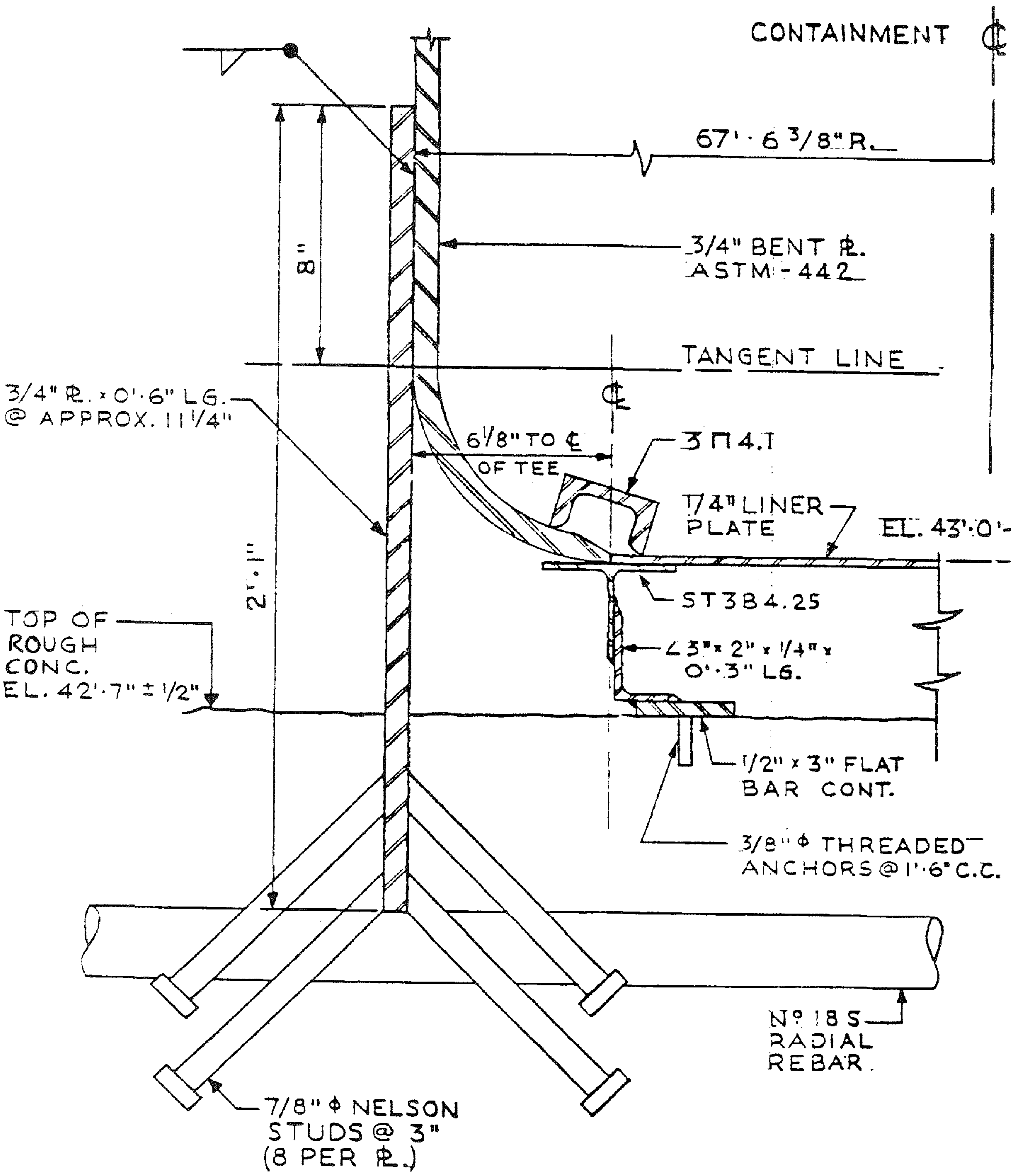
MIC. No. 1999MC3757	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-19 WELD STUD CONNECTION AT PANEL CENTER	
MIC. No. 1999MC3758	REV. No. 17A



INDIAN POINT UNIT No. 2
UFSAR FIGURE 5.1-20 WALL SECTION
MIC. No. 1999MC3759 | REV. No. 17A

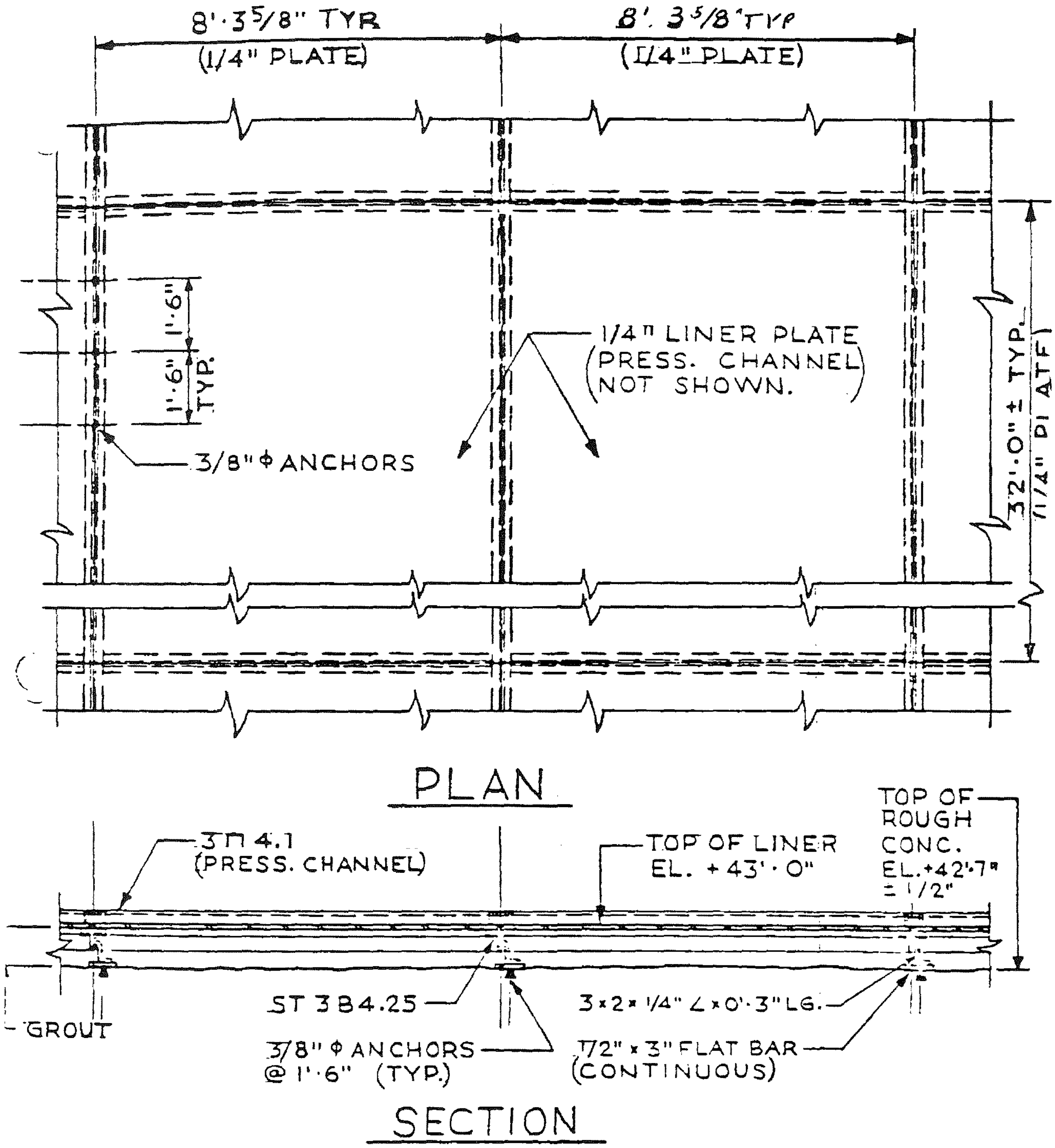


INDIAN POINT UNIT No. 2

UFSAR FIGURE NO. 5.1–21

CYLINDER BASE SLAB LINER JUNCTURE

MIC. No. 1999MC3760	REV. No. 17A

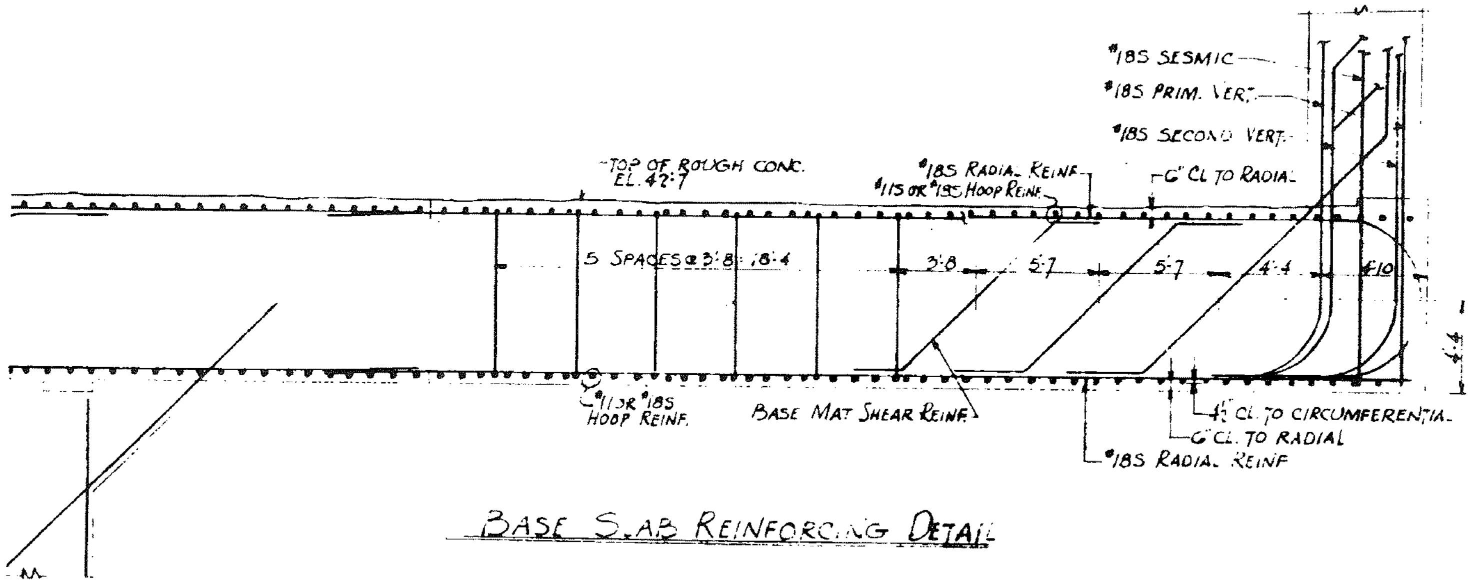


INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1–22

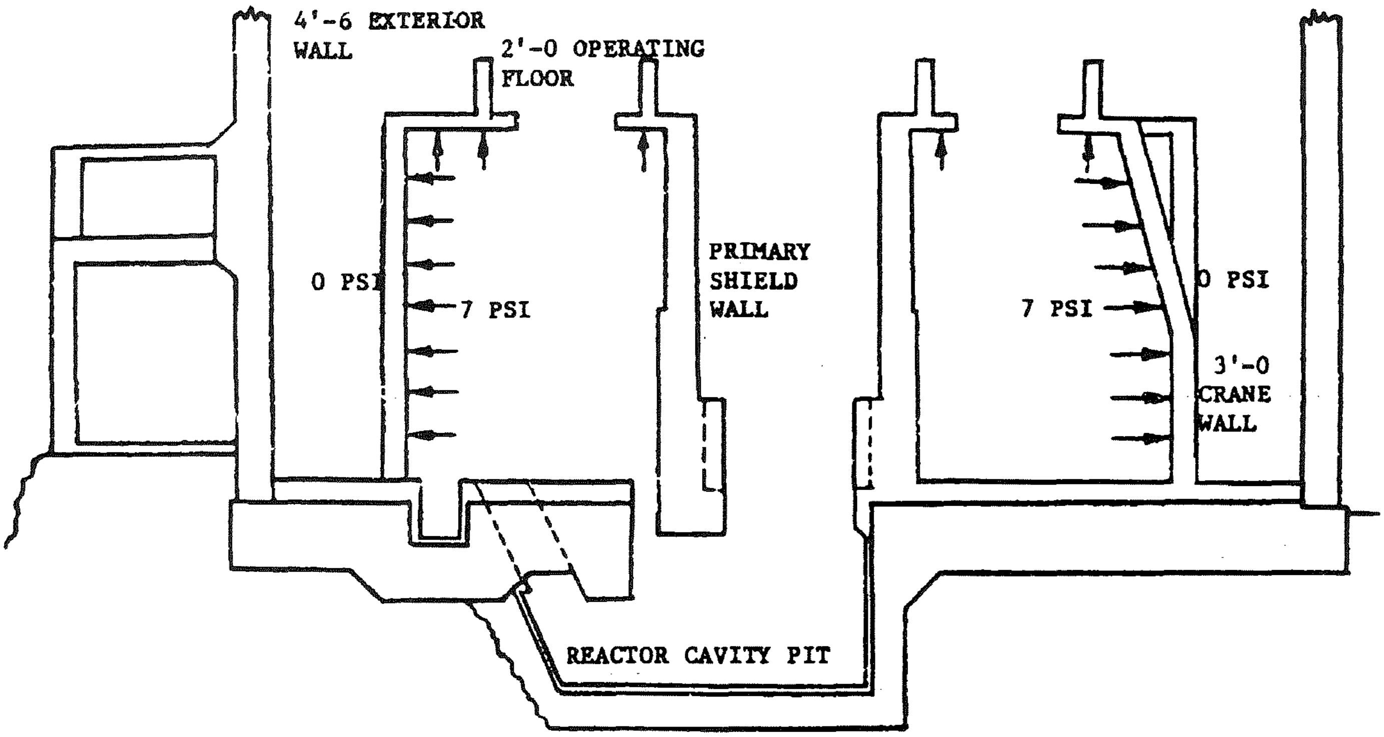
TYPICAL BASE MAT
LINER DETAIL

MIC. No. 1999MC3761	REV. No. 17A



BASE SLAB REINFORCING DETAIL

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-23 BASE SLAB REINFORCING DETAIL	
MIC. No. 1999MC3762	REV. No. 17A

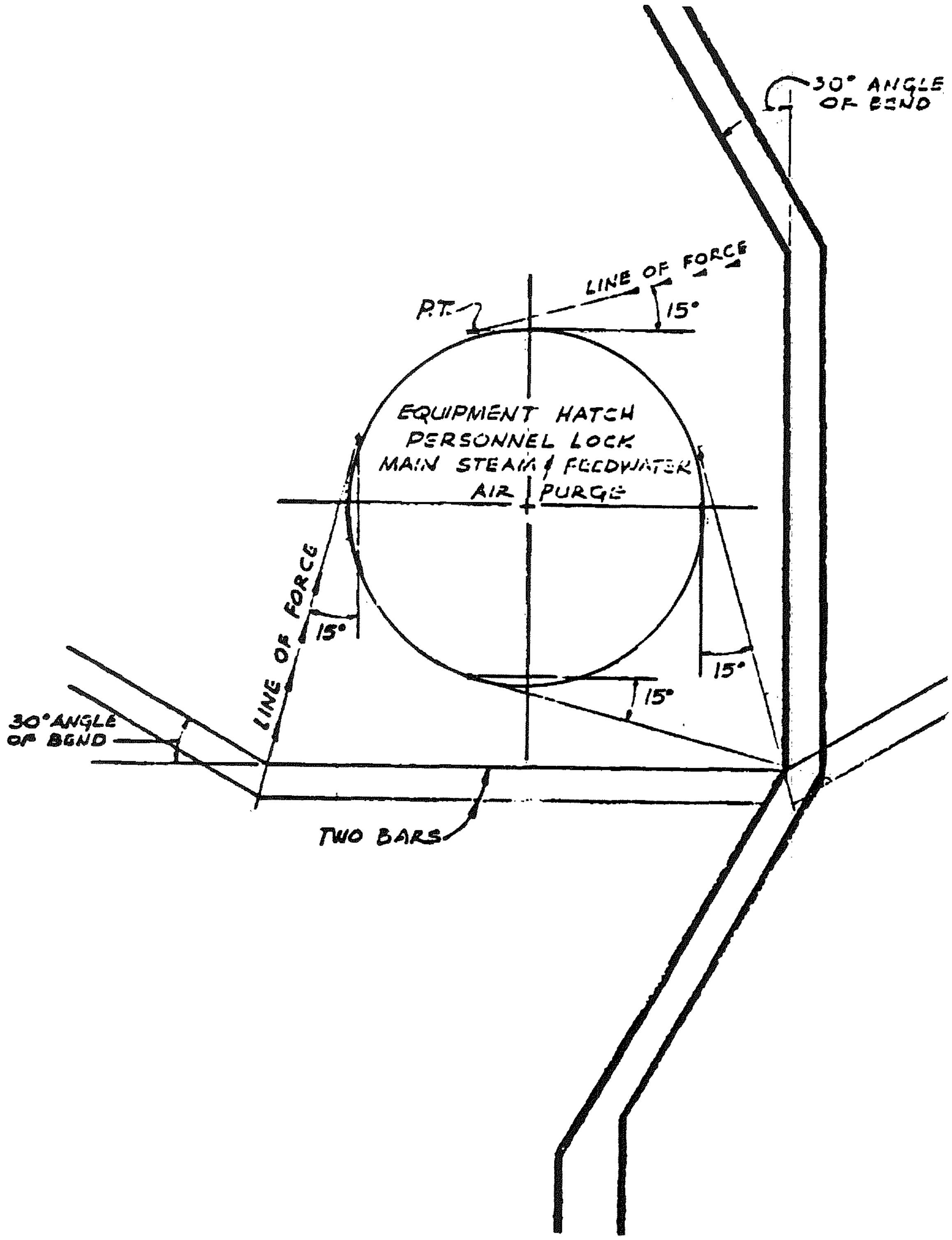


INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1-24

REACTOR CAVITY PIT

MIC. No. 1999MC3763 | REV. No. 17A

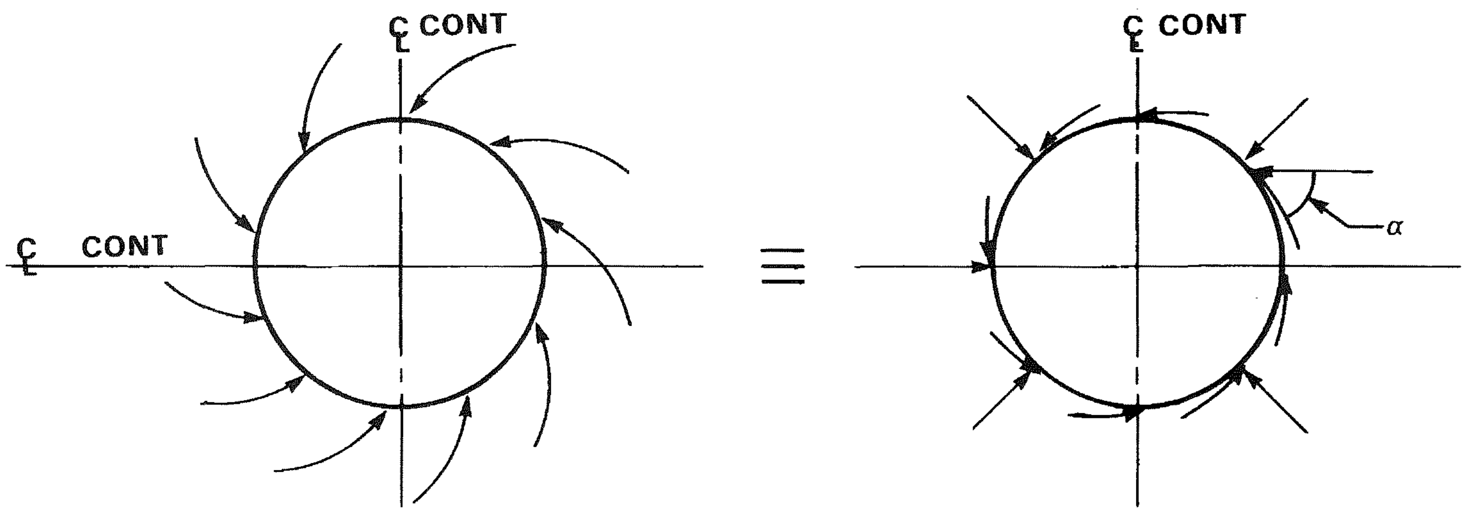


INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1-25

EQUIPMENT HATCH PERSONNEL LOCK,
MAIN STEAM AND FEEDWATER,
AIR PURGE – REBAR

MIC. No. 1999MC3764	REV. No. 17A

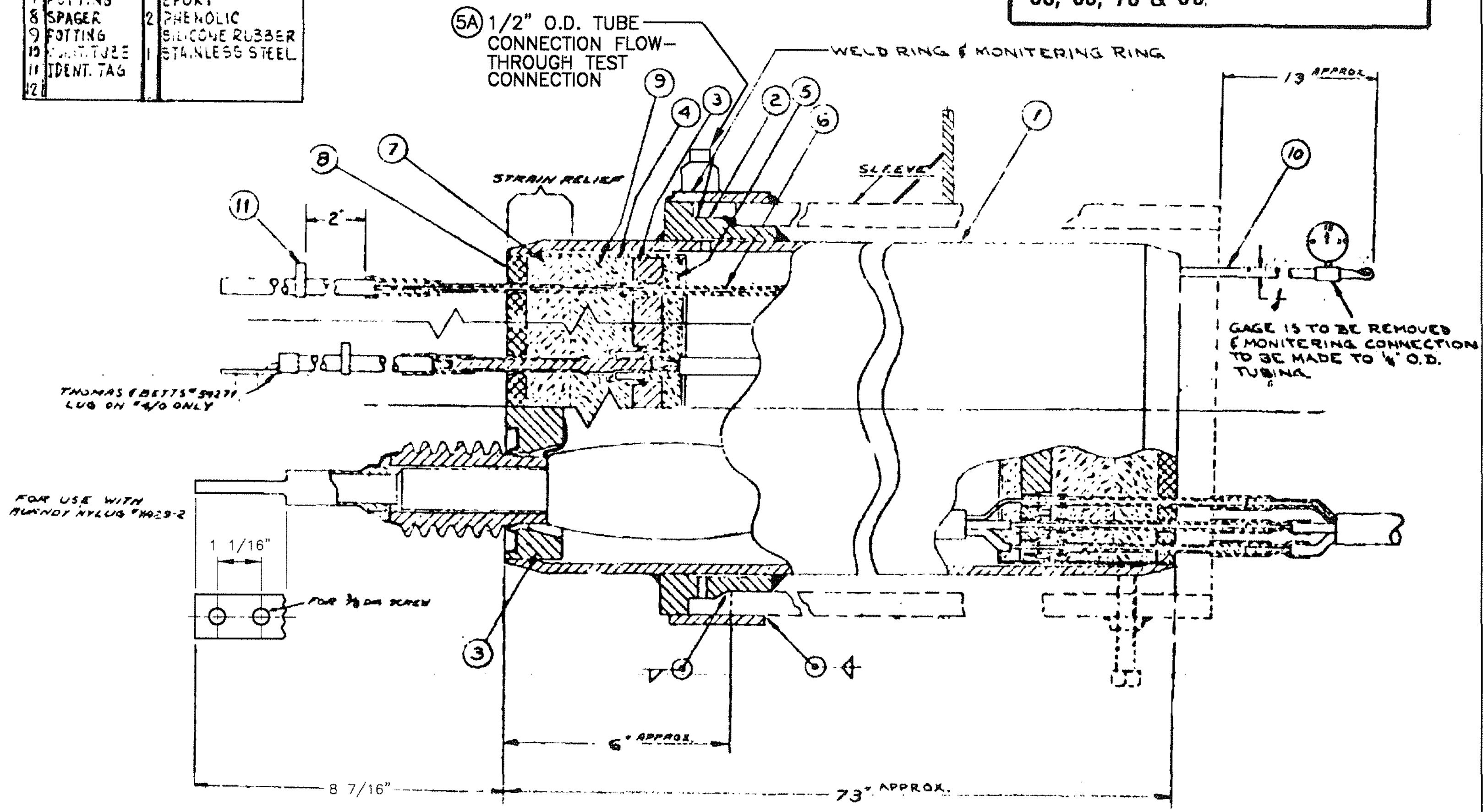


INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-26 TORSIONAL EFFECTS	
MIC. No. 1999MC3765	REV. No. 17A

THIS DWG. PERTAINS TO PENETRATIONS:

H11, 12, 13, 14, 15, 16, 17, 18, 19, 21, 22, 24, 26, 29, 31, 33, 34, 36, 37, 38, 39, 41, 42, 43, 44, 45, 47, 48, 49, 50, 51, 2, 53, 54, 55, 56, 57, 58, 59, 60, 61, 63, 64, 66, 67, 68, 69, 70 & 65.

NO	PART NAME	QY	DESCRIPTION
1	CONTAINER	1	STAINLESS STEEL
2	FLANGE	1	CARBON STEEL
3	HEADER	2	STAINLESS STEEL
4	TUBE	2	STAINLESS STEEL
5	POTTING		SILICONE RUBBER
6	INSULATION		SILICONE SLEEVING
7	POTTING		EPOXY
8	SPACER	2	PHENOLIC
9	POTTING		SILICONE RUBBER
10	[illegible] TUBE	1	STAINLESS STEEL
11	IDENT. TAG		
12			



INDIAN POINT UNIT No. 2

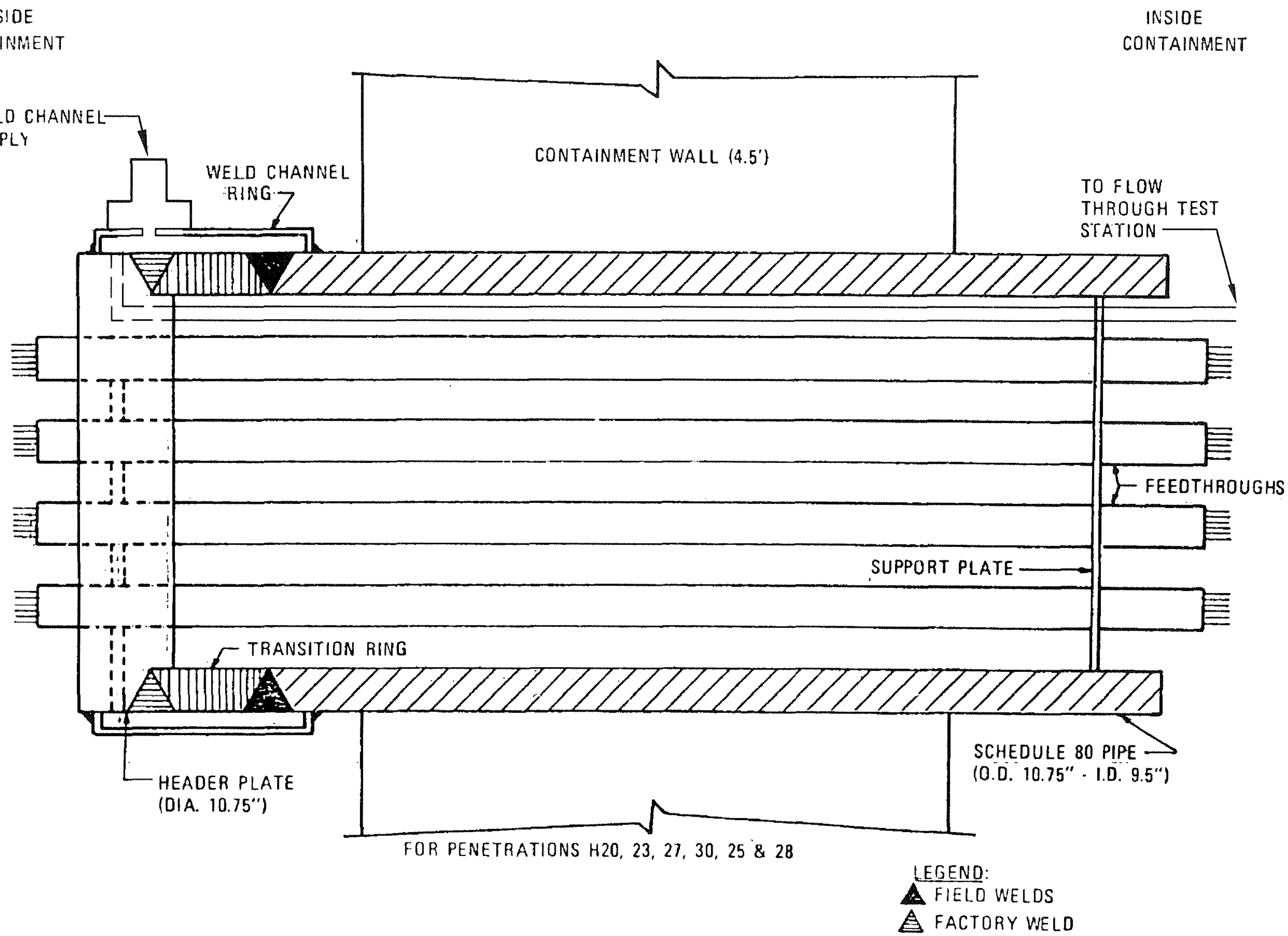
UFSAR FIGURE 5.1-27

TYPICAL ELECTRICAL PENETRATION

MIC. No. 1999MC3766

REV. No. 17A

OAGI0000215_0644

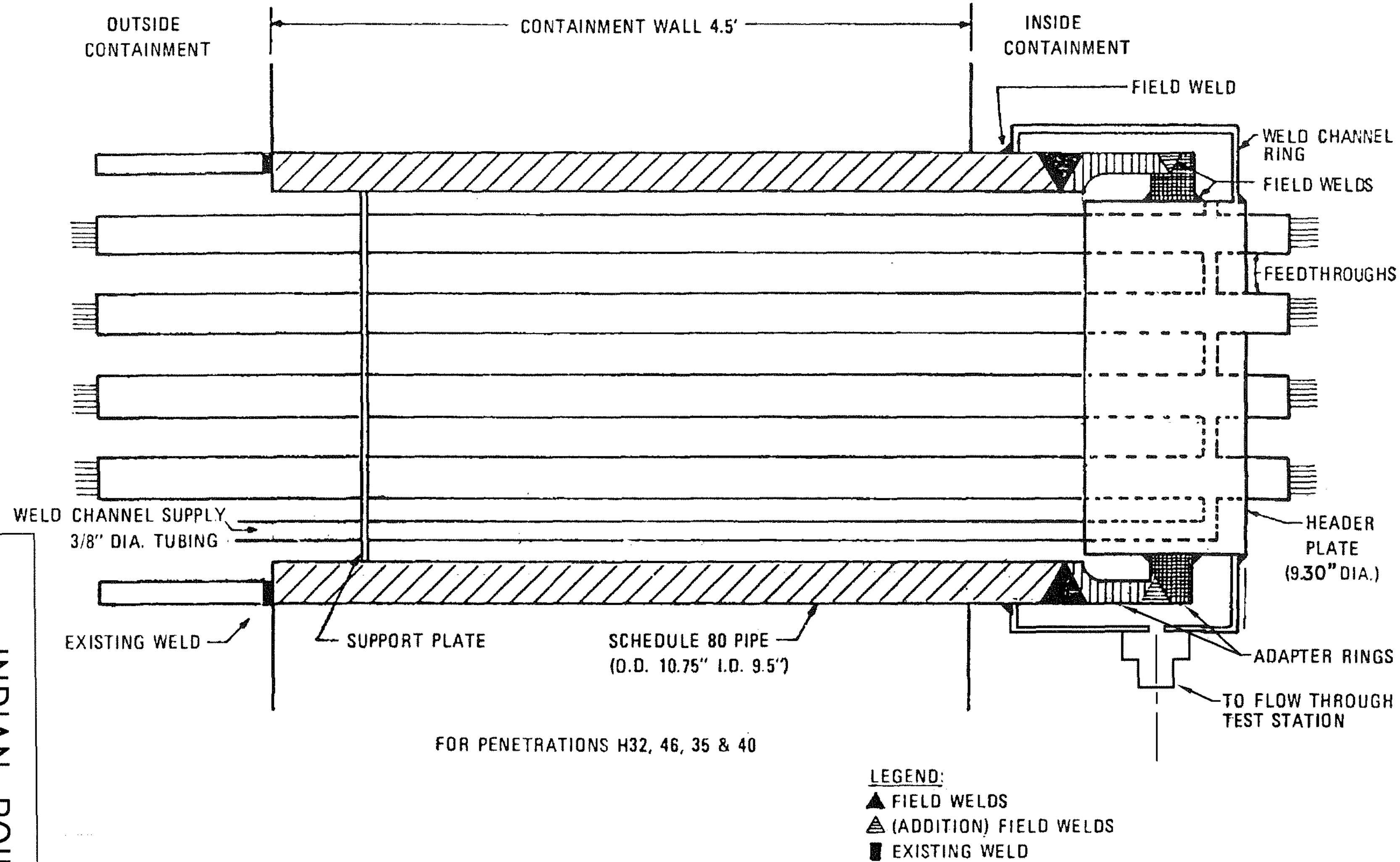


INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1-28

CONAX PENETRATIONS – OUTSIDE CONTAINMENT WELD

MIC. No. 1999MC3767	REV. No. 17A

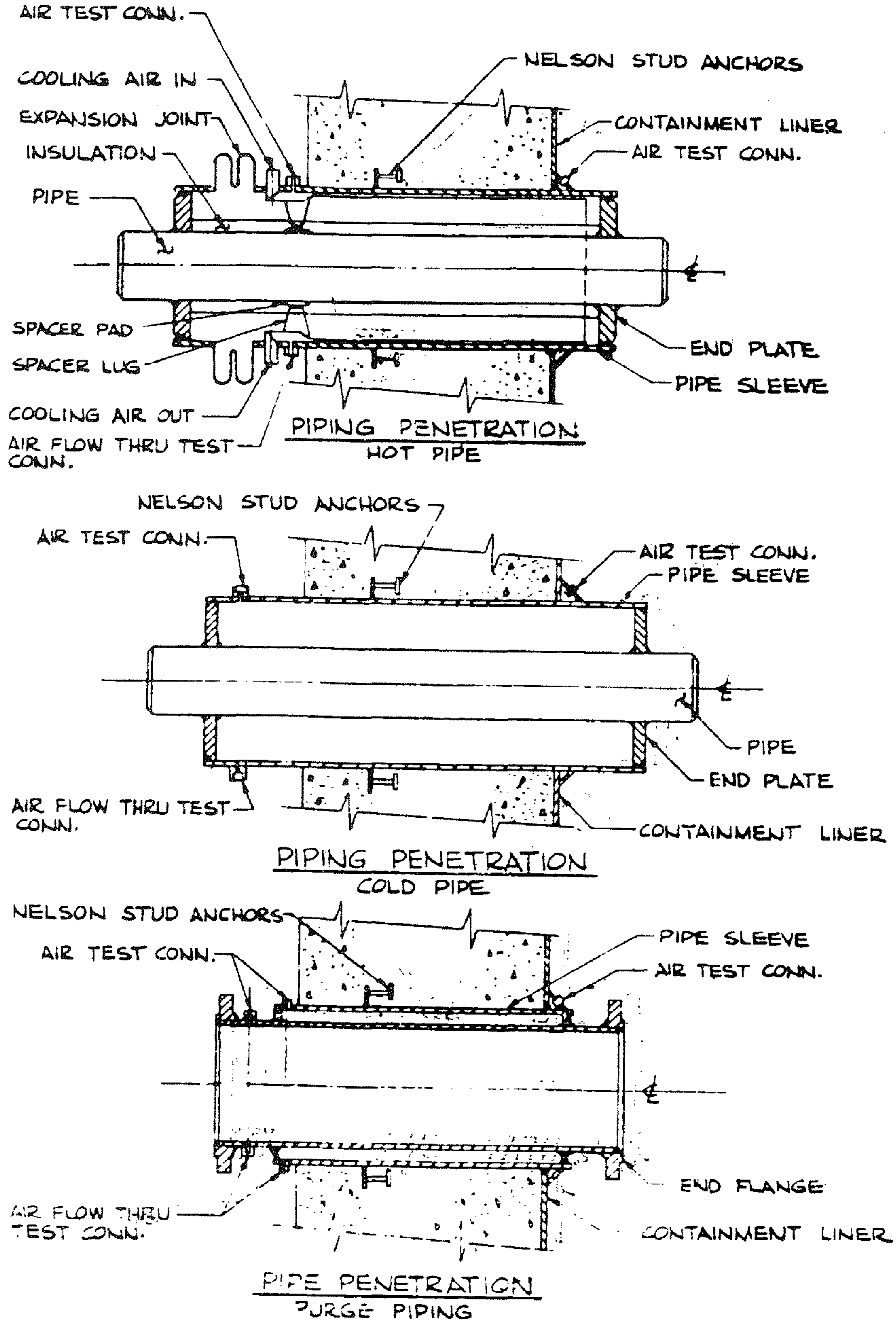


INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1-29

CONAX PENETRATIONS – INSIDE CONTAINMENT WELD

MIC. No. 1999MC3768 | REV. No. 17A

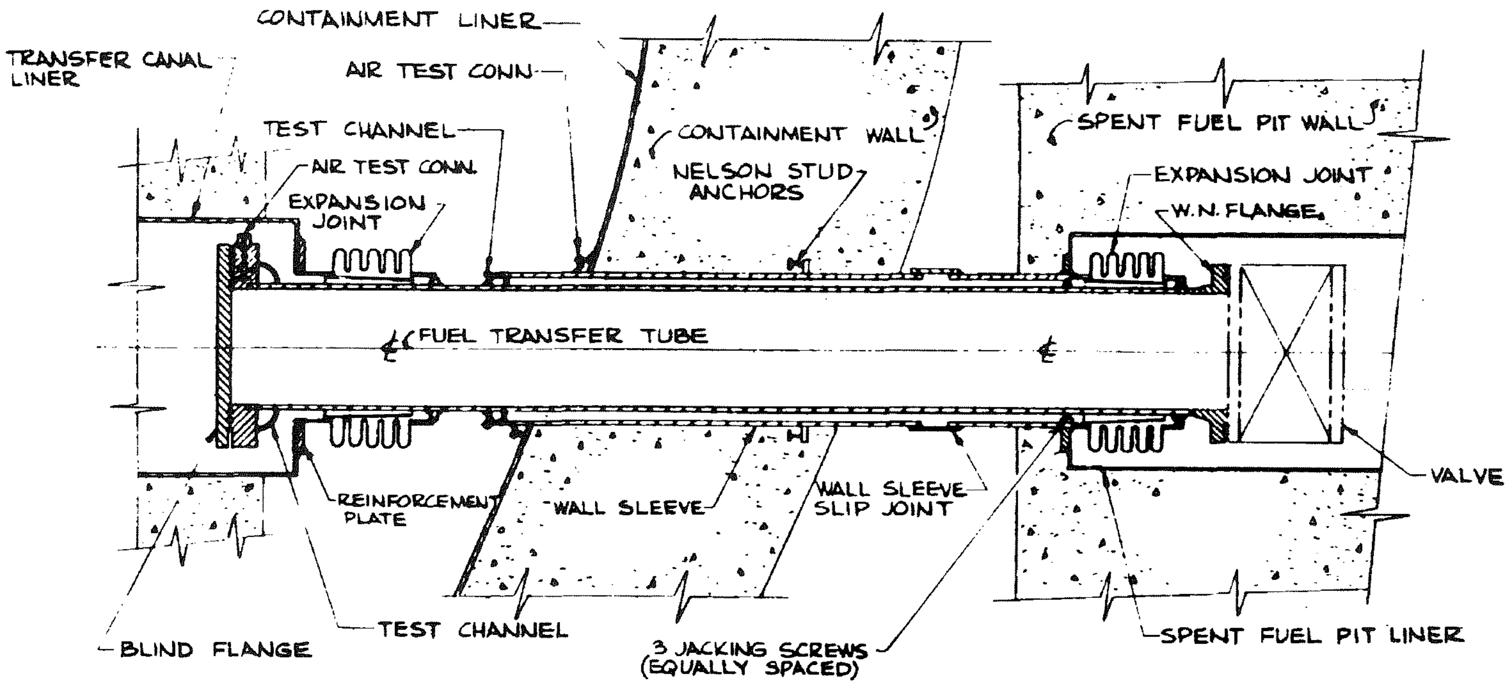


INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1–30

TYPICAL PIPING PENETRATION

MIC. No. 1999MC3769	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-31 FUEL TRANSFER TUBE PENETRATION (CONCEPTUAL DRAWING)	
MIC. No. 1999MC3770	REV. No. 17A

OAGI0000215_0648

(Sample calculation)

FORM 501

GENERAL COMPUTATION SHEET

UNITED ENGINEERS & CONSTRUCTORS INC.

NAME OF COMPANY Con Edison - Indian Point No 2 J. O. NO. 9321-06

SHEET NO. 6 OF

SUBJECT Containment - Stresses on Penetrations & Liner DATE

COMP. BY J.B.S. C'K'D BY F.X.M.

ELECTRICAL PENETRATIONS EL. 57'-0 D.A = 10.75

Sleeve t = .594

A) Compression V and Tension H

$$S' = \frac{(70.5)(837)(12)}{(1000)\,24.96}(.43) \text{ RESTRAINT}$$

$= 12.2\ ^{k}/in^2$ TENSION

Liner As/FT = (.75)(12) = 9.00

Seismic + 8(1/10)(1.414) = 2.26

Hoops + 16(1/12) = 13.70

24.96 □"/FT

$S \geq 20.2\ ^{k}/in^2$ COMPRESSION

$$\Delta_{un} = \frac{2}{3} \cdot \frac{5.375}{29 \times 10^3}\left[\overset{88.8}{5(20.2)} - 12.2\right] = 11.00 \times 10^{-3} = .011 \times .5 = 0.0055$$

(radial deformation)

$$\lambda = \sqrt[4]{\frac{3(1-.25)}{(5.375^2)(.594)^2}} = .724$$

$$S_{SLEEVE} = \frac{(.0055)(29 \times 10^3)(.750)(.724)}{\left[\underset{3.21}{(1-.25) + \frac{(.750)(5.375)(.724)}{2(.594)}}\right] 2(.594)} \quad 0.225 \times 10^3 = \underline{\underline{22.5^{ksi}}}$$

$$S_{LINER} = \frac{(.0055)(29 \times 10^3)}{5.375\left[\underset{3.21}{(1-.25) + \frac{(.750)(5.375)(.724)}{2(.594)}}\right]} = .00931 \times 10^3 = \underline{\underline{9.3^{ksi}}}$$

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-32	
CONTAINMENT-STRESSES ON PENETRATIONS AND LINER - SHEET 6	
MIC. No. 1999MC3771	REV. No. 17A

(Sample calculation)

FORM 501

GENERAL COMPUTATION SHEET

UNITED ENGINEERS & CONSTRUCTORS INC.

NAME OF COMPANY Con Edison - Indian Point #2 J. O NO. 9321-06 SHEET NO. 7 OF

SUBJECT Containment - Stresses on Penetrations & Liner DATE COMP. BY B.B.S. CH'K'D BY F.W.

ELECTRICAL PENETRATIONS

8) TENSION V & TENSION H

$S = 12.2 \, {}^{K}/in^2$

$S' \; \frac{297}{27.26} = 10.9 \, {}^{K}/in^2$

$As/FT \; (V)$

LINER $.75(12) = 9$

VERTICAL BARS $4 \times 4 = 16$

SEISMIC $8(\frac{13}{30})(1.414) = \underline{2.26}$

27.26

$\Delta_{UN} = \frac{2}{3} \cdot \frac{5.375}{29 \times 10^3} \left[\overset{71.9}{5(12.2) + 10.9} \right]$

$\Delta_{UN} = \frac{71.9}{88.8} \times .011 = .0089$

$S_{SLEEVE} = \frac{71.9}{88.8} \times 22.5 = \underline{\underline{18.2}} \; ksi$

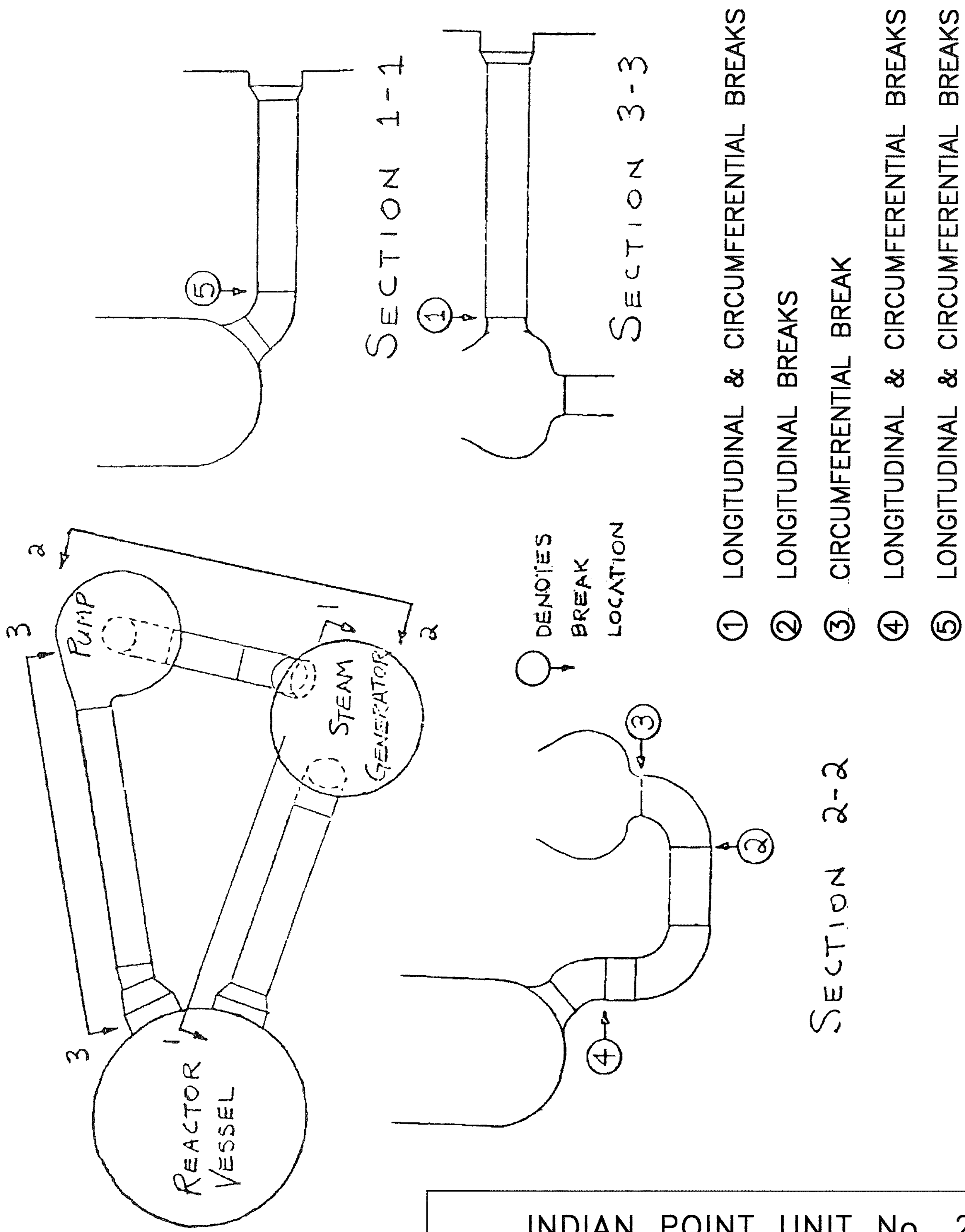
$S_{LINER} = \frac{71.9}{88.8} \times 9.3 = \underline{\underline{7.52}} \; ksi$

INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1-33

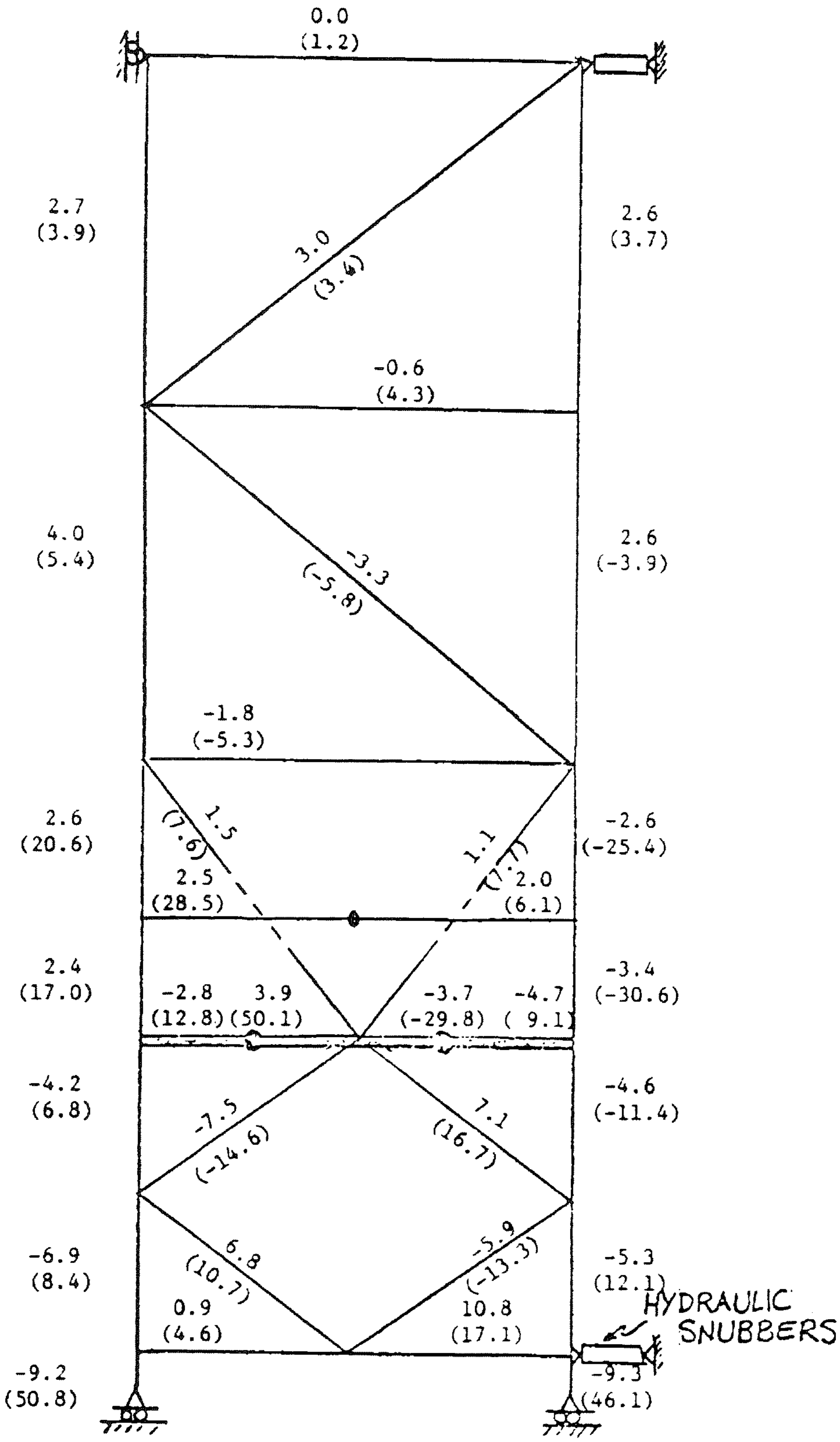
CONTAINMENT-STRESSES ON PENETRATIONS AND LINER - SHEET 7

MIC. No. 1999MC3772 | REV. No. 17A



(THIS FIGURE DEPICTS ORIGINAL HISTORICAL DESIGN DATA AND DOES NOT REFLECT THE CURRENT CONFIGURATION OR ANALYSIS.)

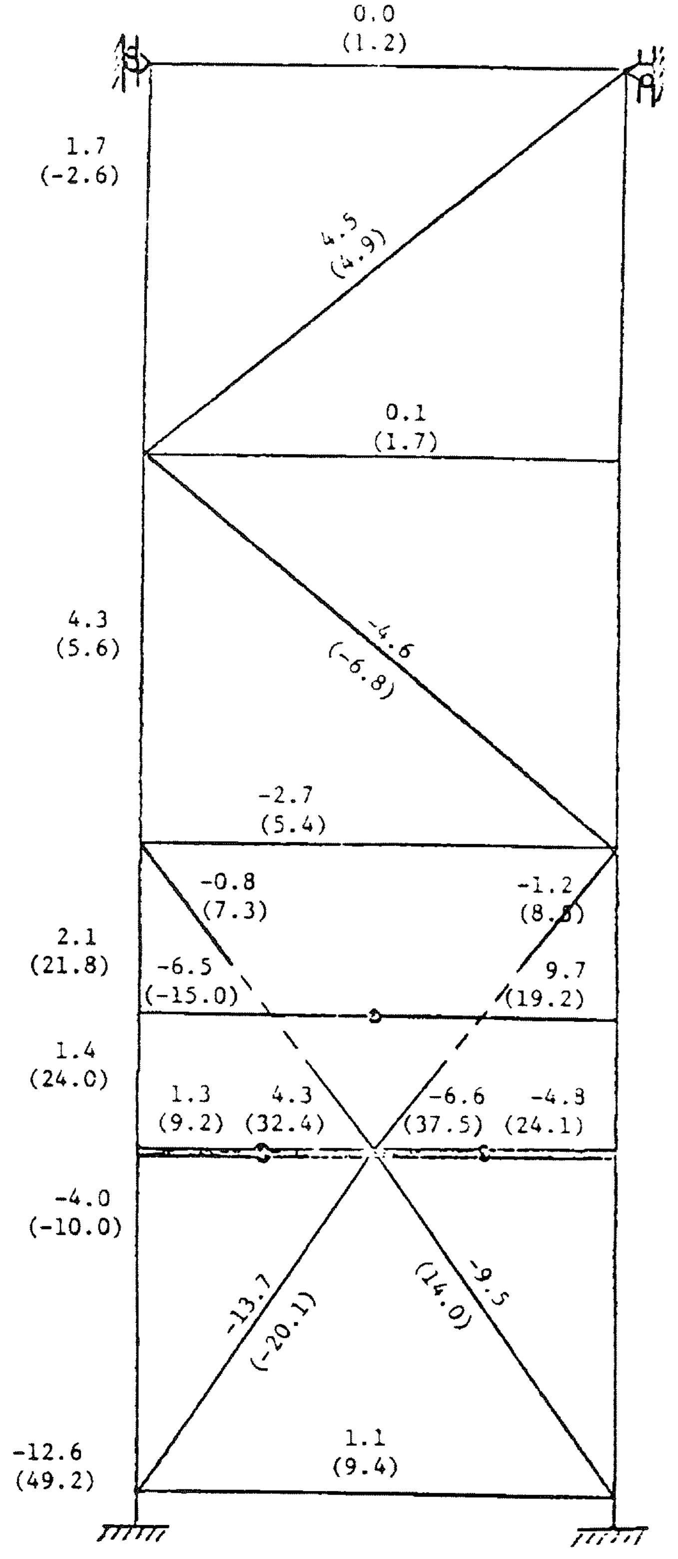
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1–34 ASSUMED PIPE RUPTURE ACCIDENT BREAK LOCATIONS	
MIC. No. 1999MC3773	REV. No. 17A



For Section location, see Figure 5.1–43

(THIS FIGURE DEPICTS ORIGINAL HISTORICAL DESIGN DATA AND DOES NOT REFLECT THE CURRENT CONFIGURATION OR ANALYSIS.)

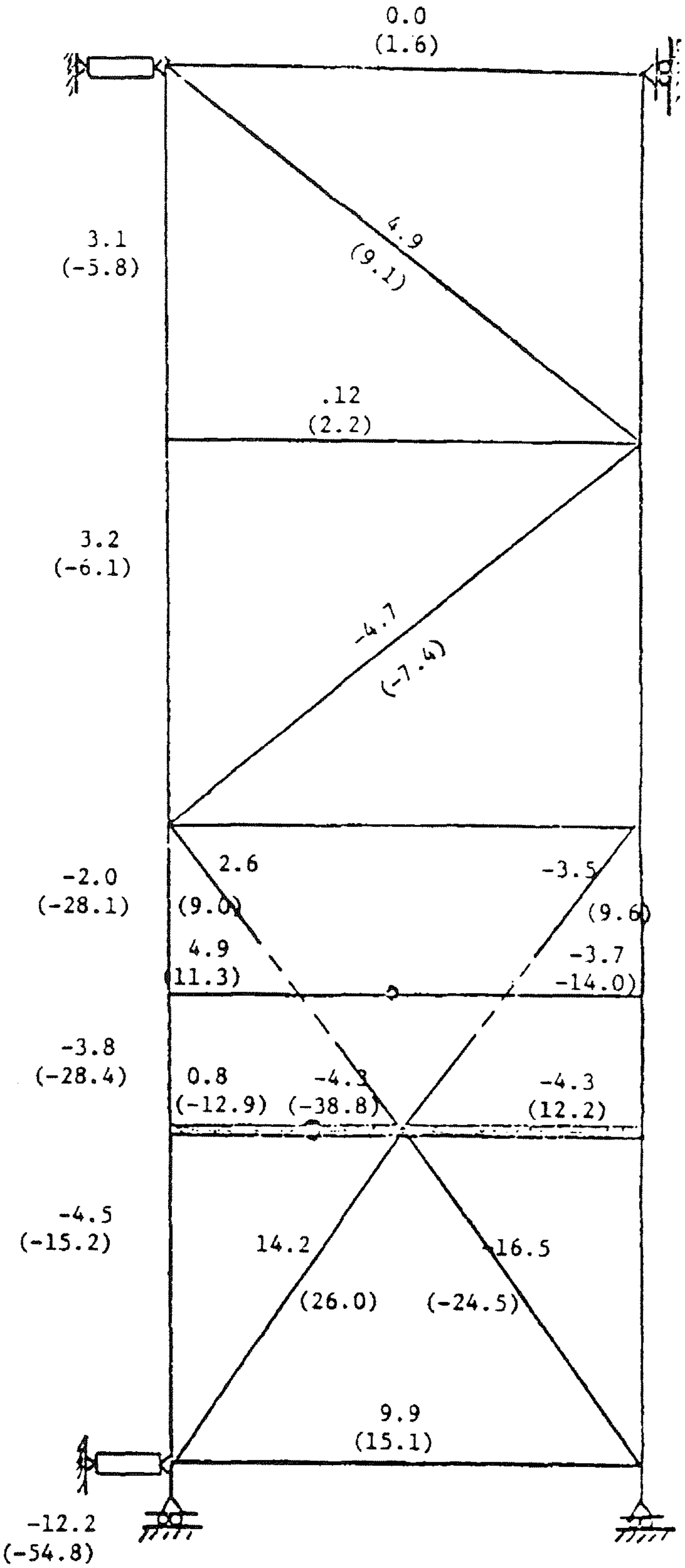
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1–35 STEAM GENERATOR SUPPORT–SECTION 1–1	
MIC. No. 1999MC3774	REV. No. 17A



For Section location, see Figure 5.1–43

(THIS FIGURE DEPICTS ORIGINAL HISTORICAL DESIGN DATA AND DOES NOT REFLECT THE CURRENT CONFIGURATION OR ANALYSIS.)

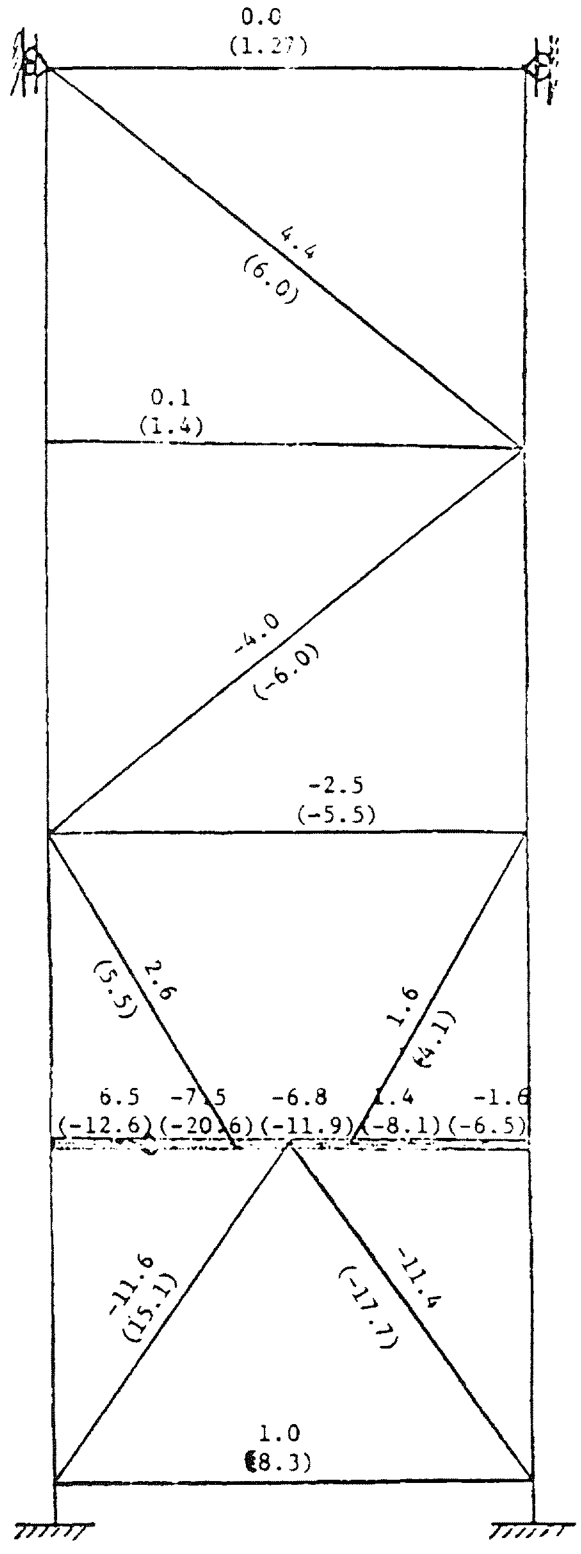
INDIAN POINT UNIT No. 2
UFSAR FIGURE 5.1–36 STEAM GENERATOR SUPPORT–SECTION 2–2
MIC. No. 1999MC3775 | REV. No. 17A



For Section location, see Figure 5.1-43

(THIS FIGURE DEPICTS ORIGINAL HISTORICAL DESIGN DATA AND DOES NOT REFLECT THE CURRENT CONFIGURATION OR ANALYSIS.)

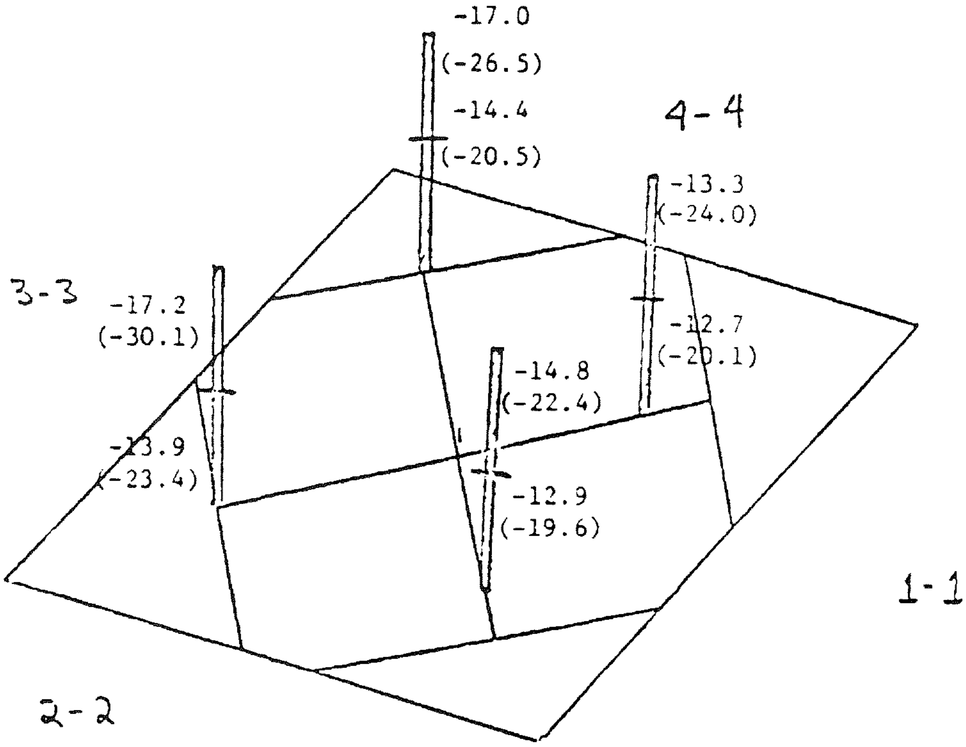
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-37 STEAM GENERATOR SUPPORT-SECTION 3-3	
MIC. No. 1999MC3776	REV. No. 17A



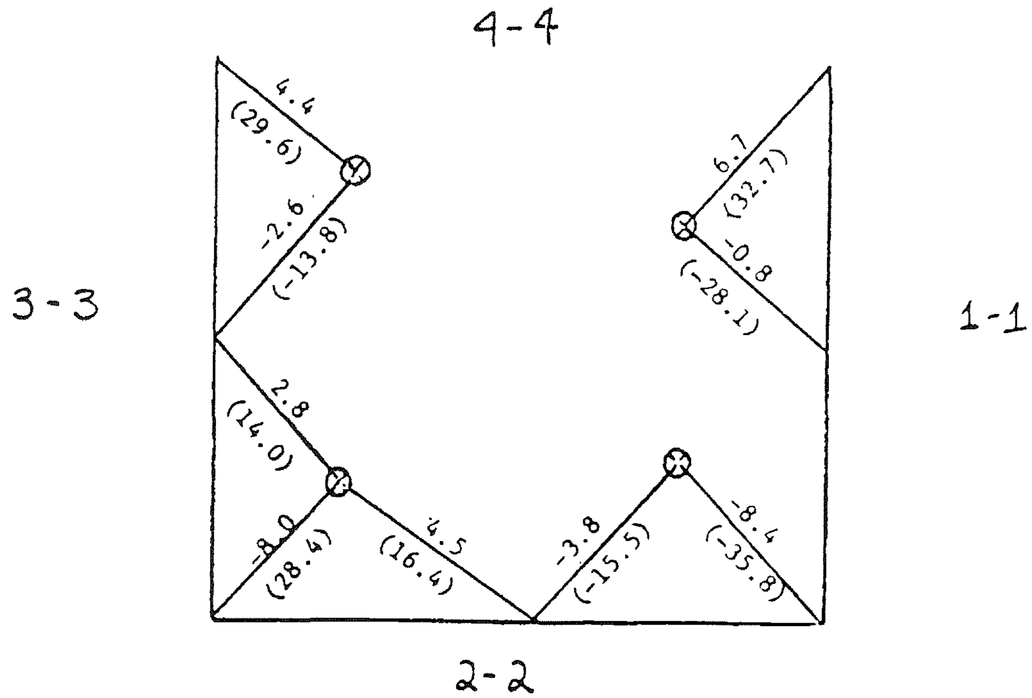
For Section location, see Figure 5.1-43

(THIS FIGURE DEPICTS ORIGINAL HISTORICAL DESIGN DATA AND DOES NOT REFLECT THE CURRENT CONFIGURATION OR ANALYSIS.)

INDIAN POINT UNIT No. 2
UFSAR FIGURE 5.1-38 STEAM GENERATOR SUPPORT-SECTION 4-4
MIC. No. 1999MC3777 | REV. No. 17A



PLAN AT EL. 60'-0"

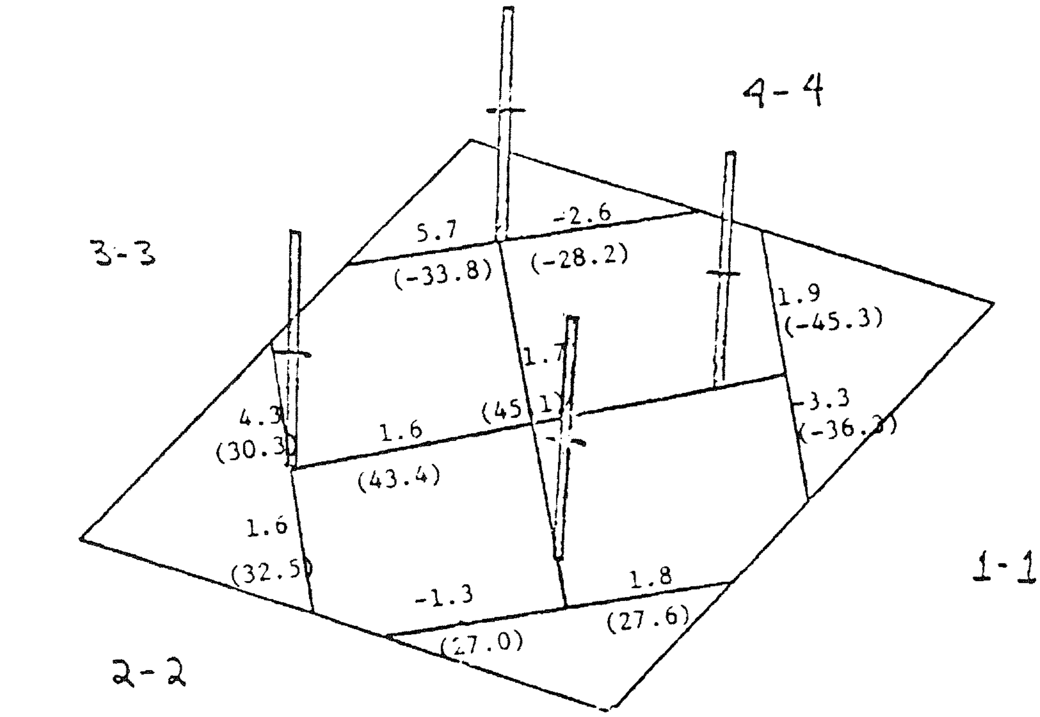


PLAN AT EL. 63'-0"

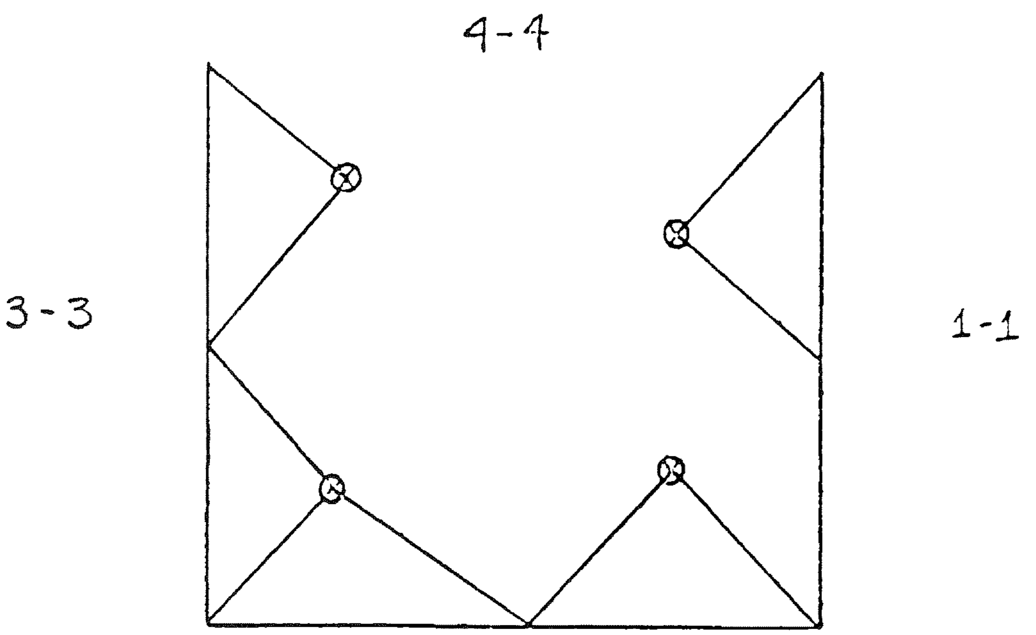
For Section location, see Figure 5.1-44

(THIS FIGURE DEPICTS ORIGINAL HISTORICAL DESIGN DATA AND DOES NOT REFLECT THE CURRENT CONFIGURATION OR ANALYSIS.)

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-39 STEAM GENERATOR SUPPORT-PLAN LOCATION ELEVATION 60 AND 63	
MIC. No. 1999MC3778	REV. No. 17A



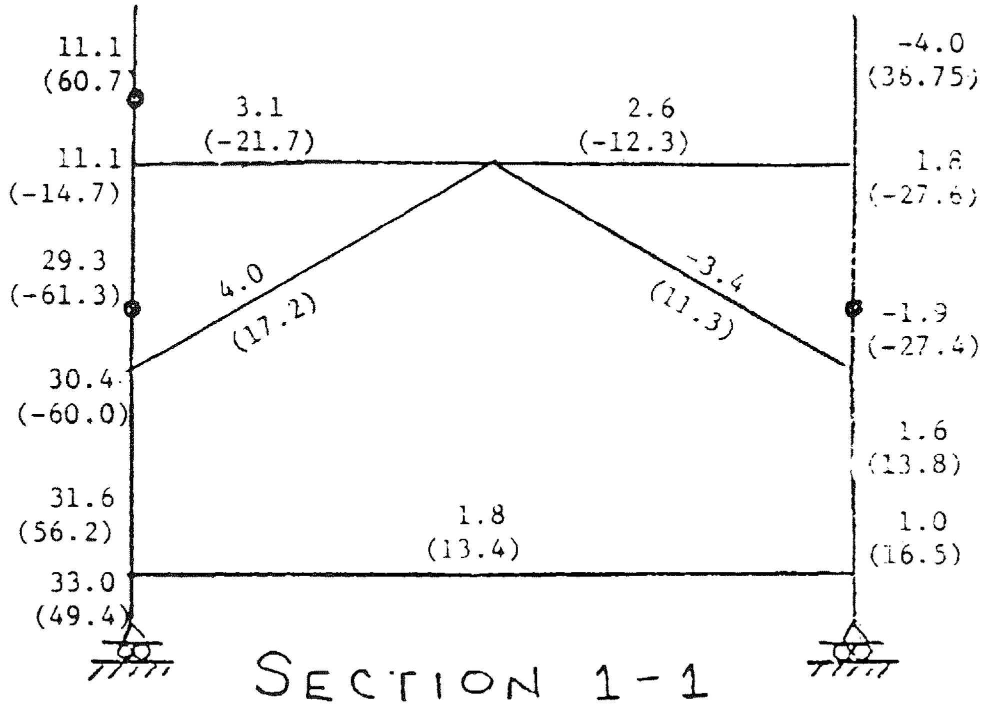
PLAN AT EL. 60'-0"



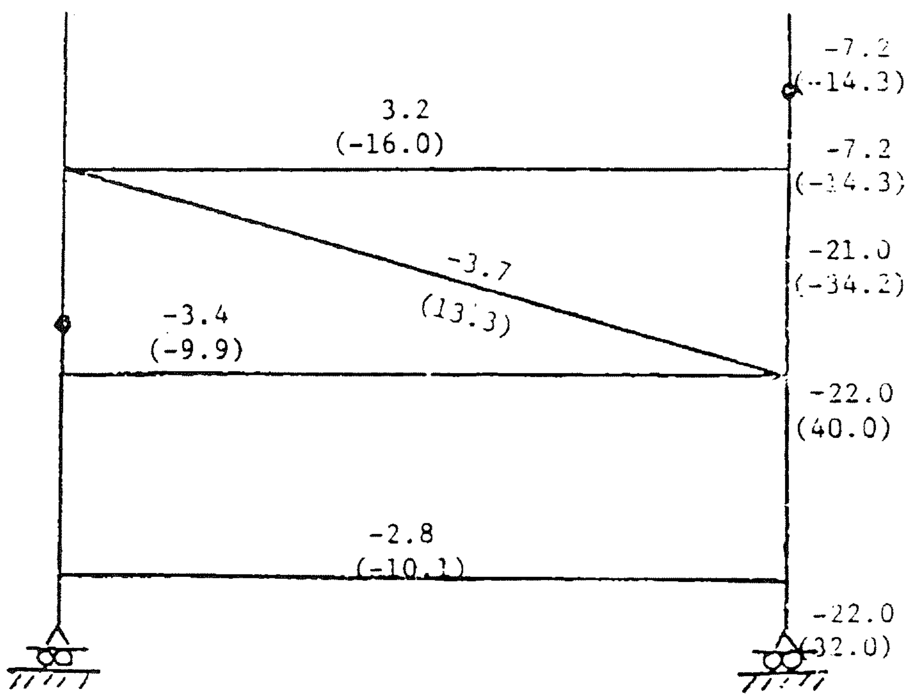
PLAN AT EL. 63'-0"

(THIS FIGURE DEPICTS ORIGINAL HISTORICAL DESIGN DATA AND DOES NOT REFLECT THE CURRENT CONFIGURATION OR ANALYSIS.)

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-40 STEAM GENERATOR SUPPORT-PLAN LOCATION ELEVATION 60 AND 63	
MIC. No. 1999MC3779	REV. No. 17A



SECTION 1-1



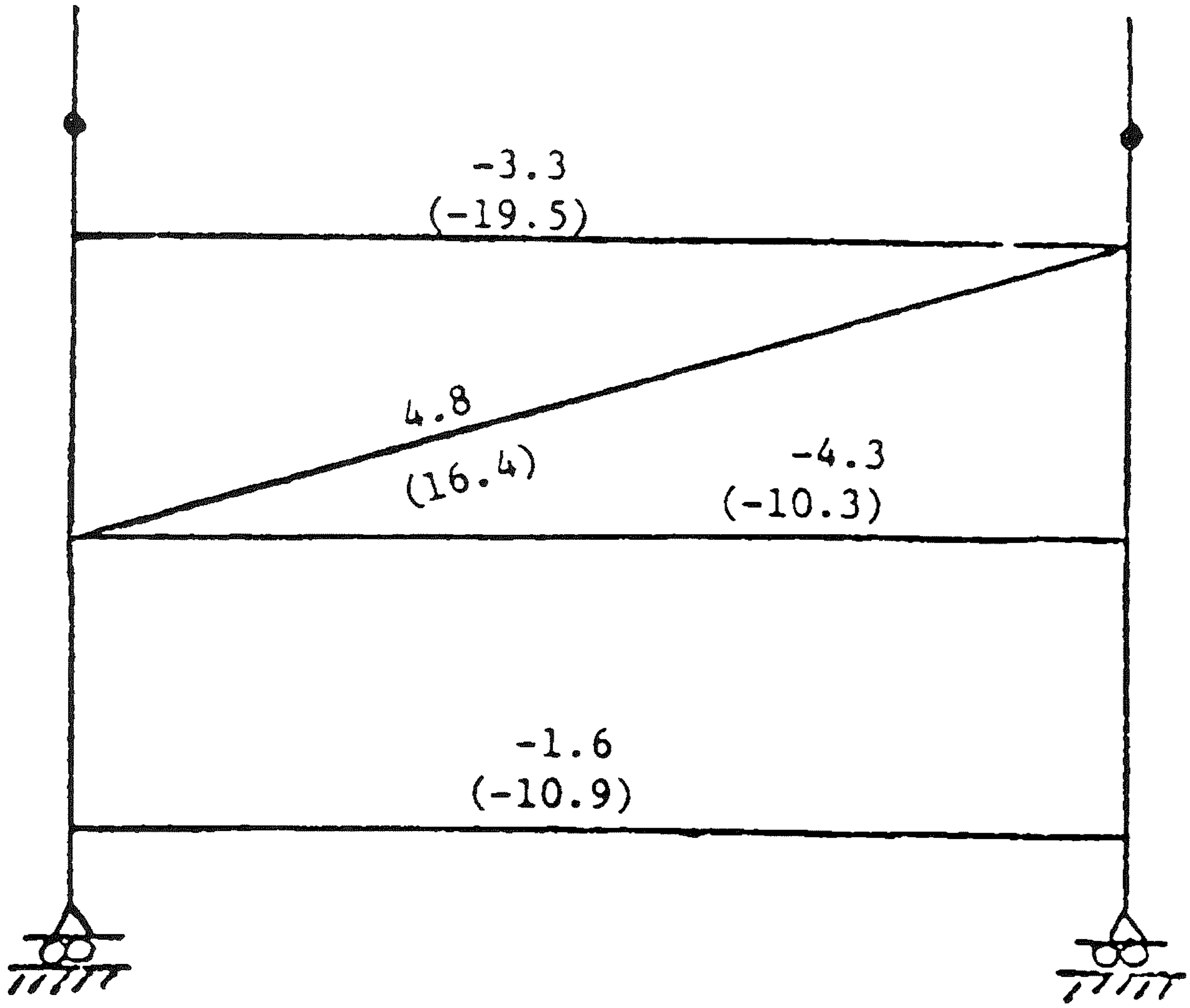
SECTION 2-2

For Section location, see Figure 5.1–44

(THIS FIGURE DEPICTS ORIGINAL HISTORICAL DESIGN DATA AND DOES NOT REFLECT THE CURRENT CONFIGURATION OR ANALYSIS.)

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1–41 PUMP SUPPORT–SECTION 2–2 AND 3–3	
MIC. No. 1999MC3780	REV. No. 17A

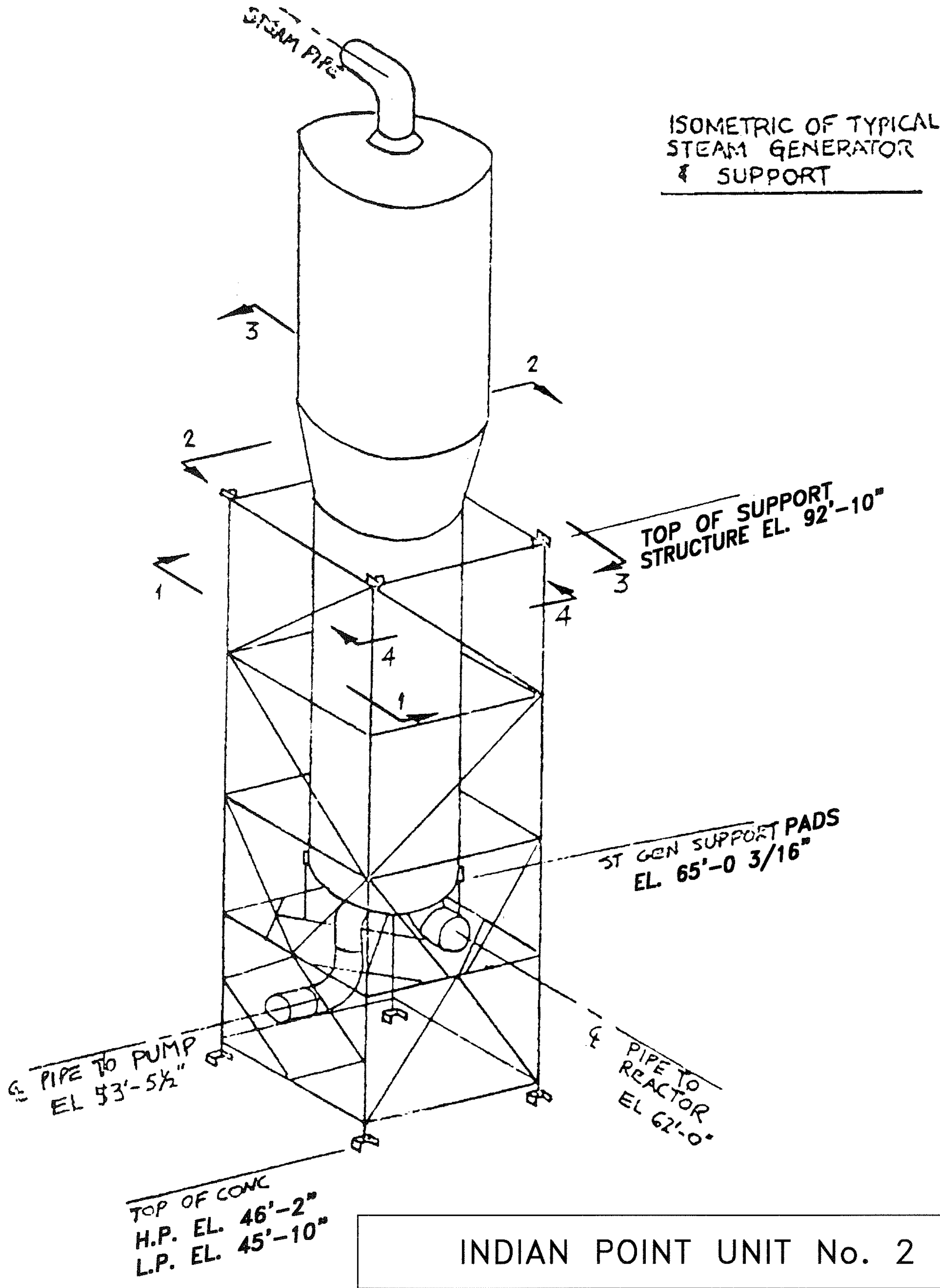
OAGI0000215_0658



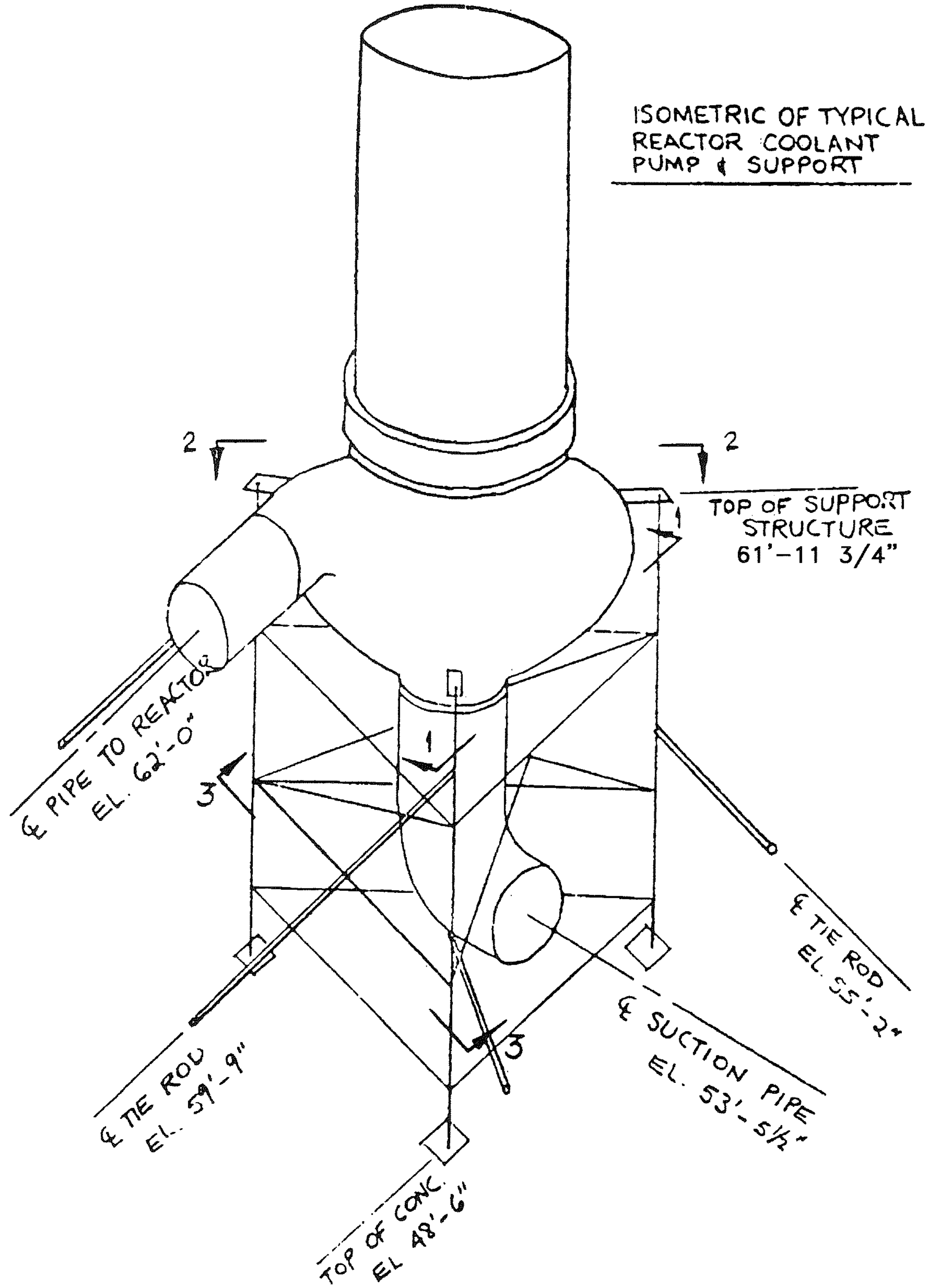
For Section location,
see Figure 5.1-44

(THIS FIGURE DEPICTS ORIGINAL HISTORICAL DESIGN DATA AND DOES NOT REFLECT THE CURRENT CONFIGURATION OR ANALYSIS.)

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-42 PUMP SUPPORT-SECTION 3-3	
MIC. No. 1999MC3781	REV. No. 17A



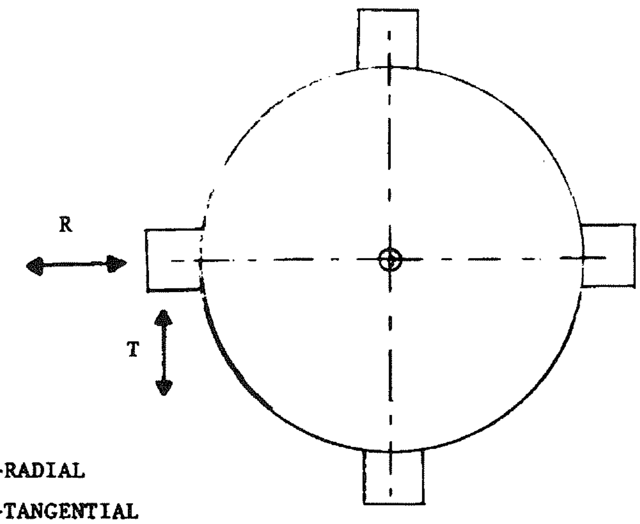
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-43 ISOMETRIC VIEW-STEAM GENERATOR SUPPORT	
MIC. No. 1999MC3782	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-44 ISOMETRIC VIEW-REACTOR COOLANT PUMP SUPPORT	
MIC. No. 1999MC3783	REV. No. 17A

MAXIMUM FORCES
ACTING ON A REACTOR
VESSEL SUPPORT

	A	B	C	D	Σ		
	REACTOR VESSEL WEIGHT & PIPING REACTION	PIPE BREAK CASE II	PIPE BREAK CASE III	EARTH-QUAKE Z & VERTICAL DIRECTION	A+B	A+C	A+D
P (lb)	934,000	-	525,000	395,000	934,000	1,459,000	1,329,000
R (lb)	322,000	-	-	-	322,000	322,000	322,000
T (lb)	140,000	1,187,000	710,000	969,000	1,327,000	850,000	1,109,000



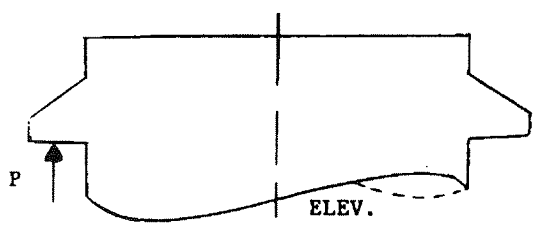
PIPE BREAK CASE II
IS BETWEEN PUMP AND
ELL ON REACTOR NOZZLE

PIPE BREAK CASE III
IS AT STEAM GENERATOR
INLET

EARTHQUAKE IN Z
DIRECTION ACTS NORMAL
TO REACTOR COOLANT
OUTLET LINE

R-RADIAL
T-TANGENTIAL
P-VERTICAL

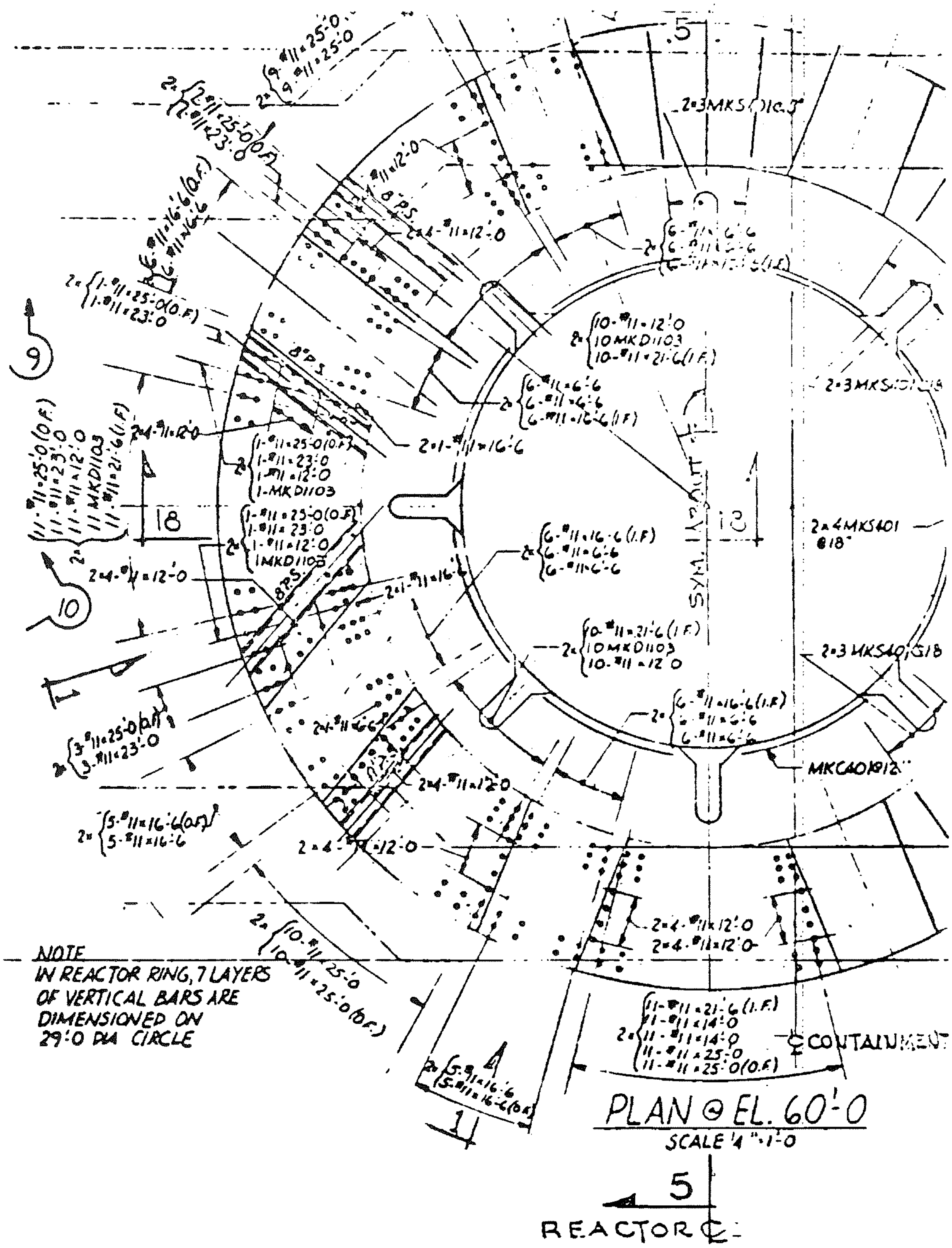
PLAN



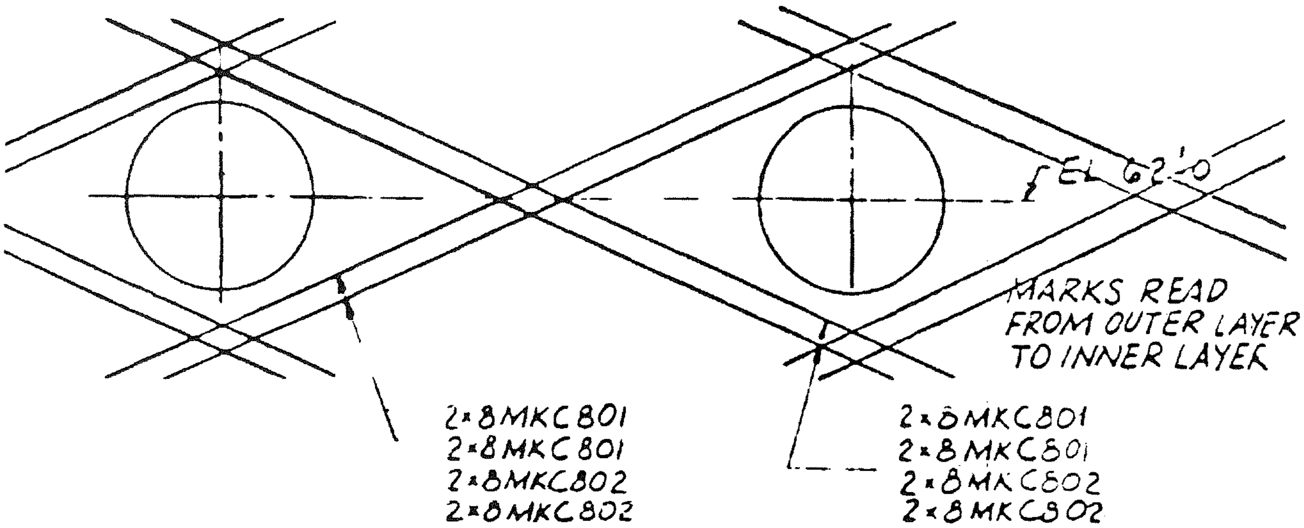
ELEV.

(THIS FIGURE DEPICTS ORIGINAL HISTORICAL DESIGN DATA AND DOES NOT REFLECT THE CURRENT CONFIGURATION OR ANALYSIS.)

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-45 MAXIMUM FORCES ACTING ON A REACTOR VESSEL SUPPORT	
MIC. No. 1999MC3784	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1–46 PLAN VIEW 60 FT – 0 IN.	
MIC. No. 1999MC3785	REV. No. 17A

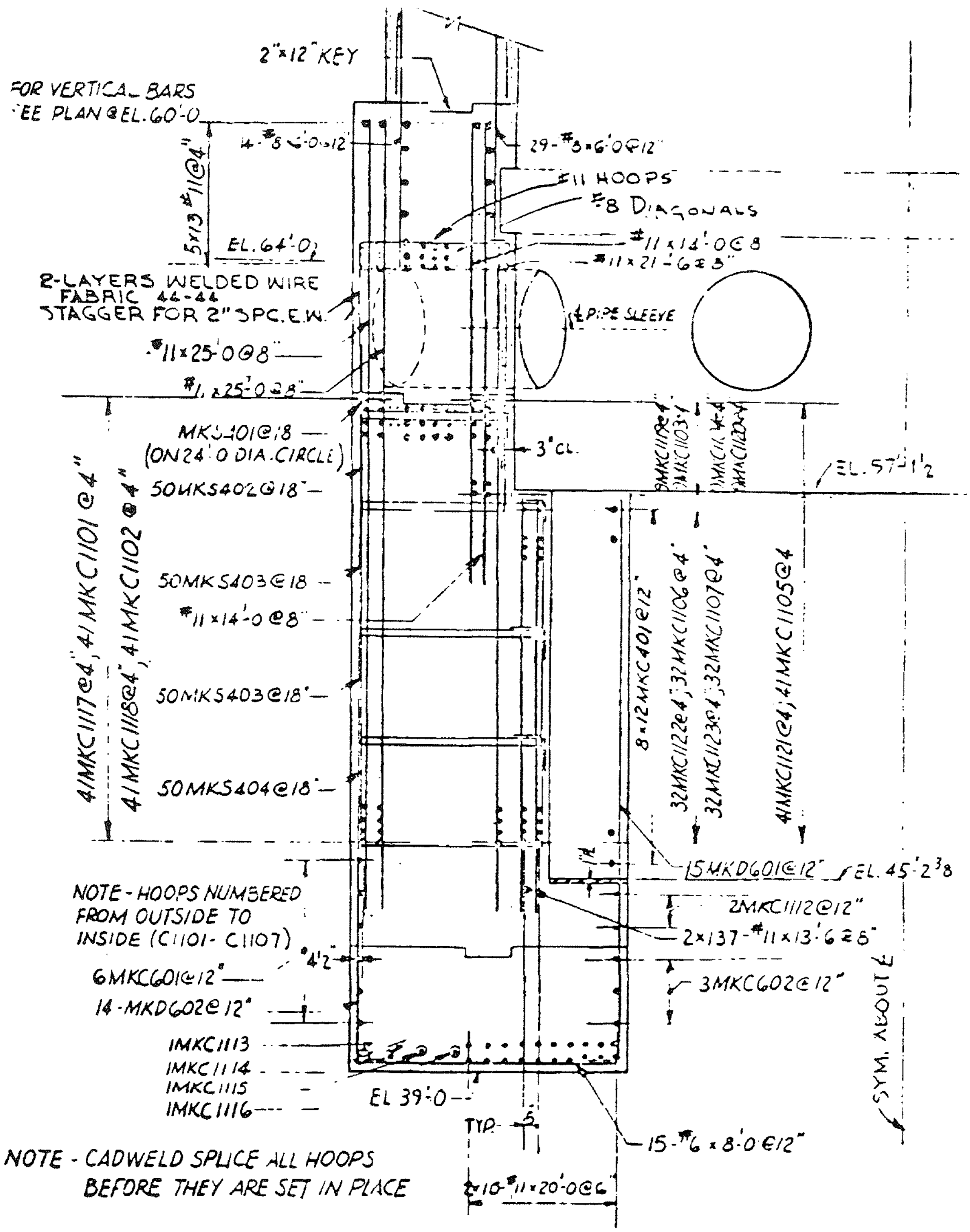


TYP. LAYER @ REACTOR RING

SCALE 1/4":1'-0

SECTION 1-1

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1–47 TYPICAL LAYER–REACTOR RING	
MIC. No. 1999MC3786	REV. No. 17A

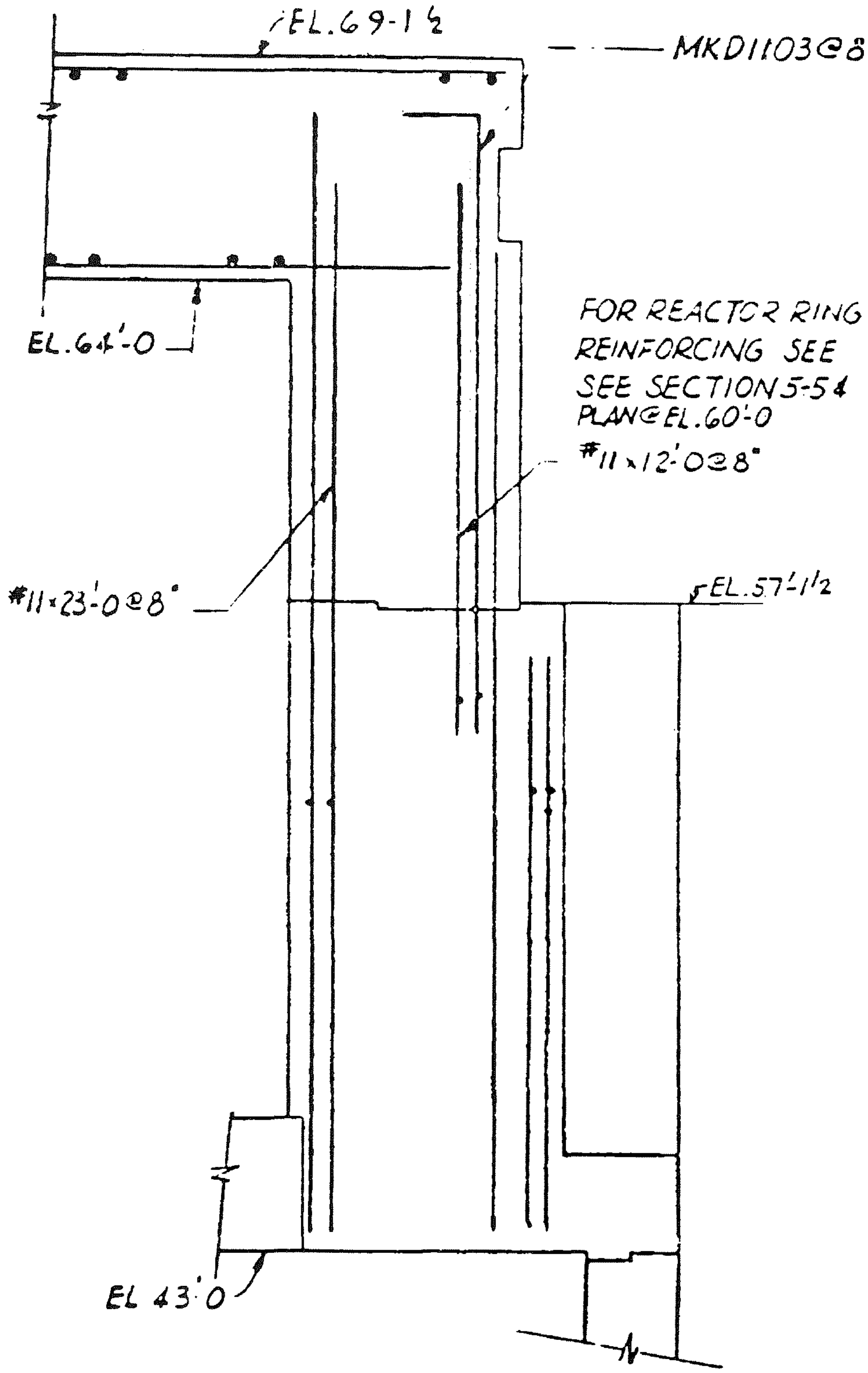


INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1-48

SECTION 5-5

MIC. No. 1999MC3787 | REV. No. 17A

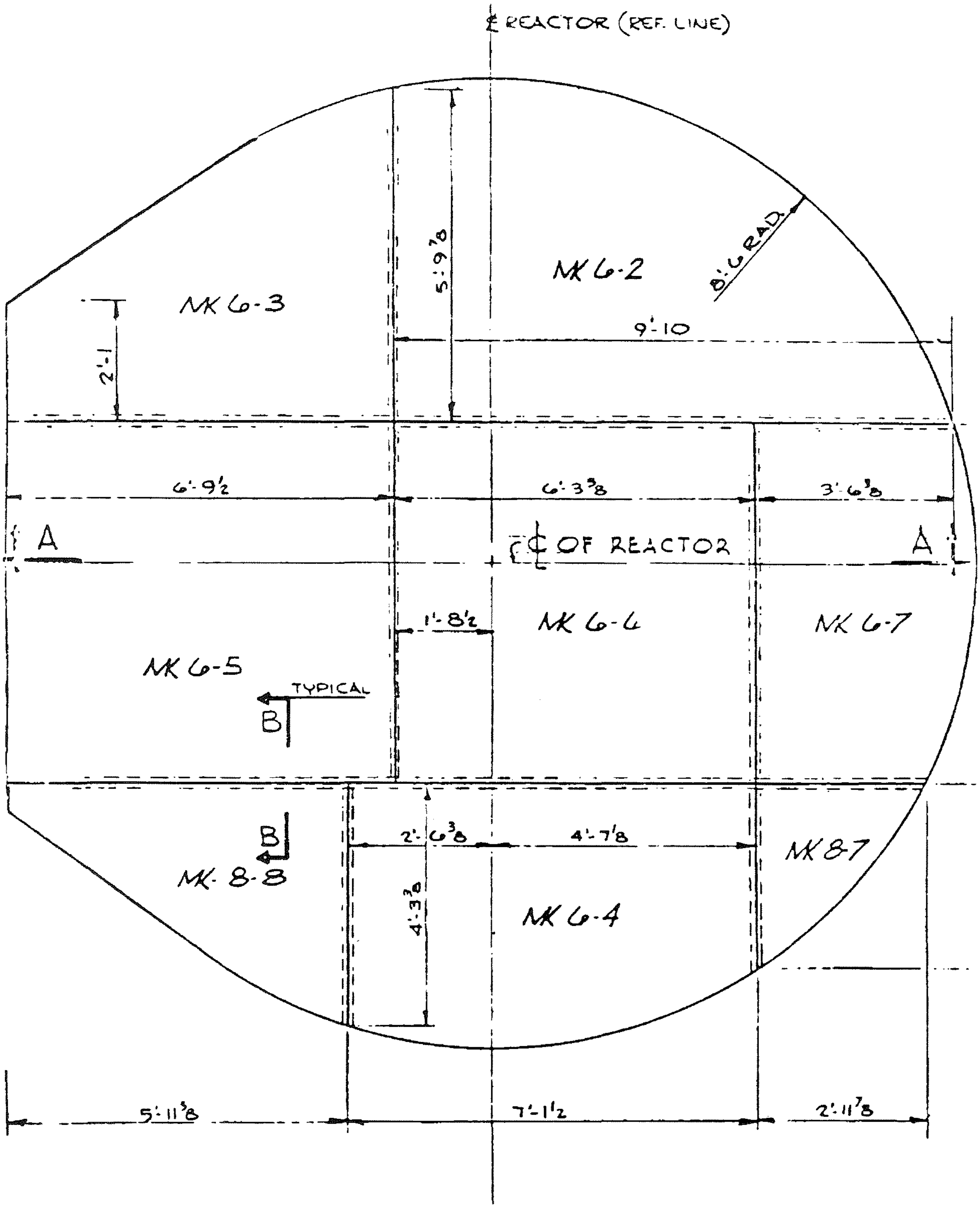


INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1-49

SECTION 18-18

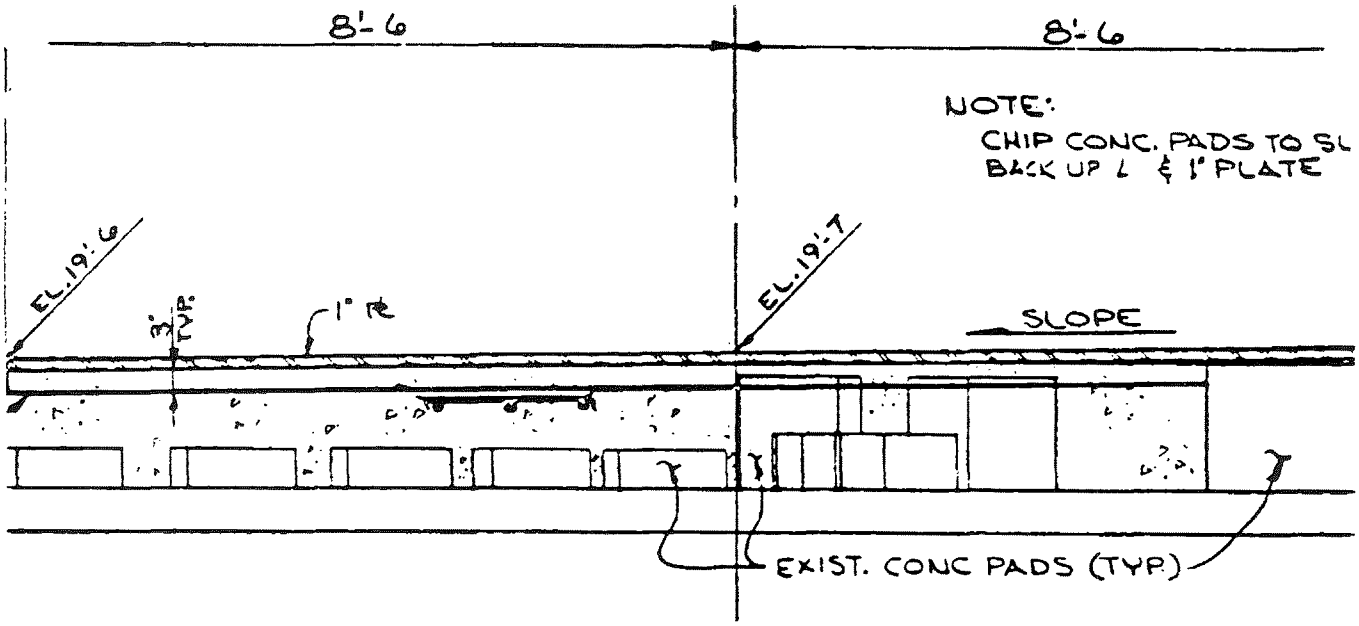
MIC. No. 1999MC3788	REV. No. 17A



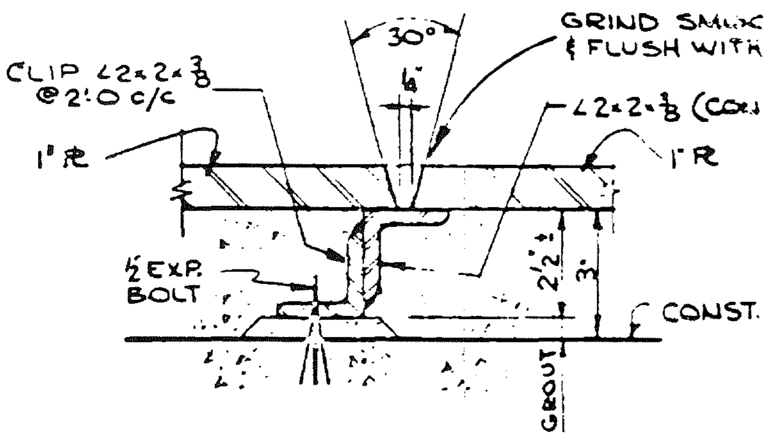
INDIAN POINT UNIT No. 2
UFSAR FIGURE 5.1-50
PLAN VIEW AT ELEVATION 19 FT.-7 IN.

MIC. No. 1999MC3789	REV. No. 17A

OAGI0000215_0667



SECTION A-A



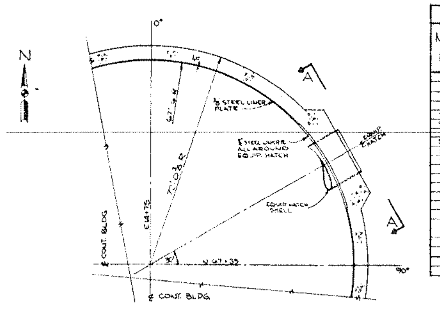
SECTION B-B

INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1-51

SECTION A-A AND
SECTION B-B

MIC. No. 1999MC3790	REV. No. 17A



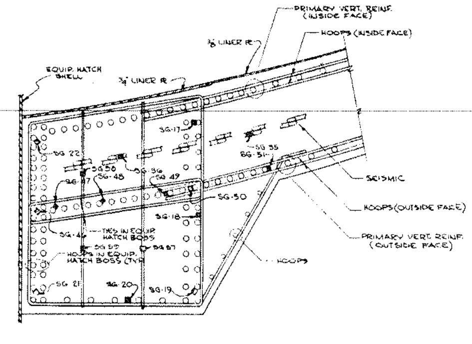
KEY PLAN

SCALE: 1/16" = 1'-0"

— CONTINUED FROM BELOW —

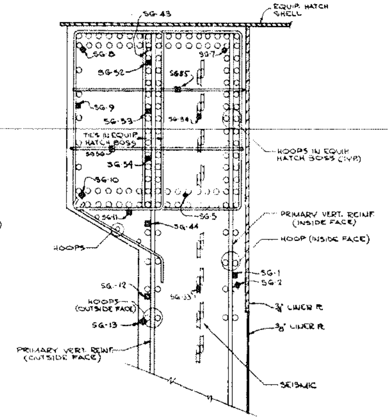
STRAIN GAUGE SCHEDULE

MARK Nº	* HORIZONTAL LOCATION LEFT	RIGHT	VERTICAL LOCATION (ELEVATION)	RESPECTIVE STRUCTURAL TYPE	REMARKS
SG-43	0'-0"	0'-0"	EL. 92'-6"	HOOP IN EQUIP. HATCH BOSS	
SG-44	0'-0"	0'-0"	EL. 85'-4"	HOOP IN EQUIP. HATCH BOSS	
SG-46	——	8'-7"	EL. 101'-6"	HOOP IN EQUIP. HATCH BOSS	
SG-47	——	9'-6"	EL. 101'-8"	PRIMARY VERT. (CENTER OF BOSS)	
SG-48	——	11'-7"	EL. 101'-1"	PRIMARY VERT. (CENTER BOSS)	
SG-49	——	14'-1"	EL. 101'-6"	PRIMARY VERT. (CENTER OF BOSS)	
SG-50	——	15'-2"	EL. 101'-5"	PRIMARY VERT. (CENTER OF BOSS)	
SG-51	——	18'-5"	EL. 101'-11"	PRIMARY VERT. (OUTSIDE FACE)	
SG-52	0'-0"	0'-3"	EL. 92'-0"	HOOP (CENTER OF EQUIP. HATCH BOSS)	
SG-53	0'-0"	0'-7"	EL. 90'-5"	HOOP (CENTER OF EQUIP. HATCH BOSS)	
SG-54	0'-0"	0'-8"	EL. 88'-4"	HOOP (CENTER OF EQUIP. HATCH BOSS)	
SG-55	0'-0"	1'-1"	EL. 91'-6"	RADIAL SHEAR BAR	4'-5 FROM LINER PL
SG-56	0'-5"	0'-0"	EL. 88'-9"	" " "	5'-6 FROM LINER PL
SG-57	——	12'-2"	EL. 100'-10"	" " "	6'-9 FROM LINER PL
SG-58	——	7'-10"	EL. 101'-3"	" " "	3'-0 FROM LINER PL
SG-59	——	10'-1"	EL. 101'-4"	" " "	5'-3 FROM LINER PL



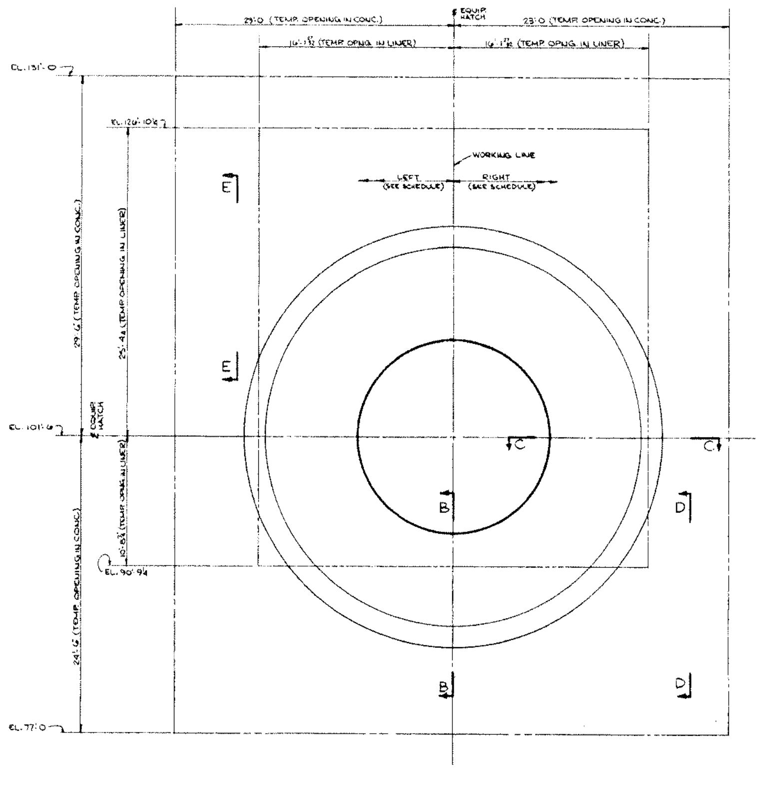
SECTION C-C

SCALE: 1/2" = 1'-0"



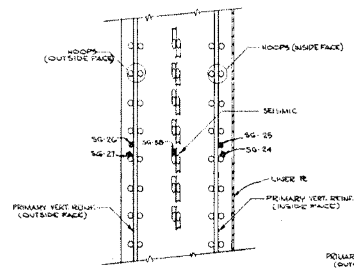
SECTION B-B

SCALE: 1/2" = 1'-0"



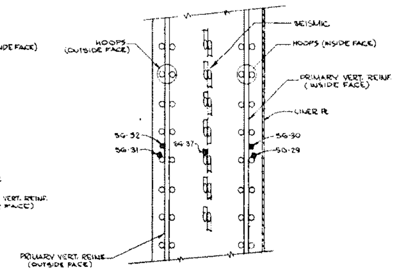
SECTION A-A (DEVELOPED ELEVATION)

SCALE: 1/4" = 1'-0"



SECTION D-D

SCALE: 1/2" = 1'-0"



SECTION E-E

SCALE: 1/2" = 1'-0"

STRAIN GAUGE SCHEDULE

MARK Nº	* HORIZONTAL LOCATION LEFT	RIGHT	VERTICAL LOCATION (ELEVATION)	RESPECTIVE STRUCTURAL TYPE	REMARKS
SG-1	0'-0"	0'-0"	EL. 83'-4"	PRIMARY VERT. (INSIDE FACE)	
SG-2	0'-0"	0'-0"	EL. 82'-9"	HOOP (INSIDE FACE)	
SG-3	0'-0"	0'-0"	EL. 87'-3"	HOOP IN EQUIPMENT HATCH BOSS	2'-7 FROM LINER PL
SG-7	0'-0"	0'-0"	EL. 92'-5"	HOOP IN EQUIPMENT HATCH BOSS	
SG-8	0'-0"	0'-0"	EL. 92'-6"	HOOP IN EQUIPMENT HATCH BOSS	
SG-9	0'-0"	0'-0"	EL. 90'-2"	TIE IN EQUIPMENT HATCH BOSS	
SG-10	0'-0"	0'-0"	EL. 86'-5"	HOOP IN EQUIPMENT HATCH BOSS	
SG-11	0'-7"	0'-0"	EL. 85'-11"	TIE IN EQUIPMENT HATCH BOSS	
SG-12	0'-0"	0'-0"	EL. 82'-7"	PRIMARY VERT. (OUTSIDE FACE)	
SG-13	0'-0"	0'-0"	EL. 82'-1"	HOOP (OUTSIDE FACE)	
SG-17	——	15'-3"	EL. 101'-6"	HOOP IN EQUIPMENT HATCH BOSS	
SG-18	——	15'-4"	EL. 100'-10"	TIE IN EQUIPMENT HATCH BOSS	
SG-19	——	15'-2"	EL. 101'-0"	HOOP IN EQUIPMENT HATCH BOSS	
SG-20	——	12'-6"	EL. 100'-5"	TIE IN EQUIPMENT HATCH BOSS	
SG-21	——	8'-10"	EL. 100'-8"	HOOP IN EQUIPMENT HATCH BOSS	
SG-22	——	8'-7"	EL. 101'-6"	HOOP IN EQUIPMENT HATCH BOSS	
SG-24	——	21'-0"	EL. 82'-9"	HOOP (INSIDE FACE)	
SG-25	——	21'-0"	EL. 83'-4"	PRIMARY VERT. (INSIDE FACE)	
SG-26	——	22'-3"	EL. 82'-7"	PRIMARY VERT. (OUTSIDE FACE)	
SG-27	——	22'-0"	EL. 82'-4"	HOOP (OUTSIDE FACE)	
SG-29	21'-0"	——	EL. 120'-3"	HOOP (INSIDE FACE)	
SG-30	21'-0"	——	EL. 120'-10"	PRIMARY VERT. (INSIDE FACE)	
SG-31	21'-7"	——	EL. 120'-9"	HOOP (OUTSIDE FACE)	
SG-32	21'-5"	——	EL. 121'-2"	PRIMARY VERT. (OUTSIDE FACE)	
SG-33	0'-0"	0'-0"	EL. 82'-9"	SEISMIC	
SG-34	0'-0"	0'-0"	EL. 89'-9"	SEISMIC	
SG-35	——	13'-10"	EL. 101'-6"	SEISMIC	
SG-36	——	17'-0"	EL. 101'-6"	SEISMIC	
SG-37	21'-0"		EL. 120'-3"	SEISMIC	
SG-38	——	21'-0"	EL. 82'-9"	SEISMIC	

— FOR CONTINUATION OF STRAIN GAUGE SCHEDULE SEE ABOVE —

* FOR CORRECT ORIENTATION OF LEFT/RIGHT DIRECTIONS SEE WORKING LINE ON SECTION A-A

GENERAL NOTES

1. WHERE STRAIN GAUGES AND RESPECTIVE STRUCTURAL TYPE LOCATIONS DO NOT COINCIDE ACCORDING TO GIVEN DIMENSIONS AND/OR GIVEN ELEVATIONS, PLACE STRAIN GAUGE ON NEAREST STRUCTURAL ELEMENT OF THE SAME TYPE.

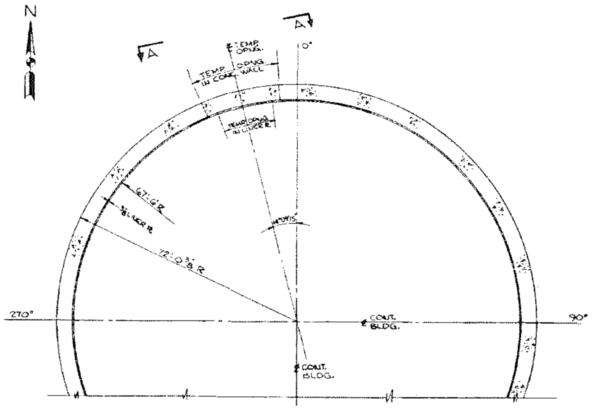
INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1-53

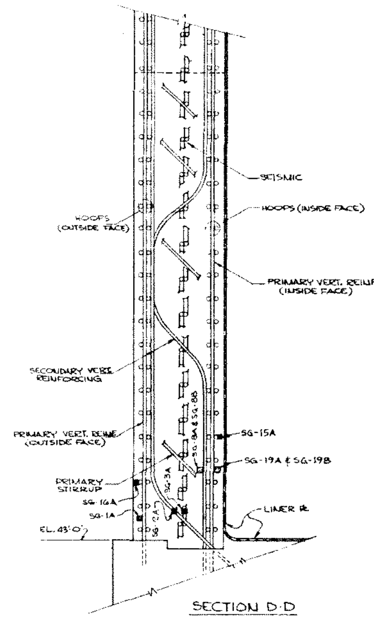
CONTAINMENT EQUIPMENT HATCH
STRAIN GAUGE TEST LOCATIONS

MIC. No. 1999MC3792

REV. No. 17A



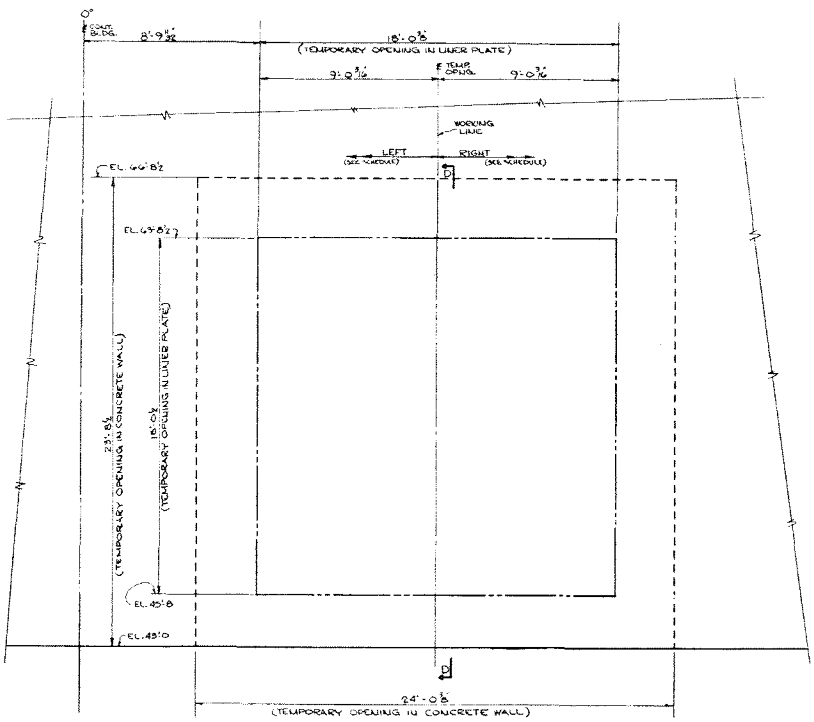
KEY PLAN
SCALE: 1/16" = 1'-0"



SECTION D-D
1/1 REDUNDANCY REQUIRED
ON GUAGES SG-8A & SG-19A

STRAIN GAUGE SCHEDULE					
MARK Nº	* HORIZONTAL LOCATION		VERTICAL LOCATION (ELEVATION)	RESPECTIVE STRUCTURAL TYPE	REMARKS
	LEFT	RIGHT			
SG-1A		0'-6"	EL. 44'-2	PRIMARY VERT. (OUTSIDE FACE)	
SG-2A	——	0'-6"	EL. 44'-2	SECONDARY VERTICAL	
SG-3A	——	0'-6"	EL. 44'-2	SEISMIC	
SG-8A	——	0'-6"	EL. 46'-6	SECONDARY VERTICAL	
SG-8B		0'-6	EL. 46'-6	SECONDARY VERTICAL	"REDUNDANT GUAGE"
SG-15A	——	0'-6"	EL. 48'-0	HOOP (INSIDE FACE)	
SG-16A	——	0'-6	EL. 45'-11	HOOP (OUTSIDE FACE)	
SG-19A	——	0'-6	EL. 46'-6	PRIMARY VERT. (INSIDE FACE)	
SG-19B		0'-6	EL. 46'-6	PRIMARY VERT (INSIDE FACE)	"REDUNDANT GUAGE"

* FOR CORRECT ORIENTATION OF *LEFT & RIGHT* DIRECTIONS SEE *WORKING LINE* ON SECTION A-A



GENERAL NOTES

1. WHERE STRAIN GAUGES AND RESPECTIVE STRUCTURAL TYPE LOCATIONS DO NOT COINCIDE ACCORDING TO GIVEN DIMENSIONS AND/OR GIVEN ELEVATIONS, PLACE STRAIN GAUGE ON NEAREST STRUCTURAL ELEMENT OF THE SAME TYPE.

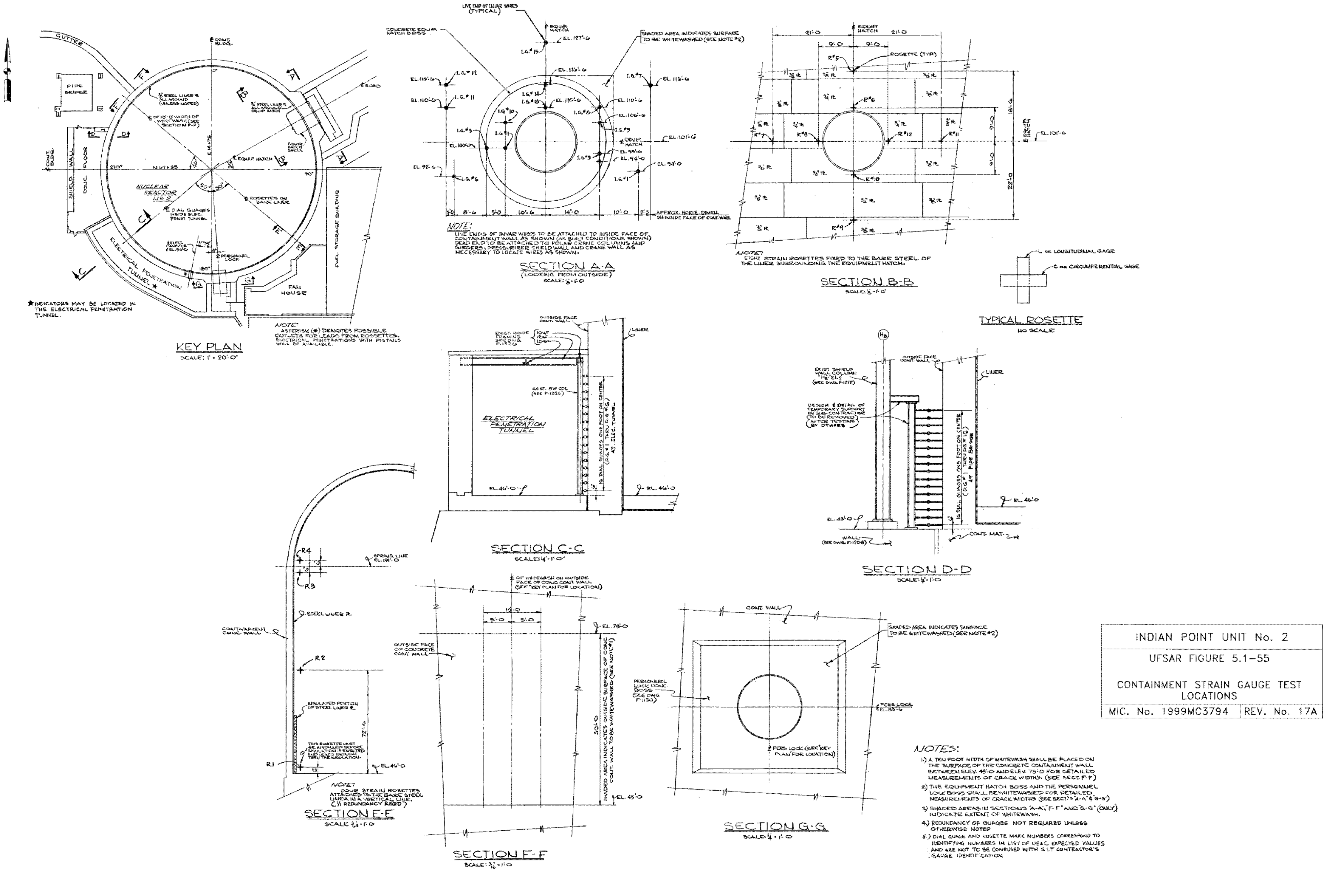
SECTION A-A (DEVELOPED ELEVATION)
SCALE: 3/8" = 1'-0"

INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1-54

CONTAINMENT TEMPORARY OPENING IN NW QUADRANT STRAIN GAUGE TEST LOCATIONS

MIC. No. 1999MC3793	REV. No. 17A

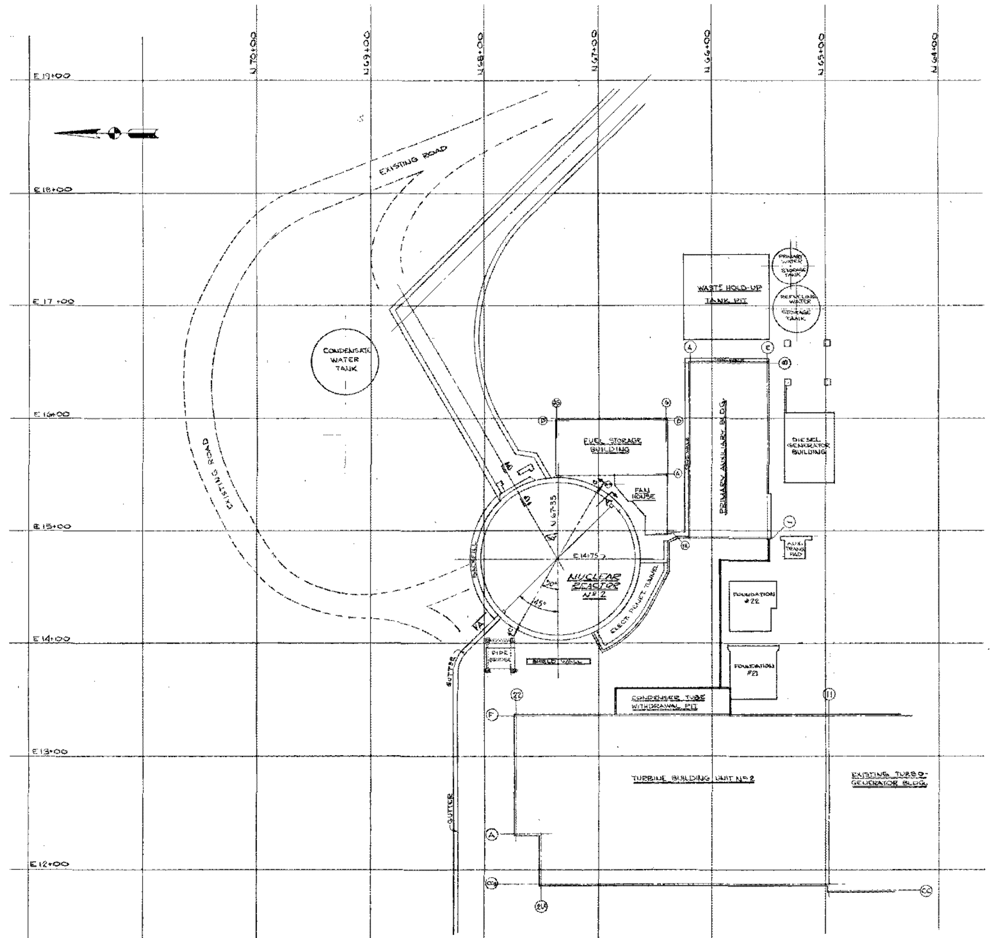


INDIAN POINT UNIT No. 2

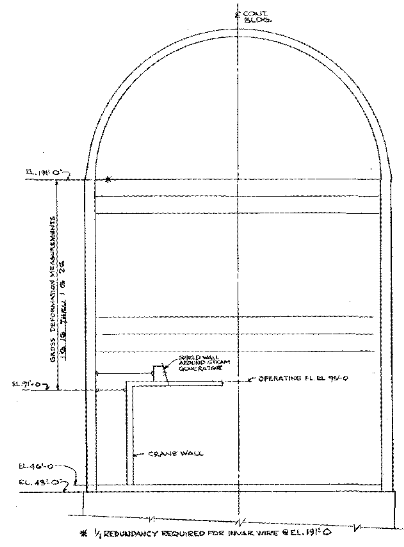
UFSAR FIGURE 5.1-55

CONTAINMENT STRAIN GAUGE TEST LOCATIONS

MIC. No. 1999MC3794 | REV. No. 17A

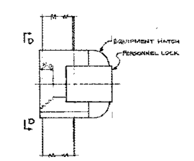


PLAN
SCALE: 1"=40'-0"

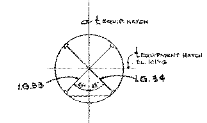


* 1/2 REDUNDANCY REQUIRED FOR INVAR WIRE @ EL. 191'-0

SECTION A-A

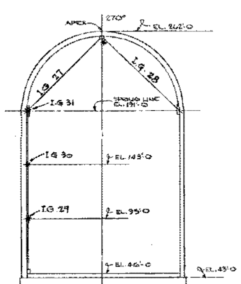


SECTION B-B



SECTION D-D

VERTICAL MEASUREMENT READINGS SHALL BE OBTAINED AT ELEVATIONS MARKED WITH AN ASTERISK (*). THE TAPE SHALL BE ATTACHED TO THE LINER BY OTHERS AND ANCHORED TO THE MAT AT EL. 46'-0.



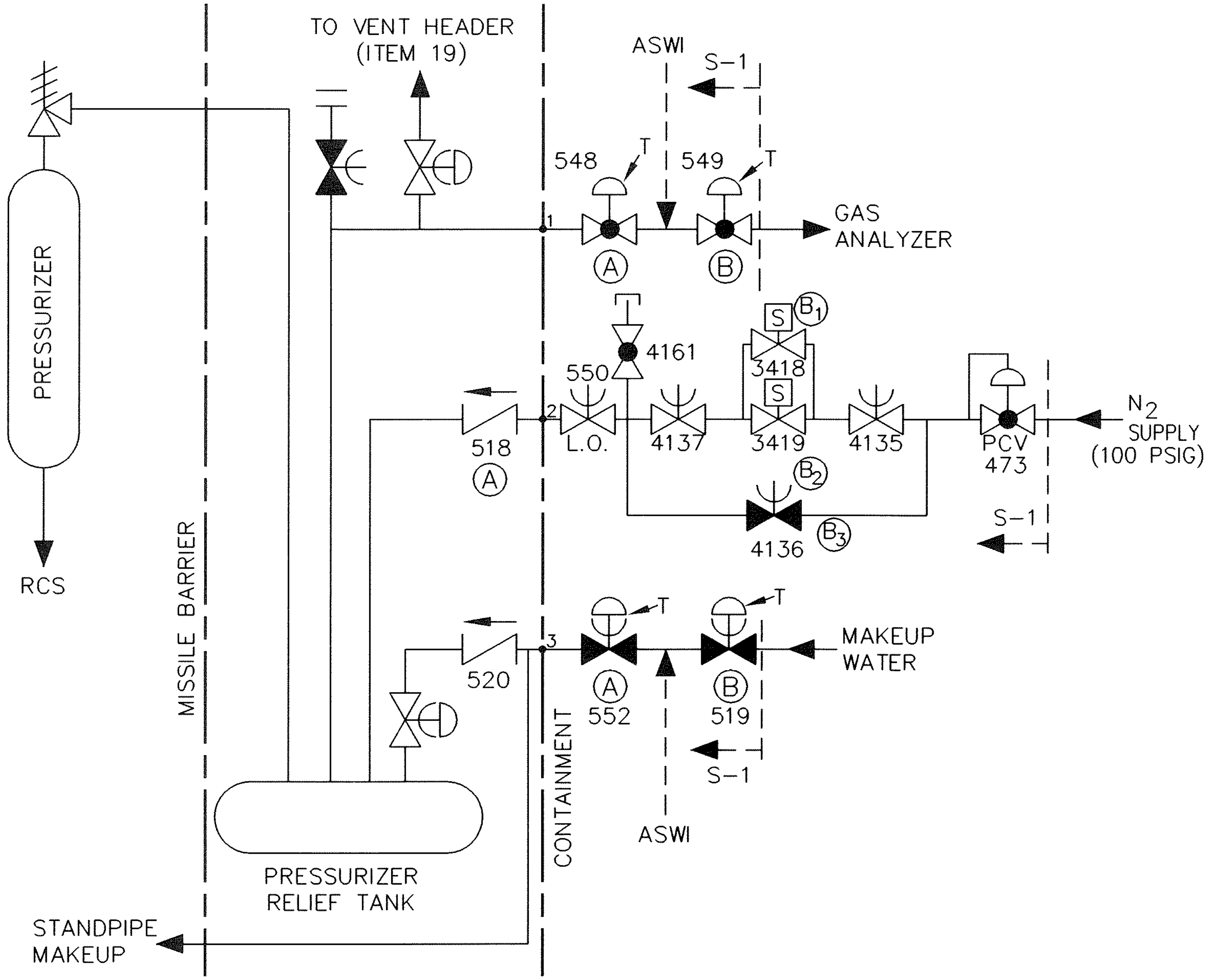
SECTION C-C
SCALE: 1"=40'-0"

NOTES

1. PROOF TEST GROSS DEFORMATION INSTRUMENTATION SHALL BE LOCATED BETWEEN ELEVATIONS SHOWN IN SECTION "A-A" AT EVERY TEN FEET ALONG VERTICAL PLANE OF CONTAINMENT BUILDING - AS SHOWN IN PLAN.
2. ALL GROSS DEFORMATION MEASUREMENTS SHALL BE OBTAINED BY AN INVAR WIRE EXTENSOMETER WITH A LINEAR POTENTIOMETER SENSING DEVICE (19 REQ'D).
3. ALL NECESSARY ATTACHMENTS TO THE LINER OR EQUIPMENT HATCH ARE BY OTHERS.
4. LOCATIONS GIVEN INDICATE THE DESIRED POSITION OF INVAR TAPE, IF INTERFERENCE IS ENCOUNTERED INSIDE THE CONTAINMENT BUILDING, TAPE SHALL BE LOCATED AS CLOSE TO GIVEN POSITION AS POSSIBLE.
5. TOTAL TRAVEL OF POTENTIOMETER SHALL BE 2" MINIMUM EXCEPT FOR VERTICAL MEASUREMENTS AT EL 95'-0, EL 143'-0 AND TOP OF DOME, RADIUS CHANGE AT EL. 95'-0 AND CONCRETE OPENING DIAMETER CHANGE AT EQUIPMENT HATCH WHERE POTENTIOMETERS WITH 1.5" TRAVEL SHALL BE USED.
6. FIELD SHALL MEASURE AND RECORD RADIUS AT EACH POINT WHERE INVAR TAPES IN SECTION A-A ARE ATTACHED TO THE LINER.
7. FIELD SHALL INSTALL THERMOCOUPLES ON OUTSIDE OF CONTAINMENT SHELL. THEY SHALL BE EMBEDDED IN CONCRETE SO NOT EXPOSED TO DIRECT RADIATION FROM SUN. AS A MINIMUM ONE (1) THERMOCOUPLE SHALL BE INSTALLED IN CYLINDER WALL BELOW AND ABOVE INSULATION AND IN THE DOME.
8. STRUCTURAL INTEGRITY TEST CONTRACTOR SHALL SUPPLY RECORDING INSTRUMENTATION NECESSARY FOR THERMOCOUPLES.

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-56	
CONTAINMENT PROOF TEST GROSS DEFORMATION MEASUREMENTS	
MIC. No. 1999MC3795	REV. No. 17A

Item 1 Pressurizer Relief Tank to Gas Analyzer
Item 2 Pressurizer Relief Tank N_2 Supply
Item 3 Pressurizer Relief Tank Makeup



Although the Pressurizer Relief Tank is missile protected, these penetrating lines can become exposed to containment atmosphere if the pressurizer discharge header is breached during the accident.

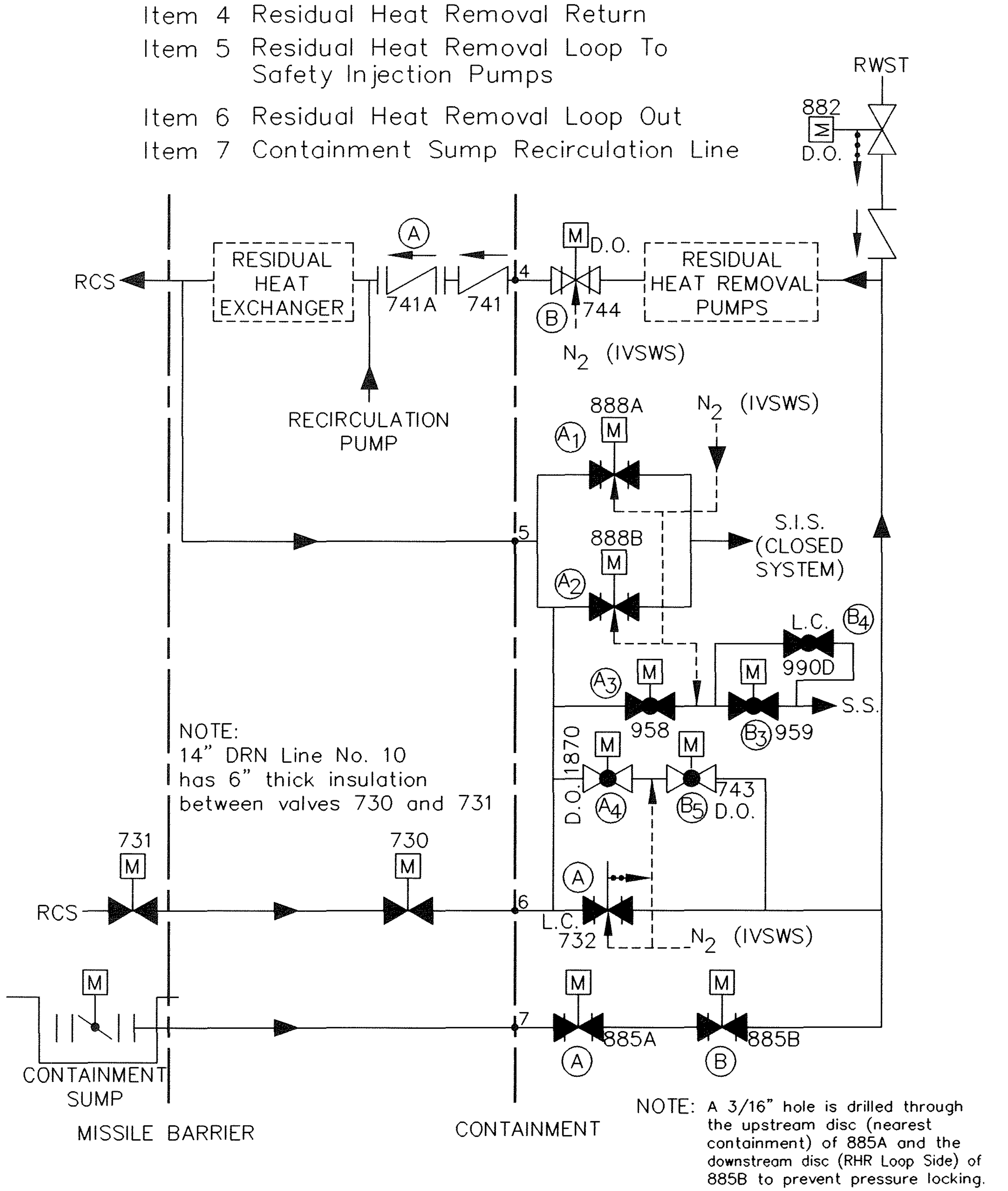
INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.2-1

CONTAINMENT ISOLATION SYSTEM
PENETRATION SCHEMATICS

MIC. No. 1999MC3411	REV. No. 17A

VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE



N_2 – Manual N_2 Pressurization

SS – Sampling System

ENTIRE SYSTEM SHOWN IS SEISMIC CLASS I DESIGN

INDIAN POINT UNIT No. 2

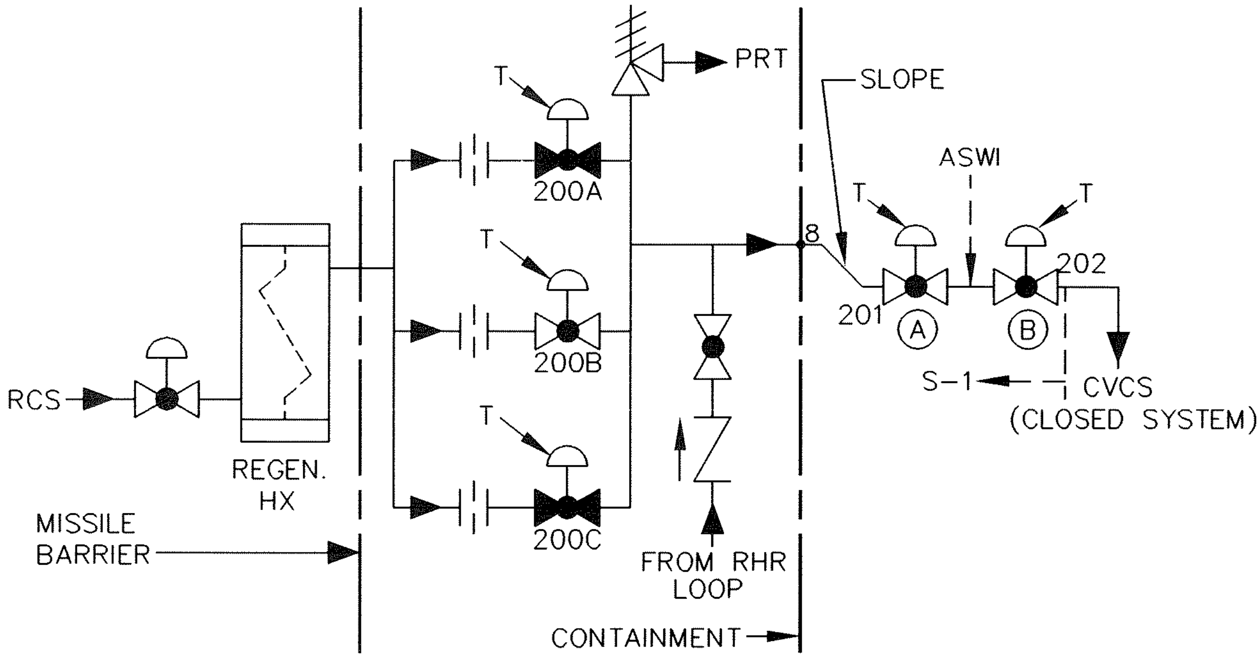
UFSAR FIGURE 5.2–2

CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS

MIC. No. 1999MC3382	REV. No. 17A

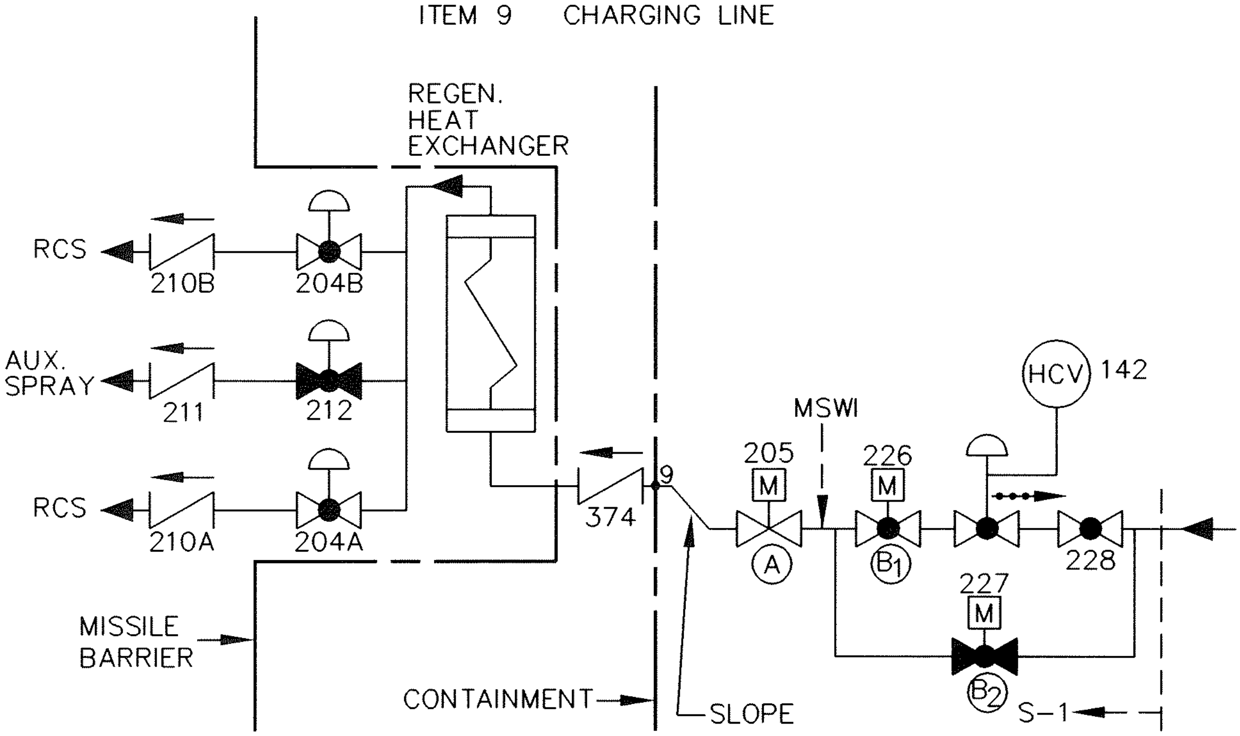
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE

ITEM 8 LETDOWN LINE



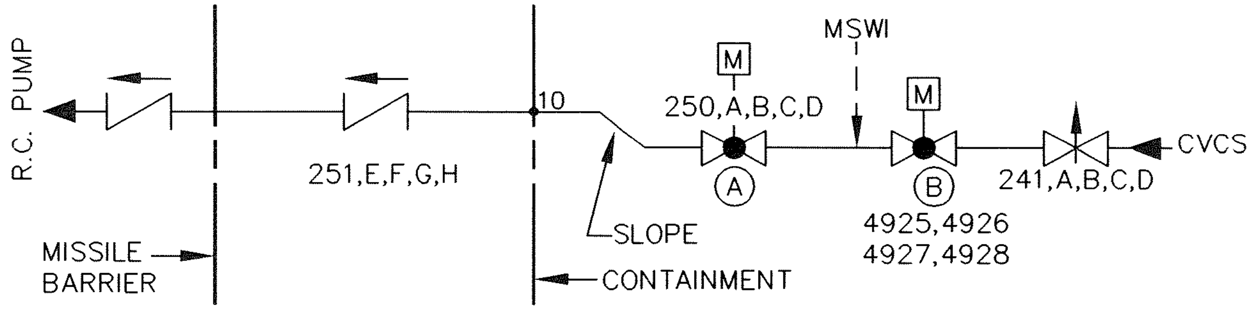
PRT – PRESSURIZER RELIEF TANK

ITEM 9 CHARGING LINE



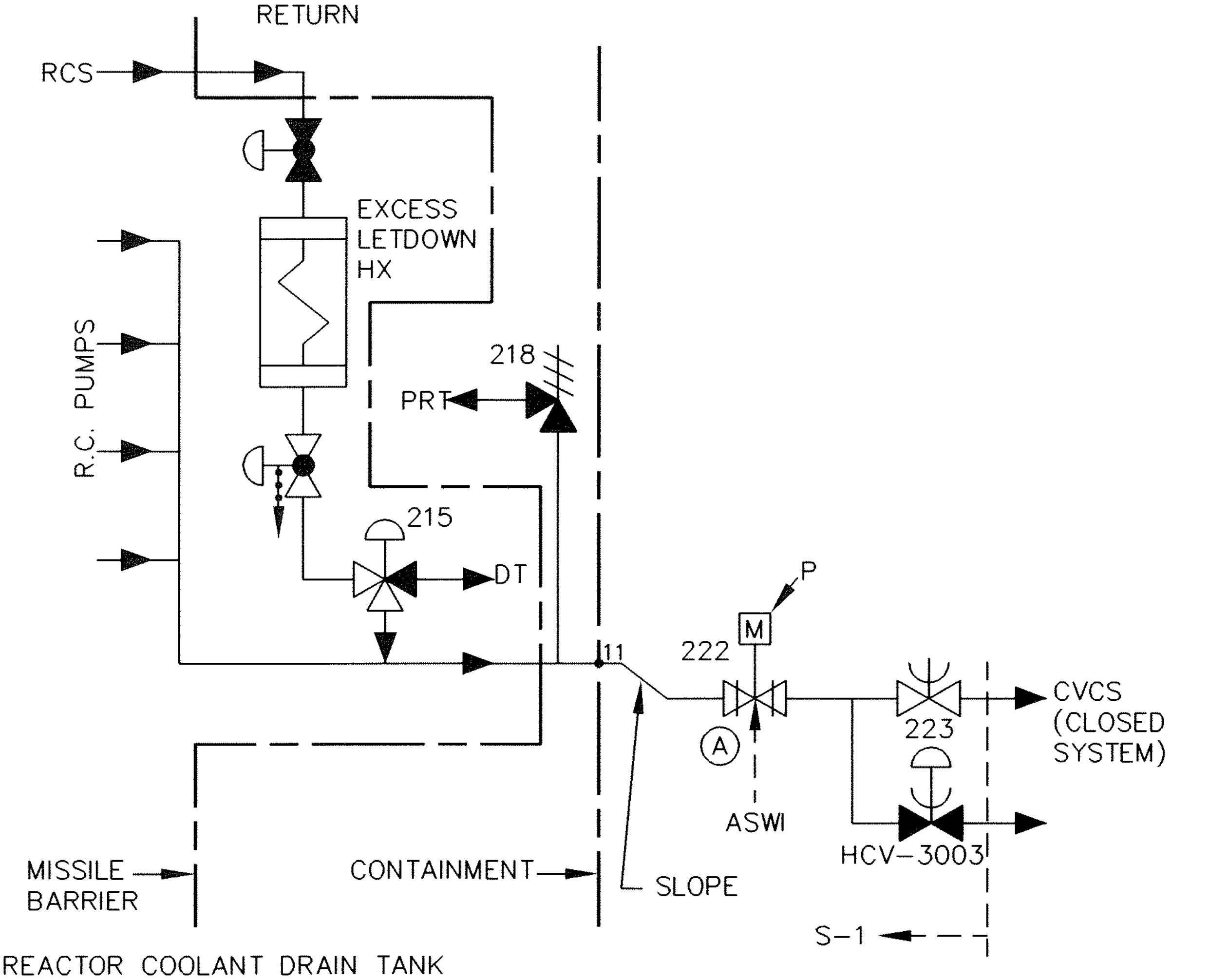
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2–3 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3383	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

ITEM 10 REACTOR COOLANT PUMP SEAL-WATER SUPPLY LINES



ENTIRE SYSTEM SHOWN IS SEISMIC CLASS 1 DESIGN

ITEM 11 REACTOR COOLANT PUMP SEAL-WATER RETURN



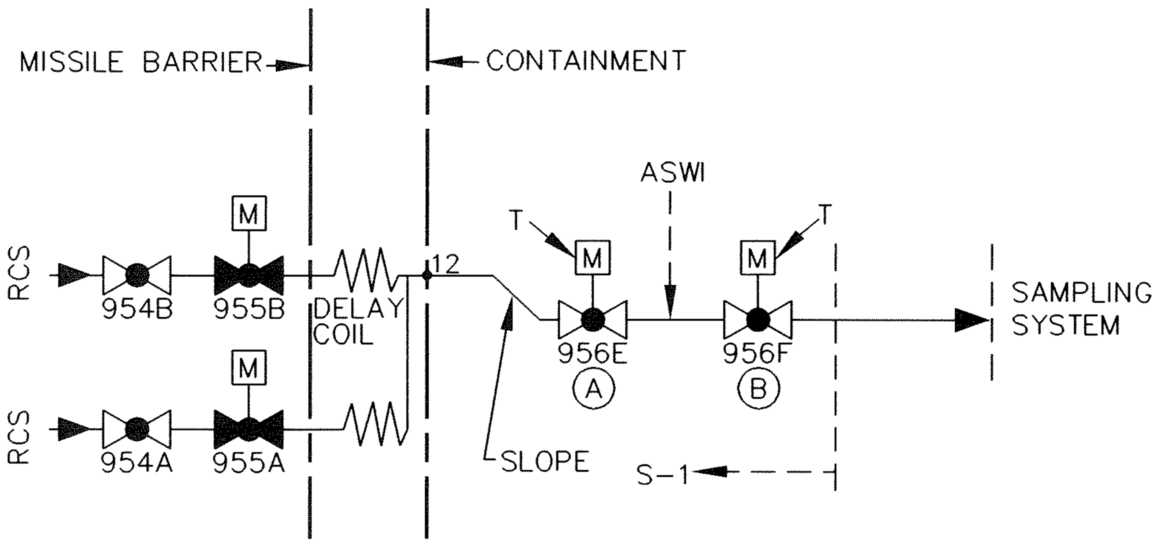
DT – REACTOR COOLANT DRAIN TANK

P – TRIPPED CLOSED BY CONTAINMENT ISOLATION SIGNAL PHASE B

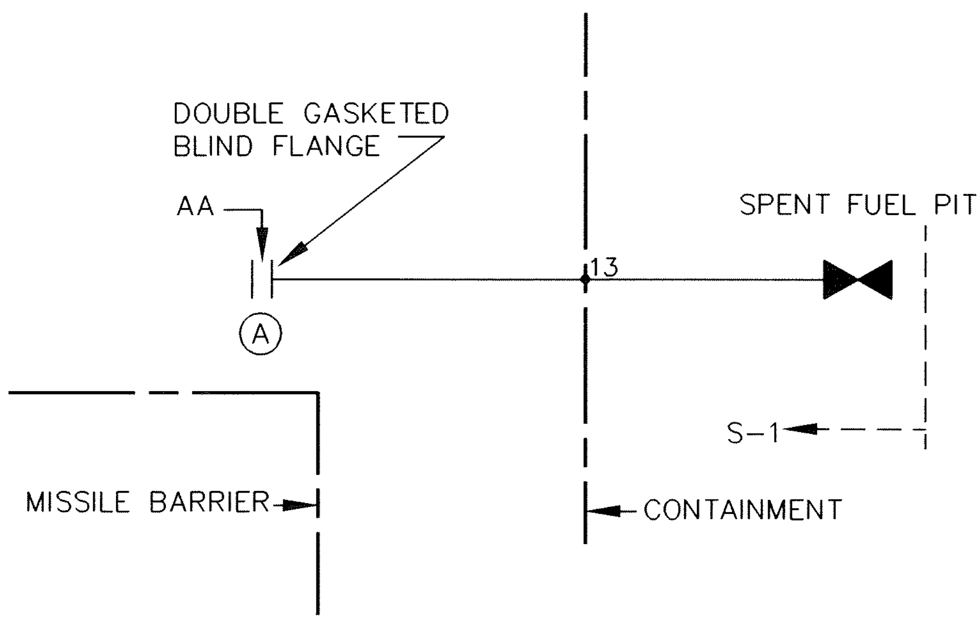
PRT – PRESSURIZER RELIEF TANK

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2-4 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3385	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

ITEM 12 REACTOR COOLANT SYSTEM SAMPLE LINES



ITEM 13 FUEL TRANSFER TUBE



INDIAN POINT UNIT No. 2

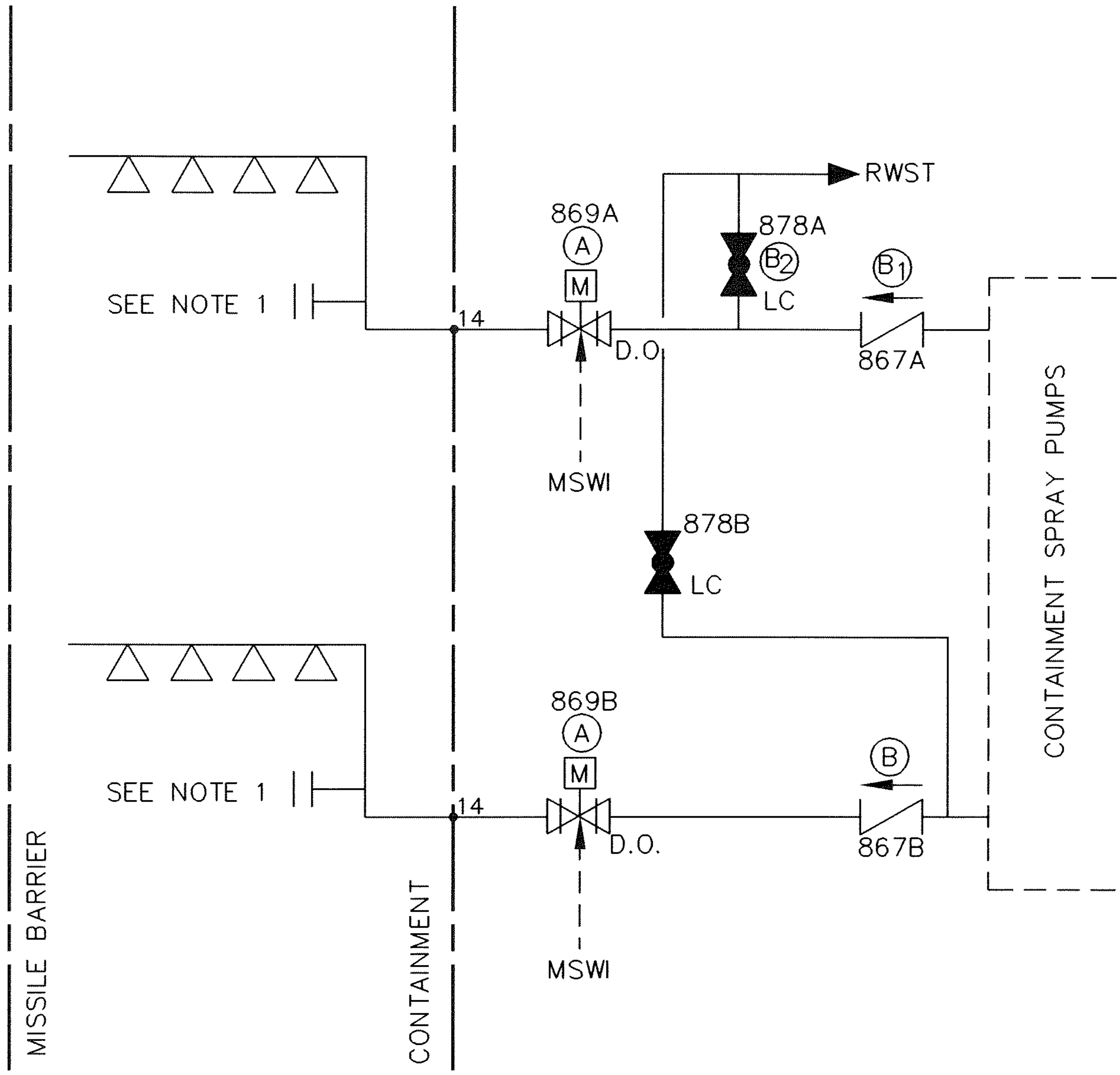
UFSAR FIGURE 5.2-5

CONTAINMENT ISOLATION SYSTEM
PENETRATION SCHEMATICS

MIC. No. 1999MC3386	REV. No. 17A

VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE

ITEM 14 CONTAINMENT SPRAY HEADERS



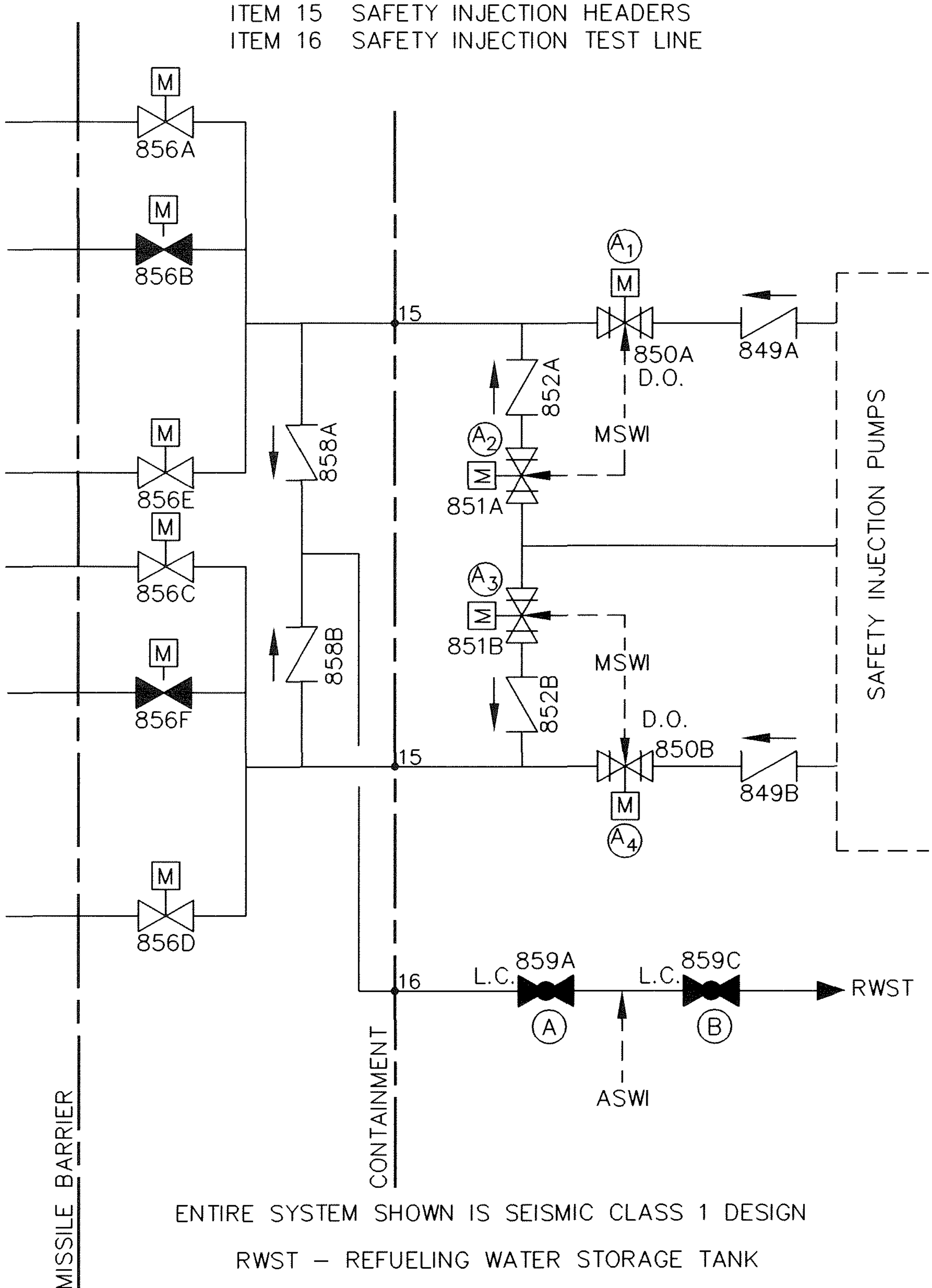
RWST – REFUELING WATER STORAGE TANK

ENTIRE SYSTEM SHOWN IS SEISMIC CLASS 1 DESIGN

NOTE 1:
FLANGED ELBOW USED DURING REFUELING OPERATIONS TO FILL REACTOR REFUELING CAVITY AND CANAL

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2–6 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3387	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

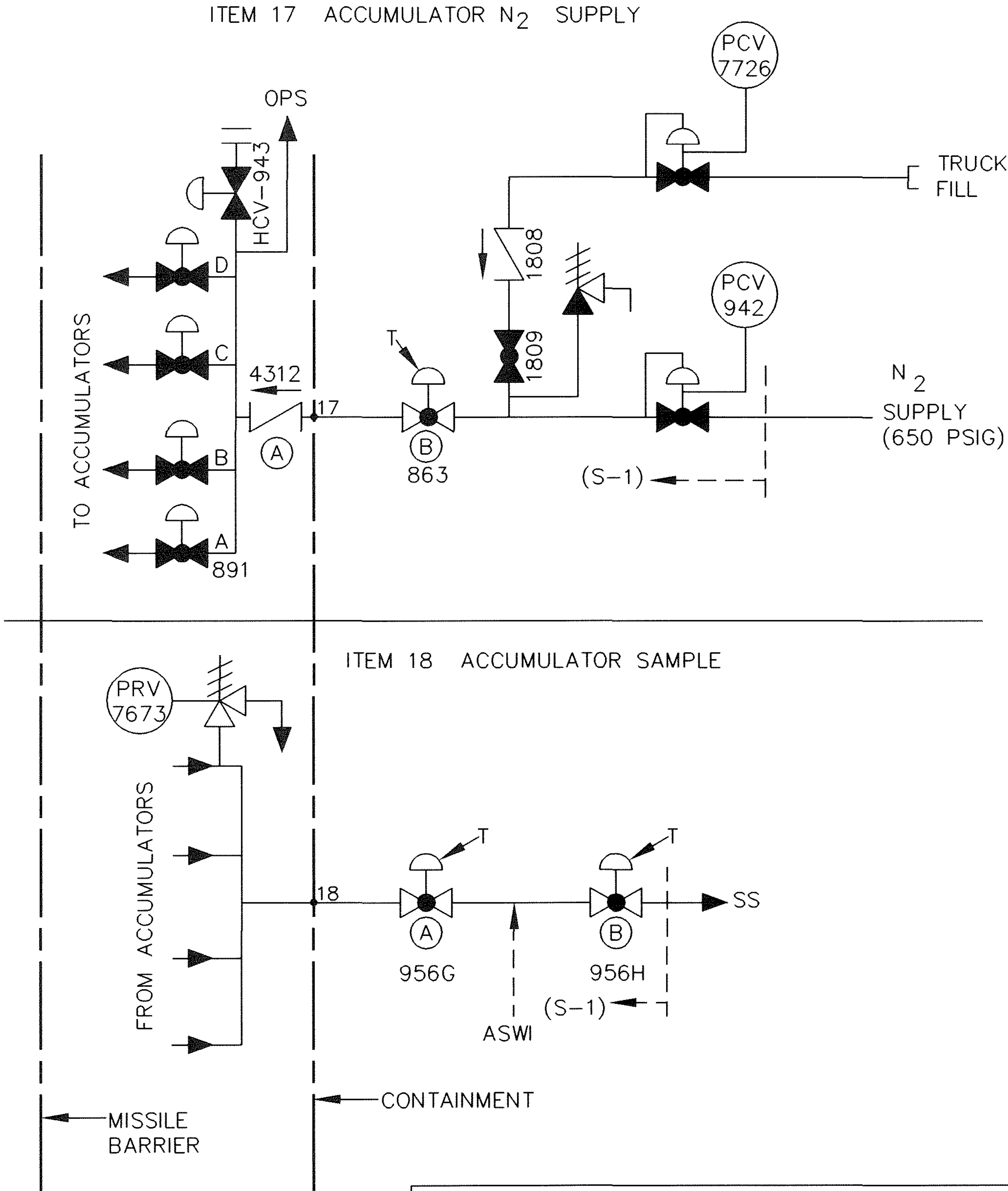
ITEM 15 SAFETY INJECTION HEADERS
ITEM 16 SAFETY INJECTION TEST LINE



ENTIRE SYSTEM SHOWN IS SEISMIC CLASS 1 DESIGN

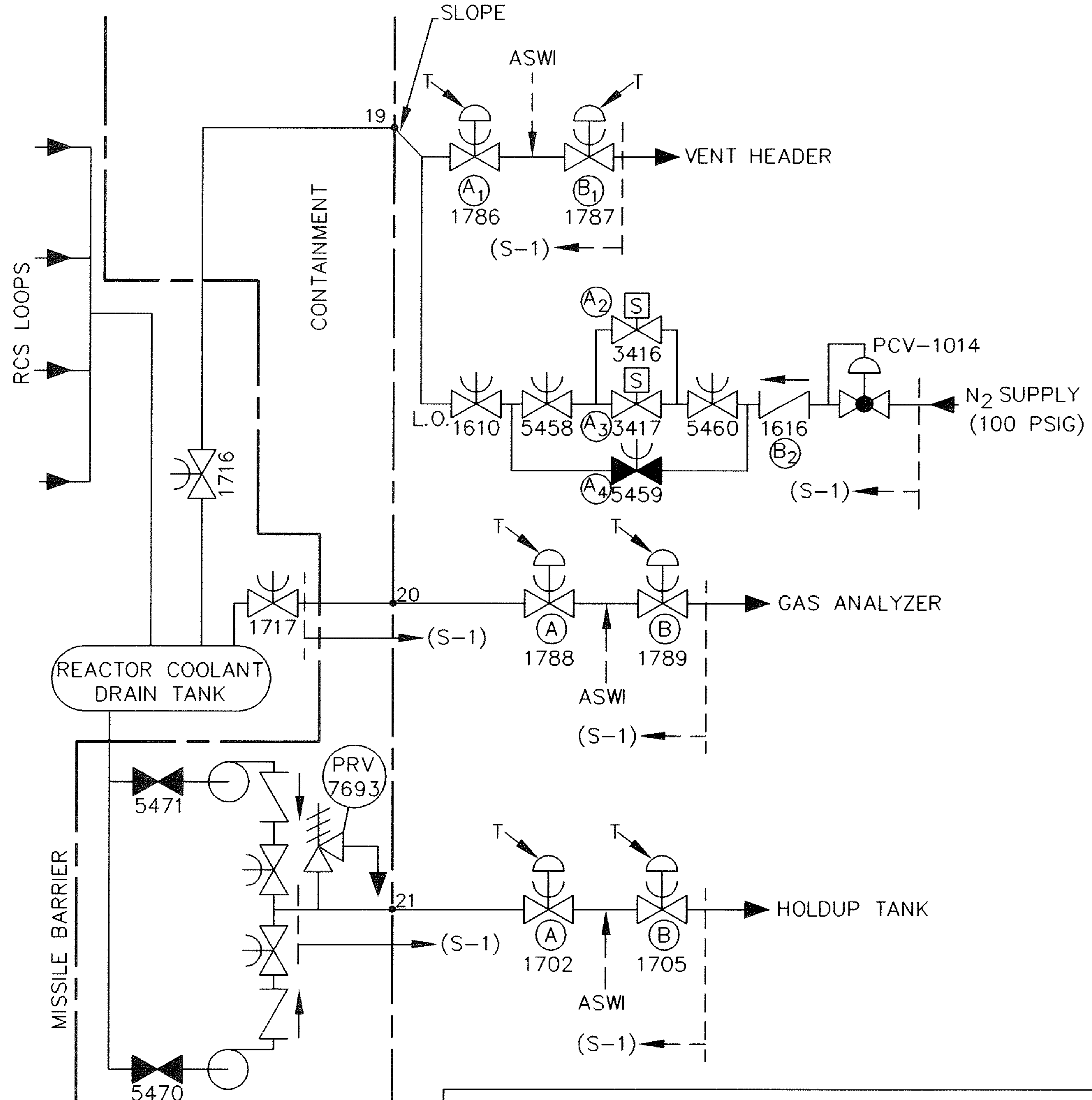
RWST – REFUELING WATER STORAGE TANK

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2–7 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3388	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2-8	
CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3389	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

ITEM 19 PRIMARY SYSTEM VENT HEADER AND N_2 SUPPLY LINE
ITEM 20 REACTOR COOLANT DRAIN TANK TO GAS ANALYZER
ITEM 21 RCDT PUMP DISCHARGE



INDIAN POINT UNIT No. 2

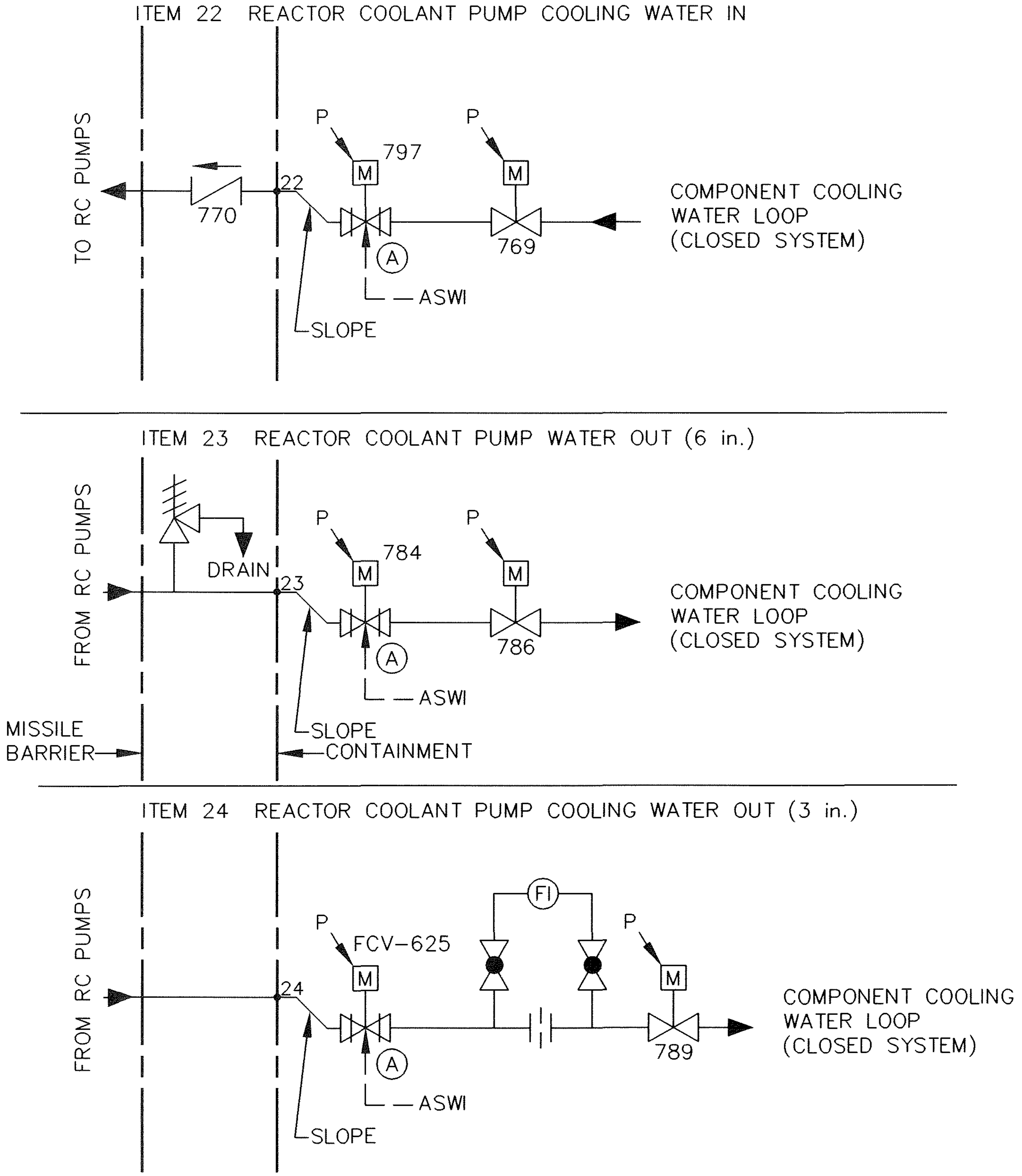
UFSAR FIGURE 5.2-9

CONTAINMENT ISOLATION SYSTEM
PENETRATION SCHEMATICS

MIC. No. 1999MC3390	REV. No. 17A

VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE

OAGI0000215_0681



ENTIRE COMPONENT COOLING WATER SYSTEM IS SEISMIC CLASS 1 DESIGN

INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.2-10

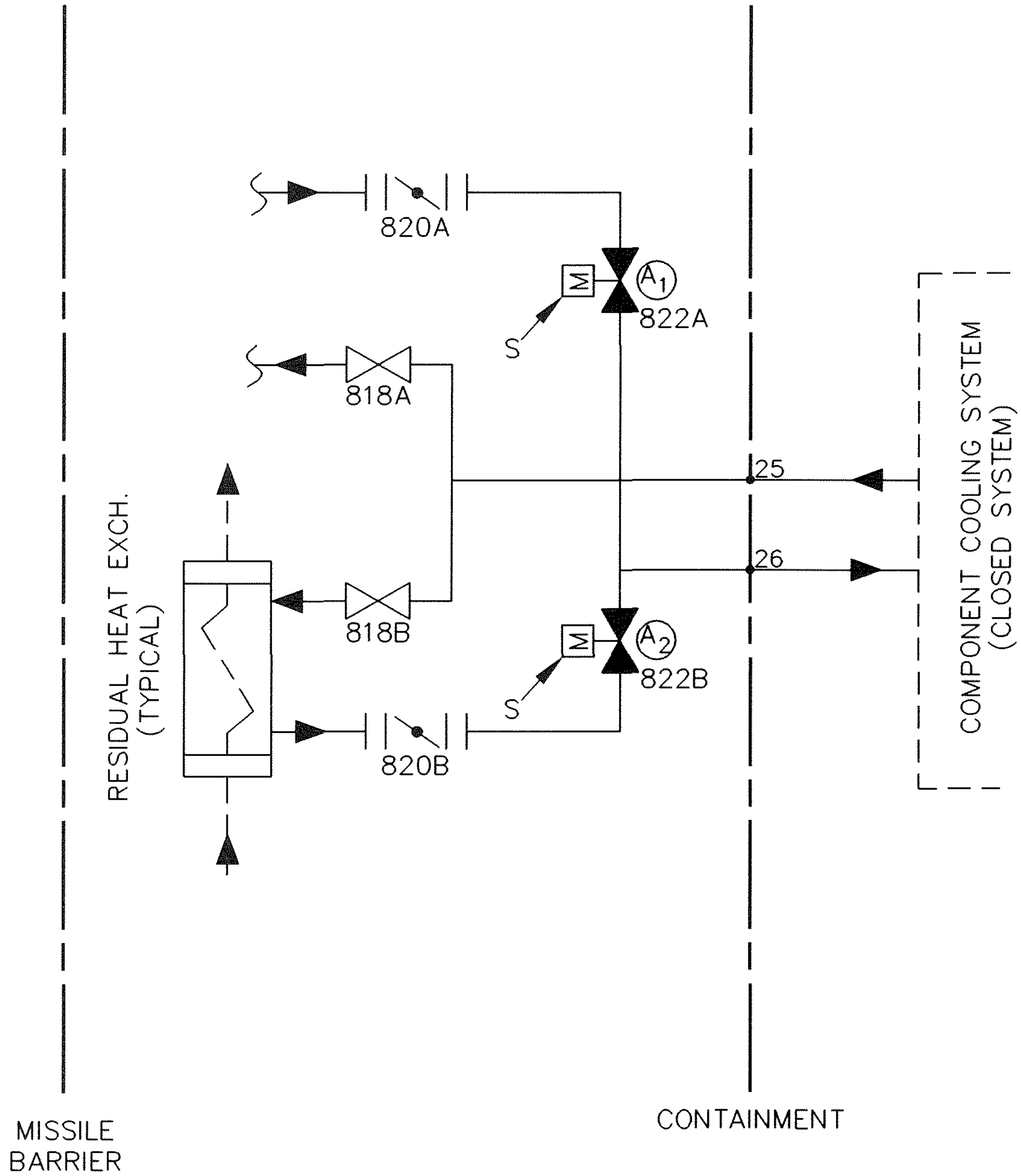
CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS

MIC. No. 1999MC3391 | REV. No. 17A

VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE

ITEM 25 RESIDUAL HEAT EXCHANGER COOLING WATER IN

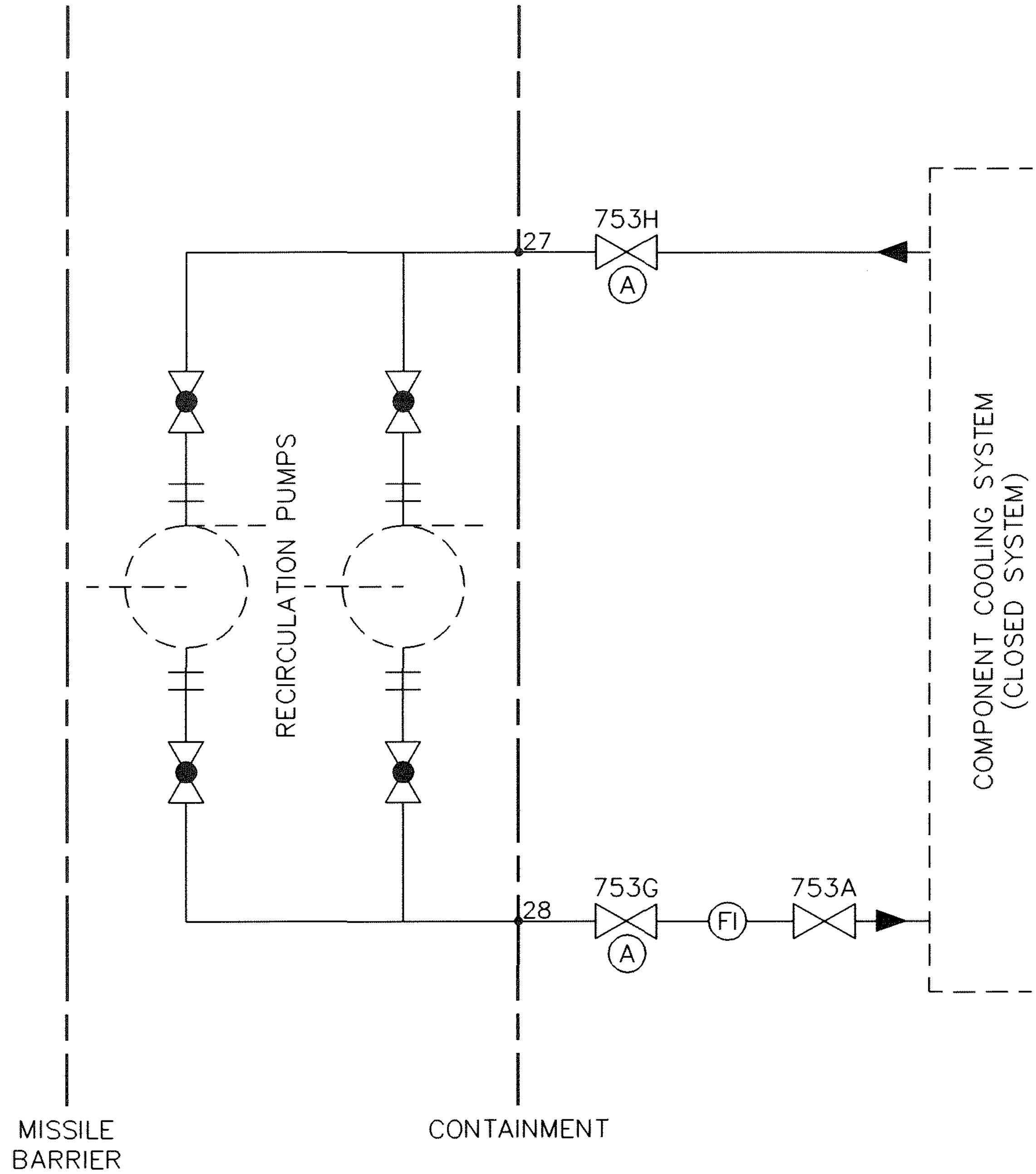
ITEM 26 RESIDUAL HEAT EXCHANGER COOLING WATER RETURN



ENTIRE SYSTEM SHOWN IS SEISMIC CLASS 1 DESIGN
S – OPEN S.I. SIGNAL, PHASE A

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2–11 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3392	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

ITEM 27 RECIRCULATION PUMP COOLING WATER SUPPLY
ITEM 28 RECIRCULATION PUMP COOLING WATER RETURN



ENTIRE COMPONENT COOLING SYSTEM IS SEISMIC CLASS 1 DESIGN

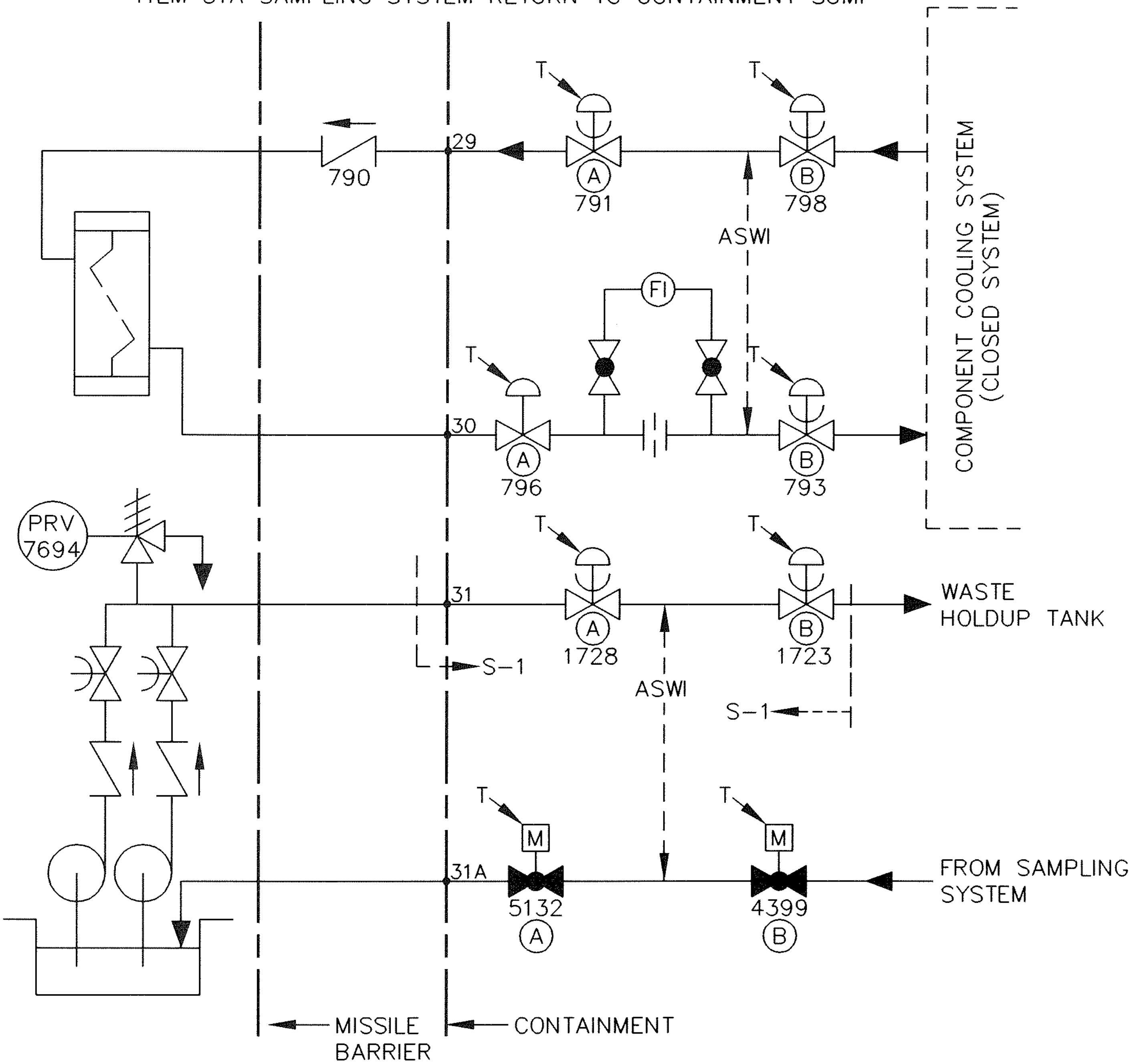
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2-12 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3393	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

ITEM 29 EXCESS LETDOWN HEAT EXCHANGER COOLING WATER IN

ITEM 30 EXCESS LETDOWN HEAT EXCHANGER COOLING WATER OUT

ITEM 31 CONTAINMENT SUMP PUMP DISCHARGE

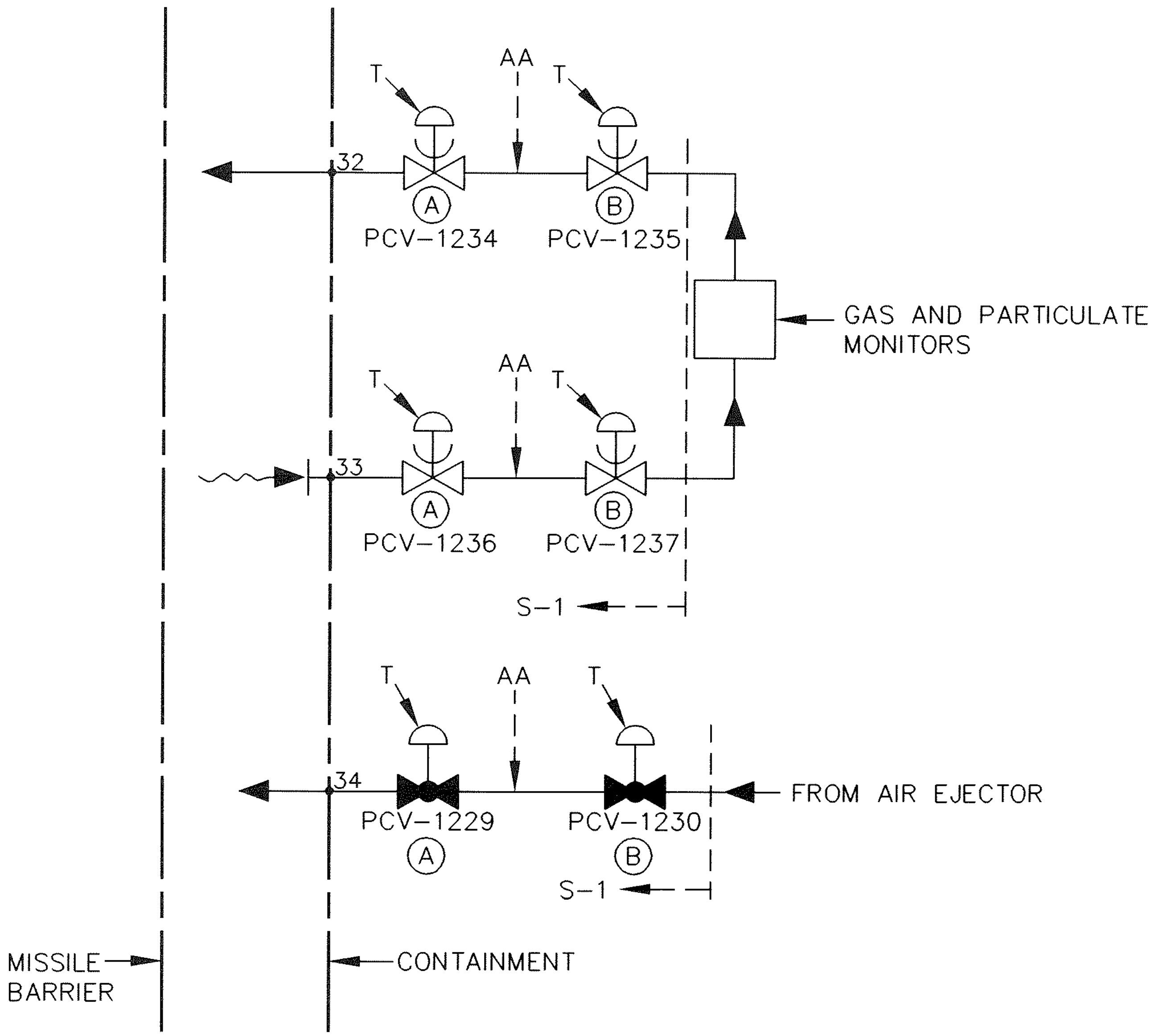
ITEM 31A SAMPLING SYSTEM RETURN TO CONTAINMENT SUMP



ENTIRE COMPONENT COOLING SYSTEM IS SEISMIC CLASS 1 DESIGN

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2-13 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3394	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

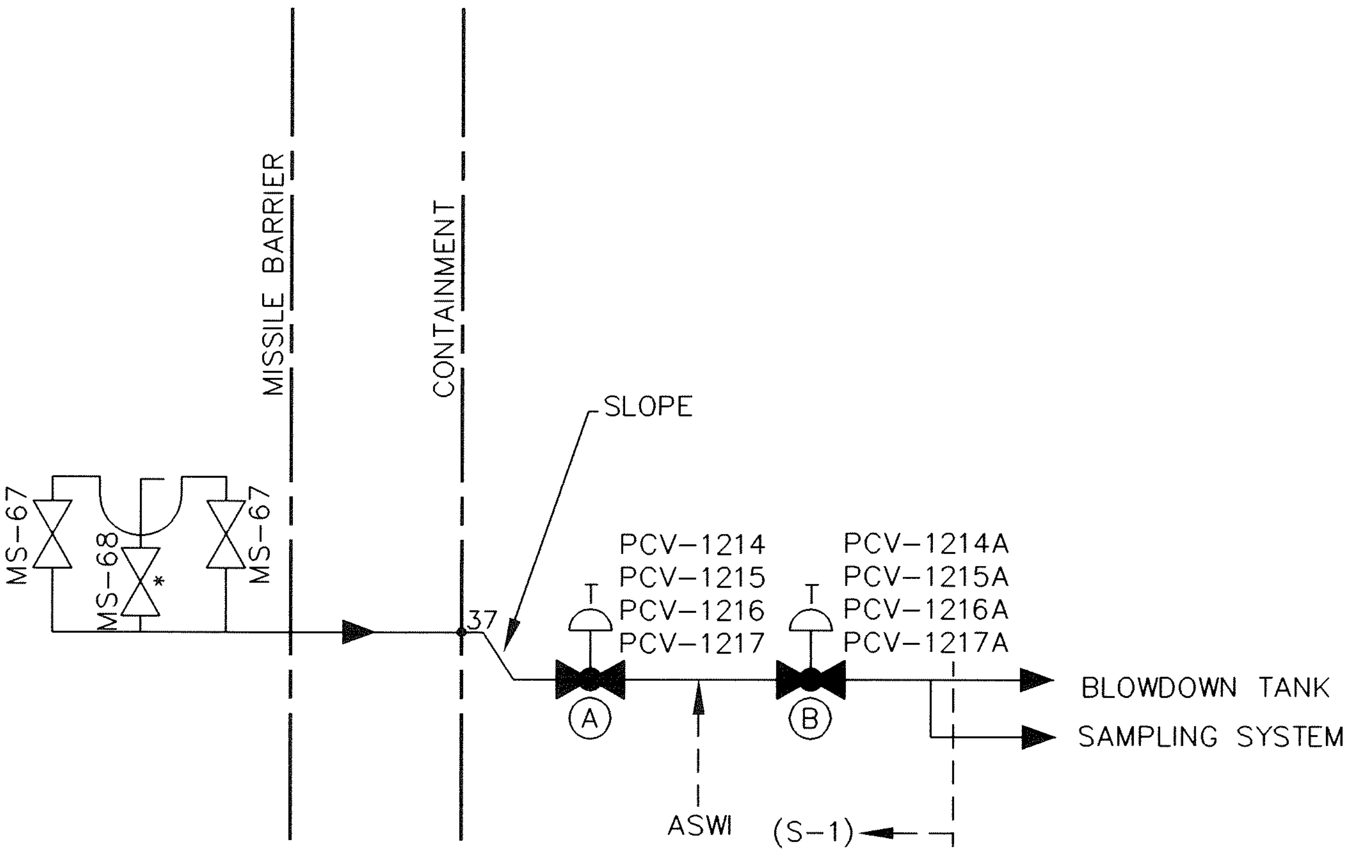
ITEM 32 CONTAINMENT AIR SAMPLE IN
ITEM 33 CONTAINMENT AIR SAMPLE OUT
ITEM 34 AIR EJECTOR DISCHARGE TO CONTAINMENT



AA – AUTOMATIC PRESSURIZATION WITH AIR FROM PENETRATION PRESSURIZATION SYSTEM

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2–14 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3395	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

ITEM 37 STEAM GENERATOR BLOWDOWN/SAMPLE (4 PENETRATIONS)



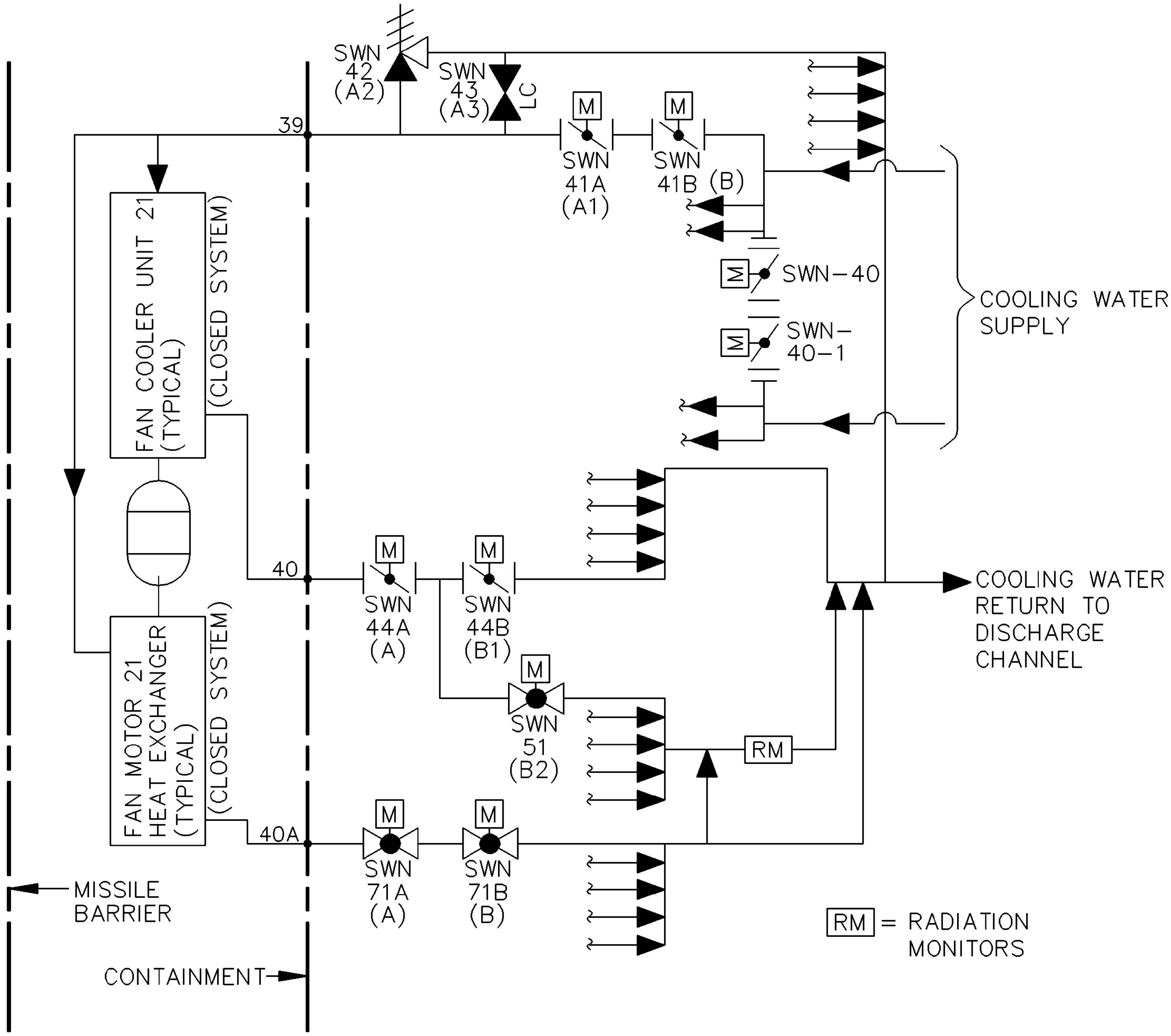
* THIS DRAIN LINE HAS BEEN REMOVED ON STEAM GENERATOR 21

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2-15 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3396	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

ITEM 39 10" VENTILATION SYSTEM COOLING WATER IN (5)

ITEM 40 10" VENTILATION SYSTEM COOLING WATER OUT (5)

ITEM 40A 2" VENTILATION SYSTEM MOTOR COOLING WATER OUT (5)



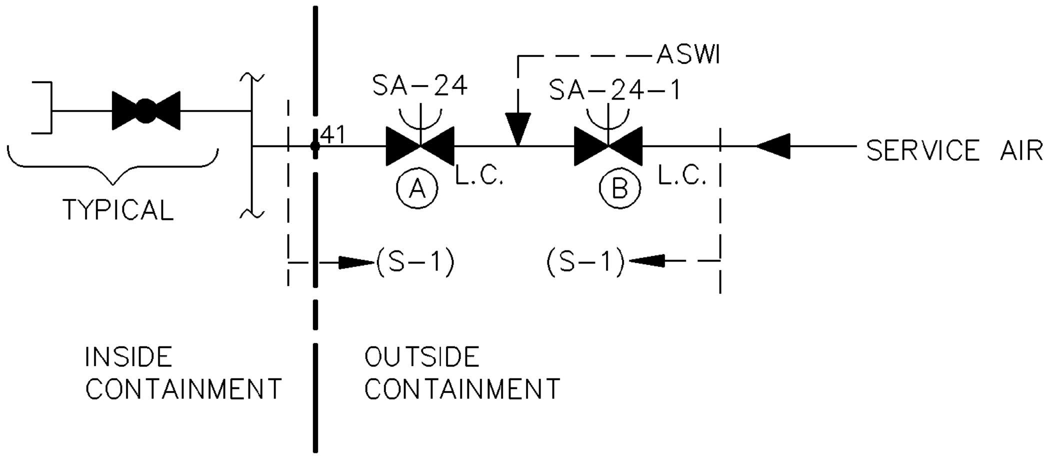
ENTIRE SYSTEM SHOWN IS SEISMIC CLASS 1 DESIGN

ALL MOV'S ARE DEENERGIZED OPEN

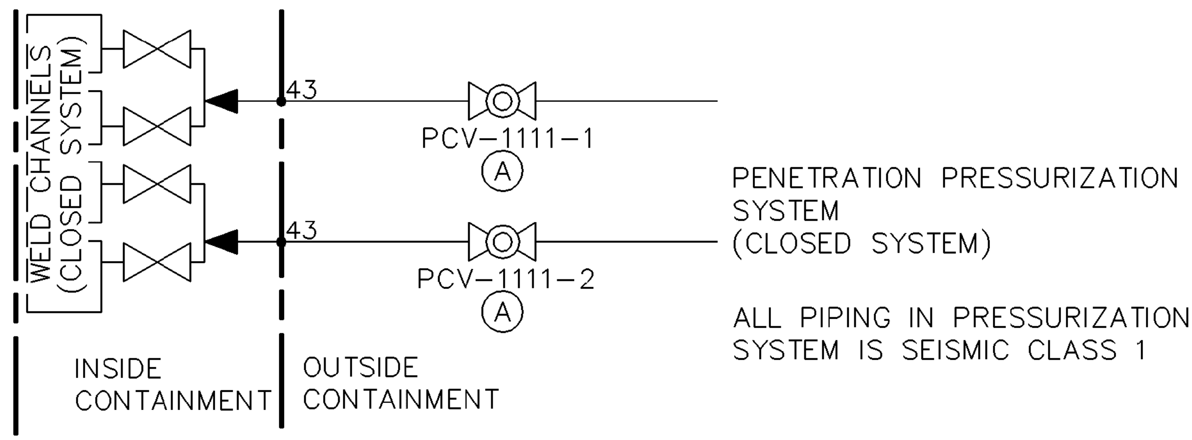
PLANT MANAGEMENT MAY DESIGNATE THE "A" OR THE "B" VALVE(S) IN THE 41, 44 AND 71 SERIES AS THE REQUIRED CONTAINMENT ISOLATION VALVE(S). DESIGNATION OF THE "B" VALVE(S) IN THE SWN-44 SERIES REQUIRES THE CODESIGNATION OF THE RESPECTIVE 51 SERIES VALVE(S) AS ADDITIONAL REQUIRED CONTAINMENT ISOLATION VALVE(S). CONTAINMENT ISOLATION VALVE SURVEILLANCE REQUIREMENTS MUST HAVE BEEN SATISFIED FOR THE DESIGNATED VALVE(S).

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2-16 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3397	REV. No. 17B
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

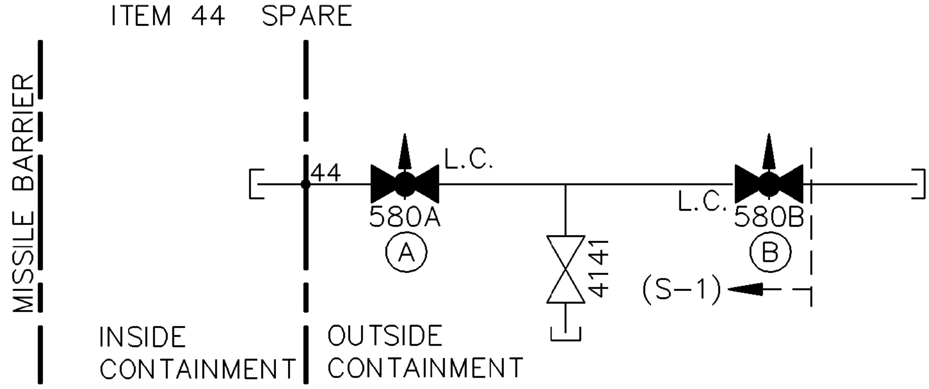
ITEM 41 SERVICE AIR



ITEM 43 WELD CHANNEL PRESSURIZATION AIR SUPPLY



ITEM 44 SPARE



INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.2-17

CONTAINMENT ISOLATION SYSTEM
PENETRATION SCHEMATICS

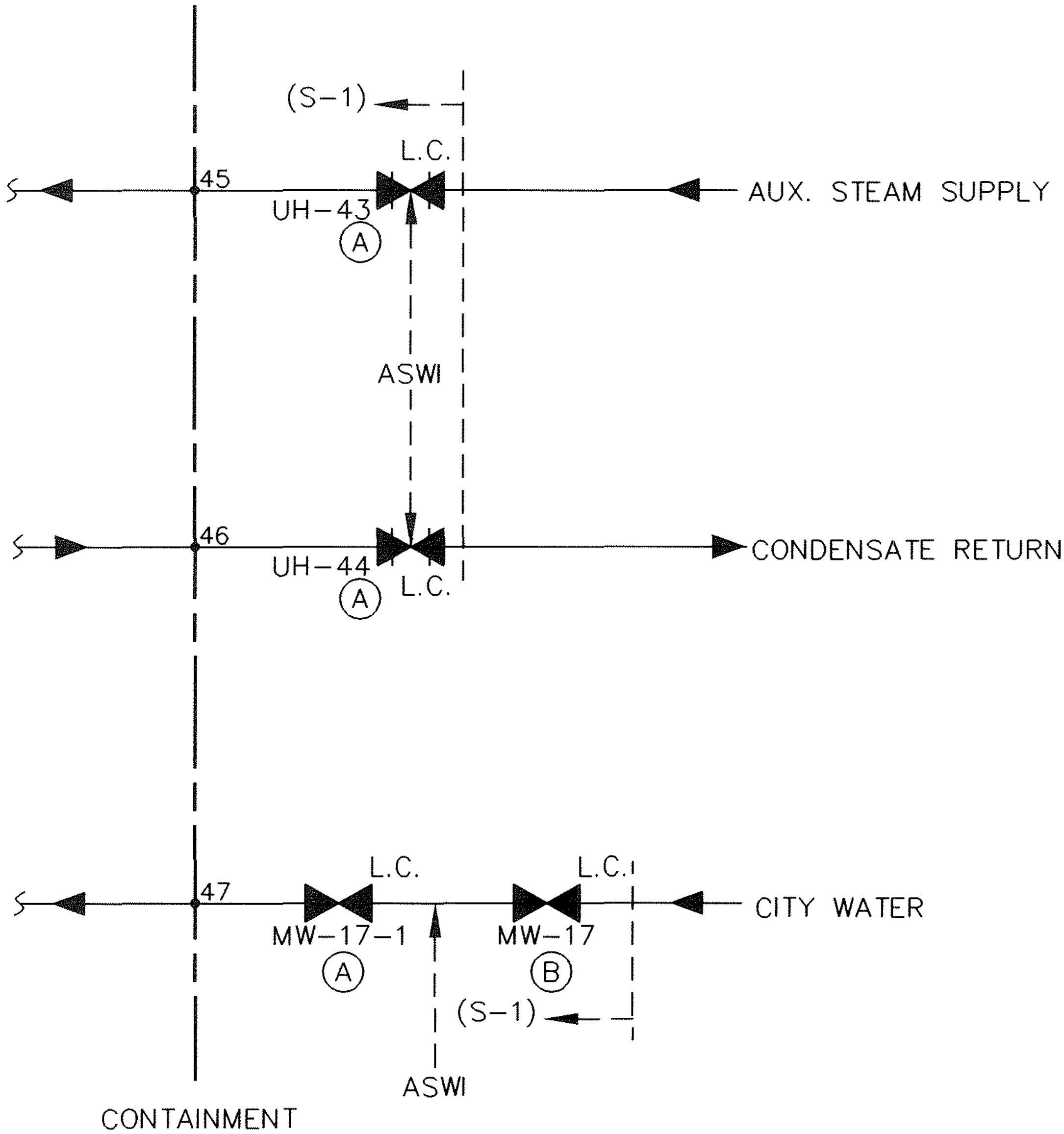
MIC. No. 1999MC3398	REV. No. 17B

VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE

ITEM 45 AUXILIARY STEAM SUPPLY

ITEM 46 AUXILIARY STEAM CONDENSATE RETURN

ITEM 47 CITY WATER



INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.2-18

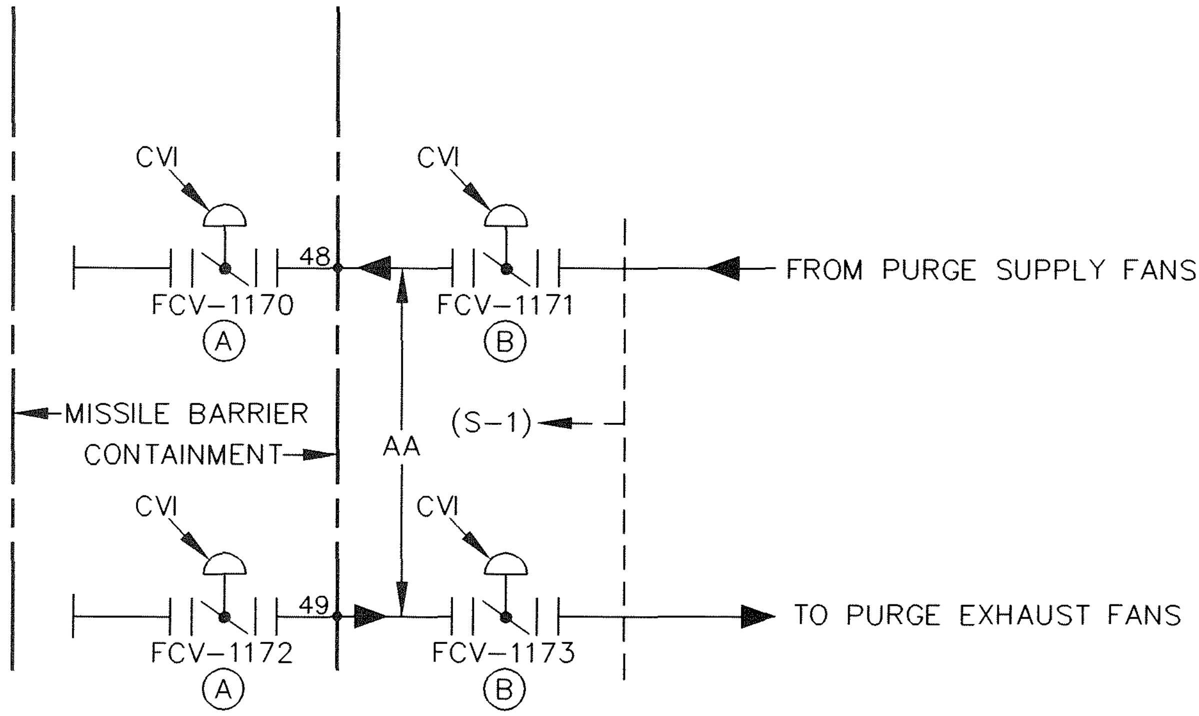
CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS

MIC. No. 1999MC3399	REV. No. 17A

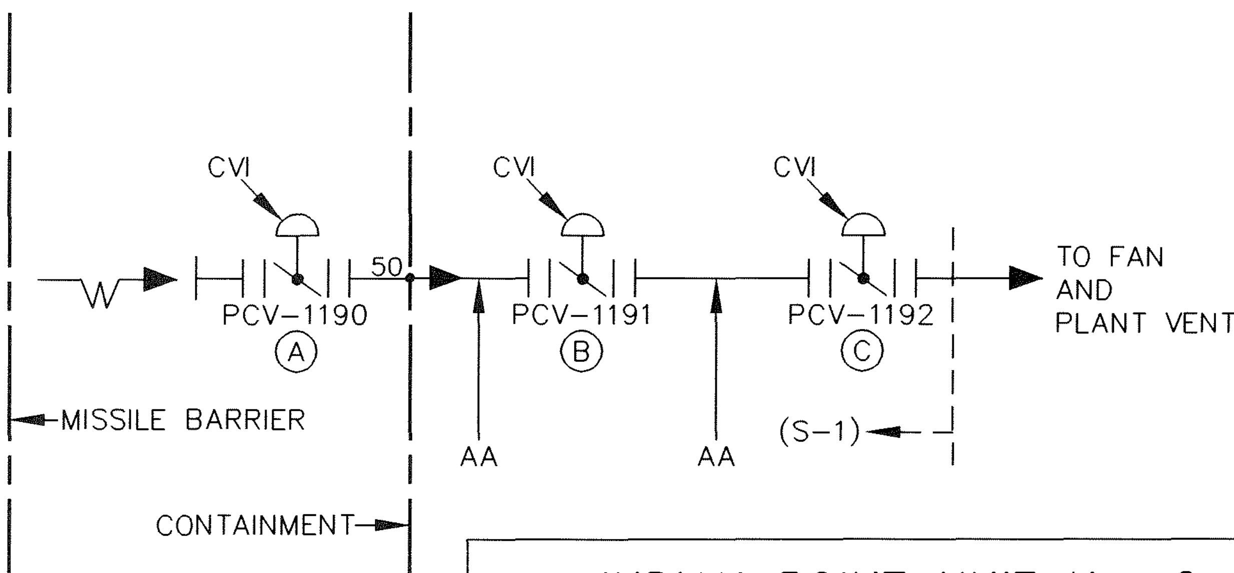
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE

OAGI0000215_0690

ITEM 48 PURGE SUPPLY DUCT
ITEM 49 PURGE EXHAUST DUCT



ITEM 50 CONTAINMENT PRESSURE RELIEF



INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.2–19

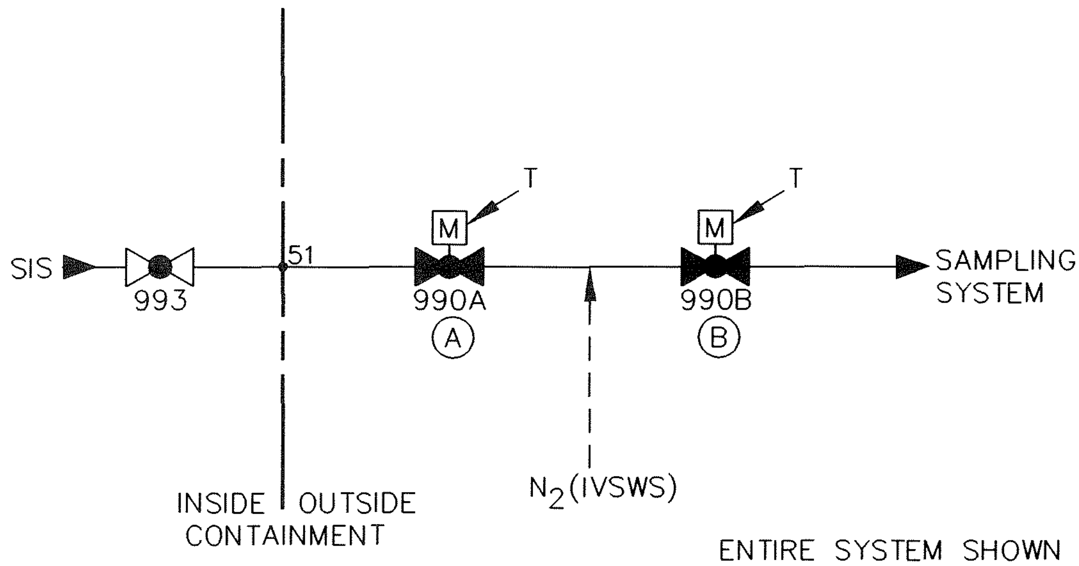
CONTAINMENT ISOLATION SYSTEM
PENETRATION SCHEMATICS

MIC. No. 1999MC3400 | REV. No. 17A

VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE

OAGI0000215_0691

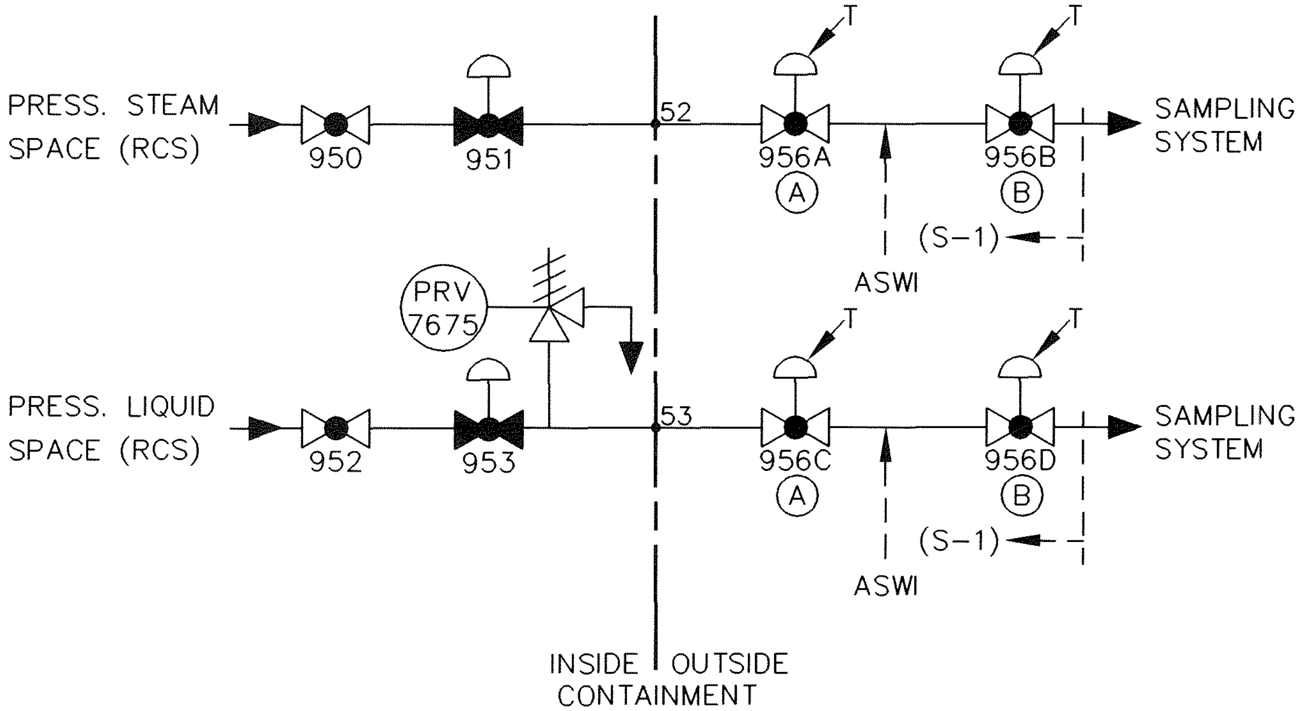
ITEM 51 RECIRCULATION PUMP DISCHARGE SAMPLE LINE



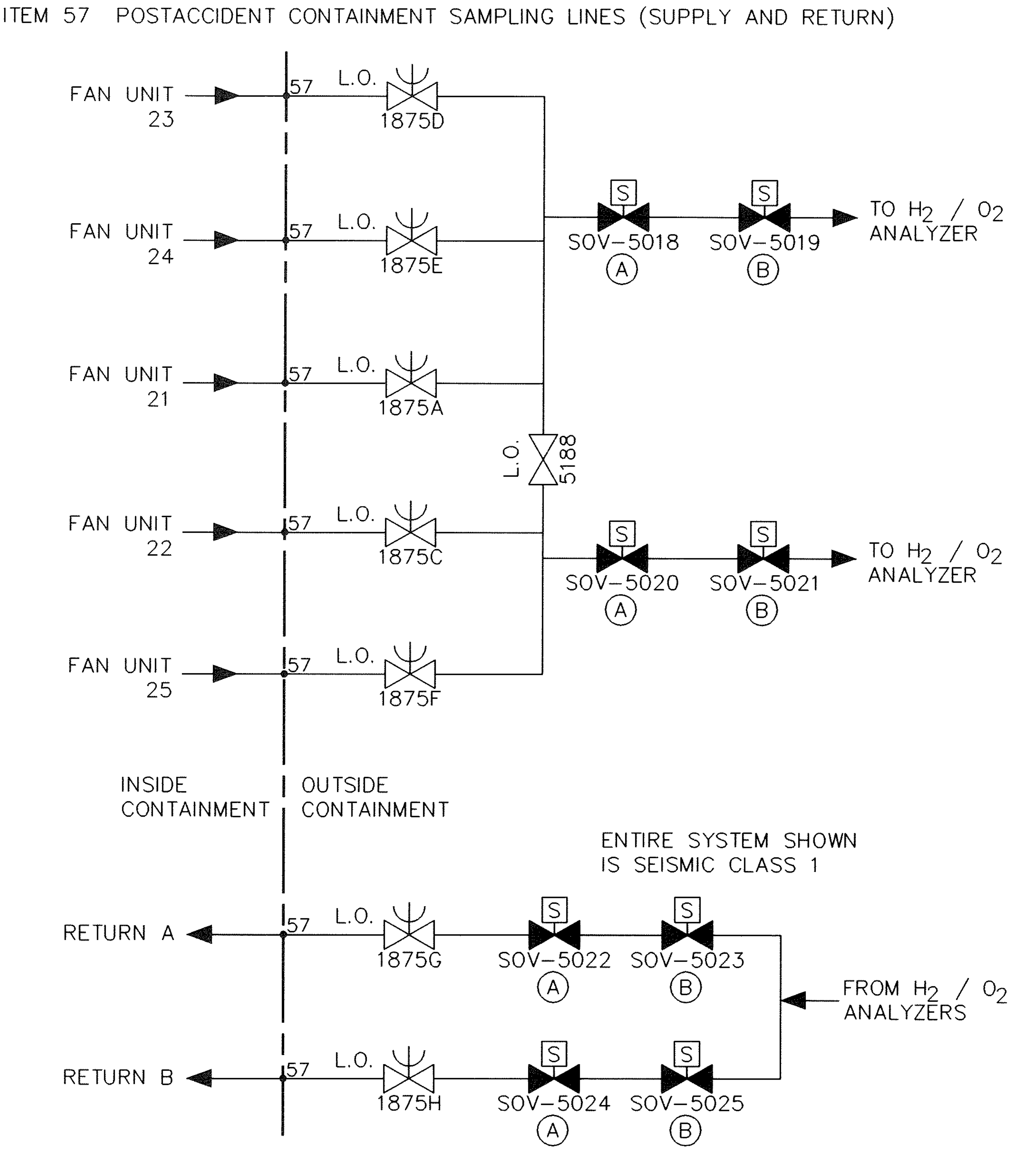
ENTIRE SYSTEM SHOWN IS SEISMIC CLASS 1

ITEM 52 PRESSURIZER STEAM SPACE SAMPLE

ITEM 53 PRESSURIZER LIQUID SPACE SAMPLE

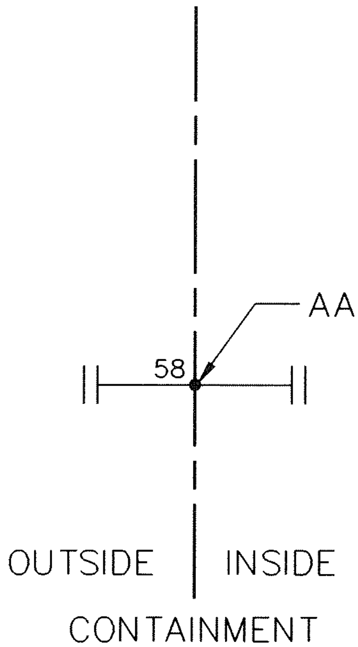


INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2–20 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3401	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	



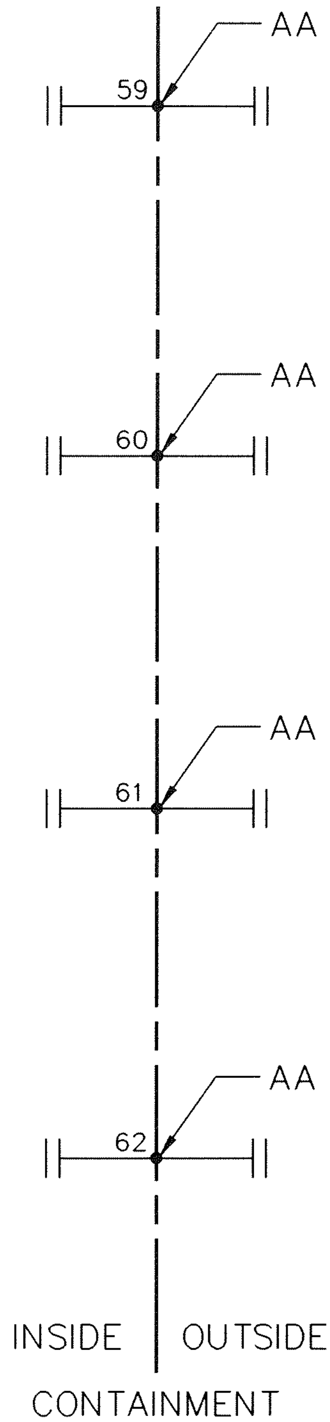
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2-22 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3403	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

ITEM 58 SPARE



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2-23 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3404	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

ITEMS 59, 60, 61, and 62 SPARE



INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.2-24

CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS

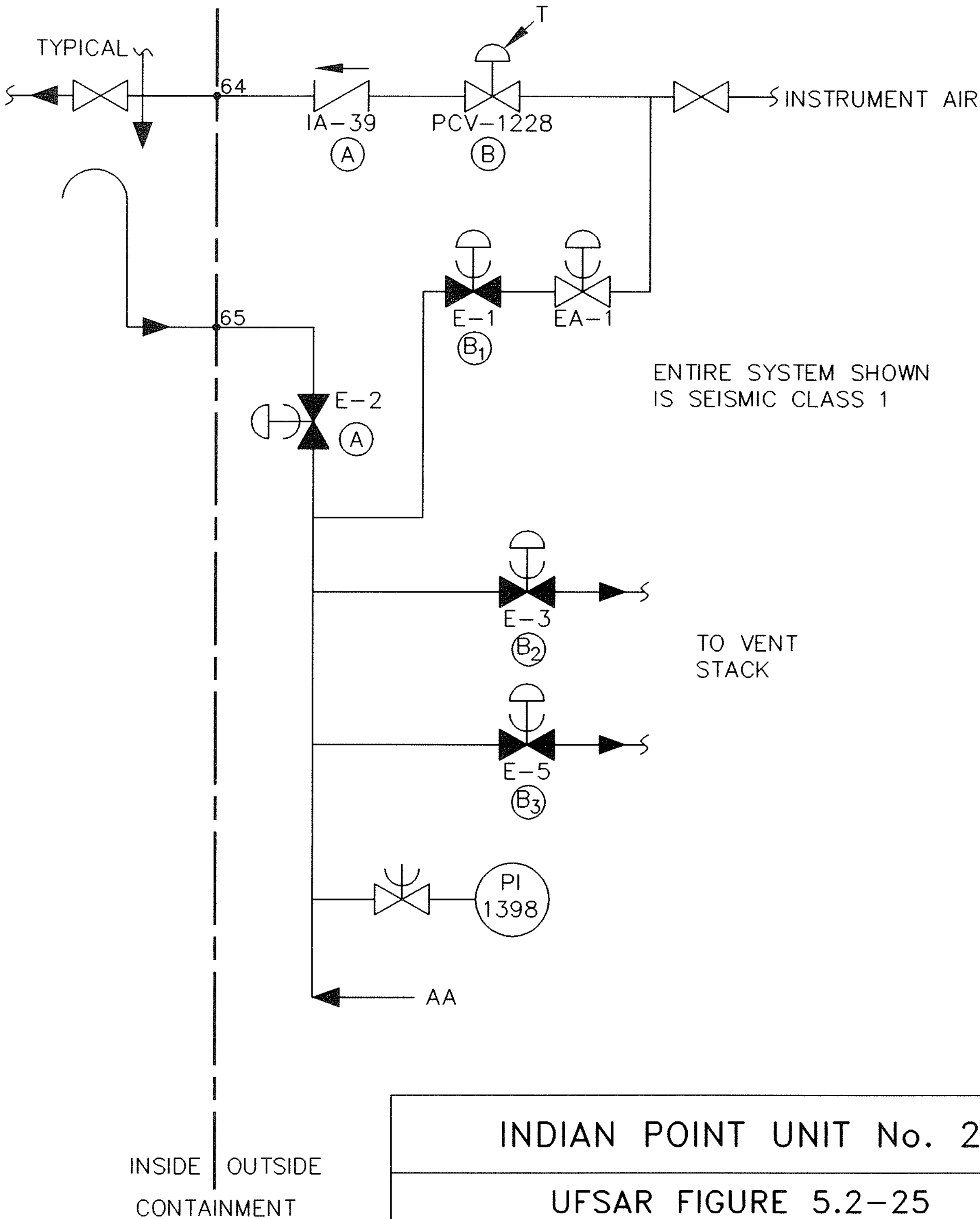
MIC. No. 1999MC3405	REV. No. 17A

VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE

POSTACCIDENT (P.A.) CONTAINMENT VENTING SYSTEM

ITEM 64 INSTRUMENT AIR/P.A. VENTING SUPPLY LINE

ITEM 65 P.A. VENTING EXHAUST LINE



INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.2-25

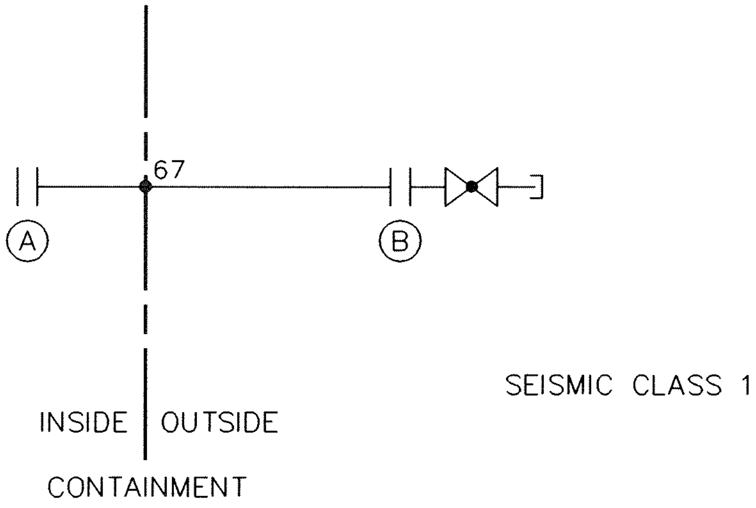
CONTAINMENT ISOLATION SYSTEM
PENETRATION SCHEMATICS

MIC. No. 1999MC3406 | REV. No. 17A

VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE

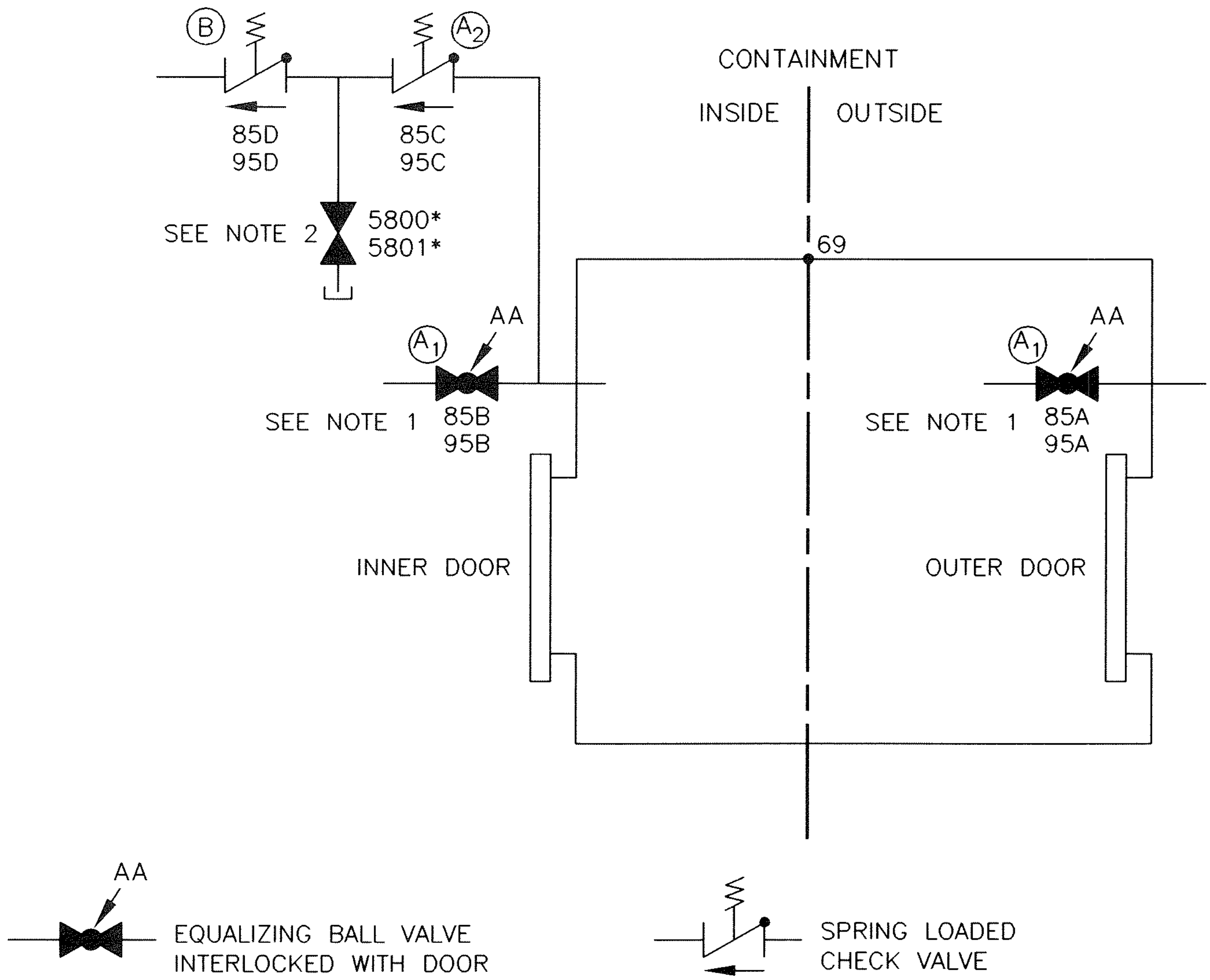
ITEM 66 DELETED

ITEM 67 CONTAINMENT LEAK TEST AIR LINE (2)



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2-26 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3407	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

ITEM 69 PERSONNEL AIR LOCK (2)



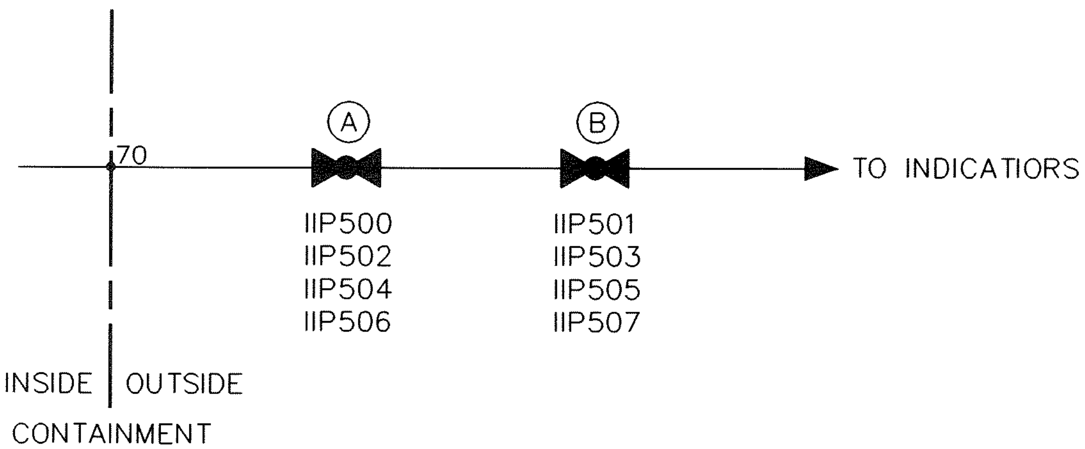
ENTIRE SYSTEM SHOWN IS SEISMIC CLASS 1 DEISGN

NOTE 1 : 85A & 95A MAY BE OPEN WHEN 85B & 95B ARE CLOSED.
85B & 95B MAY BE OPEN WHEN 85A & 95A ARE CLOSED.

2 : TEST VALVES 5800 & 5801 ARE LOCATED AT 83 FT. & 95 FT. AIRLOCKS RESPECTIVELY

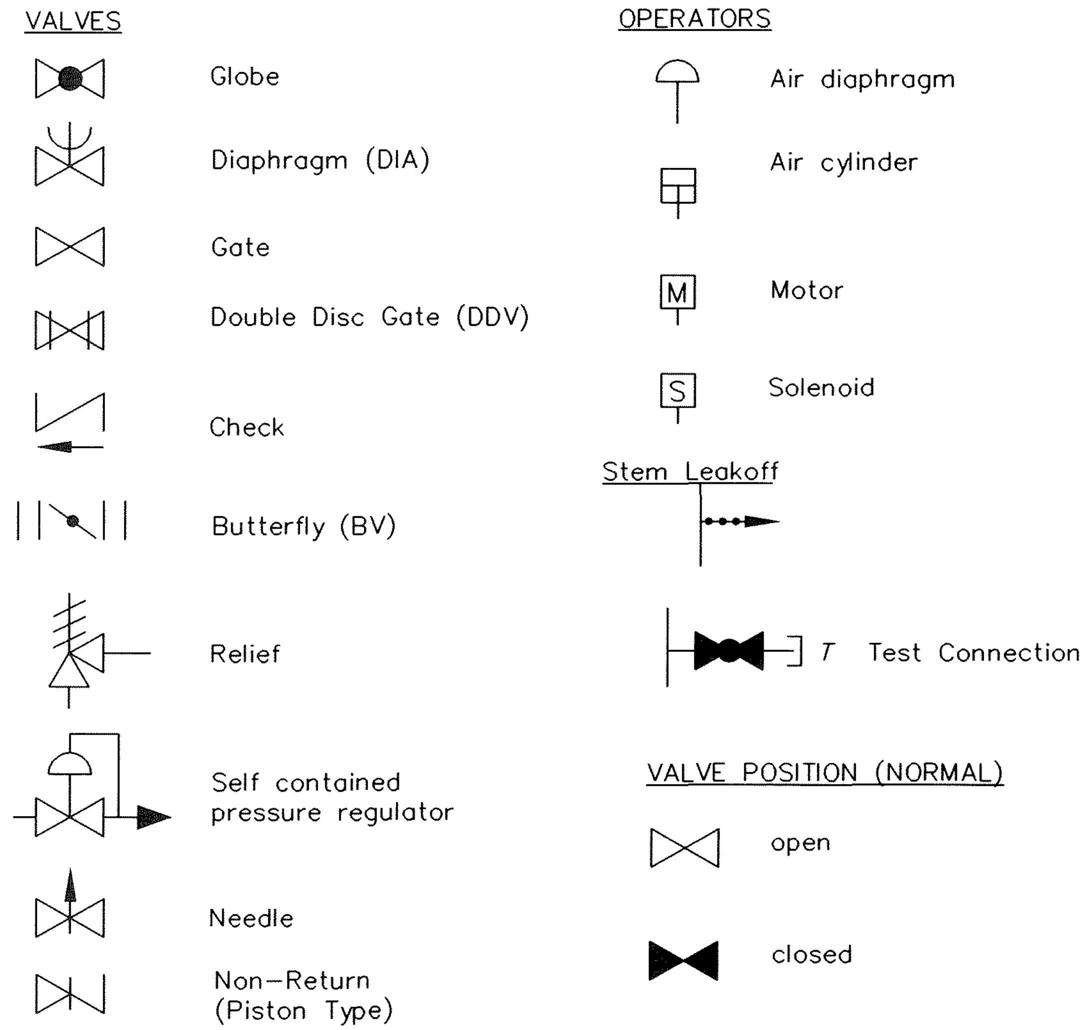
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2–27 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3408	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

ITEM 70 STEAM GENERATOR LEVEL INDICATION LINES (2)
PRESURIZER LEVEL INDICATION LINES (1)
PRESSURIZER PRESSURE INDICATION LINES (1)



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2-28 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3409	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

LEGEND FOR SYMBOLS, CONTAINMENT ISOLATION SYSTEM

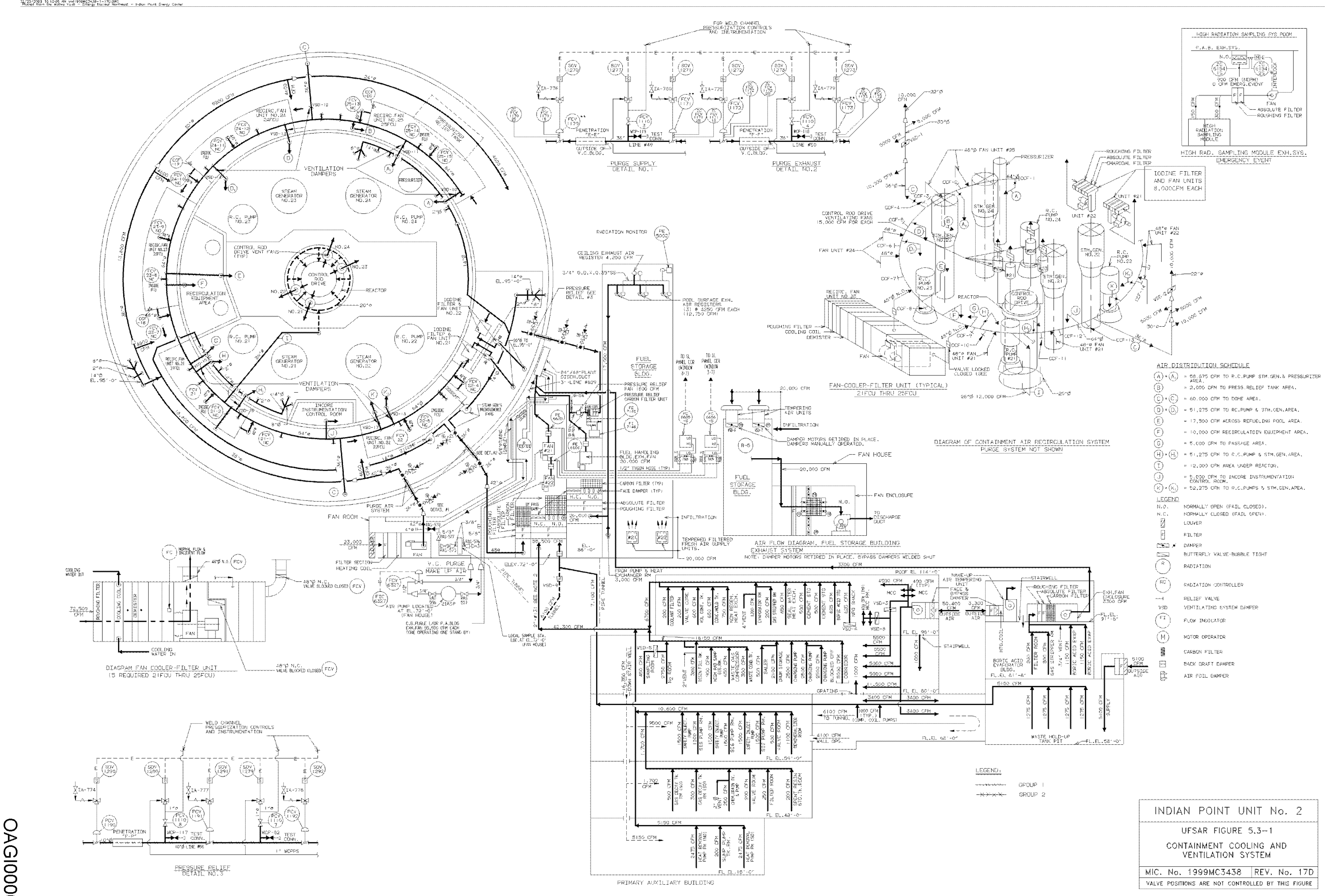


NOTATION

ASWI – AUTOMATIC SEAL WATER INJECTION
MSWI – MANUAL SEAL WATER INJECTION
AA – AUTOMATIC PRESSURIZATION WITH AIR
N_2 – MANUAL PRESSURIZATION WITH NITROGEN
LO – LOCKED OPEN
LC – LOCKED CLOSED
P – TRIPPED CLOSED BY CONTAINMENT ISOLATION SIGNAL, PHASE A
T – TRIPPED CLOSED BY CONTAINMENT ISOLATION SIGNAL, PHASE B
S–1 – SEISMIC CLASS I
S – OPEN S.I. Signal, Phase A
D.O. – DEENERGIZED OPEN
CVI – CONTAINMENT VENTILATION ISOLATION (AUTOMATIC SIGNAL)
IVSWS – ISOLATION VALVE SEAL WATER SYSTEM

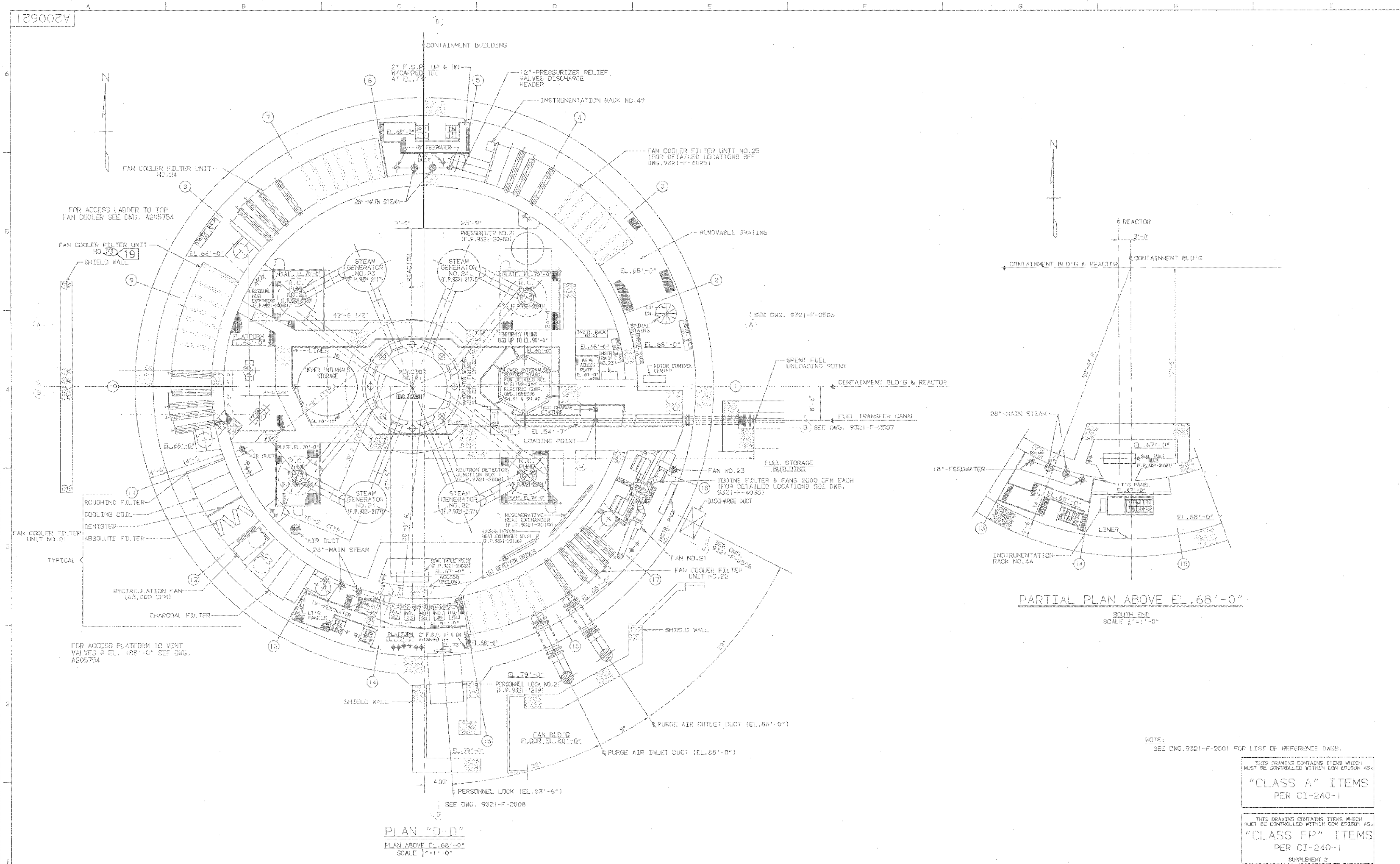
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2–29 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3410	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

OAGI0000215_0700



OAGI0000215_0701

9321-F-2501
PLAN "C-C"
PLAN ABOVE EL. 95'-0"
SCALE 1/8"=1'-0"
CONTAINMENT BLDG.
HOSE REEL 2" HOSE VALVE WITH 1" HOSE
RECIRCULATING AIR DUCT
12" PRESSURIZER RELIEF VALVES DISCH. HEADER
REMOVABLE CHECKERED PL EL. 95'-0"
LINER
POLAR CRANE RAIL
STEAM GENERATOR NO.23
STEAM GENERATOR NO.24
STEAM GENERATOR NO.21
STEAM GENERATOR NO.22
PRESSURIZER NO.22 (FP-9321-20980)
REACTOR No. 21 (BELOW)
CONTROL ROD DRIVE SHIELD
MANIPULATOR CRANE NO.2
LOWER INTERNALS LAYDOWN
UPPER INTERNALS LAYDOWN
EQUIPMENT HATCH N° 22
PERSONNEL LOCK N° 22
R.C. PUMP No.23 (BELOW)
R.C. PUMP No.24 (BELOW)
R.C. PUMP No.21 (BELOW)
R.C. PUMP No.22 (BELOW)
RECIRCULATING PUMPS NO.21&22 (BELOW)
RESIDUAL HEAT EXCHANGER NO.22(BELOW)
CATALYTIC TYPE HYDROGEN RECOMBINER (22PHR)
28" MAIN STEAM
18" FEEDWATER
SEAL TABLE NO.21
FIXED SHIELD
REMOVABLE SHIELDING
SPIRAL STAIR
(F) FOR ACCESS OPENING @ EL.+95'-0" DOWN TO PLATFORM @ EL.+78'-0" SEE DWG A205777
(E) FOR ACCESS PLATFORMS TO STEAM GENERATOR MANWAYS SEE DWG-A 205752
(D) FOR INSTALLATION OF REMOVABLE HANDRAIL AROUND C.R.D.M. BEDSPRING FRAME SEE DWG. A205752
(C) ACCESS LADDER TO STEAM GENERATOR SEE DWG. A205752
(A) FOR ACCESS PLATFORMS TO STEAM GENERATOR MANWAYS SEE DWG. A205752
(B) FOR ACCESS OPENING @ EL +95'-0" DOWN TO PLATFORM @ EL +78'-0" SEE DWG. A205777
(G) ADD ACCESS OPENING & LADDER DOWN TO COOLERS #22 & #25
(H) FOR LIFTING RIG FOR PRESSURIZER MANWAY COVER SEE DWG A205755
(I) ACCESS LADDER TO STEAM GENERATOR SEE DWG. A205752
(J) EQUIPMENT HATCH LAYDOWN AREA FOR REMOVABLE HANDRAILS SEE DWG. A205752
(K) ACCESS OPENING & LADDER DOWN TO FAN COOLER #22 & #25 SEE DWG. A205752
DISCHARGE AIR DUCT FROM IODINE FILTER & FAN UNIT NO.22
DISCHARGE DUCT
EQUIPMENT HATCH & PERSONNEL LOCK LAYDOWN
DISCHARGE AIR DUCT FROM IODINE FILTER & FAN UNIT No.21
REMOVABLE CONCRETE SLAB FOR REGENERATIVE HEAT EXCHANGER No.21 & EXCESS LETDOWN HEAT EXCHANGER No.21 (BELOW)
CONTAINMENT BLDG & REACTOR
FUEL TRANSFER CANAL
FUEL STORAGE BUILDING
SEE DWG. 9312-F-2508
DETAIL "I" 1"=1'-0"
8 X 6 TUBE SECT. (TRAILER LEG)
3/8" CABLE THRU TRAILER LEG SHACKLED TOGETHER
TRAILER LEG
SECTION "H-H" SCALE 1/4"=1'-0"
INSIDE OF CONTAINMENT BLD'G
EQUIPMENT HATCH & PERSONNEL LOCK No. 22
PORTABLE STAIR
ROADWAY
PLAN @ EL. +95'-0" 1/4"=1'0"
EXIST. INSTRUMENT LINE WITH CHANNEL PROTECTION MODIFY FLANGES OF CHANNEL TO ALLOW JIB CRANE INSTALLATION
NOTE: 3/4" □ PHILLIPS RED HEAD WEDGE ANCHORS(WS-34100) SEE NOTE "A" DWG A205752
NOTE: ALL BOLT LOCATIONS SHALL BE VERIFIED IN THE FIELD TO MATCH CRANE CONNECTION PLATE HOLES AND ADJUST AS NECESSARY
PLAN @ EL.+95'-0" 1/4"=1'-0"
THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN CON EDISON AS; "CLASS FP" ITEMS PER THE QAPD
REFERENCE DWGS.
9321-F-1002 PLOT PLAN
F-2501 PLAN "C-C" (PLAN ABOVE EL. 95'-0")
F-2502 PLAN "D-D" (PLAN ABOVE EL. 68'-0")
F-2503 PLAN "E-E" (PLAN ABOVE EL. 46'-0")
PLAN "F-F" (PLAN BELOW EL. 46'-0")
F-2506 SECTION "A-A"
F-2507 SECTION "B-B"
F-2508 SECTION "G-G"
F-2514 FUEL STORAGE BUILDING
A205752 ACCESS TO STEAM GEN. AND STEAM GEN. MANWAYS STEPS TO V.C. EQUIPT. MATCH REMOVABLE HR. AROUND C.R.D.M. FRAME AND EQUIPT., HATCH LAYDOWN AREA AND ACCESS TO FAN COOLERS IN CONTAINMENT BUILDING ACCESS TO TOP OF MONITOR TANKS IN BORIC ACID BUILDING.
NOTE: #1 CLASS "A" APPLIES ONLY TO INSTALLATION OF EXPANSION BOLTS
THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN CON EDISON AS; "CLASS A" ITEMS PER THE QAPD
SEE NOTE #1
SECTION Z-Z (HOIST NOT SHOWN) NTS
STAINLESS STEEL COLUMN DAVIT
STAINLESS STEEL COLUMN DAVIT BASE
BASE RECEPTACLE
EDGE OF CAVITY
5/8" Ø DRILLCO FLUSH MOUNT MAXI-BOLTS ASTM A 193 MATERIAL 7 1/2" EMBEDMENT (6 REQ'D)
DETAIL Z
NOTES:
1. ALL MATERIAL FOR LIFE SAFETY HOIST TO BE NON CLASS A, EXCEPT THE ANCHOT BOLTS WHICH ARE CLASS A AND MARKED (A)
2. HOIST TO BE REMOVED DURING PLANT OPERATION. IT MAY BE USED AND INSTALLED DURING PLANT OUTAGES ONLY.
3. A 1"x12"HKB II WAS SUBSTITUTED FOR THE SOUTHWEST 3/4" SHELF DRILLING SLEEVE ANCHOR ON TOP OF THE BIOSHIELD WALL FOR SG24 JIB CRANE.
JIB CRANE (BY OTHERS)
16" Ø PIPE MAST (BY OTHERS)
EL. +102'-0"
EL. 95'-0"
SECTION "M-M" N.T.S.
SECTION "P-P" 1/4"=1'-0"
3/4" □ WEDGE ANCH. (TYP.)
"SECTION N-N" N.T.S.
(CON.ED.CO.DWG.NO. A200620)
COMPUTER GENERATED DRAWING NOT TO BE HAND REVISED
THIS REVISION IS NON CLASS PER THE QAPD. UPDATED DRAWING PER CR #200106388 ISSUED FOR RECORD. P.N. 69901-AF
JK/PB 07/05/01 RCMT
25
APPROVALS
TITLE:CONTAINMENT BUILDING GENERAL ARRANGEMENT UNIT NO. 2 PLAN "C-C" - ABOVE EL. 95'-0"
- UFSAR FIGURE 5.1-2
Con Edison
STATION INDIAN PT
SCALE 1/8"=1'-0"
DWG. NO. 9321-F-2501-25



NOTE:
SEE DWG.9321-F-2501 FOR LIST OF REFERENCE DWGS.

THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN CON EDISON AS:
"CLASS A" ITEMS
PER CI-240-1

THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN CON EDISON AS:
"CLASS FP" ITEMS
PER CI-240-1
SUPPLEMENT 3

(CON.ED.CO.DWG.NO. A200621)

COMPUTER GENERATED DRAWING NOT TO BE HAND REVISED

REV	DESCRIPTION	DATE	BY	CHK'D	APP.
18	INCORPORATED DRN-05-01805 ER-IP2-04-31567	3/27/05	RML		

THIS REVISION IS NON-CLASS PER CI-240-1
UPDATED DWG. TO INCLUDE A SECOND UFSAR FIGURE NO. IN TITLE
RELEASED FOR RECORD
P/N 69311-NP
DLA/4/29/99
STRATUS ENGINEERING

TITLE: CONTAINMENT BUILDING
GENERAL ARR'GMT. UNIT NO.2 PLAN "D-D"-ABOVE EL. 68'-0" - UFSAR
FIGURE No.'s 5.1-3 and 11.2-2

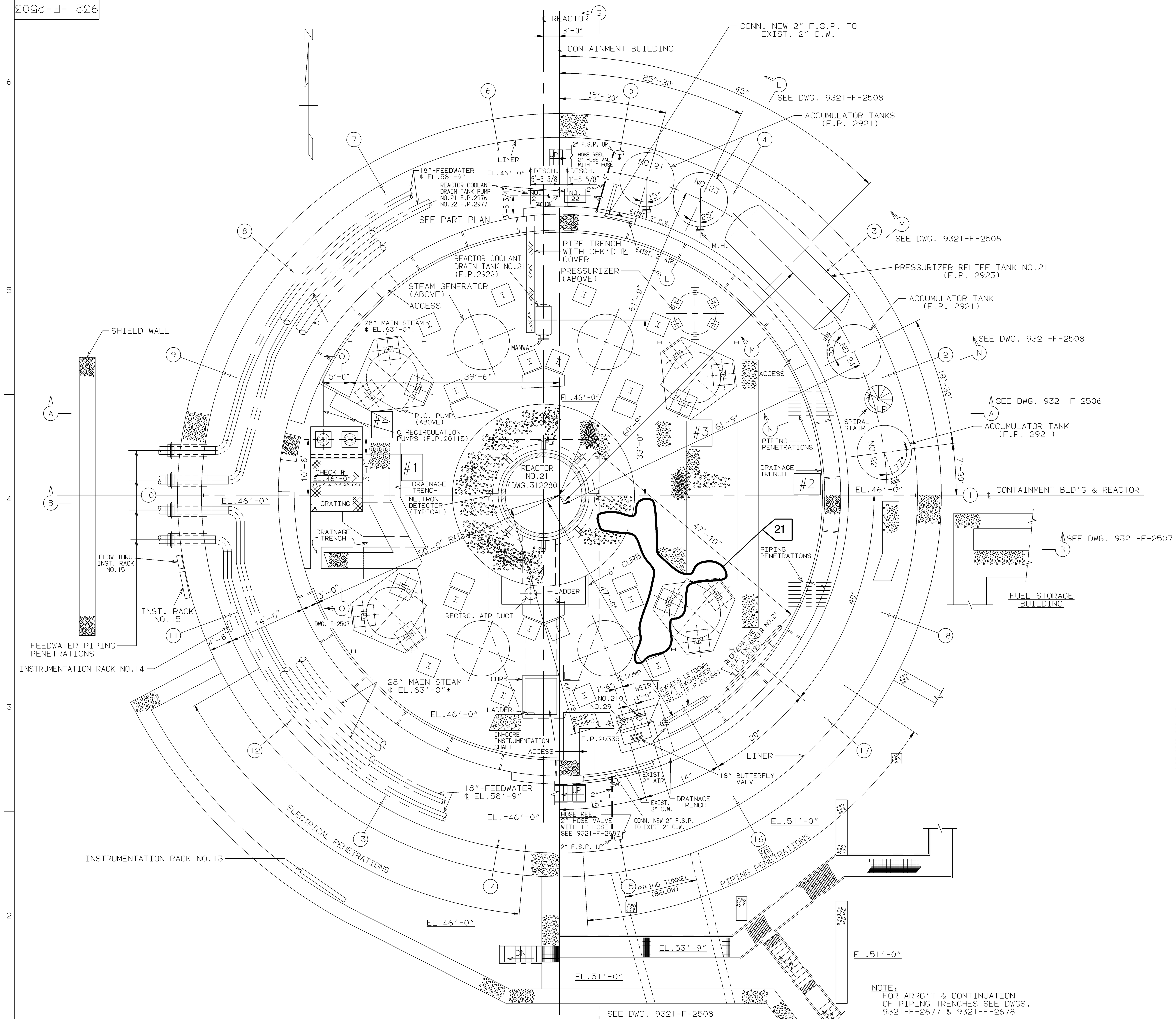
STATION: INDIAN PT

DWG. NO. 9321-F-2502-19

A200621

2/20/2004 9:39:57 AM 9321-2503-0-21.DWG

9321-F-2503



PLAN "E-E"

PLAN ABOVE EL.46'-0"

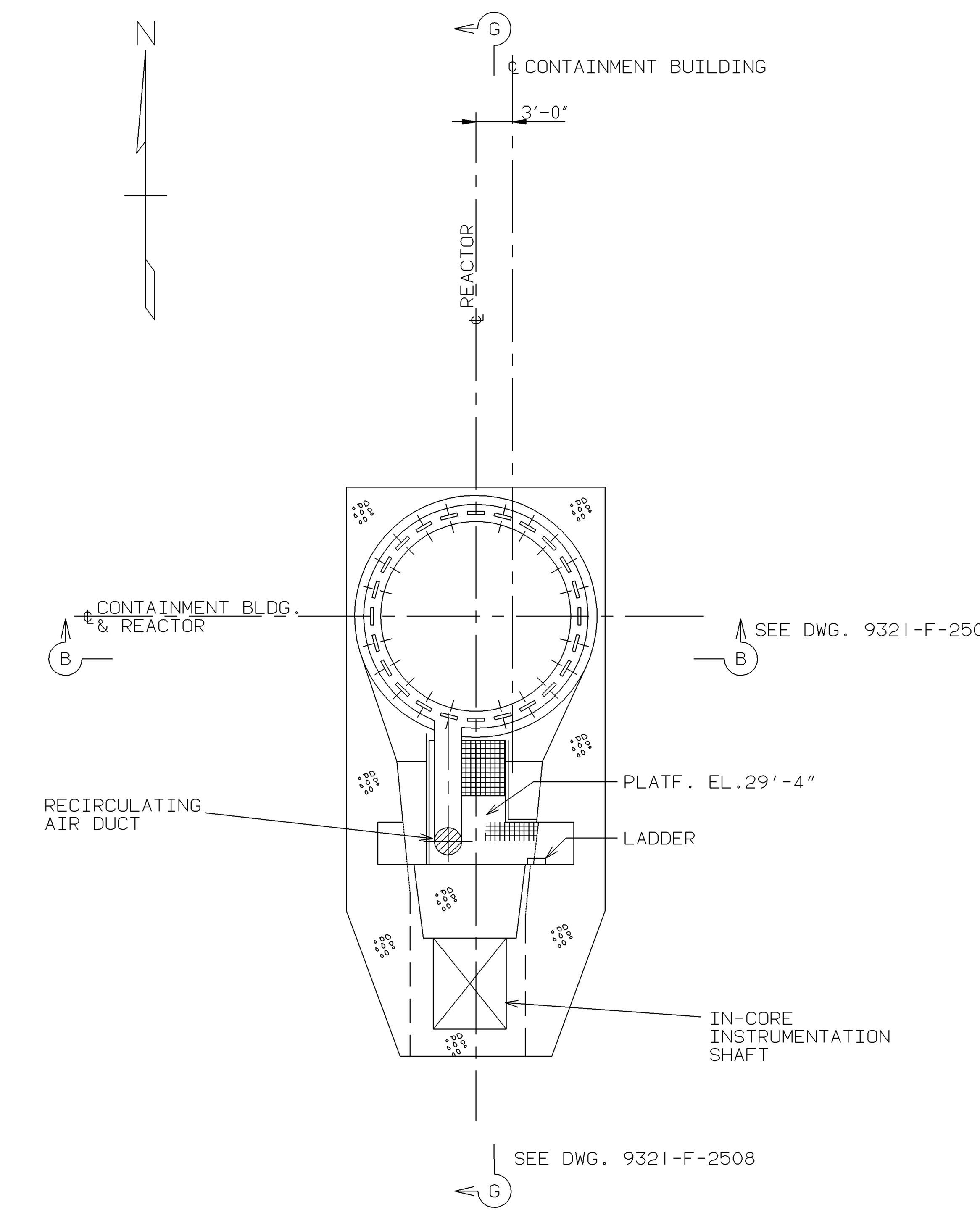
SCALE 1/8"=1'-0"

NOTE:
FOR ARRG'T & CONTINUATION OF PIPING TRENCHES SEE DWGS. 9321-F-2677 & 9321-F-2678

TRISODIUM PHOSPHATE STORAGE BASKETS
FOR DETAIL SEE DWG. 1996MB4406

#1 WALL MOUNT

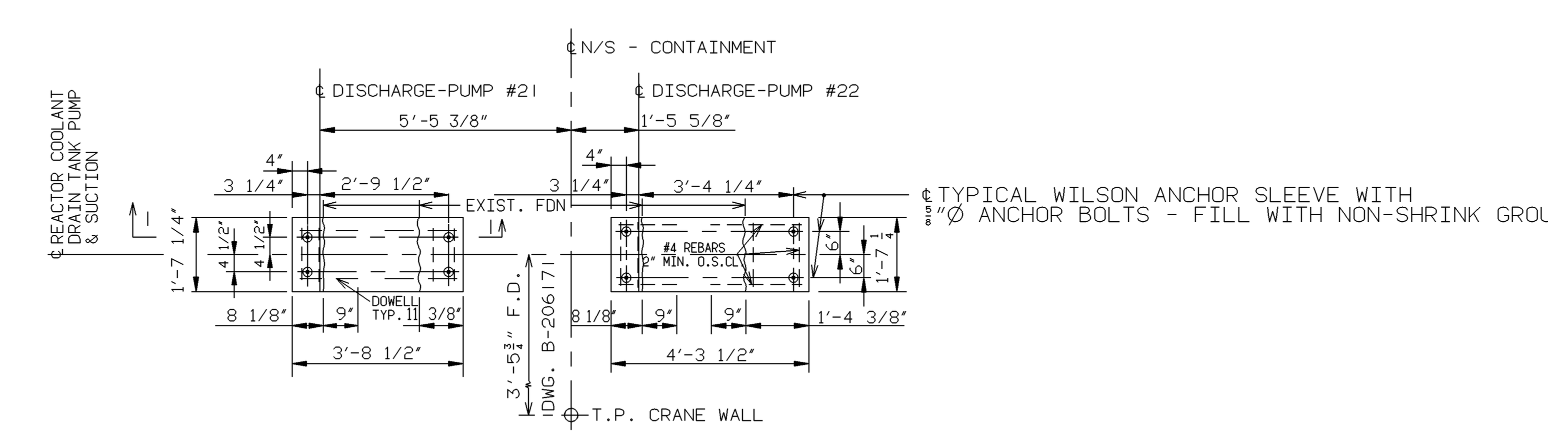
#2 #3 #4 FLOOR MOUNT



PLAN "F-F"

PLAN BELOW EL.46'-0"

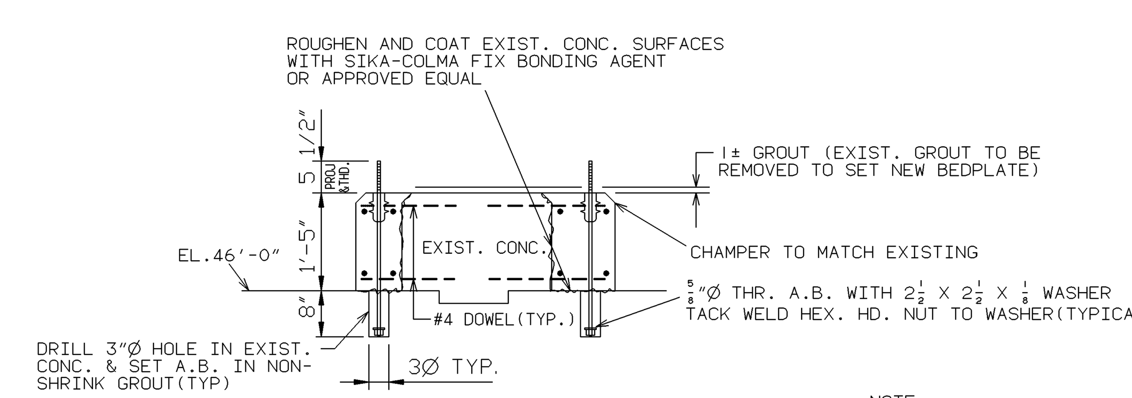
SCALE 1/4"=1'-0"



PART PLAN-FOUNDATION R.C.D.T. PUMPS

1/2"=1'-0"

NOTE:
SEE DWG. 9321-F-1124 FOR DETAILS OF EXIST. FOUNDATION. CUT AND REMOVE EXIST. ANCHOR BOLTS TO EL.47'-5" (8 TOTAL)



SECTION 1-1

1/2"=1'-0"

NOTE:
SEE DWG. 9321-F-2501 FOR LIST OF REFERENCE DWGS.

THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN ENTERGY AS "CLASS A" ITEMS PER THE QAPD.

(CON.ED.CO.DWG.NO. A200622)
COMPUTER GENERATED DRAWING NOT TO BE HAND REVISED

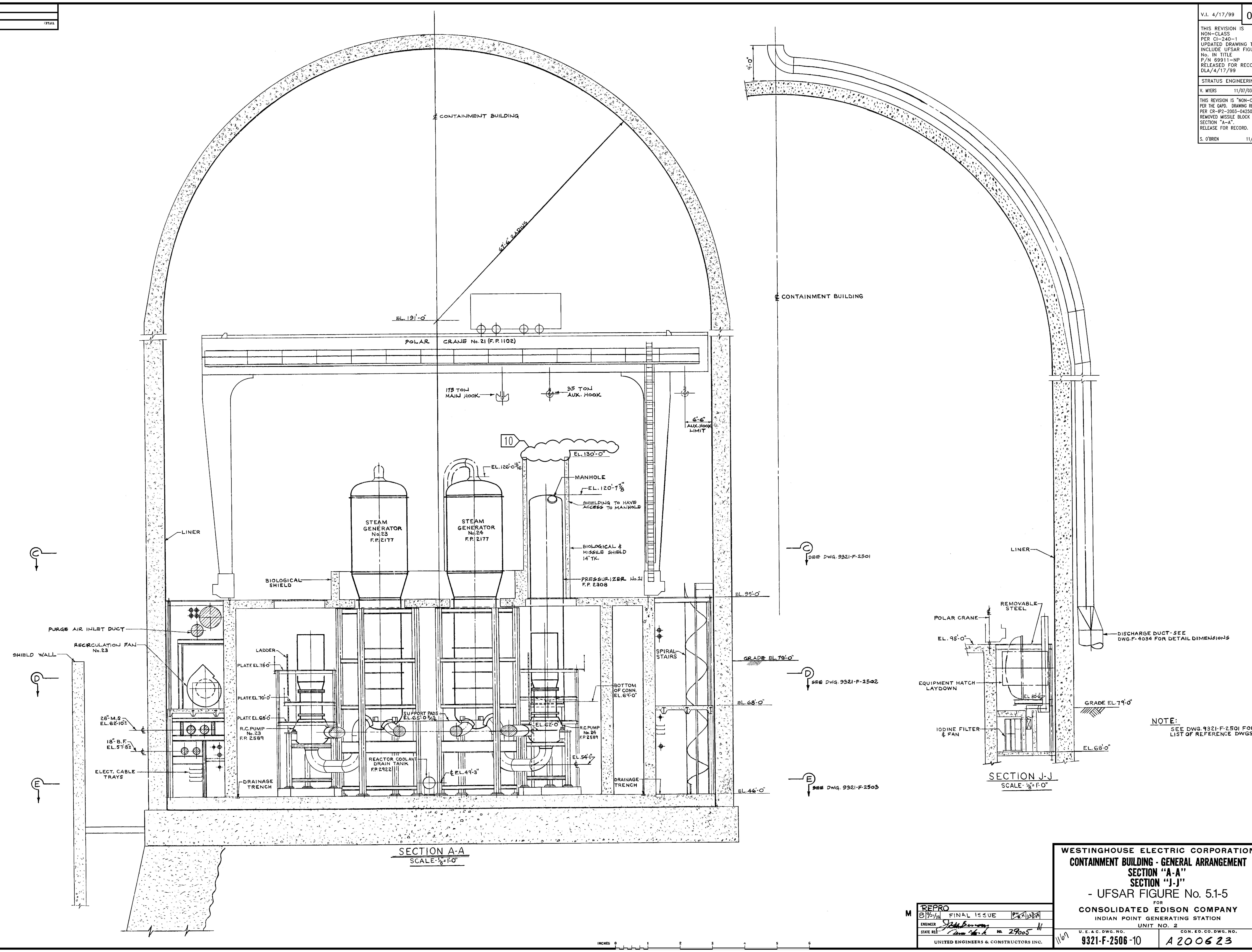
REVISION					
DES		THIS REV. IS CLASS "A" PER THE QAPD. UPDATED DWG. IN RESPONSE TO CR# 200106039 RELEASE AS-BUILT P.N. 69901-AF	DATE	CHK. SUPV. REVIEW	DESIGN ENG, DISC. ENG., ENG. MGR. APPROVALS
R. HARRIS 5/14/02	ENG	GH/MR 5/14/02			

TITLE: CONTAINMENT BUILDING-GENERAL ARRANGEMENT
UNIT NO.2 PLAN "E-E"-ABOVE EL.46'-0"
PLAN "F-F"-BELOW EL.46'-0"
- UFSAR FIGURE No. 5.1-4

DRAWN BY R. HARRIS | SCALE AS NOTED | REC'D

Con Edison | STATION INDIAN PT

DWG. NO. 9321-F-2503-21 MA



V.I. 4/17/99	09
THIS REVISION IS NON-CLASS PER CI-240-1 UPDATED DRAWING TO INCLUDE UFSAR FIGURE No. IN TITLE P/N 69911-NP RELEASED FOR RECORD DLA/4/17/99	
STRATUS ENGINEERING	
V. MYERS 11/07/03	10
THIS REVISION IS "NON-CLASS" PER THE QAPD. DRAWING REVISED PER CR-IP2-2003-04250. REMOVED MISSILE BLOCK FROM SECTION "A-A". RELEASE FOR RECORD. S. O'BRIEN 11/07/03	

M

REPRO
8 9/21/73 FINAL ISSUE
ENGINEER
STATE REG New York NO. 29005
UNITED ENGINEERS & CONSTRUCTORS INC.

WESTINGHOUSE ELECTRIC CORPORATION

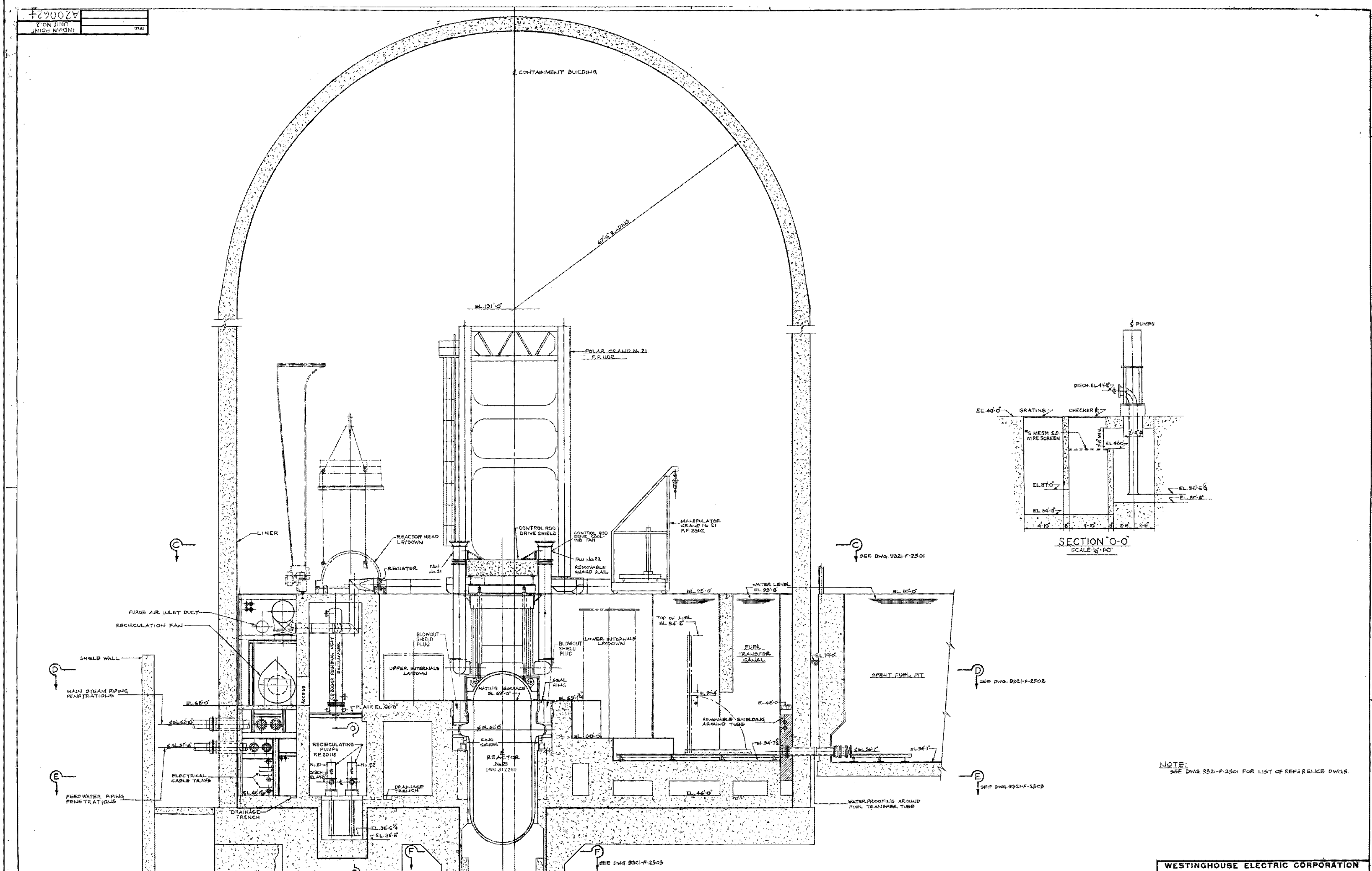
CONTAINMENT BUILDING - GENERAL ARRANGEMENT SECTION "A-A" SECTION "J-J" - UFSAR FIGURE No. 5.1-5

FOR

CONSOLIDATED EDISON COMPANY
INDIAN POINT GENERATING STATION
UNIT NO. 2

U.E.&C. DWG. NO.	CON. ED. CO. DWG. NO.
9321-F-2506-10	A200623

1167



J. SKONIECZNY 7/23/01	L. VILLANI 7/23/01	14	J. SKONIECZNY 6/21/01	L. VILLANI 6/21/01	13	V.I. 4/29/99	12
THIS REVISION IS NON-CLASS PER THE QAPD. UPDATED DRAWING PER CR # 200106298 RELEASED AS BUILT P/N 69901-AF S.L./J.S. 7/23/01			THIS REVISION IS NON-CLASS PER THE QAPD. UPDATED DRAWING PER CR # 200105699 RELEASED AS BUILT P/N 69901-AF S.L./J.S. 6/21/01			THIS REVISION IS NON-CLASS PER CI-240-1 UPDATED DRAWING TO INCLUDE A SECOND UFSAR FIGURE No. IN TITLE P/N 69911-NP RELEASED FOR RECORD DLA/4/29/99 STRATUS ENGINEERING	

WESTINGHOUSE ELECTRIC CORPORATION

14

CONTAINMENT BUILDING
GENERAL ARRANGEMENT
SECTION "B-B"

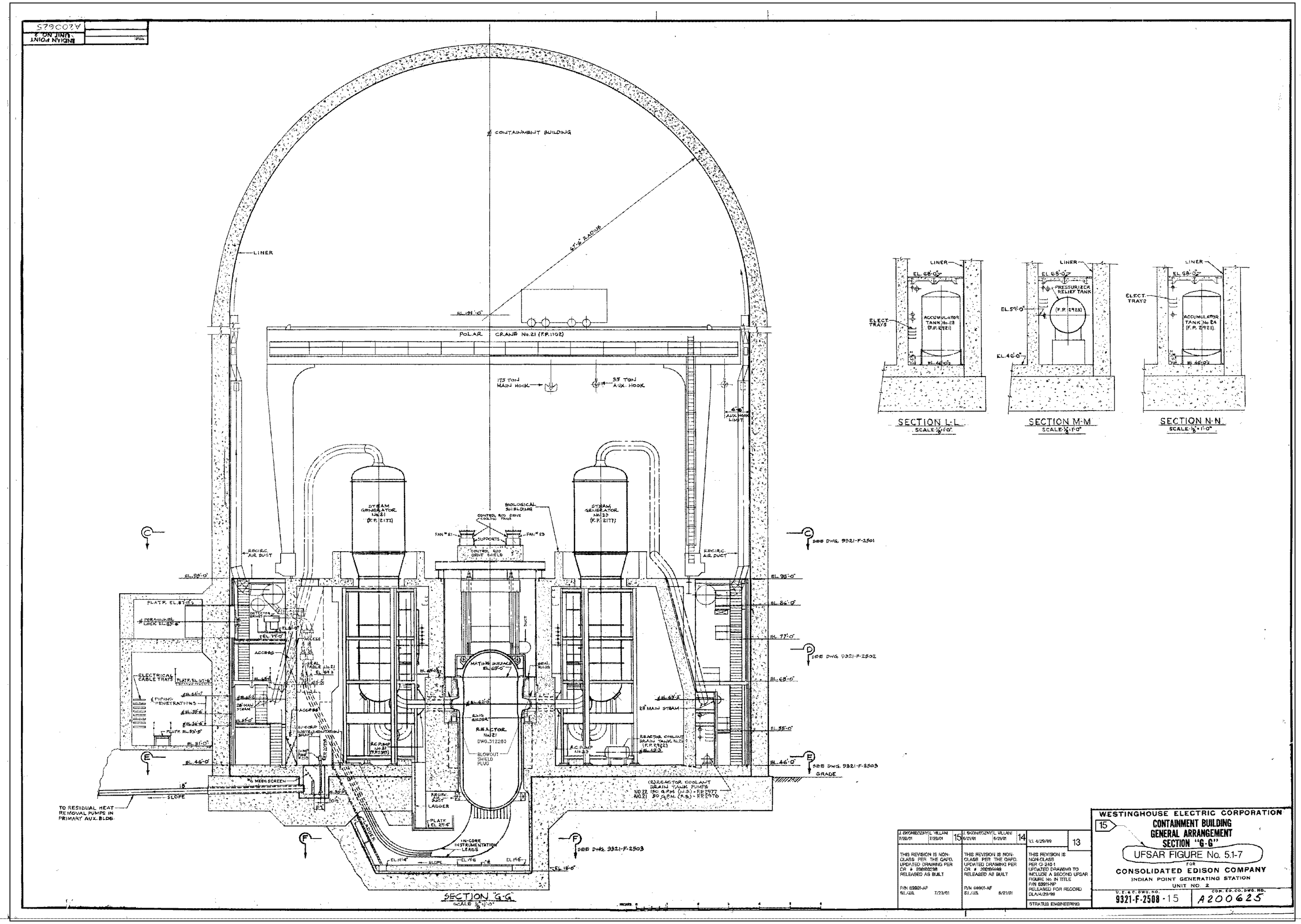
UFSAR FIGURE No. 5.1-6

FOR

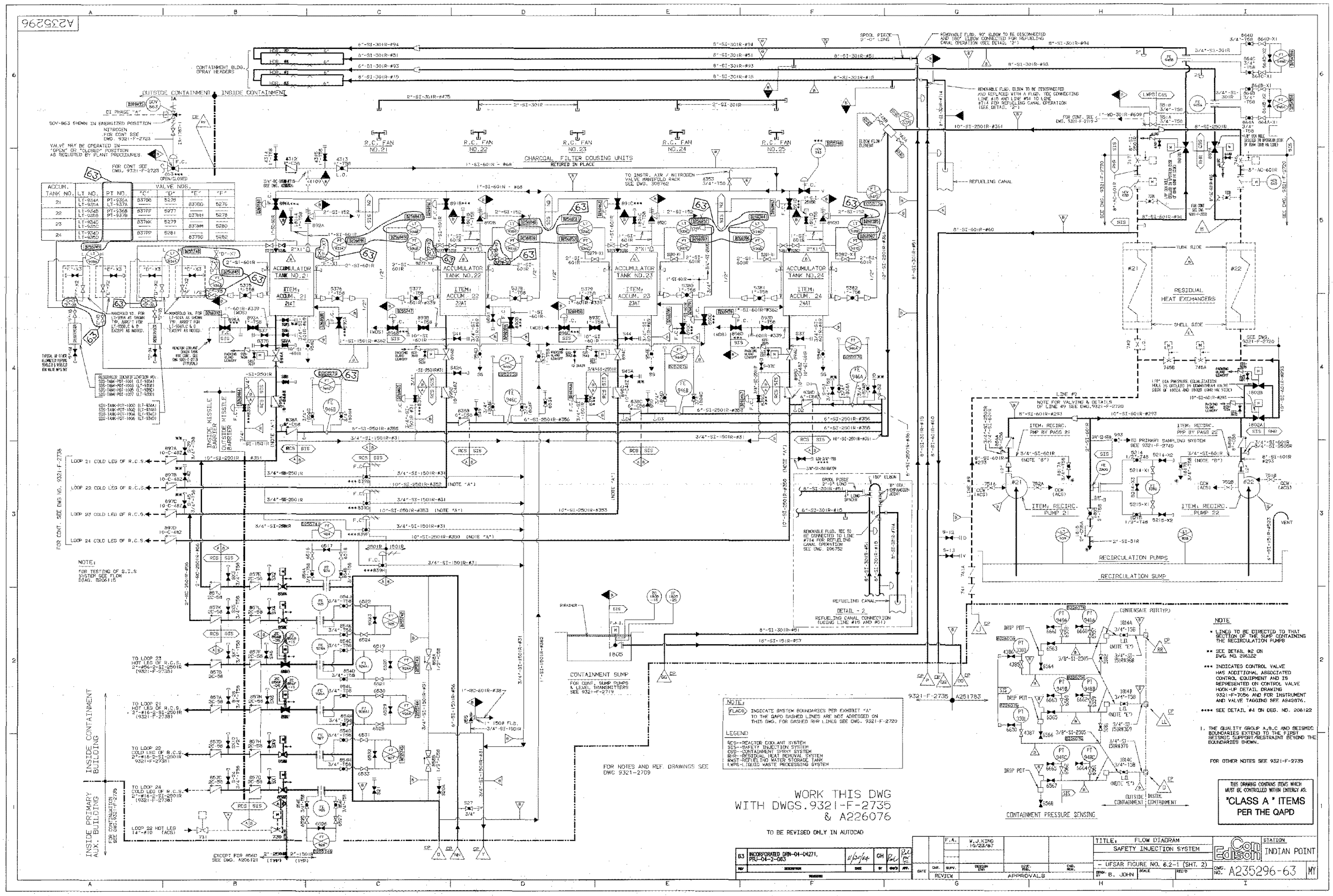
CONSOLIDATED EDISON COMPANY
INDIAN POINT GENERATING STATION
UNIT NO. 2

U.E.&C. DWG. NO. 9321-F-2507-14

CON. ED. CO. DWG. NO. A200624



INDIAN POINT UNIT NO. 2
A200625
℄ CONTAINMENT BUILDING
LINER
EL. 151'-0"
POLAR CRANE No. 21 (F.P. 1102)
175 TON MAIN HOOK
35 TON AUX. HOOK
STEAM GENERATOR No. 21 (F.P. 2177)
STEAM GENERATOR No. 23 (F.P. 2177)
BIOLOGICAL SHIELDING
CONTROL ROD DRIVE SHIELD
SUPPORTS
RECIRC. AIR DUCT
EL. 95'-0"
EL. 86'-0"
EL. 77'-0"
EL. 68'-0"
EL. 55'-0"
EL. 46'-0"
ELECTRICAL CABLE TRAYS
PIPING PENETRATIONS
28" MAIN STEAM
REACTOR No. 21
BLOWOUT SHIELD PLUG
R.C. PUMP No. 21
R.C. PUMP No. 23
REACTOR COOLANT DRAIN TANK No. 21 (F.P. 2922)
IN-CORE INSTRUMENTATION LEADS
TO RESIDUAL HEAT REMOVAL PUMPS IN PRIMARY AUX. BLDG.
SLOPE
GRADE
SEE DWG. 9321-F-2501
SEE DWG. 9321-F-2502
SEE DWG. 9321-F-2503
SECTION "G-G"
SECTION L-L
ACCUMULATOR TANK No. 23 (F.P. 2921)
ELECT. TRAYS
SECTION M-M
PRESSURIZER RELIEF TANK
SECTION N-N
ACCUMULATOR TANK No. 24 (F.P. 2921)
J. SKONIECZNY L. VILLANI 7/23/01 15
THIS REVISION IS NON-CLASS PER THE QAPD. UPDATED DRAWING PER CR # 200108298 RELEASED AS BUILT
P/N 69901-AF S.L./J.S. 7/23/01
J. SKONIECZNY L. VILLANI 6/21/01 14
THIS REVISION IS NON-CLASS PER THE QAPD. UPDATED DRAWING PER CR # 200105098 RELEASED AS BUILT
P/N 69901-AF S.L./J.S. 6/21/01
V.I. 4/29/99 13
THIS REVISION IS NON-CLASS PER QI-240-1 UPDATED DRAWING TO INCLUDE A SECOND UFSAR FIGURE No. IN TITLE P/N 69911-NP RELEASED FOR RECORD DLA/4/29/99
STRATUS ENGINEERING
WESTINGHOUSE ELECTRIC CORPORATION
15
CONTAINMENT BUILDING GENERAL ARRANGEMENT SECTION "G-G"
UFSAR FIGURE No. 5.1-7
FOR
CONSOLIDATED EDISON COMPANY
INDIAN POINT GENERATING STATION
UNIT NO. 2
W.E.&C. DWG. NO. 9321-F-2508 -15
CON. ED. CO. DWG. NO. A200625


A235296
CONTAINMENT BLDG. SPRAY HEADERS
OUTSIDE CONTAINMENT
INSIDE CONTAINMENT
R.C. FAN NO.21
R.C. FAN NO.22
R.C. FAN NO.23
R.C. FAN NO.24
R.C. FAN NO.25
CHARCOAL FILTER DOUSING UNITS RETIRED IN PLACE
ACCUMULATOR TANK NO.21
ACCUMULATOR TANK NO.22
ACCUMULATOR TANK NO.23
ACCUMULATOR TANK NO.24
RESIDUAL HEAT EXCHANGERS
TUBE SIDE
SHELL SIDE
REFUELING CANAL
ITEM: RECIRC. PUMP 21
ITEM: RECIRC. PUMP 22
RECIRCULATION PUMPS
RECIRCULATION SUMP
CONTAINMENT SUMP
INSIDE MISSILE BARRIER
OUTSIDE MISSILE BARRIER
LOOP 21 COLD LEG OF R.C.S.
LOOP 22 COLD LEG OF R.C.S.
LOOP 23 COLD LEG OF R.C.S.
LOOP 24 COLD LEG OF R.C.S.
FOR CONT. SEE DWG NO. 9321-F-2738
INSIDE CONTAINMENT BUILDING
INSIDE PRIMARY AUX. BUILDING
FOR NOTES AND REF. DRAWINGS SEE DWG 9321-F-2709
CONTAINMENT PRESSURE SENSING
NOTE:
FLAGS INDICATE SYSTEM BOUNDARIES PER EXHIBIT "A" TO THE QAPD DASHED LINES ARE NOT ADRESSED ON THIS DWG. FOR DASHED RHR LINES SEE DWG. 9321-F-2720
LEGEND
RCS--REACTOR COOLANT SYSTEM
SIS--SAFETY INJECTION SYSTEM
CSS--CONTAINMENT SPRAY SYSTEM
RHR--RESIDUAL HEAT REMOVAL SYSTEM
RWST-REFUELING WATER STORAGE TANK
LWPS-LIQUID WASTE PROCESSING SYSTEM
WORK THIS DWG WITH DWGS.9321-F-2735 & A226076
TO BE REVISED ONLY IN AUTOCAD
NOTE
* LINES TO BE DIRECTED TO THAT SECTION OF THE SUMP CONTAINING THE RECIRCULATION PUMPS
** SEE DETAIL #2 ON DWG. NO. 206122
*** INDICATES CONTROL VALVE HAS ADDITIONAL ASSOCIATED CONTROL EQUIPMENT AND IS REPRESENTED ON CONTROL VALVE HOOK-UP DETAIL DRAWING 9321-F-7056 AND FOR INSTRUMENT AND VALVE TAGGING SEE A842876.
**** SEE DETAIL #4 ON DEG. NO. 206122
1. THE QUALITY GROUP A,B,C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.
FOR OTHER NOTES SEE 9321-F-2735
THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN ENTERGY AS:
"CLASS A" ITEMS PER THE QAPD
63 INCORPORATED DRN-04-04271, PRJ-04-2-083
F.A. W.J.KING 10/23/87
REVIEW
APPROVALS
TITLE: FLOW DIAGRAM
SAFETY INJECTION SYSTEM
- UFSAR FIGURE NO. 6.2-1 (SHT. 2)
DRWN. BY B. JOHN
Con Edison
STATION INDIAN POINT
DWG. NO. A235296-63
MY

MECH.
9321-F-4022
NOTE "B"
FOR INSTRUMENT AIR SCHEMATIC SEE DWG. A206128.
FOR WELD CHANNEL PRESSURIZATION CONTROLS AND INSTRUMENTATION SEE DWG.9321-F-2726
PURGE SUPPLY DETAIL NO.1
PURGE EXHAUST DETAIL NO.2
HIGH RADIATION SAMPLING SYS ROOM
P.A.B. EXH.SYS.
900 CFM (NORM)
0 CFM EMERG.EVENT
HIGH RADIATION SAMPLING MODULE
HIGH RAD. SAMPLING MODULE EXH.SYS. EMERGENCY EVENT
IODINE FILTER AND FAN UNITS 8,000CFM EACH
FAN-COOLER-FILTER UNIT (TYPICAL)
21FCU THRU 25FCU
FANS AND DAMPERS RETIRED IN PLACE. HEATING COILS ARE OPERATIONAL.
DIAGRAM OF CONTAINMENT AIR RECIRCULATION SYSTEM
PURGE SYSTEM NOT SHOWN
EVERYTHING ON THIS DRAWING IS PART OF:
1. CONTAINMENT AIR RECIRCULATION COOLING AND FILTRATION SYSTEM-AND
2. VENTILATION/FILTRATION SYSTEM FOR AREAS CONTAINING SAFETY RELATED SYSTEMS AND SYSTEMS CONTAINING RADIOACTIVITY, LISTED IN EXHIBIT A. CI-240-1
AIR DISTRIBUTION SCHEDULE
A + A1 = 58,675 CFM TO R.C.PUMP STM.GEN.& PRESSURIZER AREA.
B = 2,000 CFM TO PRESS.RELIEF TANK AREA.
C + C1 = 60,000 CFM TO DOME AREA.
D + D1 = 51,275 CFM TO RC.PUMP & STM.GEN.AREA.
E = 17,500 CFM ACROSS REFUELING POOL AREA.
F = 10,000 CFM RECIRCULATOIN EQUIPMENT AREA.
G = 5,000 CFM TO PASSAGE AREA.
H + H1 = 51,275 CFM TO R.C.PUMP & STM.GEN.AREA.
I = 12,000 CFM AREA UNDER REACTOR.
J = 5,000 CFM TO INCORE INSTRUMENTATION CONTROL ROOM.
K + K1 = 52,275 CFM TO R.C.PUMPS & STM.GEN.AREA.
LEGEND
N.O. NORMALLY OPEN (FAIL CLOSED).
N.C. NORMALLY CLOSED (FAIL OPEN).
LOUVER
FILTER
DAMPER
BUTTERFLY VALVE-BUBBLE TIGHT
RADIATION
RADIATION CONTROLLER
RELIEF VALVE
VSD VENTILATING SYSTEM DAMPER
FLOW INDICATOR
MOTOR OPERATOR
CARBON FILTER
BACK DRAFT DAMPER
AIR FOIL DAMPER
** VALVES FURNISHED WITH DPDT LIMIT SWITCHES IN OPENED AND CLOSED POSITION.
NOTES:
1. 3"-#629 FROM POST ACCIDENT CONTAINMENT VENTING SYSTEM. SEE DWG.B208879.
2. 2"-#131 FROM GAS DECAY TANKS. SEE DWG.9321-F-2730.
3. ABSOLUTE & CHARCOAL FILTERS REMOVED IN F.C.U.'S #21 THRU 25.
4. THE QUALITY GROUP A,B,C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.
5. NORMALLY CLOSED INCIDENT FLOW CONTROL VALVES IN F.C.U.'S 21-25 ARE LOCKED CLOSED AND RETIRED IN PLACE
6. ALL F.C.U.'S INCIDENT INLET/OUTLET F.C.V.'S AND ASSOCIATED PANELS ARE RETIRED IN PLACE.
AIR FLOW DIAGRAM, FUEL STORAGE BUILDING EXHAUST SYSTEM, ALSO SHOWN ON DWG 9321-F-4060
NOTE: DAMPER MOTORS RETIRED IN PLACE, BYPASS DAMPERS WELDED SHUT
FUEL STORAGE BLDG.
FAN HOUSE
FAN ROOM
V.C. PURGE MAKE UP AIR
PIPE TUNNEL
THE CRF COOLING COILS ARE PART OF THE SERVICE WATER SYSTEM
DIAGRAM FAN COOLER-FILTER UNIT
(5 REQUIRED 21FCU THRU 25FCU)
FOR INSTRUMENTATION DIAG.SEE DWG.A242753
NOTE "A"
FOR PCV-1190, PCV-1191 & PCV-1192 PISTON ACTUATOR CONTROL SCHEMATIC SEE DWG. A206143.
FOR WELD CHANNEL PRESSURIZATION CONTROLS AND INSTRUMENTATION SEE DWG.9321-F-2726
PRESSURE RELIEF DETAIL NO.3
PRIMARY AUXILIARY BUILDING
WASTE HOLD-UP TANK PIT
REFERENCE DRAWINGS:
A228270-FCU CONTROL VA.PANEL ARRANGEMENT
A243173-FCU DAMPERS NORMAL FLOW CONTROL PANEL & LOOP DIAG
A243174-MATERIAL PARTS LIST-FCU NORMAL FLOW
A243175-FCU DAMPERS INCIDENT FLOW CONTROL PANEL & LOOP DIAG
A243176-MATERIAL PARTS LIST-FCU NORMAL FLOW
A242753-FAN COOLER FILTER UNITS INSTRUMENT DIAGRAM
312531-MODIFICATION OF RUPTURE DISC FOR FAN COOLER UNITS
THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN ENTERGY AS:
"CLASS A" ITEMS
PER THE QAPD
COMPUTER GENERATED DRAWING NOT TO BE HAND REVISED
THIS REVISION IS NON CLASS PER THE QAPD.
UPDATED DRAWING PER ER-IP2-04-16520
(CR-IP2-2003-05078).
RELEASE FOR RECORD
P.N. 69901-AF RM/TC
T. CZERNIEWSKI 04/01/04
04/23/01 SM
F.A. W.J.KING/G.C. 1-13-88
REVIEW
APPROVALS
TITLE: FLOW DIAGRAM VENTILATION
SYSTEM CONTAINMENT, PRIMARY AUX.
BLDG FUEL STG BLDG
UFSAR FIG. No.'s 5.3-1
DRWN. BY GIBBS&HILL
SCALE NONE
Con Edison
STATION
INDIAN POINT
DWG. NO. 9321-F-4022-60
MY

CHAPTER 6
ENGINEERED SAFETY FEATURES

6.0 INTRODUCTION

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

1. Core design to preclude release of fission products from the fuel (Chapter 3).

2. Retention of fission products in the reactor coolant for whatever leakage occurs (Chapters 4 and 6).

3. Retention of fission products by the containment for operational and accidental releases beyond the reactor coolant boundary (Chapters 5 and 6).

4. Optimizing fission product dispersal to minimize population exposure (Chapters 2 and 11).

The engineered safety features are the provisions in the plant that embody methods 2 and 3 above to prevent the occurrence or to ameliorate the effects of serious accidents.

The engineered safety features systems in this plant are the containment system, detailed in Chapter 5, the safety injection system, detailed in Section 6.2, the containment spray system, detailed in Section 6.3, the containment air recirculation cooling system, detailed in Section 6.4, the isolation valve seal-water system, detailed in Section 6.5, and the containment penetration and weld channel pressurization system, detailed in Section 6.6.

Evaluations of techniques and equipment used to accomplish the central objective including accident cases are detailed in Chapters 5, 6, and 14.

6.1 GENERAL DESIGN CRITERIA

Criteria applying in common to all engineered safety features are given in Section 6.1.1. Thereafter, criteria that are related to engineered safety features, but which are more specific to other plant features or systems, are listed and cross referenced in Section 6.1.2.

6.1.1 <u>Engineered Safety Features Criteria</u>

6.1.1.1 <u>Engineered Safety Features Basis for Design</u>

Criterion: Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. Such engineered safety features shall be designed to cope with any size reactor coolant piping break up to and including the equivalent of a circumferential rupture of any pipe in that boundary, assuming unobstructed discharge from both ends. (GDC 37)

The design, fabrication, testing, and inspection of the core, reactor coolant pressure boundary, and their protection systems give assurance of safe and reliable operation under all anticipated normal, transient, and accident conditions. However, engineered safety features are provided in the facility to back up the safety provided by these components. These engineered safety features have been designed to cope with any size reactor coolant pipe break up to and including the circumferential rupture of any pipe assuming unobstructed discharge from both ends as discussed in Section 14.3.3.3. They are also designed to cope with any steam or feedwater line break up to and including the main steam or feedwater lines as discussed in Section 14.2.5.

Limiting the release of fission products from the reactor fuel is accomplished by the safety injection system, which by cooling the core, keeps the fuel in place and substantially intact and limits the metal water reaction to an insignificant amount.

The safety injection system consists of high- and low-head centrifugal pumps driven by electric motors and of passive accumulator tanks that are self-energized and which act independently of any actuation signal or power source.

The release of fission products from the containment is limited in three ways:

1. Blocking the potential leakage paths from the containment. This is accomplished by:

    a. A steel-lined, reinforced-concrete reactor containment with testable, double-sealed penetrations, and liner weld channels, the spaces of which are continuously pressurized above accident pressure and which form a virtually leaktight barrier to the escape of fission products should a loss-of-coolant accident occur. (Section 6.6.2 lists those portions of the Weld Channel Pressurization System that have been disconnected because repairs have been determined not to be practical.)

    b. Isolation of process lines by the containment isolation system, which imposes double barriers in each line that penetrates the containment except for lines used during the accident. Pipes penetrating the containment are sealed as shown in Table 5.2-1. This table presents the sealing method for all containment piping penetrations and valving.

2. Reducing the fission product concentration in the containment atmosphere. This is accomplished by containment spray, which removes elemental iodine vapor and particulates from the containment atmosphere by washing action. The spray is chemically treated during the recirculation phase to enhance iodine retention.

3. Reducing the containment pressure and thereby limiting the driving potential for fission product leakage. This is accomplished by cooling the containment atmosphere by the following independent systems of approximately equal heat removal capacity that together also function to ensure the containment design criteria is maintained even with an assumed single failure:

    a. Containment spray system.

    b. Containment air recirculation cooling system.

6.1.1.2 Reliability and Testability of Engineered Safety Features

Criterion: All engineered safety features shall be designed to provide such functional reliability and ready testability as is necessary to avoid undue risk to the health and safety of the public. (GDC 38)

A comprehensive program of plant testing was formulated for all equipment, systems, and system control vital to the functioning of engineered safety features. The program consisted of performance tests of individual pieces of equipment in the manufacturer's shop, and integrated tests of the systems as a whole. Periodic tests of the actuation circuitry and mechanical components ensure reliable performance, upon demand, throughout the life of the plant.

The initial tests of individual components and the integrated test of the system as a whole complemented each other to ensure the performance of the system as designed and to prove proper operation of the actuation circuitry.

Routine periodic testing of the engineered safety features components is performed, in accordance with the Technical Specifications.

6.1.1.3 Missile Protection

Criterion: Adequate protection for those engineered safety features, the failure of which could cause an undue risk to the health and safety of the public, shall be provided against dynamic effects and missiles that might result from plant equipment failures. (GDC 40)

A loss-of-coolant accident or other plant equipment failure might result in dynamic effects or missiles. For engineered safety features that are required to ensure safety in the event of such an accident or equipment failure, protection is provided primarily by the provisions that are taken in the design to prevent the generation of missiles. In addition, protection is also provided by the layout of plant equipment or by missile barriers in certain cases. (Refer to Section 5.1.2 for a discussion of missile protection.)

Injection paths leading to unbroken reactor coolant loops are protected against damage resulting from the maximum reactor coolant pipe rupture by layout and structural design considerations. Injection lines penetrate the main missile barrier, which is the crane wall, and the injection headers are located in the missile protected area between the crane wall and the containment wall. Individual injection lines, connected to the injection header, pass through the barrier and then connect to the loops. The separation of the individual injection lines is provided to the maximum extent practicable. The movement of the injection line, associated with a rupture of a reactor coolant loop, is accommodated by line flexibility and by the design of the pipe supports such that no damage outside the missile barrier is possible.

The containment structure is capable of withstanding the effects of missiles originating outside the containment and which might be directed toward it so that no loss-of-coolant accident can result.

All hangers and anchors are designed in accordance with USAS B31.1, Code for Pressure Piping. This code provides minimum requirements on material, design, and fabrication with ample safety margins for both dead and dynamic loads over the life of the equipment. Concrete missile barriers, bumpers, walls and other concrete structures are designed in accordance with ACI 318-63, Building Code Requirements for Reinforced Concrete.

In 1989, the NRC approved changes to the design basis with respect to dynamic effects of postulated primary loop ruptures, as discussed in Section 4.1.2.4

6.1.1.4 Engineered Safety Features Performance Capability

Criterion: Engineered safety features, such as the emergency core cooling system and the containment heat removal system, shall provide sufficient performance capability to accommodate the failure of any single active component without resulting in undue risk to the health and safety of the public. (GDC 41)

Each engineered safety feature provides sufficient performance capability to accommodate any single failure of an active component and still function in a manner to avoid undue risk to the health and safety of the public.

The extreme upper limit of public exposure is taken as the levels and time periods presently outlined in 10 CFR 50.67 (i.e., 25 rem total effective dose equivalent (TEDE) at the exclusion radius for the worst two hour interval, 25 rem TEDE over the duration of the accident at the low-population-zone distance and 5 rem TEDE for the operators in the control room for the duration of the accident). The accident condition considered is the hypothetical case of a release of fission products per NUREG-1465. Also, the total loss of all outside power is assumed concurrently with this accident. With minimum engineered safety features systems functioning, the offsite exposure would be within 10 CFR 50.67 limits as discussed in Section 14.3.6.

Under these accident conditions, the containment air recirculation cooling system and the containment spray system are designed and sized so that both systems, each operating with partial effectiveness, are able to supply the necessary postaccident cooling capacity to ensure the maintenance of containment integrity, that is, keeping the pressure below design pressure at all times assuming that the core residual heat is released to the containment as steam. Partial effectiveness is defined as the operation of a system with at least one active component failure. Containment spray relies on a sufficient amount of passive trisodium phosphate stored in containment to raise the pH of the recirculating solution for continued iodine removal following an accident. The containment spray system alone is able to supply the post accident iodine removal required to restrict the offsite exposure to within 10 CFR 50.67 limits.

6.1.1.5 Engineered Safety Features Components Capability

Criterion: Engineered safety features shall be designed so that the capability of these features to perform their required function is not impaired by the effects of a loss-of-coolant accident to the extent of causing undue risk to the health and safety of the public. (GDC 42)

Instrumentation, pumps, fans, cooling units, valves, motors, cables, and penetrations located inside the containment are selected to meet the most adverse accident conditions to which they may be subjected. These items are either protected from containment accident conditions or are designed to withstand, without failure, exposure to the worst combination of temperature, pressure, and humidity expected during the required operational period.

In response to NRC Generic Letter 95-07, safety-related power-operated gate valves have been evaluated for susceptibility to pressure locking and thermal binding. The results of this evaluation identified that those potential conditions will not prevent the plant from achieving a safe shutdown, as all valves evaluated remain operable. This conclusion is based upon valve design, plant configuration during normal, accident, and post accident operating modes and sufficient actuator thrust to open the valve. The details of the system and valve evaluations are documented in Reference 1.

In response to NRC Generic Letter 96-06, isolated pipe line segments that penetrate containment have been analyzed to evaluate their susceptibility to overpressurization caused by thermal expansion of the contained fluid in the event of a design basis accident. The results show that potential overpressurization will not cause lines to fail; all remain operable. Those that are protected by safety relief devices were further evaluated for the effects of stuck-open relief valves under accident conditions. No failure modes were identified that would adversely affect the ability for safety-related systems to perform their intended functions during accidents.

The safety injection system pipes serving each loop are restrained in such a manner as to restrict potential accident damage to the portion of piping downstream of the crane wall that constitutes the missile barrier in each loop area. The restraints are designed to withstand, without failure, the thrust force of any branch line severed from the broken reactor coolant pipe and discharging fluid to the atmosphere and to withstand a bending moment equivalent to that which produces a failure of the piping under the action of free-end discharge to the atmosphere or a motion of the broken reactor coolant pipe to which the injection pipes are connected. This prevents possible failure at any point upstream from the support point, including the branch line connection into the piping header.

In 1989, the NRC approved changes to the design basis with respect to dynamic effects of postulated primary loop ruptures, as discussed in Section 4.1.2.4.

Designated valves that are located in areas that would have excessive radiation levels in the event of a release of fission products from the core are provided with capability for remote operation.

6.1.1.6 Accident Aggravation Prevention

Criterion: Protection against any action of the engineered safety features, which would accentuate significantly the adverse after effects of a loss of normal cooling shall be provided. (GDC 43)

The introduction of borated cooling water into the core results in a negative reactivity addition. The control rods insert and remain inserted.

The supply of water by the safety injection system to cool the core cladding does not produce significant metal-water reaction (<1.0-percent).

The delivery of cold safety injection water to the reactor vessel following accidental expulsion of reactor coolant does not cause further loss of integrity of the reactor coolant system boundary.

6.1.1.7 Sharing of Systems

Criterion: Reactor facilities may share systems or components if it can be shown that such sharing will not result in undue risk to the health and safety of the public. (GDC 4)

The engineered safety features at Indian Point 2 do not share systems or components with other units.

6.1.2 Related Criteria

The following are criteria, which although related to all engineered safety features, are more specific to other plant features or systems, and, therefore, are discussed in other chapters, as listed.

Name	Discussion
Quality Standards (GDC 1)	Chapter 4
Performance Standards (GDC 2)	Chapter 4
Records Requirements (GDC 5)	Chapter 4
Instrumentation and Control Systems (GDC 12) Chapter 7	
Engineered Safety Features Protection Systems (GDC 15)	Chapter 7
Emergency Power (GDC 39)	Chapter 8

REFERENCES FOR SECTION 6.1

1. NRC Letter dated May 20, 1999, Jeffrey F. Harold to A. Alan Blind, Subject: Safety Evaluation of Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," for Indian Point Nuclear Generating Unit No.2 (TAC M93473).

6.2 SAFETY INJECTION SYSTEM

At Indian Point Unit 2 the emergency core cooling function is performed by the safety injection system. Therefore, whenever the term "emergency core cooling system" or ECCS is referenced in the document it is synonymous with the safety injection system.

6.2.1 Design Basis

6.2.1.1 Emergency Core Cooling System Capability

Criterion: An emergency core cooling system with the capability for accomplishing adequate emergency core cooling shall be provided. This core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to acceptable amounts for all sizes of breaks in the reactor coolant piping up to the equivalent of a double-ended rupture of the largest pipe. The performance of such emergency core cooling system shall be evaluated conservatively in each area of uncertainty. (GDC 44)

Adequate emergency core cooling is provided by the safety injection system (which constitutes the emergency core cooling system) whose components operate in three modes. These modes are delineated as passive accumulator injection, active safety injection, and residual heat removal recirculation.

The primary purpose of the safety injection system is the automatic delivery of cooling water to the reactor core in the event of a loss-of-coolant accident. This limits the fuel clad temperature and thereby ensures that the core will remain intact and in place, with its essential heat transfer geometry preserved. This protection is afforded for:

1. All pipe break sizes up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant loop, assuming unobstructed discharge from both ends.

2. A loss of coolant associated with the rod ejection accident.

3. A steam-generator tube rupture.

The basic design criteria for loss-of-coolant accident evaluations prior to codification under 10 CFR 50.46 were as follows:

1. The cladding temperature is to be less than:

    a. The melting temperature of Zircaloy-4.

    b. The temperature at which gross core geometry distortion, including clad fragmentation, may be expected.

2. The total core metal-water reaction will be limited to less than 1-percent.

These criteria ensure that the core geometry remains in place and substantially intact to such an extent that effective cooling of the core is not impaired.

Subsequently, the basic design criteria for loss-of-coolant accident calculations have been revised to those required under 10 CFR 50.46.

For any rupture of a steam pipe and the associated uncontrolled heat removal from the core, the safety injection system adds shutdown reactivity so that with a stuck rod, no offsite power, and minimum engineered safety features, there is no consequential damage to the reactor coolant system and the core remains in place and coolable as discussed in Section 14.2.5.

Redundancy and segregation of instrumentation and components are incorporated to ensure that postulated malfunctions will not impair the ability of the system to meet the design objectives. The system is effective in the event of loss of normal station auxiliary power coincident with the loss of coolant, and is tolerant of failures of any single component or instrument channel to respond actively in the system. During the recirculation phase, the system is tolerant of a loss of any part of the flow path since backup alternative flow path capability is provided.

The ability of the safety injection system to meet its capability objectives is presented in Section 6.2.3. The analysis of the accidents is presented in Chapter 14.

6.2.1.2 Inspection of Emergency Core Cooling System

Criterion: Design provisions shall, where practical, be made to facilitate inspection of physical parts of the emergency core cooling system, including reactor vessel internals and water injection nozzles. (GDC 45)

Design provisions are made to the extent practical to facilitate access to the critical parts of the reactor vessel internals, pipes, valves, and pumps for visual or boroscopic inspection for erosion, corrosion, and vibration-wear evidence and for nondestructive test inspection where such techniques are desirable and appropriate.

6.2.1.3 Testing of Emergency Core Cooling System Component

Criterion: Design provisions shall be made so that components of the emergency core cooling system can be tested periodically for operability and functional performance. (GDC 46)

The design provides for periodic testing of active components of the safety injection system for operability and functional performance.

Power sources are arranged to permit individual actuation of each active component of the safety injection system.

The safety injection pumps and residual heat removal pumps can be tested periodically during plant operation using the minimum flow recirculation lines provided. The residual heat removal pumps are used every time the residual heat removal loop is put into operation. All remote-operated valves can be exercised, and actuation circuits can be tested either during normal operation or routine plant maintenance.

6.2.1.4 Testing of Emergency Core Cooling System

Criterion: Capability shall be provided to test periodically the operability of the emergency core cooling system up to a location as close to the core as is practical. (GDC 47)

An integrated system test can be performed when the plant is cooled down and the residual heat removal loop is in operation. This test would not introduce flow into the reactor coolant system, but would demonstrate the operation of the valves, pump circuit breakers, and automatic circuitry upon the initiation of safety injection.

Level and pressure instrumentation is provided for each accumulator tank, and accumulator tank pressure and level are continuously monitored during plant operation. Flow from the tanks can be checked at any time using test lines as described in Section 6.2.5.3.1.

6.2.1.5 Testing of Operational Sequence of Emergency Core Cooling System

Criterion: Capability shall be provided to test initially, under conditions as close as practical to design, the full operational sequence that would bring the emergency core cooling system into action, including the transfer to alternate power sources. (GDC 48)

The design provides for the capability to test initially, to the extent practical, the full operational sequence up to the design conditions for the safety injection system to demonstrate the state of readiness and capability of the system. Details of the operational sequence testing are presented in Section 6.2.5.

6.2.1.6 Codes and Classifications

Table 6.2-1 lists the codes and standards to which the safety injection system components are designed.

6.2.1.7 Service Life

All portions of the system located within the containment are designed to operate without benefit of maintenance and without loss of functional performance for the duration of time the component is required.

6.2.2 System Design And Operation

6.2.2.1 System Description

Adequate emergency core cooling following a loss-of-coolant accident is provided by the safety injection system shown in Plant Drawing 9321-2735 [Formerly UFSAR Figure 6.2-1]. Plant Drawing 235296 [Formerly UFSAR Figures 6.2-2] and Figures 6.2-2 through 6.2-5 depict how this system concept is translated into plant layout design. The system components operate in the following possible modes:

1. Injection of borated water by the passive accumulators.

2. Injection by the safety injection pumps drawing borated water from the refueling water storage tank.

3. Injection by the residual heat removal pumps also drawing borated water from the refueling water storage tank.

4. Recirculation of spilled reactor coolant, injected water, and containment spray system drainage back to the reactor from the recirculation sump by the recirculation pumps. The residual heat removal pumps provide backup recirculation capability through the independent containment sump.

The initiation signal for core cooling by the safety injection pumps and the residual heat removal pumps is the safety injection signal, which is described in Section 7.2.3.2.3.

6.2.2.1.1 Injection Phase

The principal components of the safety injection system, which provide emergency core cooling immediately following a loss of coolant are the accumulators (one for each loop), the three safety injection (high-head) pumps, and the two residual heat removal (low-head) pumps. The safety injection and residual heat removal pumps are located in the primary auxiliary building.

The accumulators, which are passive components, discharge into the cold legs of the reactor coolant piping when pressure decreases to the minimum Technical Specification value, thus rapidly ensuring core cooling for large breaks. They are located inside the containment, but outside the crane wall, therefore each is protected against possible missiles.

The safety injection signal opens certain of the safety injection system isolation valves, provides confirmatory open signals to system isolation valves that are normally open, and starts the safety injection pumps and residual heat removal pumps.

The three safety injection pumps (high-head) deliver borated water to two separate discharge headers. The flow from the discharge headers can be injected into the four cold legs and two hot legs of the reactor coolant system. The motor-operated isolation valves in the four cold-leg injection lines are open during normal plant operation. The motor-operated isolation valves in the two hot-leg injection lines are closed during normal plant operation. The hot-leg injection lines are provided for later use during hot-leg recirculation following a reactor coolant pressure boundary break. The high-head safety injection system is configured with two cold leg injection lines physically connected to the reactor coolant pressure boundary and the other two lines connected to the accumulator discharge lines upstream of the pressure boundary. Since a small break in the reactor coolant pressure boundary can include a cold leg injection line, safety injection flow capability can be limited by the resulting flow from only three intact cold leg injection lines. Depending on the assumed single failure, either two or three safety injection pumps can be operating. To maximize the fraction of safety injection flow delivered to the reactor coolant system with a broken cold leg injection line, the four cold leg injection lines are flow balanced to within an allowable range. The resulting system flow capability is sufficient for the makeup of coolant following a small break that does not immediately depressurize the reactor coolant system to the accumulator discharge pressure. Credit is not taken for operator action to isolate a broken cold leg injection line.

For large breaks, the reactor coolant system would be depressurized and voided of coolant rapidly (about 26 sec for the largest break as shown in Figure 14.3-12) and a high flow rate is required to recover quickly the exposed fuel rods and limit possible core damage as discussed in Section 14.3.3.3.1. To achieve this objective, one residual heat removal pump and two safety injection pumps are required to deliver borated water to the cold legs of the reactor coolant loops. Two residual heat removal and three safety injection pumps are available to provide for an active component failure. Delivery from these pumps supplements the accumulator discharge. Since the reactor coolant system backpressure is relatively low (rapid depressurization for large breaks), a broken injection line would not appreciably change the flows in the other injection line's delivery to the core.

The residual heat removal pumps take suction from the refueling water storage tank. In addition, the charging pumps of the chemical and volume control system are available but are not required to augment the flow of the safety injection system.
Because the injection phase of the accident is terminated before the refueling water storage tank is completely emptied, all pipes are kept filled with water before recirculation is initiated.

Water level indication and alarms on the refueling water storage tank give the operator ample warning to terminate the injection phase. Additional level indicators and alarms are provided in the recirculation and containment sumps, which also give backup indication when injection can be terminated and recirculation initiated.

6.2.2.1.2 Recirculation Phase

After the injection operation, coolant spilled from the break and water collected from the containment spray are cooled and returned to the reactor coolant system by the recirculation system.

When the break is large, depressurization occurs due to the large rate of mass and energy loss through the break to containment. In the event of a large break, the recirculation flow path is within the containment. The system is arranged so that the recirculation pumps take suction from the recirculation sump in the containment floor and deliver spilled reactor coolant and borated refueling water back to the core through the residual heat exchangers. The system is also arranged to allow either of the residual heat removal pumps to take over the recirculation function. The residual heat removal pumps would only be used if backup capacity to the internal recirculation loop is required. Water is delivered from the containment to the residual heat removal pumps from the separate containment sump inside the containment.

For small breaks, the depressurization of the reactor coolant system is augmented by steam dump from and auxiliary feedwater addition to the steam generators. For the smaller breaks in the reactor coolant system where recirculated water must be injected against higher pressures for long-term cooling, the system is arranged to deliver the water from residual heat removal heat exchanger 21 to the high-head safety injection pump suction and by this external recirculation route to the reactor coolant loops. If this flow path is unavailable, an alternate flow path is provided as indicated in Table 6.2-11. Thus, if depressurization of the reactor coolant system proceeds slowly, the safety injection pumps may be used to augment the flow-pressure capacity of the recirculation pumps in returning the spilled coolant to the reactor. In this system configuration, the recirculation pump (or residual heat removal pump) provides flow and net positive suction head to the operating safety injection pumps. To prevent safety injection pump flow in excess of its maximum allowable (i.e., runout) limit, variable flow orifices are installed at the discharge of the safety injection pumps and the hot and cold leg motor-operated isolation valves are preset with mechanical stops based on data from operational flow testing to limit system maximum flow capability.

The recirculation pumps, the residual heat removal heat exchangers, piping, and valves vital to the function of the recirculation loop are located in a missile-shielded space inside the polar crane support wall on the west side of the reactor primary shield.

There are two sumps within the containment, the recirculation sump and the containment sump. Both sumps collect liquids discharged into the containment during the injection phase of the design-basis accident.

Various flow channeling barriers are installed in the Vapor Containment, EL 46'-0" to force the recirculation flow into the Reactor Cavity Sump area, up and out the Incore Instrumentation Tunnel, through the Crane Wall via the three nominal 20 inch square openings and into the annulus area outside the Crane Wall. The recirculation flow will migrate towards the Recirculation Sump Strainer or the Containment Sump Strainer depending on which pump(s) are operating. Flow channeling barriers are installed on the reactor Cavity Platform, EL 29'-4", around the Incore Instrumentation Tunnel, on the Recirculation Sump Trenches, and at the Containment Sump. Flow channeling barrier doors are installed in the Northeast and Northwest quadrant openings of the Crane Wall. In addition, flow channeling barrier doors are installed in the North and South entrances to the Recirculation Sump area. Perforated plate is installed on the RHR Heat Exchanger Platform, EL 66'-0" to preclude debris from washing through the existing grating and into the Recirculation Sump area. Forcing the recirculation flow path into the Reactor Cavity Sump area (a low velocity zone) allows the larger debris an opportunity to settle.

The Recirculation Sump and Containment Sump strainers consist of a matrix of multi-tube top-hat modules, which are fabricated from perforated stainless steel plate and mounted in the horizontal position. The perforated plate has 3/32" diameter holes sized to limit downstream affects. The top-hat modules have four (4) layers of perforated surfaces for straining debris from the sump fluid. Typical Recirculation Sump and Containment Sump strainer top-hat modules consist of a 12-1/2" diameter outer perforated tube with a respective 10-1/2" diameter inner perforated tube and a second set of tubes, which consist of a 7-1/2" diameter outer perforated tube with a respective 5-1/2" diameter inner perforated tube. The top-hat modules feature an internal vortex suppressor, which prevents air ingestion into the piping system. Stainless steel mesh has been installed between each pair of perforated plate tubes to minimize fiber bypass through the strainers. The top-hat modules are attached to strainer water boxes. The Containment Sump Level Detection System is discussed in Section 6.7.2.13.

The recirculation Sump relies on two, connected water boxes with 249 top-hat modules in the sump pit for the purpose of preventing particles greater than 3/32" in diameter from entering the suction of the recirculation pumps. The recirculation sump strainer has effective surface area of ~3,156 square feet and an effective interstitial volume of ~476 cubic feet. Water will enter the top-hat modules through the perforated plates and flow through the stainless steel mesh inside either of the two (2) annuli flow paths within each top-hat module. Upon exiting the top-hat modules, water will flow into either of the two connected strainer water boxes, flow over the Recirculation Sump weir wall and into the Recirculation Pump Bay towards the pumps.

The Containment Sump relies on a water box with 23 top-hat modules in the Containment Sump pit for the purpose of preventing particles greater than 3/32" in diameter from entering the Containment Sump suction line to the RHR Pumps. The Containment Sump strainer has an effective surface area of ~412 square feet and an effective interstitial volume of ~65 cubic feet. Water will enter the top-hat modules through the perforated plates and flow through the stainless steel mesh inside either of the two (2) annuli flow paths within each top-hat module. Upon exiting the top-hat modules, water will flow into the strainer water box, which is connected to the Containment Sump suction line and the RHR System. The containment sump level detection system is discussed in Section 6.7.1.2.13.

The low-head external recirculation loop via the containment sump line and the residual heat removal pumps provides backup recirculation capability to the low-head internal recirculation loop. The containment sump line has two remote motor-operated normally closed valves located outside the containment and a remote motor-operated butterfly valve inside containment. The high-head external recirculation flow path via the high-head safety injection pumps is only required for the range of small-break sizes for which the reactor coolant system pressure remains in excess of the shutoff head of the recirculation pumps (or residual heat removal pumps) at the end of the injection phase.

The external recirculation flow paths within the primary auxiliary building are designed so that external recirculation can be initiated immediately after the accident. Those portions of the safety injection system outside of the containment, which are designed to circulate, under postaccident conditions, radioactivity contaminated water collected in the containment meet the following requirements:

1. Shielding to limit radiation levels.

2. Collection of discharges from pressure-relieving devices into closed systems.

3. Means to detect and control radioactivity leakage into the environs.

These criteria are met by minimizing leakage from the system. External recirculation loop leakage is discussed in Section 6.2.3.8. The radiological consequences of external recirculation loop leakage following a design basis accident are presented in Section 14.3.6.6. Detection and control of leakage from external recirculation loop components is also discussed in Section 6.7.

One pump (either recirculation or residual heat removal) and one residual heat exchanger of the recirculation system provide sufficient cooled recirculated water to keep the core flooded with water by injection through the cold-leg connections while simultaneously providing, sufficient containment recirculation spray flow to reduce containment airborne activity. Three of the five fan cooler units prevent the containment pressure from rising above design limit. Analysis demonstrates that flow will be determined by system resistance provided by the physical configuration of the recirculation piping and components, and will be hydraulically balanced such that sufficient flow is established to the core and the spray header. Only one pump and one heat exchanger are required to operate for this capability at the earliest time recirculation spray is initiated. With both recirculation (or residual heat removal) pumps in operation and both spray header valves open, a recirculation spray flow rate can be established such that no containment cooling fans (Section 6.4) are required. Likewise with five containment cooling units in operation, no containment spray is required to maintain containment pressure below its design limit. The design ensures that heat removal from the core and containment is effective in the event of a pipe or valve body rupture.

6.2.2.1.3 Cooling Water

The service water system (Section 9.6) provides cooling water to the component cooling loop, which in turn cools the residual heat exchangers, both of which are part of the auxiliary coolant systems (Section 9.3). Three non-essential service water pumps are available to take suction from the river and discharge to the two component cooling heat exchangers. Three component cooling pumps are available to discharge through their heat exchangers and deliver to the two residual heat exchangers. During the recirculation phase following a loss-of-coolant-accident, only one residual heat removal heat exchanger, one recirculation or residual heat removal pump, one non-essential service water pump, one component cooling water pump and one component cooling water heat exchanger are required to meet the core-cooling function. All of this equipment, with the exception of the residual heat exchangers and the recirculation pumps, are outside containment.

6.2.2.1.4 Changeover From Injection Phase to Recirculation Phase

Assuming that the three high-head safety injection pumps, the two residual heat removal pumps, and the two containment spray pumps (Section 6.3) are running at their maximum capacity, the time sequence, from the time of the safety injection signal, for the changeover from injection to recirculation in the core of a large rupture is as follows:

1. In approximately 15 min, sufficient water has been delivered to provide the required net positive suction head to start the recirculation pumps.

2. In approximately 20 min, (a) low-level alarms on the refueling water storage tank sound, and (b) the redundant containment and recirculation sump level indicators show the sump water level. The alarm(s) serve to alert the operator to start the switchover to the recirculation mode. The redundant containment and recirculation sump level indicators provide verification that the refueling water storage tank water has been delivered during the injection phase, as well as giving consideration to the case of a spurious (i.e., early) refueling water storage tank low-level alarm. The operator would see on the control board that the redundant sump level indications are at the appropriate points; switchover via the eight-switch sequence is performed at that time.

3. With the initiation of the eight-switch sequence (i.e., switch No. 1), only one spray pump will continue in operation. This spray pump will continue to draw from the refueling water storage tank until the level drops below 2 feet.

Recirculation pump motors are 2-ft 2-in. above the highest water level after the addition of the injected water to the spilled coolant.

The changeover from injection to recirculation takes place when the level indicator or level alarms on the refueling water storage tank indicate that the fluid has been injected. The level indicators in the containment sump will also verify that the level is sufficient within the containment. The sequence is followed regardless of which power supply is available. All switches are grouped together on the safeguard control panel. The component position lights verify when the function of a given switch has been completed. Should an individual component fail to respond, the operator can take corrective action to secure appropriate response from the backup component. The manual switchover by the operator:

1. Terminates safety injection signal in order that the control logic permits manipulation of the system (at any time following completion of the auto-start sequence).

2. Closes switches one and three (removes and isolates unnecessary loads from the diesels).

   Switch One:

   a. Trips one (i.e., pump 22) of three high-head safety injection pumps if all three are operating (no action if two are operating), and isolates the pump suction to the refueling water storage tank if the tripped pump is the middle safety injection pump (i.e. pump 22).

b. Trips one containment spray pump if both are operating (no action if one is operating).

c. Closes isolation valves at the inoperative spray pump discharge.

Switch Three:

a. Trips both residual heat removal pumps.

b. Sends close signals to isolation valves in the residual heat removal pump suction and discharge headers, which are administratively reenergized later in the sequence. (Technical Specifications require the motor operators for these valves to be deenergized.)

3. Closes switch two (establishes cooling flow for residual heat removal heat exchangers)

   a. Starts one service water pump, non-essential header (the second or third pump is given a start signal if the first or second pump fails to start).

   b. Starts one component cooling pump (the second or third pump is given a start signal if the first or second pump fails to start).

4. Isolates one RHR heat exchanger flow path. (if both are open)

5. Closes switch four (initiates internal recirculation flow).

   a. Opens valves on discharge of recirculation pumps.

   b. Starts recirculation pump 21 (if pump 21 fails to start, uses manual start on pump 22). (Pump 22 control switch is adjacent to switch four).

6. Checks flow to reactor coolant system via the low-head injection lines to ensure minimum flow requirements are established. If minimum flow requirements are established, the closes switch seven and switch eight to establish low-head recirculation.

   If minimum flow requirements are not established, then closes switch six and switch eight to establish high-head recirculation.

7. Close switch six (supplies recirculation for reactor coolant system pressures greater than 150 psig, which impedes flow via the low-head injection lines).

   a. Opens valves 888A and 888B to provide a flow path from the recirculation pump discharge to the high-head safety injection pump suction.

   b. Activates the low-pressure alarm circuit off of PT-947. If for some reason PT-947 alarm is not activated by this switch (RS-6), the operator can switch "HI HEAD PUMP LO SUCTION PRESS ALARM" to activate this alarm. This latter switch is on the safeguards panel.

c. Closes valves 842 and 843 (high-head pump test line) (if their control feed interlock switches were first placed to the "OFF" position), and sends a close signal to valves 746 and 747 (residual heat removal heat exchanger discharge).

8. Closes switch seven (removes the two running safety injection pumps from service since they are no longer needed).

9. Closes switch eight (completes the isolation of the safety injection system and containment spray system lines from the refueling water storage tank).

   a. Closes the valve on the spray test line.

   b. Sends a close signal to the valve in the safety injection pumps suction line from the refueling water storage tank, which is administratively reenergized later in the sequence. (Control power for this valve is deenergized in accordance with Technical Specifications requirements).

10. Close switch five (Establishes additional cooling capability if adequate power is available i.e. all diesel breakers are either open or racked out, or at least one breaker from each of the three diesels is racked in and closed).

   a. Starts second service water pump, non-essential header (the third pump is given the start signal if the second pump fails to start).

   b. If (a) completed, starts second component cooling pump (the third pump is given the start signal if the second pump fails to start).

   c. If (b) completed, starts recirculation pump 22 (unless already running).

Although the listed switches are manual, each automatically causes the operations listed. An indicating lamp is provided to show the operator when the operations of a given switch have been performed. In addition, lamps indicating completion of the individual functions for a given switch are provided. These lamps are adjacent to the switches. Should an individual component fail to respond, the operator can take corrective action to secure appropriate response from controls within the control room.

Remote-operated valves for the injection phase of the safety injection system (Plant Drawings 9321-2735 and 235296 [Formerly Figure 6.2-1]), which are under manual control (i.e., valves, which normally are in their ready position and do not receive a safety injection signal) have their positions indicated on a common portion of the control board. At any time during operation when one of these valves is not in the ready position for injection, it is shown visually on the board. Table 6.2-2 lists the instrumentation readouts on the control board and assessment panel, which the operator can monitor during recirculation. In addition, an audible annunciation alerts the operator to the condition.

6.2.2.1.5 Location of the Major Components Required for Recirculation

The residual heat removal pumps are located in the residual heat removal pump room, which is on the basement floor of the primary auxiliary building (elevation 15-ft). The residual heat exchangers are on a platform above the basement floor in the containment building (elevation 66-ft).

The recirculation pumps are directly above the recirculation sump in the containment building (elevation 46-ft).

The component cooling pumps and heat exchangers are located in the primary auxiliary building (elevations 68-ft and 80-ft, respectively).

The service water pumps are located at the river water intake structure, and the redundant piping to the component cooling heat exchangers is run underground, until it surfaces just prior to its penetrating the Primary Auxiliary Building exterior wall.

6.2.2.2 Steam Line Break Protection

A large break of a steam system pipe rapidly cools the reactor coolant causing insertion of reactivity into the core and the depressurization of the system. Compensation is provided by the injection of boric acid from the refueling water storage tank. The analysis of the steam line rupture accident is presented in section 14.2.5.

6.2.2.3 Components

All associated components, piping, structures, and power supplies of the safety injection system are designed to seismic Class I criteria.

All components inside the containment are capable of withstanding or are protected from differential pressure that may occur during the rapid pressure rise to 47 psig in 10 sec.

Electrical equipment that has been determined to be important to safety and located in potentially harsh environments are environmentally qualified to ensure performance of their safety function under postaccident temperature, pressure, and humidity conditions.

Emergency core cooling components are either austenitic or an equivalent corrosion-resistant stainless steel, and hence, are compatible with the spray solution over the full range of exposure in the postaccident regime. Corrosion tests performed with simulated spray indicated negligible attack, both generally and locally, in stressed and unstressed stainless steel at containment and emergency core cooling system conditions. These tests are discussed in Reference 1.

The quality standards of all safety injection system components are given in summary form in Table 6.2-3.

6.2.2.3.1 Accumulators

The accumulators are pressure vessels filled with borated water and pressurized with nitrogen gas. During normal plant operation, each of the four accumulators is isolated from the reactor coolant system by two check valves in series. Should the reactor coolant system pressure fall below the accumulator pressure, the check valves open and borated water is forced into the cold legs of the reactor coolant system. Mechanical operation of the swing-disk check valves is the only action required to open the injection path from the accumulators to the core via the cold leg.

The level of borated water in each accumulator tank is adjusted remotely as required during normal plant operations. Refueling water is added using the accumulator topping pump (or safety injection pump 22 or 23). Water level is reduced by draining to the reactor coolant drain tank or through the chemistry sampling panel. Samples of the solution in the tanks are taken at the sampling station for periodic checks of boron concentration. Pressure is adjusted by adding nitrogen as required.

The accumulators are passive engineered safety features since the gas forces injection and no external source of power or signal transmission is needed to obtain fast-acting, high-flow capability when injection is required. One accumulator is attached to each of the four cold legs of the reactor coolant system.

The design capacity of the accumulators is based on the assumption that flow from one of the accumulators spills onto the containment floor through the ruptured loop. The flow from the three remaining accumulators provides sufficient water to fill the volume outside of the core barrel below the nozzles, the bottom plenum, and approximately one-half the core.

To assure the independence of the accumulators from each other, operating procedures require that only one liquid fill valve and only one nitrogen stop valve can be open at a time when reactor temperature is equal to or greater than 350°F.

The accumulators are carbon steel, internally clad with stainless steel and designed to ASME Section III, Class C. Connections for remotely draining or filling the fluid space, during normal plant operation, are provided.

Redundant level and pressure indicators are provided with readouts on the control board. Each indicator is equipped with high- and low-level alarms.

The accumulator design parameters are given in Table 6.2-4.

6.2.2.3.2 Boron Injection Tank

The boron injection tank has been removed.

6.2.2.3.3 Refueling Water Storage Tank

In addition to its normal duty to supply borated water to the refueling canal for refueling operations, this tank provides borated water to the safety injection pumps, the residual heat removal pumps, and the containment spray pumps for the loss-of-coolant accident. During plant operation, this tank is aligned to these pumps.

The capacity of the refueling water storage tank is based on the requirement for filling the refueling canal; a minimum of 345,000 gal is required by the Technical Specifications to be maintained in the refueling water storage tank. This capacity provides an amount of borated water to assure:

1. A sufficient volume of water on the floor to permit the initiation of recirculation (246,000 gal).

2. A volume water sufficient to allow time for completing the switchover to recirculation and securing High Head Injection and Containment Spray Flow from the RWST (60,000 gal).

3. A sufficient volume of water to allow for instrument inaccuracies, additional margin, and for water that is physically unavailable from the bottom of the tank (39,000 gal).

4. The RWST water volume injected into containment, when added to accumulator discharge to the reactor coolant system, assures no return to criticality with the reactor at cold shutdown and no control rods inserted into the core.

The water in the tank is borated to ensure a minimum shutdown margin as discussed in Section 14.1.5.2.1. The maximum boric acid concentration is approximately 1.4 wt percent boric acid. At 32°F, the solubility limit of boric acid is 2.2-percent. Therefore, the concentration of boric acid in the refueling water storage tank is well below the solubility limit at 32°F. Steam heating is provided for the tank, and the outside lines are heat traced to maintain the temperature above freezing.

Each of two redundant channels of refueling water storage tank level instrumentation provide level indication and low-level alarms in the central control room. In addition, a third instrument provides local level indication.

The design parameters are presented in Table 6.2-6.

6.2.2.3.4 Pumps

The three high-head safety injection pumps for supplying borated water to the reactor coolant system are horizontal centrifugal pumps driven by electrical motors. Parts of the pump in contact with borated water are stainless steel or an equivalent corrosion-resistant material. Each safety injection pump is sized at 50-percent of the capacity required to meet the design criteria outlined in Section 6.2.1. The design parameters are presented in Table 6.2-7; Figure 6.2-6 gives the performance characteristics of these pumps.

A minimum flow bypass line is provided on each pump discharge to recirculate flow to the refueling water storage tank in the event the pumps are started with the normal flow paths blocked. Valves in the minimum flow bypass line (which are normally open) are equipped with motor operators. If either valve closes, an alarm annunciates in the control room. Power is de-energized to prevent spurious valve closure.

The safety injection pump bearing oil is cooled by CCW circulating water pumps using component cooling water as a heat sink. The CCW circulating water pumps are directly connected to the injection pump motor shaft. The pump seals are designed to operate during the injection phase without forced component cooling water flow. During the recirculation phase, cooling water is supplied by the component cooling system or alternately from the primary water system. Emergency backup is available via connections to the city water system. The two residual heat removal (low-head) pumps of the auxiliary coolant system are used to inject borated water at low pressure into the reactor coolant system. The two recirculation pumps are used to recirculate fluid from the recirculation sump back to the reactor, to the spray headers, or to suction of the safety injection pumps. The recirculation pumps will only be required to operate during the recirculation phase. In addition, the recirculation pumps are required to be operable for a period of one year. All four of these pumps are of the vertical centrifugal type driven by electric motors. The recirculation pumps are open suction, well-type pumps. Parts of the pumps, which contact the borated water solution during recirculation are stainless steel or an equivalent corrosion-resistant material. A minimum flow bypass line is provided on the discharge of the residual heat removal heat exchangers to recirculate cooled fluid to the suction of the residual heat removal pumps should these pumps be started with their normal flow paths blocked. There are two normally open motor-operated valves in this line. The control power to the two normally open motor operated valves is locked open. The emergency procedures ensure that the RHR pumps are not run in parallel for extended time periods with RCS pressure at or above their shutoff head. A minimum flow bypass, discharging back into the recirculation sump, is provided to protect the recirculation pumps should these flow paths be blocked. Valves in these lines are manually operated and are in the open position during normal plant operation. Figures 6.2-7 and 6.2-8 give the performance characteristics of these pumps. The design parameters are presented in Table 6.2-7.

The recirculation pump motors are air-to-water cooled in a similar manner as the containment cooling fan motors described in section 6.4.2.2.5, item 2. The motor fans are integral to the recirculation pump motor shafts. Cooling water to the motor heat exchanger is component cooling water. The sump water cools the pump bearings. The two auxiliary component cooling water pumps are started during the injection phase, to maintain component cooling water flow to the motor coolers; however, this function is no longer required to protect the recirculation pump motors from the containment atmosphere. The auxiliary component cooling water pumps are a part of the component cooling water system and pump data are provided in Section 9.3. The component cooling water volume constitutes a large heat sink so that the main component cooling pumps are not needed during the injection phase.

Details of the component cooling pumps and service water pumps, which serve the safety injection system, are presented in Section 9.3 and 9.6, respectively.

The pressure-retaining parts of the high-head safety injection pumps are castings conforming to ASTM A-296, Grade CA-15 or ASME SA-487, Grade CA-6NM. The pressure-retaining parts of the residual heat removal pumps and the recirculation pumps are castings conforming to ASTM A-351, Grade CF-8A (chromium content 21.0 to 22.5%) and ASTM A-351, Grade CF-8, respectively. Stainless steel forgings are procured per ASTM A-182, Grade F304 or F316, or ASTM A-336, Class F8 or F8M, and stainless steel plate is constructed to ASTM A-240, Type 304 or 316. All bolting material conforms to ASTM A-193. Material such as Monel is used at points of close running clearances in the pumps to prevent galling and to ensure continued performance ability in high-velocity areas subject to erosion.

All pressure-retaining parts of the pumps were chemically and physically analyzed, and the results were checked to ensure conformance with the applicable ASTM specification. In addition, all pressure-retaining parts of the pump were liquid-penetrant inspected in accordance with Appendix VIII of Section VIII of the ASME Boiler and Pressure Vessel Code. The acceptance standard for the liquid-penetrant test is USAS B31.1, Code for Pressure Piping, Case N-10.

The pump design was reviewed with special attention to the reliability and maintenance aspects of the working components. Specific areas include the evaluation of the shaft seal and bearing design to determine whether adequate allowances had been made for shaft deflection and clearances between stationary parts.

Where welding of pressure-containing parts was necessary, a welding procedure including joint detail was submitted for review and approval by Westinghouse. The procedure included evidence of qualification necessary for compliance with Section IX of the ASME Boiler and Pressure Vessel Code Welding Qualifications. This requirement also applied to any repair welding performed on pressure-containing parts.

The pressure-containing parts of the pump were assembled and hydrostatically tested to 1.5 times the design pressure for 30 min.

Each pump was given a complete shop performance test in accordance with Hydraulic Institute Standards. The pumps were run at design flow and head, shutoff head, and three additional points to verify performance characteristics. Where net positive suction head is critical, this value was established at design flow by means of adjusting suction pressure.

An accumulator topping pump is provided to fill the accumulators rather than using safety injection pump 22 or 23. The pump is a double-diaphragm type with a capacity of approximately 5 gpm (293 gph). It is located in the safety injection system pump area and is operated from a local key-locked push button switch. The topping pump is capable of withstanding the safe shutdown earthquake but does not operate following safety injection actuation.

6.2.2.3.5 Heat Exchangers

The two residual heat exchangers of the auxiliary coolant system are sized for the cooldown of the reactor coolant system. Table 6.2-8 gives the design parameters of the heat exchangers. During the recirculation phase following a loss-of-coolant-accident, only one residual heat removal heat exchanger is required to ensure that heat removal requirements from the core and containment are met.

The ASME Boiler and Pressure Vessel Code has strict rules regarding the wall thicknesses of all pressure-containing parts, material quality assurance provisions, weld joint design, radiographic and liquid-penetrant examination of materials and joints, and hydrostatic testing of the unit as well as requiring final inspection and stamping of the vessel by an ASME Code inspector.

The designs of the heat exchangers also conform to the requirements of the Tubular Exchanger Manufacturers Association (TEMA) for Class R heat exchangers. Class R is the most rugged class of TEMA heat exchangers and is intended for units where safety and durability are required under severe service conditions. Items such as tube spacing, flange design, nozzle location, baffle thickness and spacing, and impingement plate requirements are set forth by TEMA standards.

In addition to the above, additional design and inspection requirements were imposed to ensure rugged, high-quality heat exchangers such as the following:

1. Confined-type gaskets, general construction and mounting brackets suitable for the plant seismic design requirements.

2. Tubes and tube sheet capable of withstanding full shell-side pressure and temperature with atmospheric pressure on the tube side, ultrasonic inspection in accordance with Paragraph N-324.3 of Section III of the ASME Code of all tubes

before bending, penetrant inspection in accordance with Paragraph N-627 of Section III of the ASME Code of all welds and hot- or cold-formed parts.

3. A hydrostatic test duration of not less than 30 min, the witnessing of hydro and penetrant tests by a qualified inspector, a thorough final inspection of the unit for good workmanship and the absence of any gouge marks or other scars that could act as stress concentration points, and a review of the radiographs and of the certified chemical and physical test reports for all materials used in the unit.

The residual heat exchangers are conventional vertical shell and U-tube type units. The tubes are seal welded to the tube sheet. The shell connections are flanged to facilitate shell removal for inspection and cleaning of the tube bundle. Each unit has an SA-285, Grade C, carbon steel shell; an SA-234 carbon steel shell end cap; SA-213, Type 304, stainless steel tubes; an SA-240, Type 304, stainless steel channel; an SA-240, Type 304, stainless steel channel cover; and an SA-240, Type 304, stainless steel tube sheet.

6.2.2.3.6 Valves

All parts of valves used in the safety injection system in contact with borated water are austenitic stainless steel or an equivalent corrosion-resistant material. The motor operators on the injection line isolation valves are capable of rapid operation. All valves required for the initiation of safety injection or isolation of the system have remote position indication in the control room.

Valving is specified for exceptional tightness. All valves, except those which perform a control function, are provided with backseats that are capable of limiting leakage. The estimated leakage of backseated valves outside containment is provided in Table 6.2-9. Those valves, which are normally open are backseated, except when operational considerations do not allow. [*Note - The following valves may not be backseated based on operational requirements: 744, 850A, 850B, 851A, 851B, 883, 885A, 885B, 887A, 887B, 888A, 888B and 958.*] Normally closed globe valves are installed with recirculation flow under the seat to prevent the leakage of recirculated water through the valve stem packing. Relief valves are totally enclosed. Control and motor-operated valves, 2.5-in. and above, which are exposed to recirculation flow, are generally provided with double-packed stuffing boxes and stem leakoff-connections that are piped to the waste disposal system.

The check valves that isolate the safety injection system from the reactor coolant system are installed immediately adjacent to the reactor coolant piping to reduce the probability of a safety injection line rupture causing a loss-of-coolant accident.

A relief valve is installed in the safety injection pump discharge header discharging to the pressurizer relief tank to prevent overpressure in the lines that have a lower design pressure than the reactor coolant system. RV-855 is a thermal relief valve which protects the Safety Injection System piping and components from overpressurization due to thermal expansion of fluid in the system or from in-leakage of reactor coolant. The setpoint of RV-855 was changed to 1670 psig to ensure that the valve does not lift when operating the SI system at a pressure mear the shutoff head of the SI pumps.

The gas relief valves on the accumulators protect them from pressures in excess of the design value.

6.2.2.3.7 Motor-Operated Butterfly Valve (Containment Sump Valve)

The pressure-containing parts (body, disks) of the valves employed in the safety injection system are designed per criteria established by USAS B16.5 or MSS SP-67 specifications. The materials of construction for these parts are procured per ASTM A182, F316 or A351, Grade CF8M or CF8. All material in contact with the primary fluid, except the packing and the liner, is austenitic stainless steel or an equivalent corrosion-resistant material. The liner is EPT-NORDEL (Du Pont). The pressure-containing cast components are radiographically inspected as outlined in ASTM E-71, Class 1 or Class 2. The body and disk are liquid-penetrant inspected in accordance with ASME Boiler and Pressure Vessel Code, Section VIII, Appendix VIII. The liquid-penetrant acceptable standard is as outlined in USAS B31.1, Case N-10.

The entire assembled unit is hydrotested as outlined in MSS SP-67, with the exception that the test is maintained for a minimum period of 30 min. The motor operator is evaluated in accordance with the GL 89-10 Motor Operated Valve Program to assure its capability to meet the required stem torque for opening and closing.

The shaft material is ASTM A276, Type 316, condition B, or precipitation hardened 17-4 pH stainless steel procured and heat treated to Westinghouse specifications. These materials are selected because of their corrosion-resistant, high-tensile properties, and their resistance to surface scoring by the packing.

The motor operator is located above the maximum sump fluid level and therefore is never submerged. The motor operator is extremely rugged and is noted throughout the power industry for its reliability. The unit incorporates a hammer-blow feature that allows the motor to impact the disks away from the fore or backseat upon opening or closing. This hammer-blow feature not only impacts the disk but allows the motor to attain its operational speed.

The valve is assembled, hydrostatically tested, seat-leakage tested (fore and back), operationally tested, cleaned, and packaged per specifications. All manufacturing procedures employed by the valve supplier such as welding, repair welding, and testing are submitted to Westinghouse for approval.

The valve operator completes its cycle from one position to the other in approximately 120 sec.

Valves that must function against system pressure are designed such that they function with a pressure drop equal to full system pressure across the valve disk.

6.2.2.3.8 Motor-Operated Gate Valves

The pressure-containing parts (body, bonnet, and disks) of the valves employed in the safety injection system are designed per criteria established by USAS B16.5 or MSS SP-66 specifications. The materials of construction for these parts are procured per ASTM A182, F316 or A351, Grade CF8M or CF8. All material in contact with the primary fluid, except the packing, is austenitic stainless steel or an equivalent corrosion-resistant material. The pressure-containing cast components are radiographically inspected as outlined in ASTM E-71, Class 1 or Class 2. The body, bonnet, and disks are liquid-penetrant inspected in accordance with ASME Boiler and Pressure Vessel Code, Section VIII, Appendix VIII. The liquid-penetrant acceptable standard is as outlined in USAS B31.1, Case N-10.

When a gasket is employed, the body-to-bonnet joint is designed per ASME Boiler and Pressure Vessel Code, Section VIII, or USAS B16.5 with a fully trapped, controlled compression, spiral wound gasket with provisions for seal welding, or of the pressure seal design with provisions for seal welding. The body-to-bonnet bolting and nut materials are procured per ASTM A193 and A194, respectively.

The entire assembled unit is hydrotested as outlined in MSS SP-61, with the exception that the test is maintained for a minimum period of 30 min. The seating design is of the Darling parallel disk design, the Crane flexible wedge design, or the equivalent. These designs have the feature of releasing the mechanical holding force during the first increment of travel. Thus, the motor operator is evaluated in accordance with the GL 89-10 Motor Operated Valve Program to assure its capability to meet the required stem thrust for opening and closing. The disks are guided throughout the full disk travel to prevent chattering and to provide ease of gate movement. The seating surfaces are hard faced (Stellite No. 6 or equivalent) to prevent galling and reduce wear.

The stem material is ASTM A276, Type 316, condition B, or precipitation hardened 17-4 pH stainless steel procured and heat treated to Westinghouse specifications. These materials are selected because of their corrosion-resistant, high-tensile properties, and their resistance to surface scoring by the packing. The valve stuffing box was originally designed with a lantern ring leakoff connection with a minimum of a full set of packing below the lantern ring and a maximum of one-half of a set of packing above the lantern ring; a full set of packing is defined as a depth of packing equal to 1-1/2 times the stem diameter. An alternate packing arrangement may be installed in these valves upon approval for substitution. Experience with designs utilizing live load and graphite packing has been favorable.

The motor operator is extremely rugged and is noted throughout the power industry for its reliability. The unit incorporates a hammer-blow feature that allows the motor to impact the disks away from the fore or backseat upon opening or closing. This hammer-blow feature not only impacts the disk but allows the motor to attain its operational speed.

The valve is assembled, hydrostatically tested, seat-leakage tested (fore and back), operationally tested, cleaned, and packaged per specifications. All manufacturing procedures employed by the valve supplier such as hard facing, welding, repair welding, and testing are submitted to Westinghouse for approval.

For those valves that must function on the safety injection signal, approximately 10-sec operation is required. For all other valves in the system, the valve operator completes its cycle from one position to the other in approximately 120 sec. Operating times greater than these values are permitted on a case by case basis if properly justified by an individual safety evaluation.

Valves that must function against system pressure are designed such that they function with a pressure drop equal to full system pressure across the valve disk.

6.2.2.3.9 Manual Valves

The stainless steel manual globe, gate, and check valves are designed and built in accordance with the requirements outlined in the motor-operated valve description above with the following exceptions:

1. Alternate materials, evaluated to be equivalent, have been used in some replacement valves.

2. Liquid-penetrant inspection of the body, bonnet, and disks to ASME V Article 6 with acceptance per ASME III has been used on some replacement valves.

The carbon steel valves are built to conform with USAS B16.5. The materials of construction of the body, bonnet, and disk conform to the requirements of ASTM A105, Grade II; A181, Grade II; or A216, Grade WCB or WCC. Alternate materials, evaluated to be equivalent, have been used in some replacement valves. The carbon steel valves pass only nonradioactive fluids and are subjected to hydrostatic test as outlined in MSS SP-61, except that the test pressure is maintained for at least 30 min/in. of wall thickness. Since the fluid controlled by the carbon steel valves is not radioactive, the double packing and seal weld provisions are not provided.

6.2.2.3.10 Accumulator Check Valves

The pressure-containing parts of this valve assembly are designed in accordance with MSS SP-66. All parts in contact with the operating fluid are of austenitic stainless steel or of equivalent corrosion-resistant materials procured to applicable ASTM or WAPD specifications. The cast pressure-containing parts are radiographed in accordance with ASTM E-94 and the acceptance standard as outlined in ASTM E-71, Class 1 or Class 2. The cast pressure-containing parts, machined surfaces finished hard facings, and gasket bearing surfaces are liquid-penetrant inspected per the ASME Boiler and Pressure Vessel Code, Section VIII, and the acceptance standard is as outlined in USAS B31.1, Code Case N-10. The final valve is hydrotested per MSS SP-61, except that the test pressure is maintained for at least 30 min. The seat leakage is conducted in accordance with the manner prescribed in MSS SP-61, except that the acceptable leakage is 2 cm^3/hr-in. nominal pipe diameter.

The valve is designed with a low-pressure drop configuration with all operating parts contained within the body, which eliminates those problems associated with packing glands exposed to boric acid. The clapper arm shaft is manufactured from 17-4 pH stainless steel heat treated to Westinghouse specifications. The clapper arm shaft bushings are manufactured from Stellite No. 6 material. The various working parts are selected for their corrosion-resistant, tensile, and bearing properties.

The disk and seat rings are manufactured from forgings. The mating surfaces are hard faced with Stellite No. 6 to improve the valve seating life. The flexible disc-hinge connection permits the disc to completely contact the seat even if there is minor seat movement.

The valves are intended to be operated in the closed position with a normal differential pressure across the disk of approximately 1650 psi. The valves remain in the closed position except for testing and safety injection. Since the valve will normally not be required to operate in the open condition, hence be subjected to impact loads caused by sudden flow reversal, it is expected that this equipment will not have difficulties performing its required functions.

When the valve is required to function, a differential pressure of less than 25 psig will shear any particles that may attempt to prevent the valve from functioning. Although the working parts are exposed to the boric acid solution contained within the reactor coolant loop, a boric acid "freeze up" is not expected with this low a concentration.

The experience derived from the check valves employed in the safety injection system of the Carolina-Virginia Tube Reactor (CVTR) in a similar system has indicated that the system is reliable and workable. The CVTR emergency injection system, maintained at atmospheric conditions, was separated from the main coolant piping by one 6-in. check valve. A leak detection pit was provided in the CVTR to accumulate any leakage coming back through the check valve. A level alarm provided a signal on excessive leakage. There was a gas volume in the upper space of the loop. The pressure differential was 1500 psi and the system was stagnant. The valve was located 2 to 3-ft from the main coolant piping, which resulted in some heatup and cooldown cycling. The CVTR went critical late in 1963. Since that time and up to initial operation of Indian Point Unit 2, the level alarm in the detection pit had never gone off due to check valve leakage.

6.2.2.3.11 Relief Valves

The accumulator relief valves are sized to pass nitrogen gas at a rate in excess of the accumulator gas fill line delivery rate. The relief valves will also pass water in excess of the maximum expected leak rate of 1.0 gpm identified in Technical Specifications with leakage being from the reactor coolant system into an accumulator through an accumulator discharge line. The accumulators are provided with level and pressure alarms. Operator response to inleakage causing these alarms to actuate would preclude the need for the relief valves to perform in a water relief capacity.

The safety injection test line relief valve is provided to relieve any overpressure, that might build up in the high-head safety injection piping due to thermal expansion of fluid in the system or from leakage from the reactor coolant system past the SI header check valves. The valve will pass a nominal 15 gpm (2.25 x 10^5cm^3/hr), which is far in excess of the manufacturing design in leakage rate from the reactor coolant system of 24 cm^3/hr.

6.2.2.3.12 Leakage Limitations of Valves

Valving is specified for exceptional tightness.

Normally open valves have backseats that limit leakage as shown in Table 6.2-9. Normally closed globe valves are installed with recirculation flow under the seat to prevent stem leakage from the more radioactive fluid side of the seat.

Motor-operated valves, which are exposed to recirculation flow, are generally provided with double-packed stuffing boxes and stem leakoff connections that are piped to the waste disposal system.

The specified leakage across the valve disk required to meet the equipment specification and hydrotest requirements is as follows:

1. Conventional globe - 3 cm^3/hr-in. of nominal pipe size.

2. Gate valves - 3 cm^3/hr-in. of nominal pipe size; 10 cm^3/hr-in. for 300- and 150-lb USA Standard.

3. Motor-operated gate valves - 3 cm^3/hr-in. of nominal pipe sizes; 10 cm^3/hr-in. for 300- and 150-lb USA Standard.

4. Check valves - 3 cm^3/hr-in. of nominal pipe size; 10 cm^3/hr-in. for 300- and 150-lb USA Standard.

5. Accumulator check valves - 2 cm^3/hr-in. of nominal pipe size; relief valves are totally enclosed.

Leakage from components of the recirculation loop including valves, is given in Table 6.2-9.

6.2.2.3.13 Piping

All safety injection system piping in contact with borated water is austenitic stainless steel. Piping joints are welded except for the flanged connections at the safety injection pumps, the recirculation pumps, and valve 741A.

The piping beyond the accumulator stop valves is designed for reactor coolant system conditions (2485 psig, 650°F). All other piping connected to the accumulator tanks is designed for 700 psig and 400°F.

The safety injection pump and residual heat removal pump suction piping (210 psig at 300°F) from the refueling water storage meets net positive suction head requirements of the pumps.

The safety injection high-pressure branch lines (1500 psig at 300°F) are designed for high-pressure losses to limit the flow rate out of the branch line, which may have ruptured at the connection to the reactor coolant loop. The system design incorporates the ability to isolate the safety injection pumps on separate headers such that full flow from at least one pump is ensured should a branch line break.

The piping is designed to meet the minimum requirements set forth in (1) the USAS B31.1 Code (1955) for the Pressure Piping, (2) Nuclear Code Case N-7, (3) USAS Standards B36.10 and B36.19, and (4) ASTM Standards with supplementary standards plus additional quality control measures.

Minimum wall thicknesses are determined by the USAS Code (1955) formula found in the power piping Section 1 of the USAS Code (1955) for Pressure Piping. This minimum thickness has been increased to account for the manufacturer's permissible tolerance of −12.5-percent on the nominal wall. Purchased pipe and fittings have a specified nominal wall thickness that is no less than the sum of that required for pressure containment, mechanical strength, and manufacturing tolerance.

Thermal and seismic piping flexibility analyses have been performed. Special attention is directed to the piping configuration at the pumps with the objective of minimizing pipe imposed loads at the suction and discharge nozzles. Piping is supported to accommodate expansion due to temperature changes during the accident.

Piping between valves 730 and 731 (Line 10) has 6" thick insulation to assure operability of these valves during design basis accident conditions.

Pipe and fittings materials are procured in conformance with all requirements of ASTM and USAS specifications. All materials are verified for conformance to specification and documented by certification of compliance to ASTM material requirements. Specifications impose additional quality control upon the suppliers of pipes and fittings as listed below.

1. Check analyses are performed on both the purchased pipe and fittings.

2. Pipe branch lines 2.5-in. and larger between the reactor coolant pipes and the isolation stop valves conform to ASTM A376 and meet the supplementary requirement S6 for ultrasonic testing. Fittings conform to the requirements of ASTM A403. Fittings 2.5-in. and above have requirements for ultrasonic testing inspection similar to S6 of A376.

Shop fabrication of piping subassemblies is performed by reputable suppliers in accordance with specifications that define and govern material procurement, detailed design, shop fabrication, cleaning, inspection, identification, packaging, and shipment.

Welds for pipes sized 2.5-in. and larger are butt welded. Reducing tees are used where the branch size exceeds one-half of the header size. Branch connections of sizes that are equal to or less than one-half of the header size are of a design that conforms to the USAS rules for reinforcement set forth in the USAS B31.1 Code for Pressure Piping. Bosses for branch connections are attached to the header by means of full penetration welds.

All welding is performed by welders and welding procedures qualified in accordance with the ASME Boiler and Pressure Vessel Code, Section IX, Welding Qualifications. The shop fabricator is required to submit all welding procedures and evidence of qualifications for review and approval prior to release for fabrication. All welding materials used by the shop fabricator must have prior approval.

All high-pressure piping butt welds containing radioactive fluid at greater than 600°F temperature and 600 psig pressure or equivalent are radiographed. The remaining piping butt welds are randomly radiographed. The technique and acceptance standards are those outlined in UW-51 of the ASME Boiler and Pressure Vessel Code, Section VIII. In addition butt welds are liquid-penetrant examined in accordance with the procedure of ASME Boiler and Pressure Vessel Code, Section VIII, Appendix VIII, and the acceptance standard as defined in the USAS Nuclear Code Case N-10. Finished branch welds are liquid-penetrant examined on the outside, and where size permits, on the inside root surfaces.

A postbending solution anneal heat treatment is performed on hot-formed stainless steel pipe bends. Completed bends are then completely cleaned of oxidation from all affected surfaces. The shop fabricator is required to submit the bending, heat treatment, and cleanup procedures for review and approval-prior to release for fabrication.

General cleaning of completed piping subassemblies (inside and outside surfaces) is governed by basic ground rules set forth in the specifications. For example, these specifications prohibit the use of hydrochloric acid and limit the chloride content of service water and demineralized water.

The packaging of the piping subassemblies for shipment is done so as to preclude damage during transit and storage. Openings are closed and sealed with tight fitting covers to prevent the entry of moisture and foreign material. Flange facings and weld end preparations are protected from damage by means of wooden cover plates and securely fastened in position. The packing arrangement proposed by the shop fabricator is subject to approval.

6.2.2.3.14 Pump and Valve Motors Outside Containment

Motor electrical insulation systems are supplied in accordance with IEEE, and NEMA standards and are tested as required by standards.

Temperature rise design selection is such that normal long life is achieved even under accident loading conditions.

Criteria for motors of the safety injection system require that under any anticipated mode of operation the motor nameplate rating is not exceeded. The pump motors have a 1.15 service factor for normal operation. Design and test criteria ensure that motor loading does not exceed the application criteria.

6.2.2.3.15 Pump and Valve Motors Inside Containment

Motors for the recirculation pumps were originally specified to operate in an ambient condition of saturated steam of 270°F and 47 psig pressure for 1 day, followed by indefinite operation at 155°F and 5 psig in a steam atmosphere. These ambient conditions and operating times have been updated and are maintained by the ongoing Environmental Qualification Program discussed in Section 7.1.4. As part of this program, the recirculation pump motors are qualified to withstand containment environmental conditions following the loss of coolant accident so that the pumps can perform their required function during the recovery period (one year). These motors are of a similar design as the containment fan cooler motors. Refer to Section 6.4.2.2.5 for a description and evaluation of the motor design.

The motors for the valves inside containment are designed to withstand containment environment conditions following the loss-of-coolant accident so that the valves can perform the required function during the recovery period.

Periodic operation of the motors and tests of the insulation ensure that the motors remain in a reliable operating condition.

Although the motors that are provided only to drive engineered safety features equipment are normally run only for tests, the design loading and temperature rise limits are based on accident conditions. Normal design margins are specified for these motors to make sure the expected lifetime includes allowance for the occurrence of accident conditions.

6.2.2.3.16 Valve Motor Operators

Environmental Qualification

As part of the original plant design, a program of environmental qualifications was performed on valve motor operators important to plant safety. Tests to demonstrate the adequacy of valve motor operators to be functional after exposure to temperature, pressure, and radiation were conducted in two groups.

The first group test was the exposure of valve motor operators to both temperature and pressure. Two suppliers, Philadelphia Gear Corporation Limitorque Division and Crane

Company Teledyne Division, conducted simulated containment pressure and temperature tests as follows with pressure and temperature similar to that predicted for the incident:

1. Operator located inside a pressure vessel with the operator exposed to approximately 330°F at 90 psig.

2. Operator cycled approximately 3 times under simulated valve operating loads.

3. Pressures and temperatures reduced in step change to 285°F at 60 psig, 219°F at 20 psig, and 152°F at atmosphere or less.

4. Operator cycled approximately 3 times at each of the levels of change. Full recordings of pertinent data were taken throughout the tests.

5. Unit was examined after completion of test and operating data compared to data prior to exposure.

The second group test was the radiation test on a motor from the valve operator.

1. Two production line motors were used for this test; one exposed to 1.5 x 10^8 rads of gamma radiation for an approximate period of 1 month, the other motor used for the final comparative analysis.

2. Both units were tested for coil resistance, insulation meggering both before and after motor vibration, and reversing operations.

More recently, a program of environmental requalifications of items important to plant safety has been initiated using the "Division of Operating Reactors" or NUREG-0588 guidelines. See Section 7.1.4 for a discussion of this ongoing program.

In response to IE Information Notice 86-03, all limitorque motor operators on the EQ Master List (see Section 7.1.4) were inspected and serviced to assure that wiring, limit switches and torque switches have been environmentally qualified.

In response to the IE Bulletin 85-03, the operability of key safety Motor Operated Valves was verified with associated full differential pressure.

6.2.2.4 Electrical Supply

Details of the normal and emergency power sources for the safety injection system are presented in Chapter 8.

6.2.2.5 Protection Against Dynamic Effects

The injection lines penetrate the containment adjacent to the primary auxiliary building. For most of the routing, these lines are outside the crane wall, and hence are protected from missiles originating within these areas. Each line penetrates the crane wall near the injection point to the reactor coolant pipe. In this manner, maximum separation and hence protection it provided in the coolant loop area.

Coolant loop supports are designed to restrict the motion in one loop due to rupture in another loop to about one-tenth of an inch, whereas the attached safety injection piping can sustain a 3-in. displacement without exceeding the working stress range. The analysis assumes that the injection flow to the ruptured loop is spilled on the containment floor.

In 1989, the NRC approved changes to the design basis with respect to dynamic effects of postulated primary loop ruptures, as discussed in Section 4.1.2.4

All hangers and anchors are designed in accordance with USAS B31.1, Code for Pressure Piping. This code provides minimum requirements on materials, design, and fabrication with ample safety margins for both dead and dynamic loads over the life of the equipment. Concrete missile barriers, bumpers, walls, and other concrete structures are designed in accordance with ACI 318-63, Building Code Requirements for Reinforced Concrete. Specifically, these standards require the following:

1. All materials used are in accordance with ASTM specifications that establish quality levels for the manufacturing process, minimum strength properties, and for test requirements that ensure compliance with the specifications.

2. Welding processes and welders must be qualified for each class of material welded and for types and positions of welds.

3. Maximum allowable stress values are established, which provide an ample safety margin on both yield strength and ultimate strength.

6.2.3 Design Evaluation

6.2.3.1 Range of Core Protection

The measure of effectiveness of the safety injection system is the ability of the pumps and accumulators to keep the core flooded or to reflood the core rapidly when the core has been uncovered for postulated large area ruptures. The result of this performance is to limit sufficiently any increase in clad temperature below a value where emergency core cooling objectives are met (Section 6.2.1). The sequence of events involving safety injection actuation for small and large breaks of a reactor coolant pipe are presented in Section 14.3.2.

6.2.3.2 System Response

To provide protection for large area ruptures in the reactor coolant system, the safety injection system must respond rapidly to reflood the core following the depressurization and core voiding that is characteristic of large area ruptures. The accumulators act to perform the rapid reflooding function with no dependence on the normal or emergency power sources and also with no dependence on the receipt of an actuation signal.

The operation of this system with three of the four available accumulators delivering their contents to the reactor vessel (one accumulator spilling through the break) prevents fuel clad melting and limits metal-water reaction to an insignificant amount (<1-percent).

The function of the safety injection and residual heat removal pumps is to complete the refill of the vessel and ultimately return the core to a subcooled state. Moreover, there is sufficient excess water delivered by the accumulators to tolerate a delay in starting the pumps.

Initial response of the injection systems is automatic, with appropriate allowances for delays in the actuation of circuitry and active components. The active portions of the injection systems are automatically actuated by the safety injection signal (Chapter 7). In addition, manual actuation of the entire injection system and individual components can be accomplished from the control room. In the analysis of system performance, delays in reaching the programmed trip points and in the actuation of components are conservatively established on the basis that only emergency onsite power is available.

The starting sequence of the safety injection and residual heat removal pumps and the related emergency power equipment is discussed in sections 7.2 and 8.2.3.4 and their analyzed performance is discussed in the various Chapter 14 safety analyses.

6.2.3.3 Single-Failure Analysis

A single active failure analysis is presented in Table 6.2-10. All credible active system failures are considered. This analysis is based on the worst single failure (generally a pump failure) in both the safety injection and residual heat removal pumping systems. The analysis shows that the failure of any single active component will not prevent fulfilling the design function. The analysis of the loss-of-coolant accident presented in Section 14.3 is consistent with this single-failure analysis.

In addition, an alternative flow path is available to maintain core cooling if any part of the recirculation flow path becomes unavailable. This is evaluated in Table 6.2-11. The procedure followed to establish the alternative flow path also isolates the spilling line. A valve is provided in the containment recirculation line to the residual heat removal pumps to isolate this line should it be required.

Failure analyses of the component cooling and service water system under loss-of-coolant accident conditions are described in Sections 9.3 and 9.6, respectively.

6.2.3.4 Reliance on Interconnected Systems

During the injection phase, the high-head safety injection pumps do not depend on any portion of other systems, with the exception of the suction line from the refueling water storage tank and the component cooling loop as a heat sink for bearing and lube oil cooling. During the recirculation phase of the accident for small breaks, suction to the high-head safety injection pumps is provided by the recirculation pumps or the residual heat removal pumps. The residual heat removal (low-head) pumps are normally used during reactor shutdown operations. Whenever the reactor is at power, the pumps are aligned for emergency duty.

6.2.3.5 Shared Function Evaluation

Table 6.2-12 is an evaluation of the main components, which have been previously discussed, and a brief description of how each component functions during normal operation and during the accident.

6.2.3.6 Passive Systems

The accumulators are a passive safety feature in that they perform their design function in the total absence of an actuation signal or power source. The only moving parts in the accumulator injection train are in the two check valves.

The working parts of the check valves are exposed to fluid of relatively low boric acid concentration. Even if some unforeseen deposition accumulated, a reversed differential pressure of about 25 psi can shear any particles in the bearing that may tend to prevent valve functioning. This is demonstrated by calculation.

The isolation valve at each accumulator is only closed when the reactor is intentionally depressurized or momentarily for testing when pressurized. The isolation valve is normally open and an alarm in the control room sounds if the valve is inadvertently closed. It is not expected that the isolation valve will have to be closed due to excessive leakage through the check valves.

The check valves operate in the closed position with a nominal differential pressure across the disk of approximately 1650 psi. They remain in this position except for testing or when called upon to function. Since the valves operate normally in the closed position and are therefore not subject to the abuse of flowing operation or impact loads caused by sudden flow reversal and seating, they do not experience any wear of the moving parts, and therefore function as required.

When the reactor coolant system is being pressurized during the normal plant heatup operation, the check valves can be tested for leakage as soon as there is about 150 psi differential across the valve. This test confirms the seating of the disk and whether or not there has been an increase in the leakage since the last test. When this test is completed, the discharge line test valves are opened and the reactor coolant system pressure increase is continued. There should be no increase in leakage from this point on since increasing reactor coolant pressure increases the seating force and decreases the probability of leakage.

The accumulators can accept leakage from the reactor coolant system without effect on their availability. Table 6.2-13 indicates what inleakage rates, over a given time period, require readjusting the level at the end of the time period. In addition, these rates are compared to the maximum allowed leak rates for manufacturing acceptance tests (20 cm^3/hr i.e., 2 cm^3/hr/in).

Inleakage at a rate of 5 cm^3/hr-in., 2.5 times test, would require that the accumulator water volume be adjusted approximately once every 30 months. This would indicate that level adjustments can be scheduled for normal refueling shutdowns and that this work can be done at the operator's convenience. At a leak rate of 30 cm^3/hr-in. (15 times the acceptance leak rate), the water level will have to be readjusted approximately once every 5 to 6 months. This readjustment will take about 2 hr maximum.

The accumulators are located inside the reactor containment and protected from the reactor coolant system piping and components by a missile barrier. Accidental release of the gas charge in the accumulator would cause an increase in the containment pressure. This release of gas has been included in the containment pressure analysis for the large break loss-of-coolant accidents, (Section 14.3.3.3 and 14.3.5.1.1).

During normal operation, the flow rate through the reactor coolant piping is approximately 5 times the maximum flow rate from the accumulator during injection. Therefore, fluid impingement on reactor vessel components during operation of the accumulator is not restricting.

6.2.3.7 Emergency Flow to the Core

Special attention is given to factors that could adversely affect the accumulator and safety injection flow to the core. These factors are as follows:

1. Steam binding in the core, including flow blockage due to loop sealing.
2. Loss of accumulator water during blowdown.
3. Short circuiting of the accumulator from the core to another part of the reactor coolant system.
4. Loss of accumulator water through the breaks.

All of the above are considered in the analysis of the Loss of Coolant Accident which is discussed in Section 14.3.

6.2.3.8 External Recirculation Loop Leakage

Table 6.2-9 summarizes the maximum potential leakage from the leak sources of the external recirculation loop, which goes through the residual heat removal pumps, a residual neat exchanger, and the high-head safety injection pumps. In the analysis, a maximum leakage is assumed from each leak source. For conservatism, 3 times the maximum expected leak rate from the pump seals was assumed, even though the seals are acceptance tested to essentially zero leakage, and a leakage of 10 drops/min was assumed from each flange although each flange would be adjusted to essentially zero leakage. The total maximum potential leakage resulting from all sources is 999 cm^3/hr to the auxiliary building atmosphere and 21 cm^3/hr to the drain tank.

During external recirculation, significant margin exists between the design and operating conditions of the residual heat removal system components, as shown in Table 6.2-14. In addition, during normal plant cooldown, operation of the residual heat removal system is initiated when the primary system pressure and temperature have been reduced to below 365 psig (the upper limit to prevent RHR system overpressurization) and 350°F, respectively. Even assuming a conservative maximum RHR System pressure of 232 psig and a conservative maximum RHR System temperature of 277°F during recirculation as shown in Table 6.2-14, significant margin also exists between normal operating and accident conditions. In view of the above margins, it is considered that the leakage rates tabulated in Table 6.2-9 are conservative. The radiological consequences of external recirculation loop leakage following a design basis accident are presented in Section 14.3.6.6.

6.2.3.9 Pump Net Positive Suction Head Requirements

6.2.3.9.1 Residual Heat Removal Pumps

The net positive suction head (NPSH) of the residual heat removal pumps is evaluated for normal plant shutdown operation and the operation of both the injection and recirculation phases of the design-basis accident.

The residual heat removal pumps are used as backup to the internal recirculation pumps in the event of failures to the normal recirculation path; this duty provides the pumps with the minimum NPSH condition. For both the design case of two pumps recirculating through two heat exchangers paths, as well as the case of only one pump recirculating through one heat exchanger path, the available NPSH exceeds the NPSH required, assuming saturated fluid and no operator action to throttle back the flow.

6.2.3.9.2 Safety Injection Pumps

The NPSH for the safety injection pumps is evaluated for both the injection and recirculation phase operation of the design-basis accident. The end of injection phase operation gives the limiting NPSH requirement; the NPSH available is determined from the elevation head and vapor pressure of the water in the refueling water storage tank and the pressure drop in the suction piping from the tank to the pumps. At the end of the injection phase, 20-percent NPSH margin is available assuming all three pumps running together with two residual heat removal pumps at run-out condition.

6.2.3.9.3 Recirculation Pumps

The NPSH for the recirculation pumps is evaluated for recirculation operation. The NPSH available is determined from the elevation head of the water above the pump NPSH reference line (eye of the 1st stage impeller) in the sump. The containment water level is confirmed to be above the minimum level required for NPSH, prior to starting the recirculation pumps during the changeover from the injection phase to the recirculation phase. The RWST level is confirmed to be less than 2 feet prior to stopping the remaining operating containment spray pump and establishing simultaneous recirculation flow to the core and the spray headers. This maximizes the available NPSH to the recirculation pumps in this mode of operation.

The internal recirculation pumps are conventional vertical condensate pumps and are of double suction design, requiring less NPSH. At the initiation of the recirculation phase, adequate NPSH margin is available with one or both pumps operating at the pump design flow. When simultaneous recirculation flow to the core and spray headers is established, the available NPSH margin will be adequate at expected pump flows. Should the pump reach the runout flow, available NPSH margin will be adequate and no cavitating conditions are anticipated.

6.2.4 Minimum Operating Conditions

The Technical Specifications establish limiting conditions regarding the operability of the system when the reactor is critical.

6.2.5 Inspections and Tests

6.2.5.1 Inspection

All components of the safety injection system are inspected periodically to demonstrate system readiness.

The pressure-containing components are inspected for leaks from pump seals, valve packing, flanged joints, and safety valves during system testing.

Current requirements for safety injection system surveillance are discussed in Sections 3.5 and 5.5.6 of the facility Technical Specifications and in UFSAR Section 1.12, "Inservice Inspection and Testing Programs".

6.2.5.2 Preoperational Testing

6.2.5.2.1 Component Testing

Preoperational performance tests of the components were performed in the manufacturer's shop. The pressure-containing parts of the pump were hydrostatically tested in accordance with Paragraph UG-99 of Section VIII of the ASME Code. Each pump was given a complete shop performance test in accordance with Hydraulic Institute Standards. The pumps were run at design flow and head, shutoff head, and at additional points to verify performance characteristics. Net positive suction head was established at design by means of adjusting suction pressure for a representative pump. This test was witnessed by qualified Westinghouse personnel.

The remote-operated valves in the safety injection system are motor-operated. Shop tests for each valve included a hydrostatic pressure test, leakage tests, a check of opening and closing time, and verification of torque switch and limit switch settings. The ability of the motor operator to move the valve with the design differential pressure across the gate was demonstrated by opening the valve with an appropriate hydrostatic pressure on one side of the valve.

The recirculation piping and accumulators were initially hydrostatically tested at 150-percent of design pressure.

The service water and component cooling water pumps were tested prior to initial operation.

6.2.5.2.2 System Testing

An initial functional test of the core cooling portion of the safety injection system was conducted during the hot-functional testing of the reactor coolant system before initial plant startup. The purpose of the initial systems test was to demonstrate the proper functioning of instrumentation and actuation circuits and to evaluate the dynamics of placing the system in operation. This test was performed following the flushing and hydrostatic testing of the system.

The functional test was performed with the water level below the safety injection setpoint in the pressurizer and with the reactor coolant system initially cold and at low pressure. The safety injection system valving was set initially to simulate the system alignment for plant power operation.

To initiate the test, the safety injection block switch was moved to the unblock position to provide control power allowing the automatic actuation of the safety injection relays from low-pressure signals from the pressurizer instrumentation. Simultaneously, the breakers supplying outside power to the 480-V buses were tripped manually and operation of the emergency diesel system automatically commenced. The high-head safety injection pumps and the residual heat removal pumps were started automatically following the prescribed diesel loading sequence. The valves were operated automatically to align the flow path for injection into the reactor coolant system.

The rising water level in the pressurizer provided indication of system delivery. Flow into the reactor coolant system terminated with the filling of the pressurizer, and the operation of the safety injection systems was terminated manually in the control room.

This functional test provided information to confirm valve operating times, pump motor starting times, the proper automatic sequencing of load addition to the emergency diesels, and delivery rates of injection water to the reactor coolant system.

The functional test was repeated for the various modes of operation needed to demonstrate performance at partial effectiveness, that is, to demonstrate the proper loading sequence with two of the three emergency diesels and to demonstrate the correct automatic starting of a second pump should the first pump fail to respond. These latter cases were performed without delivery of water to the reactor coolant system, but included starting of all pumping equipment involved in each test.

The systems were accepted only after the demonstration of proper actuation and after the demonstration of flow delivery and shutoff head within design requirements.

Flow was introduced into the reactor coolant loops through the accumulator discharge line to demonstrate the operability of the check valves and remotely actuated stop valve, and to confirm length to diameter (L/D) ratios of accumulator discharge lines used in the calculation.

6.2.5.3 Postoperational Testing

6.2.5.3.1 Component Testing

Routine periodic testing of the safety injection system components and all necessary support systems at power is done. No inflow to the reactor coolant system will occur whenever the reactor coolant pressure is above the safety injection pump shutoff head. If such testing indicates a need for corrective maintenance, the redundancy of equipment in these systems permits such maintenance to be performed without shutting down or reducing load under conditions defined in the Technical Specifications. These conditions include such matters as the period within which the component is to be restored to service and the capability of the remaining equipment to meet safety limits within such a period.

Test Circuits are provided to examine periodically the leakage back through the check valves and to ascertain that these valves seat whenever the reactor system pressure is increased. The recirculation pumps are normally in a dry sump. These pumps can only be started and allowed to reach full speed with the plant at cold shutdown. Minimum flow testing of these pumps is performed during refueling operations by filling the recirculation sump and directing the flow back to the sump through the valve on the discharge of the pump. The service water and component cooling pumps not running during normal operation may be tested by alternating with the operating pumps.

The contents of the accumulators and the refueling water storage tank are sampled periodically to determine the boron concentration.

6.2.5.3.2 System Testing

System testing is conducted during plant shutdown to demonstrate proper automatic operation of the safety injection system. A test signal is applied to initiate automatic action and verification made that the safety injection and residual heat removal pumps receive start signals. The test demonstrates the operation of the valves, pump circuit breakers and automatic circuitry. Isolation valves in the injection line will be blocked closed so that flow is not introduced into the reactor coolant system. The test is considered satisfactory if control board indication and visual observations indicate all components have operated and sequenced properly.

The safety injection piping up to the final isolation valve is maintained full of borated water at refueling water concentration while the plant is in operation. The safety injection pumps recirculate refueling water through the injection lines via a small test line provided for this purpose.

Flow in each of the safety injection headers and in the main flow line for the residual heat removal pumps is monitored by a local flow indicator. Pressure instrumentation is also provided for the main flow paths of the safety injection and residual heat removal pumps. Accumulator isolation valves are blocked closed for this test.

The high pressure safety injection pumps are run and the variable orifices and injection line valves are adjusted to balance flowrates within the specified range.

The eight-switch sequence for recirculation operation is tested to demonstrate proper sequencing of valves and pumps. The recirculation pumps are blocked from starting during this test.

The external recirculation flow paths are hydrotested during periodic retests at the operating pressures. This is accomplished by running each pump, which could be used during external recirculation (safety injection and residual heat removal pumps) and checking the discharge and recirculation test lines. The suction lines are tested by running the residual heat removal pumps and opening the flow path to the safety injection pumps in the same manner as described above.

During the above test, all system joints, valve packings, pump seals, leakoff connections, or other potential points of leakage can be visually examined. Valve gland packing, pump seals, and flanges are adjusted or replaced as required to reduce the leakage to acceptable proportions. For power-operated valves, final packing adjustments are made, and the valves are put through an operating cycle before a final leakage examination is made.

The entire recirculation loop, except the recirculation line to the residual heat removal pumps, is pressurized during periodic testing of the engineered safety features components. The recirculation line to the residual heat removal pump is capable of being hydrotested during plant shutdown, and it is also leak-tested at the time of the periodic retests of the containment.

REFERENCES FOR SECTION 6.2

1. M. J. Bell, et al., Investigations of Chemical Additives for Reactor Containment Spray Systems, WCAP-7153, Westinghouse Electric Corporation, March 1968.

2. Deleted

TABLE 6.2-1
Safety Injection System – Code Requirements

Component	Code
Refueling Water Storage Tank	AWWA D100-65
Residual Heat Exchanger	
Tube Side	ASME Section III Class C
Shell Side	ASME Section VIII
Accumulators	ASME Section III Class C
Valves	USAS B16.5 (1955)
Piping	USAS B31.1 (1955)

TABLE 6.2-2 (Sheet 1 of 2)
Instrumentation Readouts On The Control Board
For Operator Monitoring During Recirculation

Valves

System	Valve No.
SIS	MOV 1802 A, B
SIS	MOV 885 A, B
SIS	MOV 889 A, B
SIS	MOV 888 A,B
SIS	MOV 866 A, B, C, D
SIS	MOV 851 A, B
SIS	MOV 856 A, B, C, D,E,F
SIS	MOV 882
SIS	MOV 842
SIS	MOV 843
ACS	MOV 744
ACS	MOV 745 A,B
ACS	MOV 746
ACS	MOV 747
ACS	MOV 1810
ACS	HCV 638
ACS	HCV 640

Instruments

System	Channel No
SIS	FI 945 A, B
SIS	FI 946 A, B, C,D
SIS	FI 924
SIS	FI 925
SIS	FI 926
SIS	FI 927
SIS	LI 938
SIS	LI 939
SIS	LI 940
SIS	LI 941
SIS	LT 3300
SIS	LT 3301
SIS	LT 3302
[Deleted]	
SIS	LT 3304
SIS	PI 922
SIS	PI 923
SIS	PI 947
ACS	FI 640
ACS	LI 628

ACS	TR 636
RCS	LI 459

TABLE 6.2-2 (Sheet 2 of 2)
Instrumentation Readouts On The Control Board
For Operator Monitoring During Recirculation

Instruments (continued)

System	Channel No.
RCS	LI 460
RCS	LI 461
RCS	LI 462

Pumps

System	Pumps
SIS	Safety Injection
SWS	Service Water
ACS	Component Cooling
CS	Containment Spray
RS	Recirculation
ACS	Residual Heat Removal

Key:

ACS - Auxiliary Coolant System
CS - Containment Spray System
RCS - Reactor Coolant System
RS - Recirculation
SIS - Safety Injection System
SWS - Service Water System

TABLE 6.2-3 (Sheet 1 of 3)
Quality Standards Of Safety Injection System Components

Residual Heat Exchanger

A. Tests and inspections
   1. Hydrostatic test
   2. Radiograph of longitudinal and girth welds (tube side only)
   3. Ultrasonic testing of tubing or eddy current tests
   4. Dye penetrant test of welds
   5. Dye penetrant test of tube to tube sheet welds
   6. Gas leak test of tube to tube sheet welds before hydro and expanding of tubes

B. Special manufacturing process control
   1. Tube to tube sheet weld qualifications procedure
   2. Welding and NDT and procedure review
   3. Surveillance of supplier quality control and product

Component Cooling Heat Exchanger

A. Tests and inspections
   1. Hydrostatic Test
   2. Dye penetrant test of welds

B. Special Manufacturing Process Control
   1. Welding and NDT and procedure review
   2. Surveillance of supplier quality control and product

Safety Injection, Recirculation, and Residual Heat Removal Pumps

A. Test and inspections
   1 Performance test
   2 Dye penetrant of pressure retaining parts[1]
   3. Hydrostatic test

B. Special manufacturing process control
   1. Weld, NDT, and inspection procedures for review
   2. Surveillance of suppliers quality control system and product

Accumulators

A. Tests and inspection
   1. Hydrostatic test
   2. Radiography of longitudinal and girth welds
   3. Dye penetrant/magnetic particle of weld

B. Special manufacturing process control
   1. Weld, fabrication, NDT, and inspection procedure review
   2. Surveillance of suppliers quality control and product

TABLE 6.2-3 (Sheet 2 of 3)
Quality Standards Of Safety Injection System Components

Valves

A. Tests and inspections
1. 200 psi and 200°F or below (cast or bar stock)
   a. Dye penetrant test
   b. Hydrostatic test
   c. Seat leakage test
2. Above 200 psi and 200°F
a. Forged valves (2-1/2-in. and larger)
   (1) Ultrasonic tests of billet prior to forging
   (2) Dye penetrant 100-percent of accessible areas after forging
   (3) Hydrostatic test
   (4) Seat leakage test

   b. Case valves
   (1) Radiograph 100-percent [2]
   (2) Dye penetrant all accessible areas [2]
   (3) Hydrostatic test
   (4) Seat leakage

3. Functional tests required for:
   a. Motor operated valves
   b. Auxiliary relief valves

B. Special manufacturing process control
1. Weld, NDT, performance testing, assembly and inspection procedure review
2. Surveillance of suppliers quality control and product
3. Special weld process procedure qualification (e.g., hard facing)

Piping

A. Tests and inspections
Class 1501 and below
Seamless or welded. If welded 100-percent radiography is required, shop-fabricated and field-fabricated pipe weld joints are inspected as follows:

2501R – 610R: 100-percent radiographic inspection and penetrant examination
301R – 302R: 20-percent random radiographic inspection
151R – 152R: 100-percent liquid penetrant examination

B. Special manufacturing process control
Surveillance of suppliers quality control and product

Refueling Water Storage Tank

A. Tests and inspections
1. Vacuum box test of tank bottom seams