3.8 DESIGN OF SEISMIC CATEGORY I STRUCTURES

3.8.1 Concrete Containment

3.8.1.1 Description of the Containment

3.8.1.1.1 General Description

The design, analysis, and construction of the containment structure is similar to, and takes full advantage of Stone & Webster Engineering Corporation's (SWEC'S) experience in, the designs for the following plants:

1. Connecticut Yankee Atomic Power Company's Nuclear Power Plant - Unit No. 1 (Docket No. 50-213).

2. Virginia Electric and Power Company's Surry Power Station - Units 1 and 2 (Docket No. 50-280 and 50-281).

3. Maine Yankee Atomic Power Company's Maine Yankee Atomic Power Station - (Docket No. 50-309).

4. Beaver Valley Power Station - Unit 1 (BVPS-1) (Docket No. 50-334).

5. Virginia Electric and Power Company's North Anna Power Station - Units 1 and 2 (Docket No. 50-338 and 50-339).

The containment structure is a heavily reinforced concrete, steel-lined vessel with a flat base mat, cylindrical walls, and a hemispherical dome. The arrangement of the containment structure is shown on Figures 3.8-1, 3.8-2, 3.8-3, 3.8-4, 3.8-5, 3.8-6 and 3.8-7. The containment is not structurally integral with any of the structures surrounding it (Figure 3.8-8). A "shake space" is provided between the containment and the adjacent structures to accommodate relative structural movement.

The base mat is a soil-bearing concrete slab 10 feet thick, without projections below its lower surface. A 4-inch thick (minimum) layer of porous concrete sub-base underlies the mat and consists of coarse aggregate bound with a water-cement paste. This concrete was made by omitting the fine aggregate from a standard concrete mix. The mix was designed to have a 28-day compressive strength of 1,000 psi minimum. The results of previous laboratory tests have indicated that this concrete provides adequate drainage for the emergency seepage removal system described in the following paragraph.

The exterior surface of the concrete shell and foundation mat, shown on Figure 3.8-9, has a continuous waterproofing membrane to protect the containment structure against water seepage during flood stages resulting from the probable maximum flood described in Section 2.4.3. As a supplementary safety factor, water relief systems are provided

in the floor of the two instrument pits at el 690 feet-11 inches. The pits are located in the mat outside of the containment wall. A sump extends into the mat from the bottom of the pit to a point above the bottom reinforcement. From the bottom of the sump, a vertical 6-inch pipe projects through the reinforcement into the underlying porous concrete. In the event of a flood and unexpected leakage through the membrane, the vertical pipe would allow the water to rise in the sump where it would sound an alarm in the control room after reaching a predetermined height. The water would then be removed by a sump pump to prevent buildup of pressure under and behind the steel liner. The instrument pits are enclosed by the waterproofing membrane protecting the containment structure.

The inside diameter (ID) of the containment cylinder is 126 feet, and the cylinder wall is 4 feet-6 inches thick.

The distance from the top of the mat to the inside of the dome crown is approximately 185 feet. The dome has a thickness of 2 feet-6 inches and an inside radius of 63 feet.

The inside faces of the containment wall, dome, and mat are lined with steel liner plates which act as a leaktight membrane. The liner plate for the walls is 3/8-inch thick and for the dome is 1/2-inch thick. A 1/4-inch plate is used over the base mat.

The containment is provided with a personnel air lock whose ID is 7.0 feet, and an equipment hatch whose ID is 14 feet-6 inches, and contains a 5 feet-0 inch ID x 12 feet-6 inch long emergency air lock. Other penetrations consist of hot and cold process pipes, the fuel transfer tube, and electrical penetrations.

3.8.1.1.2 Reinforcing Steel Arrangement

The foundation mat of the containment structure is strengthened with top and bottom layers of reinforcing, as shown on Figure 3.8-10. Bottom mat reinforcing is placed in a rectangular grid pattern with layers at 90 degrees to each other. Reinforcing for the top of the mat consists of concentric circular bars combined with radial bars. The reinforcing pattern for the top of the mat is arranged to permit a uniform spacing of the vertical wall rebars which extend into the mat. Splices in adjacent parallel rebars in the mat are staggered 4 feet wherever possible.

Hoop tension in the cylinder is resisted by horizontal bars located near the outer and inner surfaces of the wall. All horizontal circumferential bars, including those in the dome, have their splices staggered 3 feet in both circumferential and meridional directions wherever possible.

Longitudinal or meridional tension in the cylinder wall is resisted by rows of vertical bars placed near the interior and exterior faces of the wall. Vertical bars are placed in groups of not more than

20 bars of equal length. These groups are arranged so that no adjacent group in the same or opposite face of the wall have splices closer than 6 feet apart vertically wherever possible. At the temporary construction opening, both the vertical and horizontal reinforcement splices are staggered a minimum of 12 inches.

When the containment is pressurized, radial shear in lower elevations of the containment walls exceeds the capacity of unreinforced concrete. To resist the large radial shear near the base of the wall, flat steel bars, inclined at approximately 45 degrees with the horizontal, are welded to the vertical reinforcing as shown on Figure 3.8-10. A report on tests of concrete specimens containing these shear assemblies was presented to the U.S. Atomic Energy Commission during February 1970 (SWEC 1969). The results proved the adequacy of the design. The welded flat bars are terminated at a level above the mat where the radial shear load is reduced. Above this level, the radial shear is resisted by Z-type reinforcing steel, where required. In the lower portion of the containment wall, deformed studs welded to the liner resist splitting of the concrete in the plane of the vertical rebar as shown on Figure 3.8-10.

Tangential shear (Vu) resulting from the earthquake loading is resisted by the concrete and diagonal reinforcing bars. A maximum allowable tangential shear stress (Vc) is assigned to the concrete (Section 3.8.1.4.1). Stresses in excess of Vc are resisted by inclined reinforcing bars anchored in the mat. The spacing between these diagonal bars is increased as the design tangential shear decreases at higher wall elevations. No diagonal rebars are required above the spring line.

The dome reinforcing consists of horizontal layers of circumferential hoop bars and layers of meridional rebar extending from the vertical reinforcing of the cylindrical wall. Layers are located near both the inner and outer faces of the concrete. The radial pattern of the meridional reinforcing steel terminating in the containment dome results in a high degree of redundancy of reinforcing steel in the dome. Bars are terminated beyond a point where there is more than twice the amount of steel required for design purposes. The rate of convergence of these bars and the low stress requirements, dictated by the arrangement, produce a relatively low bond stress. In a limited number of cases where bars are terminated close to the apex of the dome, anchorage stresses are more critical. These bars are hooked or mechanically secured to provide the required anchorage. Near the crown, the meridional rebars are welded to a concentric ring cast in the concrete as shown on Figure 3.8-11.

A typical detail of the reinforcing steel at the junction of the containment mat and cylinder is shown on Figure 3.8-10, and Figure 3.8-12 shows a typical detail of the junction of the cylinder and the dome.

Section 3.8.1.4 describes the reinforcing steel arrangement at the equipment hatch and the personnel air lock openings as shown on Figures 3.8-13 and 3.8-14.

Minimum concrete cover for all principal reinforcing steel of the containment structure equals or exceeds the requirements of ACI 318-71.

Anchorage for terminated bars is developed either by bond or by means of a mechanical anchorage. In the case of anchorage by bond in a biaxial tension zone, the development length of the bars is increased 25 percent above that required by ACI 318-71 for reinforcement terminating in a tension zone.

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

3.8.1.1.3 Containment Liner, Penetrations, and Access Openings

The containment liner consists of a vertical cylindrical portion capped by a hemispherical dome and closed at the bottom by a mat liner portion. The liner pressure boundary includes embedments, insert plates, and penetrations.

The liner plate acts as a leaktight membrane under conditions that can be encountered throughout the operating life of the plant. The liner plate is anchored to the concrete containment at sufficiently close intervals so that the overall deformation of the liner is essentially the same as that of the concrete containment.

All welded seams in the mat, cylindrical liner wall, hemispherical dome, and liner penetrations are covered with continuously welded test channels in a manner similar to those installed at BVPS-1 (Docket No. 50-334). These channels are zoned into test areas by dams welded to the ends of the sections of the channels. The channels are used to check tightness of welds during liner erection. Test ports are provided for each zone of the leak chase channels. After testing during erection, plugs are installed in the test fittings. Plugs remain in place during subsequent leak-rate testing. Test channel plugs are 1/8-inch NPT pipe plugs with socket hex heads. The plugs incorporate tapered pipe threads which are self-locking. Unlike set screws with untapered machine threads, these plugs require no significant torque to assure that they remain in place. Hence, there are no procedure requirements for torquing or tightening channel test plugs.

For channels mounted inside the containment structure (that is, those on the floor mat, penetration and cylinder wall), plugs are accessible throughout the life of the plant. For channels mounted on

the exterior of the dome, plugs are covered by concrete after testing. As added assurance that plugs will not be accidentally disturbed during concrete placement, dome plugs are seal welded in place after testing. Should access to dome test channel zones be required during the operating life of the plant, holes could be drilled through the dome liner. These holes would then be tapped and provided with a type NPT plug after testing.

Surfaces of carbon steel components of the liner, penetrations, and access openings which are exposed to the atmosphere inside or outside of the containment are painted. The coating is comprised of a prime coat of inorganic zinc primer applied in the fabricator's shop and touched up in the field after erection. Finish paint is an epoxy coating system.

Unpainted surfaces covered by concrete are located such that they are exposed to a limited oxygen supply which will only support insignificant corrosion. No need exists for a corrosion allowance.

The liner plate consists of the following:

1. Cylindrical Portion and Dome

   The cylindrical portion of the liner is a vertical circular cylinder attached to the foundation mat at its base. The top of the cylindrical portion is closed by a hemispherical dome. The liner dimensions are given in Section 3.8.1.1.1.

   The 3/8-inch-thick liner served as the internal form for the cylindrical portion of the concrete containment during construction. All liner seams are double-butt welded, except for the lower 30 feet of the cylindrical shell liner or where joint access was limited to one side; these liner plates were welded using a backing plate, as shown on Figure 3.8-15.

   The 1/2-inch-thick hemispherical steel dome liner plate served as an internal form for the containment reinforced concrete dome during construction. All seams in the liner dome were double-butt welded as shown on Figure 3.8-15.

   The wall-to-dome liner junction is a double-butt welded joint, as shown on Figure 3.8-15.

   The bottom-to-wall liner plate junction knuckle joint is made of 3/8-inch-thick plates. The end of the knuckle plate in contact with the mat is attached to steel bridging plates, as shown on Figure 3.8-16,

   The knuckle joint (Figure 3.8-16), including the 7/8-inch-thick skirt, was designed as a unit for the same conditions as the rest of the liner. The skirt and the

surrounding concrete isolate the knuckle-shaped section of the liner from a large part of the loads due to temperature and pressure.

2. Mat Liner Plate

The 1/4-inch thick bottom liner plates were assembled in place and were continuously welded at their periphery to steel bridging plates which were cast in the reinforced concrete base mat.

Except at the in-core instrumentation area and the sump areas, the bottom liner plate is overlayed with an approximately 2-foot-thick reinforced concrete slab that is anchored to the bottom concrete mat (Figure 3.8-17). This 2-foot-thick slab provides anchorage and support for equipment located in the base of the containment structure. It will insulate the mat liner from temperature effects and prevents damage to the liner from internal missiles.

The 3/4-inch-thick liners in the instrumentation and sump areas are anchored to the containment foundation using 5/8-inch diameter by 20-inch long deformed anchors or 5/8-inch diameter by 6 9/16-inch long headed concrete anchor studs.

3. Embedments

In areas where the transfer of loads through the floor liner plate is required, bridging bars with a rectangular cross section are used. As shown on Figures 3.8-18 and 3.8-19, internal structures are anchored firmly to the concrete base mat by lengths of 4 1/2 by 6-inch-steel bridging bars, which are placed horizontally and extend through the bottom plate liner. The main reinforcing bars are welded to the bottom faces and joined by cadwelds to the top faces of the bridging bars, thus providing continuity of reinforcing through the bottom liner.

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

4. Insert and Overlay Plates

The loads derived from support of piping, platforms, or other miscellaneous equipment are transferred to the containment concrete wall through insert plates and their anchors. Sufficient anchorage is provided such that the liner plate adjacent to insert plates is isolated from loads

applied to brackets or attachments, and leaktight integrity is maintained.

Overlay plates are welded directly to the liner plate for the attachment of supports for small (under 3 inches in diameter) piping, conduit, and cable trays.

5. Anchors

The steel containment liner is anchored to the concrete wall and dome with concrete anchors. The shell anchorage attachments in the lowest 16 feet of the liner wall cylinder are deformed anchor bars 5/8-inch in diameter and 3 feet-10 inches long. The remainder of the cylinder and the dome are anchored by 5/8-inch diameter headed anchor studs, 6 9/16 inches long. Tests conducted for one stud manufacturer indicate that, with the manufacturer's recommended depth of embedment of the stud in concrete, the ultimate strength of the stud material can be developed in direct tension. These tests also show that shear failure occurs in the stud adjacent to the weld connecting the stud to the plate (TRW 1975).

3.8.1.1.3.1 Penetrations

Penetrations are used to carry piping, mechanical systems, and electrical services through the containment wall. Isolation systems for piping which penetrates the containment are described in Section 6.2.

Containment penetrations are anchored to and transfer loads to the reinforced concrete containment wall. These penetrations can be classified as follows:

1. Piping System Penetrations

All containment piping penetrations consist of basic containment inserts, each of which consists of a pipe sleeve, approximately 6 feet long with heavy reinforcing plates near both ends, plus additional items as required for the individual services. All containment piping penetration inserts are anchored in the reinforced concrete containment wall so that loads can be transferred from the piping to the reinforced concrete.

For cold penetrations, the piping is welded to a reinforcing plate which is anchored to the containment concrete wall so the loads can be transferred from the piping to the concrete wall, as shown on Figure 3.8-20.

Each penetration insert carrying hot piping (fluid continually over 200°F) is equipped with a water-cooled

cooling unit located on the inside of the penetration encompassing that length of the sleeve which is covered by concrete. Cooling water is supplied by the component cooling water system described in Section 9.2.2.1.

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

The cooling water system, as described in Section 9.2.2, has inlet and outlet lines located on the atmospheric end of the cooler which circulates water through the unit. The cooling water circulation pipes do not require any secondary penetration of the containment structure.

The cooling water system limits the radial heat flow, resulting from convection and radiation from the hot pipe penetration, thereby maintaining the temperature of the concrete in contact with the sleeve within allowable limits. The system also controls the longitudinal heat flow, resulting from conduction from the same heat source, thereby maintaining the temperature of the liner and the temperature gradient along the sleeve within allowable limits.

For hot penetrations, transition from the process pipe to the sleeve is by an integral forging, as shown on Figure 3.8-20.

For multiple penetrations per single sleeve, a forged head links the sleeve with the piping, as shown on Figure 3.8-20. The sleeve is welded to a reinforcing plate which is anchored to the containment concrete wall so the loads can be transferred from the piping to the wall through the forged head and sleeve.

2. Mechanical System Penetrations - Fuel Transfer Tube Enclosure

A fuel transfer tube penetration is provided for the transferring of fuel between the refueling canal in the containment structure and the spent fuel pool in the fuel building. The penetration consists of a stainless steel pipe installed inside an enclosure. As shown on Figure 3.8-24, the inner pipe acts as the fuel transfer tube and connects the refueling canal with the fuel pool. The enclosure is welded to the containment liner and provision is made, by use of a leak chase ring, for leak testing of all welds essential to the integrity of the penetration. Bellows expansion joints are provided on the enclosure to compensate for any differential movement between the two buildings. The enclosure consists of the three sets of bellows, plus the connecting sleeves.

There are two main requirements for the fuel transfer tube enclosure:

a. The bellows must accommodate the maximum deflections, including rotation and offset, between the spent fuel pool and the containment refueling canal.

b. The end of the enclosure inside the containment must withstand pressures and temperatures of the test and loss-of-coolant accident (LOCA) conditions.

The bellows expansion joints were selected to accommodate deflections caused by thermal expansion, seismic motions, and radial movement of the containment wall due to peak internal pressure and temperature.

3. Electrical Service Penetrations

Electrical conductors penetrating the containment structure range in size from No. 16 AWG thermocouple leads to 750 MCM copper cables for 4kV power circuits. The sleeves are welded into the containment liner reinforcement plate with a test channel around the seal weld, as shown on Figure 3.8-21 for periodic leak testing.

The basic electrical penetration is installed in a 12- or 18-inch steel pipe. The header plate of the penetration or the unitized penetration assembly has been welded to this pipe, and a test channel surrounds the weld. A permanent system of pressure monitoring for each electrical penetration has been installed. This permits either continuous or periodic tests for leaktightness.

3.8.1.1.3.2 Access Openings

The containment has the following access openings:

1. Equipment Hatch

The equipment hatch is a single closure penetration approximately 8 feet-4 inches in length with an ID of 14 feet-6 inches. The equipment hatch cover is mounted inside the containment structure and is provided with a hoist with two point suspension and a sliding rail for storage. A positive locking device is furnished to prevent circular swing. The cover is double-gasketed with a leakage test tap between the O rings.

The emergency air lock is a subassembly of the equipment hatch consisting of a double-closure removable penetration 12 feet-6 inches long and 5 feet in diameter attached to the removable equipment hatch cover by a bolted

flanged connection with double O rings. A leakage test tap is located between the "O" rings. A 30-inch-diameter opening is located at each end of the lock for personnel access. The doors swing toward the center of the containment vessel and a viewport approximately 4 inches in diameter is provided at each end of the lock. The doors are mechanically interlocked so that one door cannot be operated unless the other is sealed. However, provisions exist for deliberately violating the interlock by use of special tools or keys. An electromechanical interlock is provided to prevent operation of a door when a pressure differential greater than 0.5 psi exists across the bulkhead.

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

Each door is equipped with a valve for equalizing the pressure across the door. At no time will the equalizing valves on both doors be open at the same time and in no case shall an equalizing valve be open on one door while the other door is operating. The equalizing valves are capable of equalizing pressure differential in the emergency air lock within 2.5 minutes (maximum).

The air lock will be operated manually. This includes the operation of the mechanism and the swinging of the doors. In addition to normal operation, it is possible to operate each door from a remote location by manual means only. The inside door is operable from outside the containment structure, and the outside door is operable from inside the containment structure.

Both doors of the air lock are designed to withstand the containment test pressure of 52 psig. Also, each door is designed to withstand 8 psia pressure within the containment structure with full atmosphere on the other side.

The interior door is provided with an additional securing device to withstand the maximum test pressure inside the air lock when the containment structure is at 8 psia.

All shafts penetrating the door or bulkhead have double packing and a test connection to permit periodic leak testing between the seals.

Three pressure gages are provided: two penetrate the bulkhead at the reactor end of the air lock and measure containment vessel pressure and interior air lock pressure,

and one penetrates the bulkhead at the opposite end of the air lock and measures interior air lock pressure.

Test connections are provided for periodic leak testing between the double seals on each door.

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

The emergency air lock will be unbolted from the equipment hatch and secured on a cart for removal before the equipment hatch is used.

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

2. Personnel Air Lock

The personnel air lock is a double-closure penetration, 7 feet wide (ID) and 15 feet long. Each closure is flanged and double-gasketed with a leakage test tap between the O rings. The enclosed space between the O rings is pressurized to containment design pressure to test for leakage through the access door when it is locked in place. The entire personnel air lock can be independently pressurized to containment design pressure for testing.

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

Each door is furnished with a pressure-equalizing, manually-operated valve installed on each side of both doors, which allows equalizing at an adjustable rate by the person entering or leaving the air lock. These valves are capable of equalizing the pressure in the personnel air lock within 5 minutes. As a safety device, a normally-open differential pressure switch, which prevents the opening of the door latches before 0.5 psi is obtained, is installed for each of the equalizing valves.

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

Also provided are two electrical penetration connections, a 3-inch capped emergency air connection, a walkway, and two folding waiting benches inside the air lock. Each electrical penetration has at least six spare terminals for telephone and other circuits.

3.8.1.2 Applicable Codes, Standards, and Specifications

Structural design, materials, materials quality control, fabrication, construction testing, and in-service inspection, where applicable, conform to the following codes, standards, and specifications unless otherwise stated.

3.8.1.2.1 General

3.8.1.2.1.1 American Concrete Institute

ACI codes, standards, and specifications are presented in Table 3.8-1.

An exception is taken to the pressure tests called for in ACI 318-71, Section 6.3.2.4. The following shall be substituted: pipes which contain liquid, gas, or vapor, and which are embedded in concrete will be pressure-tested to the applicable ASME, ANSI, or Fire Protection and Plumbing Code requirements.

3.8.1.2.1.2 American Society of Mechanical Engineers

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, Nuclear Vessels, 1971 Edition with addenda through Winter 1972, is used as a guide in the selection of materials, fabrication (including welding), nondestructive examination, and inspection of the steel containment liner and mat embedments.

ASME materials specifications for materials used in construction are presented in Table 3.8-2.

Applicability of the design provisions of ASME III is discussed in Section 3.8.1.2.3.

3.8.1.2.1.3 American Institute of Steel Construction

Structural design, materials and fabrication conform to AISC's Specification for the Design, Fabrication, and Erection of Structural

Steel for Buildings (February 12, 1969), with or without including Supplement No. 1 (November 1, 1970) and Supplement No. 2 (December 8, 1971), with the following exception:

An exception is taken to the provision that plate on continuous bar washer at least 5/16 inch thick are required to cover long slots in the outer plies of joints with ASTM A325 bolts, as called for in the Specification for Structural Joints Using ASTM A325 or A490 Bolts, dated April 18, 1972. For these applications, standard circular washers have been demonstrated to be adequate.

3.8.1.2.1.4 American Welding Society

The welding of structural and reinforcing steel meets the requirements of AWS D1.1-1972, 1973, 1975, or 1980, "Structural Welding Code," and AWS D12.1-61, "Recommended Practice for Welding Reinforcing Steel, Metal Inserts, and Connections in Reinforced Concrete Construction."

3.8.1.2.1.5 Occupational Safety and Health Standards

Applicable Safety and Health Standards are in conformance with U.S. Department of Labor, Occupational Safety and Health Administration, October 18, 1972, and the proposed Walking-Working Surfaces, September 6, 1973.

3.8.1.2.1.6 American National Standards Institute

Applicable ANSI codes, standards, and specifications are given in Table 3.8-3.

Exceptions to ANSI N45.2.5-1974 are discussed in Section 1.8, Table 1.8-1, under Regulatory Guide 1.94, Rev. 1.

3.8.1.2.1.7 American Society for Testing and Materials

Applicable ASTM materials specifications are presented in Table 3.8-4. Subsequent revisions to the standards and specifications listed are considered acceptable.

3.8.1.2.1.8 Handbook for Concrete and Cement, Corps of Engineers, U.S. Army

Specifications for cement and water stops are provided in CRD C 119, "Method of Test for Flat and Elongated Particles in Coarse Aggregate

CRD," C513, "Specification for Rubber Water Stop," and CRD C 572, "Specification for Polyvinyl Chloride Water Stop."

3.8.1.2.1.9 U.S. Nuclear Regulatory Commission Regulatory Guides

U.S. Regulatory Commission (USNRC) Regulatory Guides, listed in Table 3.8-5, are applicable to the extent described in Section 1.8.

3.8.1.2.2 Structural Specifications

All Category I structural specifications are written to comply with the applicable codes, standards, specifications, and procedures given in this Final Safety Analysis Report.

Table 3.8-6 is a summary of the principal plant structural specifications that are prepared for Seismic Category I materials.

3.8.1.2.3 Steel Liner, Penetrations, and Access Openings

Steel liner, penetrations, and access openings are in conformance with the codes presented in Table 3.8-7 to the extent discussed in this section, and further clarified in the ASME Code Baseline Document.

The specification for the steel liner, penetrations, and access openings is written to stipulate the extent of applicability of the codes, standards, specifications, and procedures defined herein. Table 3.8-8 is a summary of the materials called for in this specification.

The personnel air lock and the emergency air lock are designed, fabricated, and stamped to ASME III, Class MC.

The equipment hatch is designed and fabricated to ASME III. The equipment hatch is not code stamped as it cannot be pressure tested as a pressure vessel (that is, 45 psig internal pressure), but is tested during the final structural acceptance test of the concrete primary reactor containment.

The fuel transfer tube bellows assemblies are designed, fabricated, and code stamped ASME III, Class MC, as addressed by Code Case 1330-3 (special ruling), Special Equipment Requirement Section III, and Code Case 1177.

To provide additional assurance of the bellows design, duplicate bellows assemblies were tested in accordance with ASME III, Winter 1974 Addendum, paragraph NE-3365.2(e)(2), which was later modified as follows:

"The 15 percent maximum convolution pitch in accordance with paragraph NE-3365.2(c) for unreinforced bellows may be exceeded provided the bellows remain within the elastic range."

These requirements provide assurance that the bellows assemblies will function as designed for the life of the containment liner.

The process piping penetrations are designed, fabricated, and code stamped ASME III, Class 2 NPT. This also complies with the process piping system of which these are a part.

The penetrations are analyzed to Class 2 with further analysis to the more stringent Class MC requirements since MC is invoked as a guide.

The electrical penetrations through the liner are only sleeves which have electrical canisters welded on the outer end. Thus, the sleeves form a part of the containment boundary extension to provide installation of the electrical canisters. The electrical canisters are designed and fabricated to IEEE Standard 317-1976, which invokes ASME III Class MC. ASME III Section NE-3700 states "The rules for design of electrical and mechanical penetration assemblies shall be the same as for vessels (see NE-3300) except that the design and the materials performing electrical conducting and insulating functions of electrical penetration assemblies need not meet the requirements of this code." Thus, the electrical penetrations are designed and fabricated using ASME III Class MC as a guide, which is compatible with the remainder of the containment liner.

Thus, those parts of the liner to which Class MC directly applies are designed, fabricated, and stamped Class MC. Those to which all provisions of the code cannot be applied, for the reasons described, are not stamped. The resulting product complies to the intent of ASME III, (that is, to provide a high quality leaktight containment boundary).

3.8.1.3 Loads and Load Combinations

3.8.1.3.1 Containment Structure Shell and Mat

The containment structure is designed to have ultimate load capacity, as modified by the safety provisions of ACI 318-71, of not less than that required to satisfy the following structural loading criteria:

$$U = 1.0D + 1.5P + 1.0\ (T + TL) \qquad (3.8\text{-}1)$$

$$U = 1.0D + 1.25P + 1.0\ (T + TL) + 1.25E \qquad (3.8\text{-}2)$$

$$U = 1.0D + 1.0P + 1.0\ (T + TL) + 1.0E' \qquad (3.8\text{-}3)$$

$$U = 1.0D + 1.0T + 1.0WT \qquad (3.8\text{-}4)$$

where:

U = Required strength to resist design loads,

D = Dead load of structure and equipment, including efect of earth, hydrostatic forces, ice, and snow loads, when their effect increases the resultant stresses.

P = Pressure load resulting from the design basis accident (DBA)(45 psig).

T = Load due to maximum temperature gradient through the concrete shell and mat for normal operating conditions.

TL = Load exerted by the liner, when it is exposed to the temperature associated with the pressure resulting from a DBA (280°F) (Section 6.2.1), or other significant pipe break event.

E = Loading from the 1/2 SSE (Section 3.7).

E′ = Loading from the SSE (Section 3.7).

WT = Loading due to tornado (Section 3.3).

The four equations represent a conservative combination of loads selected among events, which may occur during the life of the plant. Because the design basis accident (DBA) for the containment structure is a LOCA, no direct interaction with a ruptured pipe appears in Equations 3.8-1 through 3.8-3.

Flooding after a LOCA is negligible.

Design controlled by the listed loading combinations will remain substantially elastic under the service conditions (pre-cracking of the containment shell concrete under structural acceptance test loads is presumed). Structural performance during the structural acceptance test is predicted by analysis using the following load combination:

$$D + L + P_t + T_t \qquad (3.8\text{-}5)$$

where:

$P_t + T_t$ represent pressure and temperature effects expected during the test. Monitoring of the structure during the test is described in Section 3.8.1.7.

The earthquake loadings include consideration of simultaneous excitation from the horizontal and vertical earthquake motions as described in Section 3.7B.3.6.

3.8.1.3.2 Steel Liner

Load combinations which control the design of the concrete structure are not the most severe for the liner components.

The liner plate, anchors, insert plates, overlay plates, and embedments have been designed for the load combinations presented in Table 3.8-9. Because of its specialized function, pressures and temperatures resulting from a LOCA are considered design loads under the provisions of Subsections NE-3100 and NE-3200 of the ASME III Code. The analysis of the post-DBA condition is not necessary since the lowest post-DBA pressure is between atmospheric and normal operating pressure.

During the severe operational condition, the minimum design pressure of the containment can occur. This pressure is the minimum operating air pressure minus the pressure drop due to the maximum containment cool-down situation, which results from accidental operation of the quench spray subsystem (Section 6.2.2). Assuming that the containment air is at 108°F and is suddenly cooled to the quench spray water temperature of 45°F, the vacuum would be increased by about 1.4 psi. This condition results in the minimum possible containment pressure of 8.0 psia, and the minimum temperature of 45°F.

The barometric pressure change due to the maximum hypothetical tornado is 3.0 psia. This atmospheric disturbance causes a decrease in the atmospheric pressure, which decreases the differential between the atmospheric pressure, and the containment ambient pressure. Since this situation decreases the potential for stresses in the containment liner, it is not analyzed.

3.8.1.3.3 Penetrations and Access Openings

Portions of penetrations and access openings which are not backed by concrete are analyzed to the loads and load combinations discussed in Section 3.8.2. Piping penetrations meet the more stringent requirements for their intended piping function (ASME Subsection NC) or the containment function. The containment function is discussed in Section 3.8.2.

3.8.1.4 Design and Analysis Procedures

3.8.1.4.1 Containment Structure

The containment structure consists of a hemispherical dome, a cylindrical shell, and a circular mat supported by an elastic subgrade. The design, analysis, and construction of the containment structure are similar to other plants designed and SWEC, as listed in Section 3.8.1.1.1. The containment structure is analyzed and designed for the load combinations given in Section 3.8.1.3.1.

The analysis of the mat for axisymmetric loading is accomplished using the program MAT 6 which is described in Appendix Section 3A.1.8. This program treats the mat as a symmetrically loaded circular plate on an elastic foundation. The general method is described by Zhemochkin and Sinitzin (1962). The program is set up so that the foundation stiffness can be formulated either through Boussinesq's approach or through a Winkler-type assumption. Zhemochkin's method is modified to account for a finite depth of elastic foundation, that is, the distance between mat and underlying rock. In addition, the cylindrical containment wall, crane wall, and primary shield wall are considered as elastic constraints, which are determined by applying compatibility conditions at the shell mat interface.

For the purpose of calculating the elastic constraint of the containment, the base of the cylinder is assumed to be completely cracked vertically and cracked horizontally to the neutral axis of the transformed section. At this location, the cylinder has a hoop stiffness of the circumferential rebars and the meridional bending stiffness of the transformed section.

A short distance above the mat, the meridional bending moment becomes so small that the entire meridional cross section is in a state of tension. Above this plane, the cylinder is assumed to be completely cracked horizontally and vertically.

Thus, the elastic constraint is determined from a cylinder which is divided into a short shell having the properties of a cross section completely cracked vertically and partially cracked horizontally, and a long shell having the properties of a cross section completely cracked horizontally and vertically.

Seismic analysis of the containment structure described in Section 3.7B.2 provides the dynamic loads imposed on the mat. Since these loads are asymmetric, the mat is analyzed using the finite difference computer program SHELL 1 which is described in Appendix Section 3A.1.2.

The discontinuity forces at the mat-shell junction, calculated as a part of the mat analysis, are used as boundary conditions for the analysis of the cylindrical shell which is performed using the program SHELL 1. The seismic analysis of the containment structure provides the accelerations to which the containment structure would be subjected. These accelerations are applied as static loading to the shell. The tangential shear caused by the seismic loading is resisted by the concrete or by the concrete and a system of diagonal rebars. The maximum allowable tangential shear stress carried by the concrete (Vc) is assumed not greater than 40 psi. This value of Vc was selected on the basis of SWEC nonproprietary reports No. SWND-5, dated November 1969 and submitted to the USAEC on December 3, 1969, and SWND-5S, dated March 1970 and submitted to the USAEC on April 10, 1970. Stresses in excess of Vc are resisted by inclined reinforcing

bars. The specified design compressive strength of the concrete carrying the tangential shear is not less than 3,000 psi with coarse aggregate not smaller than size No. 467, as given in ASTM C33-71. In areas of severe reinforcement congestion, coarse aggregate not smaller than size No. 7 is used.

The diagonal reinforcement required to supplement the tangential shear ability of the concrete consists of No. 14 rebars anchored in the mat and spaced horizontally 2 feet on centers, both ways, at the base of the wall at el 690 feet-11 inches. The spacing between these diagonal bars is increased as the design tangential shear decreases at higher wall elevations. No diagonal rebars are required above the spring line.

The requirements for diagonal reinforcement necessary for carrying the (Vu-Vc) shear force are determined by an analysis based on a paper by Prof. M. J. Holley, Jr. (1969) of MIT. The bar net vertical, horizontal, and diagonal reinforcing is treated as a continuum. This allows the establishment of a strain relationship between horizontal, vertical, and diagonal reinforcement. The total force in the diagonals is taken as the sum of forces determined under a symmetric loading due to internal pressure and vertical earthquake load (Vu=0) and involving only horizontal and vertical strains, plus forces determined under an asymmetric stress distribution due to seismic shear (Vu-Vc) and involving only shear strains.

Two temperature conditions are considered in the analysis of the containment:

1. Temperature under normal operating condition.

2. Temperature associated with the DBA. This is the transient temperature which, when combined with the coincident internal pressure, will produce the most adverse effects on the reactor containment structure.

Under normal operating conditions, the temperature gradient to produce the worst stress resultant is used.

The effect of the liner temperature increase associated with the DBA condition on the concrete containment shell is determined by the following procedure. For this condition, it is assumed that the temperature of the liner increases and the concrete remains at its ambient temperature. Liner expansion is limited by the concrete shell and a pressure develops between the concrete shell and the liner. The equivalent pressure exerted by the liner on the concrete shell is given by the expression:

$$Pe = LH \times hL/RL \tag{3.8-6}$$

where:

Pe = equivalent pressure,

hL = thickness of liner,

RL = radius of liner, and

LH = circumferential liner stress.

At the junction of the mat and shell, it is assumed that the mat prevents radial movement of the shell; therefore, at this location, the circumferential stress in the liner used to determine equivalent pressure is:

LH = alpha X Es X delta T (3.8-7)

where:

Es = Young's modules of the steel liner,

alpha = coefficient of thermal expansion of the liner, and

delta T = change in temperature due to DBA.

A short distance above the mat, where the effect of the mat to shell discontinuity is negligible, the liner and concrete shell expand due to the DBA pressure and temperature of the liner. Free radial displacement of the liner due to temperature would be larger than the displacement of the reinforced concrete shell due to the DBA pressure. Thus, the liner will be constrained under pressure by the concrete shell. The resulting pressure exerted by the liner is determined from the expression for equivalent pressure given before. The liner stress is determined by the following procedure:

Expressions for stresses in the shell and liner are written in terms of the meridional and circumferential liner strains. These are inserted into the force equilibrium equations resulting in an explicit solution for liner strains from which liner stresses are determined.

The effects of creep and shrinkage of concrete are not important considerations in the analysis and design of a pre-stressed concrete containment. Shrinkage results in meridional and radial displacements which are the opposite of the displacements caused by the principal loads, temperature and internal pressure. Consequently, the effects of creep and shrinkage can be safely ignored.

Cracking is an important consideration in the analysis and design of a reinforced concrete containment. For this reason, stiffness of the concrete is adjusted for the extent of cracking present under various design conditions. When the concrete is completely cracked, calculation of the stiffness of the structure uses only the properties of the reinforcing steel. The steel liner is assumed to make no contribution to the structural integrity of the containment shell. However, the resultant composite action due to the anchorage of the steel liner to the concrete shell does contribute, and adds to the conservatism of the containment design.

The penetrations through the containment wall are grouped into the following three categories for the purposes of analysis and design:

1. 12-inch diameter (nominal) or less

   No special or additional reinforcing is provided. The principal wall reinforcement is located to avoid interference with the penetration.

2. All piping penetrations larger than 12-inch diameter (nominal)

   Reinforcing bars terminated at penetrations are replaced by at least twice the number of bars, one-half of these being placed on each side of the opening. Diagonal reinforcing bars are also provided around openings to take shear and diagonal tension. The anchorage length of the additional bars that frame the openings is determined by using a conservative value for bond stress. This method is consistent with established practice and pressure-tested at the plants listed in Section 3.8.1.1.1.

3. Personnel access and equipment access hatches

   Penetrations for the equipment and personnel access hatches are analyzed using the 3-dimensional finite element capability of the computer program STRUDL II which is discussed in Appendix Section 3A.1.1.

   The thickened ring beam and cylinder wall for both hatches are assumed to be cracked, and to have the extensional stiffness of the reinforcing bars only. The analysis shows that sizeable tangential (in plane) shears exist in the wall near the ring beam. These shears are resisted by reinforcing bars which are placed parallel to the typical earthquake shear bars.

   The ring beam is designed to resist the axial tension and shears resulting from the loading criteria listed in Section 3.8.1.3. The axial tension is assumed to be resisted by the reinforcing bars only. The shears,

including torsional shear, are resisted entirely by stirrups placed radially around the penetrations.

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

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

3.8.1.4.2 Steel Liner

The containment liner plate is analyzed and designed for the load combinations specified in Section 3.8.1.3.2.

The thermal analysis of the wall-to-mat transition area of the liner, including the knuckle region and support skirt, is done using the computer program TAC2D which is described in Appendix Section 3A.1.5. The results of this analysis along with the forces obtained from the analysis of the cylindrical portion of the liner are used as input to the computer program SHELL 1 to analyze this region. SHELL 1 is described in Appendix Section 3A.1.2.

The cylindrical and dome portions of the liner plate are analyzed using the computer program SHELL 1. This analysis provides the forces in the liner plate which are used for the design of the plate and as boundary conditions for the analysis of the wall-to-mat transition area of the liner as well as the analyses of all the penetrations.

The liner is considered firmly anchored to the containment concrete under all loading conditions, and liner loads are determined from the analysis of the containment structure assuming displacement compatibility between the liner and the inside face of the concrete containment wall or dome. Section loads developed from the SHELL 1 analysis, adjusted by hand analysis to include seismic responses, are proportioned according to the relative stiffness of the liner and concrete, taking into account the extent of concrete cracking. Membrane tension, compression, and shear distributions in the liner are thus provided.

The analysis of the liner adjacent to the penetrations is performed as a part of the analysis of the penetrations. The analysis is accomplished with the computer program ASAAS which is described in

Appendix Section 3A.1.4. The model includes the penetration and reinforcement collar as well as the liner up to a distance of approximately three times the radius of the penetration from the center of the penetration.

The loads derived from the support of piping, conduit, or miscellaneous equipment are transferred to the containment concrete wall through insert plates and their anchors. Sufficient anchorage is provided to ensure that the liner plate adjacent to the insert plates is isolated from the loads applied to brackets or attachments, and the leaktight integrity of the liner plate is maintained. Manual calculations are used to determine the resultant anchor loads.

The anchors for the liner plate are evaluated by manual calculations. Since the forces in the liner are usually compressive (test condition is an exception), the anchor studs are spaced throughout the liner wall and dome to prevent buckling. Pitch dimensions are determined by the procedure set forth by Timoshenko and Gere (1961) for a cylindrical shell under combined axial and uniform lateral pressure. Compressive forces which tend to buckle the liner are obtained from the liner plate analysis. These forces are transformed into the axial load and lateral pressure required by the referenced text.

Shear loads for the anchors are determined by assigning an area of influence to each anchor. Boundary loads on each area are calculated from the membrane force distributions. Equilibrium of each area requires that shear loads be transferred to the anchor. The anchor is designed for these loads.

3.8.1.5 Structural Acceptance Criteria

3.8.1.5.1 Containment Structure

The containment structure is designed for the loads and load combinations presented in Section 3.8.1.3.1. Allowable stresses, unless otherwise defined, are in accordance with ACI 318-71. Details of the design conform to ACI 318-71 and the additional requirements discussed in Section 3.8.1.4.

The tangential shear stress, Vu, resulting from earthquake loading is resisted by the concrete and by diagonal reinforcing bars. As discussed in Section 3.8.1.4.1, the maximum allowable tangential shear stress, Vc, carried by the concrete is assumed to be not greater than 40 psi.

3.8.1.5.2 Steel Liner

The load combinations and the corresponding allowable stresses used in the analysis and design of the steel liner plates and anchors are presented in Table 3.8-9.

Piping penetrations meet the requirements for the process piping systems of which they are a part. In addition, those portions which are not concrete backed and which form a part of the containment boundary meet the requirements discussed in Section 3.8.2.5. Other steel components not backed by concrete also meet the requirements discussed in Section 3.8.2.5.

3.8.1.6 Materials, Quality Control, and Special Construction Techniques

For applicable codes, standards, and specifications, refer to Section 3.8.1.2. The quality control program for the fabrication and construction of the containment complies with the requirements of Appendix B of 10 CFR 50 and the applicable Regulatory Guides listed in Section 3.8.1.2. Exceptions to these guides are indicated in this Section (3.8.1) or in Section 1.8.

The description of the Quality Control Program is given in Section 17.1.

3.8.1.6.1 Concrete

The concrete and its constituents meet the requirements of Regulatory Guide 1.55 with the exceptions given in Section 1.8.

3.8.1.6.1.1 Materials

Cement conforms to the requirements of Portland Cement of ASTM C 150, Type II. Fine and coarse aggregates conform to the requirements of ASTM C33.

Air-Entraining Admixtures conform to the requirements of "Standard Specification for Air-Entraining Admixtures for Concrete," ASTM C260, when tested in accordance with "Standard Method of Testing Air-Entraining Admixtures for Concrete," ASTM C233.

The concrete mixes used in the containment structure yield a unit weight of at least 135 lbs/ft^3, air- and oven-dried, in accordance with ASTM C567.

The compressive strength of the concrete in the mat, shell, and dome is at least 3,000 psi at 28 days with a maximum slump of 4 inches except as indicated in Section 1.8.

3.8.1.6.1.2 Quality Control

Concrete protection for reinforcement, preparation, and cleaning of construction joints, concrete mixing, delivering, placing, and curing, meets the requirements of Regulatory Guide 1.55 with the exceptions given in Section 1.8.

Curing and protection of freshly deposited concrete conform to ACI 301, Chapter 12, except that curing compounds are not used on surfaces to which additional concrete is to be bonded, and where wood and/or metal forms are used, and remain in place for curing, the forms are kept wet as required to prevent their opening at the joints and drying out of the concrete.

Concrete batching conforms to ASTM C94, ACI 301, and ACI 304.

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

Concrete strength tests of the job concrete are performed in accordance with ACI 301, Chapter 16, supplemented by the following: not less than two sets of compression test specimens are made for each strength of concrete placed during the first 2 days of placing concrete. Thereafter, one set of test specimens is made for each 8-hour shift, or for every 150 yd^3 for Seismic Category I structures and for every 250 yd^3 for other structures, whichever is less, of each mix design of concrete placed in any 1 day. In addition, one set of specimens is made whenever, for any reason, the material, method of concreting, or proportioning is changed.

The test specimens for compressive strength are 6-inch diameter and 12-inch long cylinders. Each set consists of three specimens; one is tested at 7 days and two at 28 days age.

The strength level of the concrete is considered satisfactory if the averages of all sets of three consecutive strength test results equal or exceed the specified compressive strength f'c and no individual strength test result falls below the specified compressive strength f'c by more than 500 psi.

If any strength test results fail to meet the criteria of ACI 318 as specified in the previous paragraph, specimens of hardened concrete are obtained and tested in accordance with "Method of Obtaining and Testing Drilled Cores and Sawed Beams of Concrete" (ASTM C42). The concrete specimens are evaluated in accordance with ACI 318, Chapter 4. If necessary, strength evaluation of the concrete member and load tests are made in accordance with ACI 318, Chapter 20.

Statistical quality control of the concrete is maintained by a computer program based on the article "Application of Computers in the Evaluation of Quality Control of Concrete" in ACI Publication SP-16, "Computer Application in Concrete Design and Technology." This program analyzes compression test results in accordance with methods established by ACI 214.

3.8.1.6.1.3 Construction Techniques

In general, concrete in the wall and dome of the containment structures is placed in approximately 6-foot-high lifts around the circumference. Each lift is constructed in approximately 18-inch layers placed at such a rate that concrete surfaces do not reach their initial set before additional concrete is placed.

Forms are used on the exterior of the concrete dome to a line about 50 degrees above the horizontal. The permanent steel liner serves as the inner form for concrete. For the area where exterior forms are used, the concrete joints lie in horizontal planes. Above the 50 degree line, the dome concrete is placed in three placements without the use of exterior forms.

3.8.1.6.2 Reinforcing Steel

3.8.1.6.2.1 Materials

Reinforcing bars size No. 11 and smaller conform to the requirements of ASTM A615 Grade 40 or 60. Reinforcing bar sizes No. 14 and 18 are grade 50 per ASTM A615 as modified by the special mechanical and chemistry requirements presented in Table 3.8-10.

Cadweld T-Series reinforcing steel splices are full tension splices manufacturered by Erico Products, Inc., Cleveland, Ohio, and are used to splice No. 14 and 18 reinforcing bars. In restricted areas, reinforcing bars are butt-welded in a manner conforming to the requirements of AWS D12.1-61. Cadweld splices are made in accordance with the instructions issued by the manufacturer.

3.8.1.6.2.2 Quality Control

Reinforcing Bars

For the No. 14 and 18 bars used in the containment structure, ingots and billets are traced with identifying heat numbers. Bundles of bars are tagged with a heat number as they come off the rolling mill. A mark is rolled into bars to identify them as possessing the chemical and mechanical qualities specified.

Inspectors from SWEC witness, on a random basis, the pouring of the heats and the physical and chemical tests performed by the manufacturer of the reinforcing bars.

Bars not meeting specification are rejected.

Mill test reports showing actual chemical and mechanical properties are furnished to the owner for each heat of steel used in making all reinforcing steel furnished.

Cadweld Splices

Splicing complies with the requirements of Regulatory Guide 1.10, Revision 1, dated January 2, 1973 with the exceptions given in Section 1.8.

3.8.1.6.2.3 Construction Techniques

General

Fabrication and placing of reinforcing steel conform to the requirements of Paragraph 5.4 of ACI 301 and Section 7.14 of ACI 318-71.

Tack welding of reinforcing steel is not permitted.

3.8.1.6.3 Structural Steel

3.8.1.6.3.1 Material

Steel specifications invoked for structural framing, brackets, and attachments are discussed in Section 3.8.1.2.2.

3.8.1.6.3.2 Quality Control

In general, main members, columns, baseplates, bracing, trusses, girts, and bolts larger than 1 inch in diameter are traceable to a specific heat number. For the above items in the auxiliary building roof framing, various field fabricated minor framing, and all clip angles, seats, stiffeners, gusset plates, and weld filler metal, traceability to a specific heat number is confirmed in the suppliers' shops or upon receipt at the jobsite. For bolts 1 inch in diameter and smaller, lot identification is similarly confirmed. The storage and issuance of these materials for construction is controlled in a manner which assures that only QA Category I materials are installed in QA Category I applications.

3.8.1.6.3.3 Construction Techniques

Structural steel material, erection, and fabrication tolerances are in accordance with the AISC "Specification for the Design Fabrication, and Erection of Structural Steel for Buildings."

Welding of structural steel is in accordance with AWS D 1.1-72.

3.8.1.6.4 Steel Liner

The materials for the reactor containment liner and mat embedments, including penetrations, are listed in Table 3.8-8 and meet the applicable requirements of the ASME B&PV Code, Sections II, III, and VIII.

3.8.1.7 Testing and Inservice Surveillance Requirements

3.8.1.7.1 Concrete Containment Structural Acceptance Test

In order to demonstrate that the concrete containment structure responds satisfactorily to required internal pressure loads, a structural acceptance test is performed after the liner is completed, the last concrete is poured and cured, and all penetrations, sleeves, and hatches are installed. The test equals or exceeds the requirements of Regulatory Guide 1.18.

The containment is subjected to an internal pressure equal to at least 1.15 times the containment design pressure. The pressure test commences at atmospheric pressure and is raised to the maximum test pressure in four or more approximately equal pressure increments. The containment is depressurized in the same number of decrements. At each pressurization and depressurization level, the pressure is held constant for at least 1 hour before the deflection and strain measurements are recorded.

Radial deflection is measured along six meridians around the circumference at 13 feet-6 inches above the top of the mat at mid-height between the mat and the springline, and at the springline of the dome. Vertical deflections are measured at the springline and the apex of the dome. Radial deflections for the largest opening with a ring edge beam are also measured.

Mapping of cracks is performed on exterior surfaces of the containment at locations selected prior to start of the pressure application. Mapping is performed at four locations: One near the base-wall intersection, one at the mid-height of the wall, one at the springline of the dome and one around one quadrant of the equipment hatch.

Environmental conditions are measured and monitored to permit the evaluation of their contribution to the response of the containment. The testing sequence is repeated if the test pressure drops for unexpected reasons to or below the next lower pressure level, or if significant modification or repairs are made to the containment following the test.

The anticipated deflections are computed by taking into account the interaction of liner, reinforcements, and concrete including the effects of concrete cracking. A final test report will be prepared following the test. The report will contain the information outlined in Regulatory Guide 1.18.

3.8.1.7.2 Steel Liner and Penetrations

3.8.1.7.2.1 Initial Structural Acceptance Test

The liner plate and penetrations are tested in conjunction with the concrete containment as discussed in Section 3.8.1.7.1.

3.8.1.7.2.2 Leakage Rate Test

A containment integrated leak rate test is performed initially and periodically to verify that the leakage rate of the containment is within allowable limits. The general test description is covered in Chapter 16 and Section 6.2.6.

The leak rate is performed in accordance with 10 CFR 50, Appendix J.

3.8.1.7.2.3 Piping System Penetrations

The unsleeved and sleeved piping system penetrations are tested in conjunction with the containment structural acceptance test. The spare process piping penetrations are capped off for the containment structural integrity test. The piping penetrations weld seams are capable of being locally leak tested periodically to verify continued integrity.

3.8.1.7.2.4 Mechanical System Penetrations

The fuel transfer tube enclosure is sealed off with a blind flange. The blind flange is pressure tested during the containment structural integrity test and has double gaskets capable of being locally tested.

3.8.1.7.2.5 Access Openings

The personnel air lock and emergency air lock are self-contained pressure vessels and are pressure tested independently. The equipment hatch opening is tested in conjunction with the containment structural acceptance test and is locally tested periodically thereafter.

3.8.2 Steel Containment

This section, as outlined in Regulatory Guide 1.70, Revision 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," regarding a steel containment, is not applicable to the BVPS-2 containment structure because a steel-lined reinforced concrete containment is being used, as described in Section 3.8.1. Certain steel components in the containment system, described in Section 3.8.1, are classified in accordance with the ASME B&PV Code, Section III, as Class MC components except as discussed later in this section; these are the personnel air lock, the equipment hatch, the emergency air lock, the fuel transfer tube enclosure, and the piping and electrical penetrations subject to pressure-induced stresses.

The ASME III Class MC requirements were not intended to apply to concrete backed pressure vessels. However, ASME III Division 2 was not available for use in the design, material selection, fabrication, and erection of the containment vessel. Hence, ASME III Division 1

and ASME VIII were used as guides. The extent to which ASME III Subsection NE applies to various containment components is discussed in Section 3.8.1.2.3.

This section addresses itself to the requirements of the USNRC Standard Review Plan (SRP), NUREG-75/087, Section 3.8.2, dated November 24, 1975.

3.8.2.1 Description of the Containment

For the description of the containment and its steel components, refer to Section 3.8.1.1.

3.8.2.2 Applicable Codes, Standards, and Specifications

The basic code for the Class MC steel components is the ASME B&PV Code, Section III, Division 1, 1971 Edition (including all addenda through Winter 1972). This code provides the requirements of the components, for materials, design, fabrication, examination, and testing. For other applicable codes, standards, and specifications, refer to Section 3.8.1.2.

3.8.2.3 Loads and Load Combinations

The loads presented and defined in NUREG-75/087, SRP Section 3.8.2 are considered where applicable in the design of the ASME B&PV Code, Section III, Class MC steel components. Discussion of the various load combinations and allowable stresses is presented in Section 3.8.2.5.

3.8.2.4 Design and Analysis Procedures

The design and analysis of the Class MC components are in accordance with the requirements of Subsection NE of the ASME B&PV Code, Section III, including the applicable portions of Appendix A.

3.8.2.5 Structural Acceptance Criteria

The design is such that the stress and strain limits, as defined in NE-3000 of the ASME B&PV Code, Section III, are satisfied for pressure loads in combination with mechanical loads and thermal loads. The load combinations and allowable stresses are in accordance with the requirements of NUREG-75/087, SRP Section 3.8.2.

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

The materials meet the requirements in the ASME B&PV Code, Section III, Division 1, Subsection NE, for Class MC components, including the impact test requirements described by the Liner Specification. The materials are listed in Table 3.8-2.

The quality assurance program for fabrication and erection is in accordance with the requirements of Subsection NE of the ASME B&PV Code, Section III, Division 1

3.8.2.7 Testing and In-Service Inspection Requirements

Testing and ISI requirements of Class MC components are in accordance with 10 CFR 50, Appendix J. The personnel hatch, and the emergency air lock are tested and stamped Class MC in accordance with the requirements of the ASME B&PV Code, Section III, Subsection NE.

The equipment hatch is designed and fabricated to Class MC requirements, but is not code stamped. The electrical penetrations are designed and fabricated to Class MC requirements, but an ASME Code data report, (ANI) third party inspection and ASME Code stamping are not required. The Class MC components are tested during the containment structural acceptance test as described in Section 3.8.1.7.

The fuel transfer tube expansion bellows are tested in accordance with Code Case 1330-3 (special ruling) Special Equipment Requirements Section III.

3.8.3 Concrete and Structural Steel Internal Structures of Steel or Concrete Containments

3.8.3.1 Description of Internal Structures

The containment internal structures, shown on Figures 3.8-1, 3.8-2, 3.8-3, 3.8-4, 3.8-5, 3.8-6 and 3.8-7, consist of heavily reinforced concrete walls and slabs which are designed to support the principal nuclear steam supply equipment. The interior concrete also provides radiation shielding for equipment and operating personnel, supplies protection from missiles resulting from component failure, provides restraint for various piping systems, and acts as a jet impingement barrier during postulated pipe breaks.

The reinforced concrete primary shield wall forms the reactor cavity at the center of the containment structure. It surrounds and provides lateral restraint for the reactor vessel and its structural steel support.

Located concentrically to the primary shield wall is the reinforced concrete crane wall, which is supported by reinforced concrete columns extending from the foundation mat.

Extending approximately radially between the primary shield wall and the crane wall are reinforced concrete walls, which separate the internals into cubicles. Reinforced concrete slabs at the floor of three cubicles provide structural platforms for steam generator and reactor coolant pump (RCP) supports. In addition, the pressurizer is located in and supported by its own cubicle.

Structural supports for the reactor vessel steam generators, RCPS, and pressurizer are described in Section 5.4.14.

The refueling cavity, located above the reactor vessel, and the fuel transfer canal are stainless steel-lined reinforced concrete structures.

The overhead polar crane is located above the operating floor of the containment and spans approximately 106 feet. The crane is supported at the top of the reinforced concrete crane wall in such a way that it is restrained both horizontally and vertically against seismic dislodgement.

The auxiliary crane is located on the operating floor of the containment, between steam generator SG21A and the equipment hatch. The crane is supported vertically by the operating floor, and restrained laterally by bracing attached to the crane wall.

3.8.3.2 Applicable Codes, Standards, and Specifications

Codes, standards, specifications, and USNRC Regulatory Guides, as applicable to the internal structures of the containment, are provided in Sections 3.8.1.2, 3.8.1.6, and 5.4.14.

3.8.3.3 Loads and Loading Combinations

Loads considered in the design of the containment internals include those encountered during normal plant operation, severe environmental and extreme environmental events, and those sustained during abnormal plant conditions including the LOCA and other high-energy pipe rupture events.

Interior concrete structures and structural steel within the containment are designed to withstand the pressure buildup resulting from the most severe accident for each cubicle, including the LOCA discussed in Sections 6.2.2 and 15.6.5. The blowdown of a postulated rupture of the primary coolant pipe is assumed to be either in one of the three steam generator cubicles or within the primary shield wall. Since the volume of each of these cubicles is less than the entire containment structure, differential pressures exist between the interior and exterior of the cubicle until full pressurization of the containment is attained. All structural components, walls, floors, and beams enclosing these cubicles are designed to withstand these differential pressures.

Section 3.6B describes the protection provided against the dynamic effects associated with a postulated rupture of piping.

Missile generation and design of barriers to resist missile hazards are described in Section 3.5.

The reinforced concrete portions of the containment internal structure are designed for the loads and loading combinations which follow. These loading combinations compare with those presented in SRP 3.8.3. The accident loads considered in loading combinations 3.8-7 through 3.8-9, listed as follows, control the design of the internal concrete structures. These equations are equal to or more

conservative than their counterparts in ACI 349 or SRP 3.8.3. Loading combinations and strength limits for concrete structures are:

$$U = 0.75\ (1.4D + 1.7CL) \quad (3.8\text{-}1)$$

$$U = 1.4D + 1.7L \quad (3.8\text{-}2)$$

$$U = 1.4D + 1.7L + 1.9E \quad (3.8\text{-}3)$$

$$U = 0.9D + 1.4E \quad (3.8\text{-}4)$$

$$U = 1.0D + 1.0L + 1.0E' \quad (3.8\text{-}5)$$

$$U = 0.9D + 1.0E' \quad (3.8\text{-}6)$$

$$U = 1.0D + 1.0L + 1.0Ta + 1.0Ra + 1.5Pa \quad (3.8\text{-}7)$$

$$U = 1.0D + 1.0L + 1.0Ta + 1.0Ra + 1.25Pa + 1.0(Yr + Yj + Ym) + 1.25E \quad (3.8\text{-}8)$$

$$U = 1.0D + 1.0L + 1.0Ta + 1.0Ra + 1.0Pa + 1.0(Yr + Yj + Ym) + 1.0E' \quad (3.8\text{-}9)$$

$$U = 0.75\ (1.4D + 1.7L + 1.9E + 1.7To + 1.7Ro) \quad (3.8\text{-}10)$$

where:

U = Required section strength based on strength design methods as described in ACI 318-71.

D = Dead loads or their related internal moments and forces, including any permanent equipment loads.

L = Live loads or their related internal moments and forces, including any movable equipment loads and other loads which vary with intensity and occurrence.

CL = Construction live load.

E′ = Loads generated by the safe shutdown earthquake (SSE).

E = Loads generated by the one-half SSE.

Ta = Thermal loads under accident conditions generated by a postulated break.

Ra = Pipe and equipment reactions under accident conditions generated by a postulated break.

Pa = Differential pressure load generated by a postulated break including an appropriate dynamic load factor applied to the peak of the pressure-time curve.

Yr = Load on the structure generated by the reaction on the broken high energy pipe during the postulated break including an appropriate dynamic load factor applied to the peak of the reaction-time curve.

Yj = Jet impingement load on a structure during the postulated break, including an appropriate dynamic load factor applied to the peak of the jet-time curve.

Ym = Missile impact load on a structure during the postulated break, including an appropriate dynamic load factor applied to the peak of the missile impact-time curve.

Note: Except for the fuel building, To has a negligible effect on the concrete design.

The structural steel portions of the containment internal structure are designed for the loads, loading combinations, and strength limits described as follows:

$$S = D + L \qquad (3.8\text{-}11)$$

$$S = D + L + E \qquad (3.8\text{-}12)$$

$$1.6S = D + L + E' \qquad (3.8\text{-}13)$$

$$1.6S = D + E' \qquad (3.8\text{-}14)$$

$$1.6S = D + L + Ta + Ra \qquad (3.8\text{-}15)$$

l $$1.6S = D + L + Ta + Ra + Pa + 1.0\ (Yj + Yr + Ym) + E \qquad (3.8\text{-}16)$$

$$1.7S = D + L + Ta + Ra + Pa + 1.0\ (Yj + Yr + Ym) + E' \qquad (3.8\text{-}17)$$

$$1.5S = D + L + E + To + Ro \qquad (3.8\text{-}18)$$

where:

To = Thermal effects and loads during normal operation or shutdown conditions based on the most critical transient or steady state condition.

Ro = Pipe reactions during normal operation or shutdown conditions based on the most critical transient or steady state condition.

S = The required section strength based on the elastic working stress methods and allowable stresses defined in Part 1 of the AISC (1979) Specification as referenced in Section 3.8.1.2. The 33-1/3 percent increase in allowable stresses due to seismic loadings allowed by AISC is not used in determining the valvue of S.

Note: For load combinations 3.8-16 and 3.8-17, the plastic section modulus of steel shapes may be used computing the required section strength, S, except for those sections which do not meet the AISC criteria for compact sections.

In general, for both concrete and steel load combinations:

1. Peak values of Pa, Ta, Ra, Yj, Yr, and Ym are used unless a time-history analysis is performed to justify otherwise.

2. Local stresses may be exceeded under the concentrated loads Yr, Yj, and Ym, provided that there will be no loss of function of any safety-related system.

3. For structural steel loading combinations equations 3.8-11 through 3.8-17, the elastic modulus in bending is used.

3.8.3.4 Design and Analysis Procedures

The cylinders, beams, columns, walls, and slabs which comprise the containment internal structure provide multiple load paths to transfer applied loads to the containment mat. Distribution of loads through the members was accomplished by stiffness formulations using space frame, finite difference, and finite element techniques based on elastic behavior of the redundant structures. In limited local areas of walls and slabs where it was determined that overall structural integrity would not be impaired, and that deflections would not be excessive, the yield line theory was used to analyze the effects of accident (pipe whip, missile, or jet impingement) loads.

Using principles of engineering mechanics, appropriate for the geometry of the structure, overlapping models were selected such that boundary conditions assigned to one model were confirmed by deflections and loads generated by adjacent models.

Loads were applied directly to the models as static surface forces, body forces, or concentrated loads. Pressure time histories developed using the procedures described in Section 6.2 were converted to equivalent static loads considering peaks of the forcing functions and dynamic load factors appropriate to the stiffness of the structures and the duration and shape of the impulsive loads. Seismic forces and reactions developed using the techniques described in Section 3.7B.2 were applied as body forces, boundary loads, and concentrated reactions from equipment and piping using appropriate DLFs.

Accident loads were similarly applied, primarily as concentrated loads, at restraints and pipe whip target locations. Jet impingements were treated similarly to pressures.

Internal loads are transferred to the containment foundation mat through the primary shield wall and columns below the crane wall and steam generator cubicle slabs. The primary shield wall and the columns share the distribution of loads approximately equally. The relative stiffnesses of the 10 foot containment foundation mat and the support elements were considered in determining the load distribution. Where appropriate, cracked concrete sections were used to refine the loads distribution.

Computer programs STRUDL and SHELL 1, described in Appendix 3A, are the primary analytical codes used.

The design of the internals concrete conforms to the requirements of ACI 318-71 in a conservative manner. Reinforcing steel ratios are determined in accordance with the procedures outlined in ACI 318 and the principal reinforcement patterns are located in the direction of tensile forces.

tensile forces. Splice and anchorage requirements comply with ACI 318 and, where biaxial tension fields exist, the development lengths required by Section 12.5 of ACI 318 are increased by a minimum of 25 percent.

Structural steel is designed in accordance with the AISC (1969) Specification.

Structural design and analysis of specific internals components comply with the general procedures described. The reactor support system, steam generator supports, RCP supports, and pressurizer supports have been designed and analyzed as discussed in Section 5.4.14.

The primary shield wall provides the major structural support for the refueling cavity and also shares the loads from most other internals components. Loads from these structures are transferred to the primary shield walls through integrally cast, heavily reinforced slabs and walls. Direct loads on the primary shield wall include internal and external asymmetric pressures, equipment reactions, seismic loads, and pipe reactions and jet impingements resulting from primary coolant loop and other high energy line breaks. These loads were developed and applied as discussed in the general procedure, including appropriate consideration of dynamic loading effects. Where loads transfer became particularly complex, local computer analyses were used.

Loads developed in the meridional reinforcement of the primary shield wall are transferred directly through the base liner by means of bridging bars. The reinforcement is developed within the base mat. Compression and shear loads at the base are transferred in bearing.

During normal operation, a thermal gradient exists within the primary shield wall due to thermal input from the reactor and heat of gamma and neutron radiation. The design of the primary shield wall includes the effect of the gradient. Thermal input from a LOCA is not a significant design factor. Due to the relatively short duration of LOCA high temperatures, only a small thickness of the primary shield wall near each surface is heated.

Secondary shield walls (that is, radial and crane walls, refueling cavity walls, the operating floor, and intermediate floors) were designed as described in the general procedures. In general, all components were designed to remain elastic under the applied loads. When these components serve as missile barriers, local loads may exceed elastic limits. Analysis of missile barriers is discussed in Section 3.5.

The upper crane wall extends above the operating floor to support the polar crane. It is designed by the methods of ACI 318 to accommodate all dynamic loads from the polar crane, and to provide support for numerous high energy pipe restraints.

3.8.3.5 Structural Acceptance Criteria

Design and analysis of interior concrete structures comply with ACI 318, using the strength design methods, except where limit analysis is used. The basic criterion for concrete strength design is expressed as:

Required Strength ≤ Calculated Strength

The required strength is expressed in terms of design loads (that is, loads or load combinations multiplied by appropriate load factors). Calculated strength is computed using ACI 318-71 including the appropriate capacity reduction factors. Acceptance criteria for limit analysis are discussed in Section 3.8.3.

Design for horizontal shear forces in the plane of internal structure walls complies with the requirements of Section 11.16 of ACI 318 which incorporates the combined effects of shear and tensile stresses into the nominal permissible shear stress, Vc, allowed to be carried by the concrete.

Design of interior steel structures is based either on elastic working stress design methods using normal working stress levels given in Part 1 of the AISC (1969) Specification or on the plastic design methods of Part 2 of the AISC (1969) Specification.

The loading criteria specified in Section 3.8.3.3 also specify the allowable limits which constitute the structural acceptance criteria.

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

Material and quality controls used for the reactor containment internal structures are described in Sections 3.8.1.2 and 3.8.1,6, except that a higher strength concrete mix was used. The 90-day minimum specified compressive strength of concrete is 5,000 psi.

Materials and quality control comply with the requirements of ACI 318-71 for concrete and with the AISC (1969) specification for structural steel. Quality control complies with the requirements of ANSI N45.2.5-1974 and Regulatory Guides 1.55 and 1.94, except as discussed in Section 1.8 and 3.8.1.2.1.6.

Welding of reinforcing bars complies with the requirements of AWS D12.1-61. Welding of structural steel complies with the requirements of AWS D1.1.

Materials and quality control for the steel supports of the reactor coolant system are as discussed in Sections 3.9B.3.4 and 5.4.14.

The Quality Assurance Program is described in Section 17.1.

3.8.3.7 Testing and Inservice Inspection Requirements

No full-scale structural testing or inservice inspection of the reactor containment interior structure is planned. The materials used in the construction of the internals are tested as described in Sections 3.8.1.2 and 3.8.1.6.

3.8.4 Other Seismic Category I Structures

3.8.4.1 Description of the Structures

This section describes other Seismic Category I structures which perform a safety-related function. The primary intake structure, which was designed, constructed, and licensed with Beaver Valley Power Station - Unit 1 (BVPS-1) and which serves both BVPS-1 and Beaver Valley Power Station - Unit 2 (BVPS-2), is not included herein.

Seismic Category I structures other than the containment structure and other safety-related structures not classified as Seismic Category I because of other design provisions are included herein.

The plot plan and the general arrangement of these structures and equipment within these structures are shown on the figures referenced in Table 3.8-12. A seismic shake space is provided between buildings as required to allow independent movement of structures under earthquake loading.

The design of Seismic Category I structures to resist tornado loads is discussed in Sections 3.5 and 3.7B.

3.8.4.1.1 Auxiliary Building

The auxiliary building is a Seismic Category I structure, consisting of a basement and three upper stories is approximately 120 feet wide by 145 feet long by 63 feet high. It is located adjacent to, and south of, the fuel and decontamination building, and adjacent to and west of the service building and main steam valve and cable vault area.

The auxiliary building is supported on a reinforced concrete foundation mat. The roof and walls of the top story are predominantly steel framed with metal siding and metal roof deck, with the exception of the ventilation core area, component cooling surge tank cubicle, and the air conditioning room which are reinforced concrete. The remainder of the structure is reinforced concrete.

Concrete walls and floors are designed to provide tornado protection for safety-related equipment and piping and to provide biological shielding where required. The top story steel structure is not designed to provide tornado protection.

3.8.4.1.2 Safeguards Area

The safeguards area is a Seismic Category I structure approximately 60 by 106 feet at the base and approximately 59 feet high. It is located adjacent to, and east of, the containment structure.

The safeguards area is supported on a reinforced concrete foundation mat and the remainder of the structure is reinforced concrete.

The concrete structure of the safeguards area provides tornado protection for the engineered safety features pumps, valves, and piping penetrations.

3.8.4.1.3 Main Steam Valve and Cable Vault Area

The main steam valve and cable vault area is a Seismic Category I multi-level structure, approximately 94 feet wide (at its widest part) by 138 feet long by 77 feet high, located adjacent to, and south of, the containment structure.

The main steam valve and cable vault area is supported on a reinforced concrete foundation mat. The remainder of the structure is reinforced concrete, with the exception of one portion of roof that is steel framed with metal roof deck.

The concrete structure provides tornado protection for safety-related systems, including the main steam isolation valves.

The steel frame roof portion is non-Seismic Category I and is not designed for seismic or tornado loadings.

3.8.4.1.4 Fuel and Decontamination Building

The fuel and decontamination building is a Seismic Category I L-shaped structure with the main portion approximately 44 by 110 by 71 feet high. It is located adjacent to, and west of, the containment building and north of, and adjacent to, the auxiliary building.

The fuel and decontamination building is supported on a continuous reinforced concrete foundation mat. The roof and walls are of reinforced concrete. The fuel and decontamination building contains the new and spent fuel and associated fuel handling facilities. The spent fuel pool has a cask loading area, storage rack area, and transfer canals, all of which are lined with stainless steel plate. The cask loading area is separated from the storage rack area by a stainless-steel-lined reinforced concrete wall. An opening, 25 feet

high by 2 feet wide, in this wall permits fuel transfer from the storage racks to the shipping cask. The shipping cask cannot be moved over the storage racks because the crane which lifts it is located such that it cannot travel over the storage racks.

Reinforced concrete is provided for biological shielding where required. Safety-related equipment and the spent fuel are protected against tornadoes and tornado-generated missiles.

3.8.4.1.5 Control Building

The control building is a Seismic Category I three-story structure built adjacent to, and as an extension of, the BVPS-1 control room. It is approximately 69 by 89 by 45 feet high. The control building is located west of the fuel building and auxiliary building and is connected to the auxiliary building by the electrical cable tunnel.

The control building is supported on a reinforced concrete foundation mat. The roof and walls are of reinforced concrete. The top story of the control building contains the control room, computer room, and the heating, ventilation and air conditioning equipment room. The lower two stories house switchgear, cable spreading areas, and other associated equipment.

The concrete structure is designed to provide tornado protection.

3.8.4.1.6 Diesel Generator Building

The diesel generator building is a Seismic Category I structure located east of the service building and adjacent to, and north of, the east end of the turbine building. It is a two story structure approximately 78 by 88 by 57 feet high.

The diesel generator building is supported on a reinforced concrete foundation mat. The roof and walls are of reinforced concrete.

The concrete structure is designed to provide tornado protection.

3.8.4.1.7 Service Building

The service building is a Seismic Category I structure located adjacent to, and north of, the turbine building and west of the diesel generator building. It is a four-story structure approximately 54 by 186 by 70 feet high.

The service building is supported on a reinforced concrete foundation mat. The roof and walls of the top story are steel framed with metal siding and roof deck. The remainder of the structure is reinforced concrete.

The reinforced concrete walls and floors are designed to protect against tornado and tornado-generated missiles. The top story steel

framed structure is non-Category I and is not designed for seismic or tornado loadings.

3.8.4.1.8 Electrical Cable Tunnel

The cable tunnel is a Seismic Category I subsurface structure extending approximately 82 feet from the auxiliary building to the control building.

The foundation mat, walls, and roof of the cable tunnel are reinforced concrete.

The concrete structure of the cable tunnel provides tornado protection for safety-related electrical systems.

3.8.4.1.9 Service Water Valve Pit

There are two Seismic Category I service water valve pits. One is approximately 14 by 20 by 15 feet high and is located adjacent to, and north of, the safeguards area. The other is approximately 24 feet by 36 feet by 18 feet high and is located northwest of the fuel and decontamination building. They are subsurface structures.

The service water valve pits are supported on reinforced concrete foundation mats. The walls and roofs are of reinforced concrete.

The concrete structures of the service water valve pits provide tornado protection for their contents.

3.8.4.1.10 Primary Demineralized Water Tank Enclosure

The primary demineralized water tank enclosure is a Seismic Category I structure located east of the safeguards area and south of the refueling water storage tank. It is approximately 38 by 40 by 46 feet high.

The primary demineralized water tank enclosure is supported on a reinforced concrete foundation mat which also supports the tank. The roof and walls of the enclosure are of reinforced concrete.

The reinforced concrete structure is designed to provide tornado protection.

3.8.4.1.11 Emergency Outfall Structure

The emergency outfall structure is a Seismic Category I dual-chambered overflow weir structure. It is approximately 21 by 35 by 24 feet high, located about 1,900 feet west of the center of the containment building.

The emergency outfall structure is constructed of reinforced concrete. It protects the ends of the emergency service water lines

from missile impact and maintains proper hydraulic head within the service water system.

The structure is designed to remain functional when subjected to the postulated tornado and tornado-generated missile loadings.

3.8.4.1.12 Equipment Hatch Platform

The equipment hatch platform is a Seismic Caregory I structure located adjacent to, and northeast of, the containment. It is approximately 29 by 31 by 49 feet high.

The equipment hatch platform is supported on a reinforced concrete foundation mat. The walls and slabs are of reinforced concrete.

The concrete structure is designed to provide tornado-generated missile protection for the containment equipment hatch.

3.8.4.1.13 Refueling Water Storage Tank and Chemical Addition Tank Enclosure

The refueling water storage tank (RWST) and chemical addition tank (CAT) are located east of the safeguards area. The RWST and CAT pad and surrounding shield wall are Seismic Category I. The wall is approximately 56 by 67 by 16 feet high.

The RWST and CAT are supported on a reinforced concrete foundation mat. The walls of the surrounding structure are of reinforced concrete.

The concrete structure is designed to provide biological shielding.

3.8.4.1.14 Pipe Trenches

There are two Seismic Category I pipe trenches. One is approximately 42 feet long by 10 feet wide by 13 feet deep. It connects the service building and the safeguards area. The other is approximately 7 feet wide by 6 feet deep with one portion enlarging to approximately 14 feet wide by 8 feet deep. It has a total length of approximately 164 feet, connecting the piping area of the electrical cable with the fuel and decontamination building.

The pipe trenches are reinforced concrete subsurface structures, with the top of the trench covers approximately level with the ground grade.

3.8.4.1.15 Pipe Trenches for BVPS-1 and BVPS-2 Crosstie Piping

The fuel building for BVPS-1 is connected to the waste handling and auxiliary buildings of BVPS-2 by a pipe trench. This pipe trench, approximately 6 feet wide by 4 feet deep, runs due east from the

BVPS-1 fuel building to the west wall of the BVPS-2 condensate polishing building.

The BVPS-1 turbine building is connected to the BVPS-2 pipe trench, just north of the electrical cable tunnel, by a pipe trench. This pipe trench, 8 feet to 9 feet wide by approximately 6 feet deep, runs north from the BVPS-2 pipe trench and then west to the BVPS-1 turbine building.

The crosstie pipe trenches are reinforced concrete subsurface structures with the top of the trench covers approximately level with the ground grade.

The crosstie pipe trenches are non-Seismic Category I. With the exceptions described in Section 3.7B.2, the seismic design requirements are in accordance with Regulatory Guide 1.143.

3.8.4.1.16 Waste Handling Building

The waste handling building is located adjacent to, and west of, the turbine building. It is a four-story, plus basement, structure approximately 40 by 112 by 77 feet high.

The waste handling building is supported on a reinforced concrete foundation mat. The roof and walls of the top two stories are steel

framed with metal siding and roof deck. The remainder of the structure is reinforced concrete.

The waste handling building is designed to provide biological shielding where required. The waste handling building is non-Seismic Category I. With the exceptions described in Section 3.7B.2, the seismic design requirements are in accordance with Regulatory Guide 1.143.

3.8.4.1.17 Condensate Polishing Building

The condensate polishing building is L-shaped with the main portion being approximately 44 by 141 feet and a maximum of 93 feet high. It is located adjacent to, and west of, the waste handling building. The condensate polishing building consists of a basement and three upper stories.

The condensate polishing building is supported on a reinforced concrete foundation mat. The roof, walls, and floor slabs of the building are reinforced concrete.

The condensate polishing building is non-Seismic Category I. With the exceptions described in Section 3.7B.2, the seismic design requirements are in accordance with Regulatory Guide 1.143.

3.8.4.1.18 Gaseous Waste Storage Tank Enclosure

The gaseous waste storage tank (GWST) enclosure is located north at the fuel building. It is an in-ground one story structure 37 by 52 by 15 feet high, with an at-grade entrance of 11 by 18.25 by 10 feet high.

The GWST enclosure is supported on a reinforced concrete foundation mat. The walls, roof, and interior structures are also reinforced concrete.

The GWST enclosure is designed to provide biological shielding where required. The GWST enclosure is non-Seismic Caregory I. With the exceptions described in Section 3.7B.1 and 3.7B.2, the seismic design requirements are in accordance with Regulatory Guide 1.143.

3.8.4.1.19 Turbine Building

The turbine building is located adjacent to, and south of, the auxiliary and service buildings and adjacent to, and east of, the waste handling building. It is approximately 135 by 275 by 115 feet high.

The turbine building and major equipment, including the turbine generator, are supported on reinforced concrete spread footings and foundation mats. The ground floor is a reinforced concrete slab.

The mezzanine floor consists of partial area platforms of steel grating or reinforced concrete supported by steel framing. The operating floor is a reinforced concrete slab supported by steel framing. The turbine building is steel framed. The building is enclosed with insulated metal siding and roof deck. The turbine support, located near the center of the building, is a reinforced concrete structure.

The turbine building steel frame is designed to remain in place with the siding blown off during a tornado. The turbine building is a non-Seismic Category I structure. Seismic design of the structure is discussed in Section 3.8.4.4.

3.8.4.1.20 Other Structures

Other plant structures, including the natural draft cooling tower, alternate intake structure, discharge flume, cooling tower pumphouse, transmission towers and temporary construction and office buildings are sufficiently remote from the Seismic Category I and other safety-related structures to preclude damage that would impair the ability of the safety-related structures to remain functional.

3.8.4.2 Applicable Codes, Standards, and Specifications

Unless indicated otherwise, the material properties and design and construction methods used for other Category I structures are based upon the appropriate sections of the applicable codes, standards, specifications, and USNRC regulations and Regulatory Guides listed in Sections 3.8.1.2 and 3.8.1.6.

3.8.4.3 Loads and Load Combinations

Loads considered in the design of other Category I structures include those encountered during normal plant operation, severe and extreme environmental events, and those sustained during abnormal plant conditions including high-energy pipe rupture events.

Other Seismic Category I structures are designed to the applicable loading criteria given in Section 3.8.3.3, for concrete and steel internal structures of concrete containment. Loads and the applicable load combinations for which each structure is designed depend on the conditions to which that particular structure could be subjected.

In addition to the loads and load combinations referenced, other Seismic Category I structures are designed for the wind and tornado loads described in Section 2.3.1.2 as applicable. The reinforced concrete portions of other Seismic Category I structures and foundations are designed for the following load combinations and allowable limits in addition to those given in Section 3.8.3.3:

$$U = (0.75)\ (1.4D + 1.7L + 1.7W) \qquad (3.8\text{-}1)$$

$U = 0.9D + 1.3W$ (3.8-2)

$U = 1.0D + 1.0L = 1.0W$ (3.8-3)

$U = 0.9D + 1.0W_t$ (3.8-4)

$U = 1.0D + 1.0F$ (3.8-5)

where:

U = Required section strength based on strength design methods, as described in ACI 318-71,

D = Dead loads or their related internal moments and forces, including any permanent equipment loads, include hydrostatic pressure in accordance with Section 9.3.5 of ACI 318-71,

L = Live loads or their related internal moments and forces, including any movable equipment loads and other loads which vary with intensity and occurrence, includes earth pressure in accordance with Section 9.3.4 of ACI 318-71,

W = Loads generated by the design wind,

W_t = Loads generated by the design tornado, including loads due to the tornado wind pressure, the tornado-created differential pressure, and to tornado-generated missiles,

F = Maximum flood or precipitation load.

In addition to the loading criteria given in Section 3.8.3.3, other Seismic Category I steel structures are designed for the following load combinations and allowable limits:

1. $1.33S = D + W + L$ (3.8-6)

2. $1.6S = D + W_t$ (3.8-7)

where:

S = The required section strength based on the elastic working stress methods and allowable stresses defined in Part 1 of the AISC (1969) Specification referenced in Section 3.8.1.2.

3.8.4.4 Design and Analysis Procedures

Design and analysis procedures conform to the requirements of ACI 318-71 and the AISC Specification for the design, fabrication,

and erection of steel structures, except as stated in Section 3.8.4.5. If biaxial tension fields exist, splice and anchorage requirements comply with ACI 318.

Earthquake forces on the structures are determined by dynamic analysis and then applied statically in the design of the structures. The analytical techniques used to determine the forces are given in Section 3.7B.2.

For the turbine building, dynamic analyses as described in Section 3.7B.2 are performed for independent plane frames at four column lines, two oriented north-south, and two oriented east-west. The masses of appropriate tributary areas are taken into consideration for each frame analyses. Because the earthquake forces calculated are substantially less than the forces calculated for tornado loading on the respective frames, no further seismic analysis is required.

Concrete structures are supported by their foundations. The reactions obtained from the analyses of these structures are applied as loads to the foundations.

Steel structures are considered to be supported by the concrete structures or foundations to which they are anchored. The reactions obtained from the analysis of a steel structure are applied as loads to the supporting concrete.

Material quality control procedures (Section 3.8.4.6) assure that the material supplied has the required strength.

3.8.4.5 Structural Acceptance Criteria

Steel structures are designed for the loads, loading combinations and strength limits given in Sections 3.8.3.3 and 3.8.4.3.

Concrete structures are designed by the strength design method and conform to the requirements of ACI 318-71. Load factors are as given in Section 3.8.4.3. The basic criterion for strength design is expressed as:

Required Strength ≤ Calculated Strength

Concrete members and sections of concrete members are proportioned to meet this criterion. The required strength is expressed in terms of design loads or their related internal moments and forces. Design loads are defined as loads that have been multiplied by their appropriate load factors. Calculated strength is that computed by the provisions of ACI 318, including the appropriate capacity reduction factors. Capacity reduction factors are given in Section 9.2 of ACI 318.

The loading criteria given in Section 3.8.4.3 also specify the allowable limits which constitute the structural acceptance criteria.

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

Materials and quality controls used for Seismic Category I structures other than the containment are described in Sections 3.8.1.2, 3.8.1.6, and 3.8.3.6. The 28-day minimum compressive strength of the concrete is predominantly 3,000 psi in these structures, with 4,000 and 5,000 psi being used in a limited number of areas. No special construction techniques are used for the structures described in Section 3.8.4.1.

3.8.4.7 Testing and In-service Inspection Requirements

No full-scale structural testing or in-service inspection is anticipated for the structures described Section 3.8.4.1. Materials used in the construction are tested, as described in Section 3.8.4.6.

3.8.4.8 Masonry Walls

BVPS-2 has no safety-related masonry walls. In addition, BVPS-2 has no nonsafety-related masonry walls located in those portions of structures classified as Seismic Category I, except in the auxiliary building at el 710 ft-6 in. and 773 ft-6 in. The block walls at el 710 ft-6 in. are only two feet high and an analysis was performed to verify their structural integrity for a seismic event. Analysis verified that failure of the block walls at el 773 ft-6 in. will not adversely affect the steel framing at this elevation, and will not impact any safety-related equipment. Seismic and QA classification of structures is provided in Table 3.2-2.

3.8.5 Foundations

3.8.5.1 Description of the Foundations

Foundations for Seismic Category I structures consist of reinforced concrete mats designed for service on soil. Section 2.5 defines the geological and seismic design parameters. There are no unique foundations or concrete support features used in the design.

Sufficient space is provided between various Seismic Category I foundations to prevent contact during an SSE.

3.8.5.1.1 Containment Structure Foundation

The containment structure is supported on a flat, circular, reinforced concrete mat (Section 3.8.1.1) 10 feet thick and 142 feet in diameter. It is reinforced with both top and bottom layers of reinforcing steel as described in Section 3.8.1.1.

The containment structure is founded approximately 50 feet below plant grade. Lateral loads and forces are transmitted to the soil in bearing over the vertical area of the engaged portion of the structure. Lateral stability is not dependent on the transfer of horizontal shear through the waterproofing membrane.

3.8.5.1.2 Foundation For Other Seismic Category I Structures

The foundations for other Seismic Category I structures are concrete mats that vary in thickness depending on the requirements for shear and bending moment. Reinforcing is provided in two directions at both top and bottom faces. The principal reinforcement in the foundations consists of No. 11 bars. Lateral loads and forces are transmitted to the founding soil by friction.

Sections of the foundations of other Seismic Category I structures, as well as the relationship between the various foundations, are shown on the figures listed in Table 3.8-12.

3.8.5.2 Applicable Codes, Standards, and Specifications

The design codes, standards, and specifications for foundations and concrete supports are given in Sections 3.8.1.2 (for containment foundation) and 3.8.4.2 (for other Seismic Category I foundations).

3.8.5.3 Loads and Load Combinations

The loads and load combinations used in the design of the foundations are the same as those used in the design of the superstructures, as discussed in Sections 3.8.1.3 (for containment foundation) and 3.8.4.3 (for other Seismic Category I foundations).

3.8.5.4 Design and Analysis Procedures

The containment structure foundation is analyzed as a part of the analysis of the entire containment structure (Section 3.8.1.4).

Other Seismic Category I structure foundations are analyzed as a part of the entire structure analyses (Section 3.8.4.4). In the analysis of these foundations, the founding soil is assumed to behave in a uniformly elastic manner.

The predicted static and dynamic settlement of Seismic Category I structures is discussed in Section 2.5.4. The effect of the predicted settlement on these structures is insignificant.

3.8.5.5 Structural Acceptance Criteria

Seismic Category I foundations are designed as integral parts of the structures they support. Structural acceptance criteria are given in Sections 3.8.1.5 (for the containment structure) and 3.8.4.5 (for the other Seismic Category I structures).

Differential settlement of Seismic Category I structures is discussed in Section 2.5.4.

Factors of safety against sliding and overturning are presented in Table 3.8-13.

3.8.5.6 Materials, Quality Control, Special Construction Techniques

Sections 3.8.1.2 and 3.8.1.6 describe materials, quality control, and specifications for the containment structure foundation.

Materials, quality control, and specifications for foundations of other Category I structures are described in Sections 3.8.3.6, 3.8.4.6, and 3.8.5.2.

There are no special construction techniques required by the design.

The Quality Assurance Program is described in Section 17.1.

3.8.5.7 Testing and Inservice Inspection Requirements

The entire containment structure, including its foundation, undergoes structural acceptance testing (Section 3.8.1.7). Except for this test, no other testing or inservice inspection of foundation systems is planned.

3.8.6 References for Section 3.8

ACI Publication SP-16, Computer Application in Concrete Design and Technology, Article: Application of Computers in the Evaluation of Quality Control of Concrete.

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TABLE 3.8-1

AMERICAN CONCRETE INSTITUTE
CODES, STANDARDS, AND SPECIFICATIONS*

Code Number	Title
ACI 211.1-70	Recommended Practice for Selecting Proportions for Normal Weight Concrete
ACI 214-65	Recommended Practice for Evaluation of Compression Test Results of Field Concrete
ACI 301-72	Specification for Structural Concrete for Buildings
ACI 304-73	Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete
ACI 305-77	Recommended Practice for Hot Weather Concreting
ACI 306-72	Recommended Practice for Cold Weather Concreting
ACI 309-72	Recommended Practice for Consolidation of Concrete
ACI 318-71	Building Code Requirements for Reinforced Concrete
ACI 347-68	Recommended Practice for Concrete Formwork

NOTE:

*Codes, standards, and specifications are applicable as discussed in Section 3.8.

TABLE 3.8-2

AMERICAN SOCIETY OF MECHANICAL ENGINEERS
MATERIAL SPECIFICATIONS[1,2]

ASME Specification Number	Title
SA-105	Specifications for Forged or Rolled Steel Pipe Flanges, Forged Fittings, and Valves and Parts for High-Temperature Service
SA-182	Specification for Forged or Rolled Alloy Steel Pipe Flanges, Forged Fittings, and Valves and Parts for High-Temperature Service
SA-193	Specification for Alloy Steel Bolting Materials for High-Temperature Service
SA-194	Specification for Carbon and Alloy Steel Nuts for Bolts for High-Pressure and High-Temperature Service
SA-213	Specification for Seamless Ferritic and Austenitic Alloy Steel Boiler, Superheater, and Heat Exchanger Tubes
SA-234	Specification for Piping Fittings of Wrought Carbon Steel and Alloy Steel for Moderate and Elevated Temperatures
SA-240	Specification for Stainless and Heat-Resisting Chromium and Chromium-Nickel Steel Plate, Sheet, and Strip for Fusion-Welded Unfired Pressure Vessels
SA-312	Specification for Seamless and Welded Austenitic Stainless Steel Pipe
SA-333	Specification for Seamless and Welded Steel Pipe for Low-Temperature Service
SA-350	Specification for Forged or Rolled Carbon and Alloy Steel Flanges, Forged Fittings, and Valves and Parts for Low-Temperature Service
SA-358	Specification for Electric-Fusion-Welded Austenitic Chromium-Nickel Alloy Steel Pipe for High-Temperature Service
SA-403	Specification for Wrought Austenitic Stainless Steel Piping Fittings

TABLE 3.8-2 (Cont)

ASME Specification Number	Title
SA-508	Specification for Quenched and Tempered Vacuum Treated Carbon and Alloy Steel Forgings for Pressure Vessels
SA-516	Specifications for Carbon Steel Plates for Pressure Vessels for Moderate and Lower Temperature Service
SA-537	Specification for Carbon-Manganese-Silicon Steel Plates, Heat Treated, for Pressure Vessels

NOTES:

1. Boiler and Pressure Vessel Code, Section III, Material Specifications, Part A-Ferrous.
2. Codes and specifications are applicable as discussed in Section 3.8.

TABLE 3.8-3

AMERICAN NATIONAL STANDARDS INSTITUTE
CODES, STANDARDS, AND SPECIFICATIONS*

Code Number	Title
ANSI N512-1974	Protective Coatings (Paints) for the Nuclear Industry
ANSI N45.2.2-1972	Packaging, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants
ANSI N45.2.5-1974	Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction of Nuclear Power Plants
ANSI N45.4-1972	Leakage Rate Testing of Containment Structures for Nuclear Reactors
ANSI/ANS 56.8-1987	Containment System Leakage Testing Requirements (This document used only as a guideline.)
ANSI N101.4-1972	Quality Assurance for Protective Coatings Applied to Nuclear Facilities

NOTE:

*Codes, standards, and specifications are applicable as discussed in Section 3.8.

TABLE 3.8-4

AMERICAN SOCIETY FOR TESTING AND MATERIALS SPECIFICATIONS*

Specification Number	Title
ASTM A 29-67	Specification for General Requirements for Steel Bar, Carbon and Alloy, Hot Rolled and Cold Finished
ASTM A 36-72	Specification for Structural Steel
ASTM A 105-71	Specifications for Forged or Rolled Steel Pipe Flanges, Forged Fittings, and Valves and Parts for High-Temperature Service
ASTM A 106-71	Specification for Seamless Carbon Steel Pipe for High-Temperature Service
ASTM A 123-71	Specification for Zinc Hot-Galvanized Coatings on Products Fabricated from Rolled, Pressed, and Forged Steel Shapes, Plates, Bars, and Strip
ASTM A 131-71	Specification for Structural Steel for Ships
ASTM A 153-73	Specification for Zinc-Coating (Hot-Dip) on Iron and Steel Hardware
ASTM A 182-72	Specification for Forged or Rolled Alloy-Steel Pipe Flanges, Forged Fittings, and Valves and Parts for High-Temperature Service
ASTM A 184-72	Specification for Fabricated Deformed Steel Bar Mats for Concrete Reinforcement
ASTM A 193-71	Specification for Alloy-Steel and Stainless Steel Bolting Materials for High-Temperature Services
ASTM A 194-71	Specification for Carbon and Alloy Steel Nuts for Bolts for High-Pressure and High-Temperature Service
ASTM A 240-72	Specification for Heat-Resisting Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Fusion-Welded Unfired Pressure Vessels
ASTM A 242-70A	Specification for High-Strength Low-Alloy Structural Steel

TABLE 3.8-4 (Cont)

Code Number	Title
ASTM A 262-70	Standard Recommended Practices for Detecting Susceptibility to Intergranular Attack in Stainless Steels
ASTM A 276-72	Standard Specification for Stainless and Heat-Resisting Steel Bars and Shapes
ASTM A 307-68	Specification for Carbon Steel Externally and Internally Threaded Fasteners
ASTM A 325-71A	Specification for High Strength Bolts for Structural Steel Joints Including Suitable Nuts and Plain Hardened Washers
ASTM A 384-62	Recommended Practice for Safeguarding Against Warpage and Distortion During Hot-Dip Galvanizing of Steel Assemblies
ASTM A 385-62	Recommended Practice for Providing High Quality Zinc Coatings (Hot-Dip)
ASTM A 441-70	Specification or High-Strength Low-Alloy Structural Manganese-Vanadium Steel
ASTM A 490-71	Specification for Quenched and Tempered Alloy Steel Bolts for Structural Steel Joints
ASTM A 500-74	Standard Specification for Cold-Formed Welded and Seamless Carbon Steel Structural Tubing in Rounds and Shapes
ASTM A 501-74	Standard Specification for Hot-Formed Welded and Seamless Carbon Steel Structural Tubing
ASTM A 515-74	Specification for Pressure Vessel Plates, Carbon Steel, for Intermediate- and High-Temperature Service
ASTM A 516-72	Specification for Pressure Vessel Plates, Carbon Steel, for Moderate and Higher-Temperature Service
ASTM A 543-74	Specification for Pressure Vessel Plates, Alloy Steel, Quenched and Tempered Nickel-Chromium-Molybdenum
ASTM A 572-72	Specification for High-Strength Low-Alloy Columbium-Vanadium Steels of Structural Quality

TABLE 3.8-4 (Cont)

Code Number	Title
ASTM A 588-71	Specification for High-Strength Low-Alloy Structural Steel with 50,000 psi Minimum Yield to Four Inches Thick
ASTM A 615-68	Specification for Deformed and Plain Billet-Steel Bars for Concrete Reinforcement
ASTM A 706-74	Specification for Low-Alloy Steel Deformed Bars for Concrete Reinforcement
ASTM C 42-68	Obtaining and Testing Drilled Cores and Sawed Beams of Concrete
ASTM C 33-71	Specification for Concrete Aggregates
ASTM C 87-69	Test for Effect of Organic Impurities in Fine Aggregates on Strength of Mortar
ASTM C 88-73	Test for Soundness of Aggregates by Use of Sodium Sulfate or Magnesium Sulfate
ASTM C 94-74	Specification for Ready-Mixed Concrete
ASTM C 109-73	Test for Compressive Strength of Hydraulic Cement Mortars (Using 2 in Cube Specimens)
ASTM C 114-69	Chemical Analysis of Hydraulic Cement
ASTM C 115-73	Test for Fineness of Portland Cement by the Turbidimeter
ASTM C 117-69	Test for Materials Finer than No. 200 Sieve in Mineral Aggregates by Washing
ASTM C 123-69	Test for Lightweight Pieces in Aggregate
ASTM C 127-73	Test for Specific Gravity and Absorption of Coarse Aggregate
ASTM C 128-68	Test for Specific Gravity and Abosrption of Fine Aggregate
ASTM C 131-69	Test for Resistance to Abrasion of Small Size Coarse Aggregate by Use of the Los Angeles Machine
ASTM C 136-71	Test for Sieve or Screen Analysis of Fine and Coarse Aggregates

TABLE 3.8-4 (Cont)

Code Number	Title
ASTM C 138-74	Test for Unit Weight, Yield, and Air Content (Gravimetric) of Concrete
ASTM C 142-71	Test for Clay Lumps and Friable Particles in Aggregates
ASTM C 143-71	Test for Slump of Portland Cement Concrete
ASTM C 150-74	Specification for Portland Cement
ASTM C 151-74	Test for Autoclave Expansion of Portland Cement
ASTM C 172-71	Sampling Fresh Concrete
ASTM C 186-73	Test for Heat of Hydration of Hydraulic Cement
ASTM C 191-71	Test for Time of Setting of Hydraulic Cement by Vicat Needle
ASTM C 192-69	Making and Curing Concrete Test Specimens in the Laboratory
ASTM C 204-74	Test for Fineness of Portland Cement by Air Permeability Apparatus
ASTM C 227-71	Test for Potential Alkali Reactivity of Cement-Aggregate Combinations (Mortar Bar Method)
ASTM C 231-74	Test for Air Content of Freshly Mixed Concrete by the Pressure Method
ASTM C 233-73	Specification for Testing Air-entraining Admixtures in Concrete
ASTM C 235-68	Test for Scratch Hardness of Coarse Aggregate Particles
ASTM C 260-73	Specification for Air - Entraining Admixtures for Concrete
ASTM C 266-70	Test for Time of Setting of Hydraulic Cement by Gillmore Needles
ASTM C 289-71	Test for Potential Reactivity of Aggregates (Chemical Method)

TABLE 3.8-4 (Cont)

Code Number	Title
ASTM C 295-65	Recommended Practice for Petrographic Examination of Aggregates for Concrete
ASTM C 311-68	Standard Methods of Sampling and Testing Fly Ash for Use as an Admixture in Portland Cement Concrete
ASTM C 451-72	Test for False Set of Portland Cement (Paste Method)
ASTM C 494-71	Specification for Chemical Admixtures for Concrete
ASTM C 535-69	Test for Resistance to Abrasion of Large Size Coarse Aggregate by Use of the Los Angeles Machine
ASTM C 566-67	Test for Total Moisture Content of Aggregate by Drying
ASTM C 567-71	Test for Unit Weight of Structural Lightweight Concrete
ASTM C 586-69	Test for Potential Alkali Reactivity of Carbonate Rocks for Concrete Aggregates (Rock Cylinder Method)
ASTM C 618-72	Specification for Fly Ash and Raw Calcined National Pozzo Laws for Use in Portland Cement Concrete
ASTM C 637-73	Specification for Aggregate for Radiation-Shielding Concrete
ASTM C 642-69	Test for Specific Gravity, Absorption, and Voids and Hardened Concrete
ASTM C 666-71	Test for Resistance of Concrete to Rapid Freezing and Thawing
ASTM D 75-71	Sampling Aggregates
ASTM D 512-67	Tests for Chloride Ion in Water and Waste Water
ASTM D 994-71	Specification for Preformed Expansion Joint Filler for Concrete (Bituminous Type)
ASTM D 1752-67	Specification for Preformed Sponge Rubber and Cork Expansion Joint Fillers for Concrete Paving and Structural Construction
ASTM D 2842-69	Method of Tests for Water Absorption of Rigid Cellular Plastics

TABLE 3.8-4 (Cont)

Code Number	Title
ASTM E 165-71	Recommended Practice for Liquid Penetrant Inspection Method
ASTM E 208-69	Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels
ASTM F 436-76b	Specification for Hardened Steel Washers for Use with High-Strength Bolts

NOTE:

*Codes, standards, and specifications are applicable as discussed in Section 3.8.

TABLE 3.8-5

USNRC REGULATORY GUIDES*

Regulatory Guide Number	Title
1.10	Mechanical (Cadweld) Splices in Reinforcing Bars for Category I Concrete Structures
1.15	Testing of Reinforcing Bars for Category I Concrete Structures
1.17	Protection of Nuclear Power Plant Against Industrial Sabotage
1.18	Structural Acceptance Test for Concrete Primary Reactor Containments
1.19	Nondestructive Examination of Primary Containment Liner Welds
1.29	Seismic Design Classification
1.31	Control of Ferrite Content in Stainless Steel Weld Metal
1.44	Control of the Use of Sensitized Stainless Steel
1.46	Protection Against Pipe Whip Inside Containment
1.54	Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants
1.55	Concrete Placement in Category I Structures
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants
1.61	Damping Values for Seismic Design of Nuclear Power Plants
1.63	Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants
1.69	Concrete Radiation Shields for Nuclear Power Plants
1.76	Design Basis Tornado for Nuclear Power Plants

TABLE 3.8-5 (Cont)

Regulatory Guide Number	Title
1.84	Design and Fabrication Code Case Acceptability - ASME Section III, Division 1
1.85	Materials Code Case Acceptability - ASME Section III, Division 1
1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis
1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants
1.117	Tornado Design Classification
1.122	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components

NOTE:

*USNRC Regulatory Guides are discussed in Section 1.8.

TABLE 3.8-6

PRINCIPLE PLANT STRUCTURAL SPECIFICATIONS
FOR SEISMIC CATEGORY I MATERIALS*

Seismic Category I Materials	Specifications
Furnish Reinforcing Steel	
Reinforcing Bars	ASTM A 615, Grade 40, 50 Special Chemistry, and 60, and ASTM A 706
Bar and Rod Mats	ASTM A 184
Detailing and Fabrication	ACI 315 and ACI 318
Furnish Radial Shear Bar Assemblies	
Inclined Flat Shear Bars	ASTM A 242 Type 2, ASTM A 441, ASTM A 572, or ASTM A 588 having a minimum yield strength of 50 ksi
Reinforcing Bars	ASTM A 615, Grade 50, and ASTM A 29
Filler Metal for Welding	AWS D1.1 Table 4.1-1, low hydrogen
Fabrication	AWS D1.1 and AWS D12.1
Mixing and Delivering Concrete Materials	
Portland Cement	ASTM C 150, Type II
Fly Ash	ASTM C 618, Class F (except that the loss of ignition shall not be more than 6 percent and available alkalies as Na O shall not exceed 1.5 percent)
Air Entraining Agent	ACI 301, ASTM C 233, ASTM C 260
Water Reducing Agent	ASTM C 494
Aggregates	ASTM C 33, ASTM C 227, ASTM C 289
Proportioning Concrete	ACI 301, ACI 304, ACI 211.1
Batching and Mixing Concrete	ASTM C 94, ACI 301, ACI 304
Delivery	ACI 301, ACI 304
Hot Weather Requirements	ACI 305
Cold Weather Requirements	ACI 306

TABLE 3.8-6 (Cont)

Seismic Category I Materials	Specifications
Concrete Testing Services	
Gather Aggregate Samples	ASTM D 75
Test Water and Ice	ASTM C 109, ASTM C 151, ASTM C 191
Test Fine Aggregates	ASTM C 136, ASTM C 29, ASTM C 40, ASTM C 142, ASTM C 117, ASTM C 123, ASTM C 227, ASTM C 289, ASTM C 128, ASTM C 88, ASTM C 295, ASTM C 33, ASTM C 87
Test Coarse Aggregates	ASTM C 136, ASTM C 29, ASTM C 131, ASTM C 535, ASTM C 88, ASTM C 227, ASTM C 289, ASTM C 127, ASTM C 142, ASTM C 117, ASTM C 123, ASTM C 295, ASTM C 33, ASTM C 235, CRD C 119
Test Cement	ASTM C 150
Test Calcium Aluminate Cement	ASTM C 114, ASTM C 115, ASTM C 109, ASTM C 191, ASTM C 204
Design Concrete Mixes	ACI 211.1
Test Concrete	
Compressive Strength	ACI 301, ASTM C 31, ASTM C 39
Air Dry Weight	ASTM C 567
Slump	ASTM C 143
Air Content	ASTM C 231
Unit Weight	ASTM C 138
Fly Ash	ASTM C 311, ASTM C 618
Hardened Concrete	ASTM C 42
Test Grout	ASTM C 109
Evaluation of Concrete Strength	ACI 301, ACI 214
Placing Concrete and Reinforcing Steel	
Cadweld Splices	See Section 3.8.1.6
Welded Splices	See Section 3.8.1.6
Formworks	ACI 301, ACI 347
Placing Reinforcement	ACI 301, ACI 318
Concrete Protection for Reinforcing	ACI 318
Concrete Construction, Expansion, and Contraction Joints	ACI 301

TABLE 3.8-6 (Cont)

Seismic Category I Materials	Specifications
Water Stops	CRD C 513 (except that ozone resistance requirements do not apply), CRD C 572
Anchor Bolts and Miscellaneous Steel	ASTM A 307, ASTM A 36, ASTM A 193, ASTM A 194, 1 and AISC Specifications
Inserts, Sleeves, and Pipes	ACI 301, ACI 318, AISC Specifications
Concrete Placing	ACI 301, ACI 304, ACI 305, ACI 306, ACI 318
Cold Weather Requirements	ACI 306
Hot Weather Requirements	ACI 305
Vibration of Concrete	ACI 309
Finishing of Concrete Lift Surfaces	ACI 301
Depositing Underwater	ACI 301
Watertight Concrete	ACI 301
Repair of Surface Defects	ACI 301
Finishing of Formed and Flat Surfaces	ACI 301
Curing Materials	ASTM C 171, ASTM C 309 - Type I
Structural Steel	
Structural Steel	ASTM A 36
Filler Metal	AWS D1.1 Table 4.1-1, Low Hydrogen
Bolts, Nuts, and Washers	ASTM A 325 - Type I, ASTM A 490, ASTM A 307
Welding	AWS D1.1
Miscellaneous Steel	
Steel	ASTM A 36, ASTM A 515
Bolts, Nuts, Washers	ASTM A 307 - Grade B, ASTM A 325
Glavanizing	ASTM A 123, ASTM A 153, ASTM A 384, ASTM A 385
Welded Studs	AWS D1.1
Pipe Handrails	ASTM A 106- Grade B Schedule 40, ASTM A 501 Schedule 40
Ladders	ASTM A 36

TABLE 3.8-6 (Cont)

Seismic Category I Materials	Specifications
Category I Concrete Embedments	
Steel	ASTM A 36, ASTM A 242, ASTM A 516, Grade 70, ASTM A 543 Class 1
Bolts, Nuts, and Washers	ASTM A 36, ASTM A 307, ASTM A 325, ASTM A 490, ASTM A 193 Grade B7 ASTM A 588, ASTM A 194 Grade 7, ASTM F 436
Welded Studs	AWS D1.1
Stainless Steel	ASTM A 240 Type 304, ASTM A 276, Type 304

NOTE:

*Specifications are applicable as discussed in Section 3.8.

TABLE 3.8-7

STEEL LINER, PENETRATIONS, AND ACCESS OPENING CODES STANDARDS, AND SPECIFICATIONS*

Code	Description
ASME Boiler and Pressure Vessel Code	Section III, Division I; Section II, VIII, V, and IX
IEEE Standard 317-1976	Electrical Penetration Assemblies in Containment Structures for Nuclear Fueled Power Generating Stations. Institute of Electrical and Electronics Engineers.
10 CFR 50 Appendix J	"Reactor Containment Leakage Testing for Water Cooled Power Reactors." Details of Type A tests (performed on liner plate) and Type B tests (performed on electrical penetration and access openings) are presented in Section 6.2.6.

NOTE:

*Codes, standards, and specifications are discussed in Section 3.8.

TABLE 3.8-8

SUMMARY OF MATERIALS[1] CALLED FOR
IN SPECIFICATIONS FOR
STEEL LINER, PENETRATIONS, AND ACCESS OPENINGS[2]

Item	Specifications[1]
Liner plates (1/4-inch to 1 1/4-inch inclusive)	SA537, Grade B, quenched and tempered, NDTT not higher than -10°F
Backing plates, shear lugs, bridging plates	SA516, Grade 60, fine grained, normalized, NDTT not higher than -10°F
Piping penetration sleeves without coolers	SA333, Grade 6, or SA516, Grade 60, fine grained, normalized, NDTT not higher than -10°F
Penetrations 73, 74, 75 sleeves	SA537, Grade B quenched and tempered, NDTT not higher than -10°F
Piping penetration sleeves with coolers	SA312, Type 304, seamless or welded or SA240, Type 304 or SA358, Type 304
Piping penetration coolers	SA312, Type 304, welded or SA358, Type 304
Piping penetration cooler end rings	SA240, Type 304
Carbon steel pipe and electrical penetration sleeves	SA333, Grade 6, fine grained, normalized, NDTT not higher than -10°F
Stainless steel pipe	SA312, Type 304 or SA312, Type 316
Pipe penetration end caps 4-inch diameter and under	SA105, Grade II, normalized
Pipe penetration end caps over 4-inch diameter	SA234, Grade WPB, normalized
Gas testing channels and angles	ASTM-A131, Grade C
Equipment hatch, emergency air lock, and personnel air lock, plate over 2 1/2 inches thick	SA516, Grade 60 or Grade 70, fine grained normalized, NDTT not higher than -10°F

TABLE 3.8-8 (Cont)

Item	Specifications[1]
Equipment hatch, emergency air lock, and personnel air lock plate, 2 1/2 inches thick and under	SA537, Grade B, quenched and tempered, NDTT not higher than -10°F or SA516, Grade 60 or Grade 70, fine grained normalized, NDTT not higher than -10°F
Carbon steel forgings and embedments	SA350, Grade LF1 or LF2, NDTT not higher than -10°F; acceptable alternate material electro-slag remelt processed steel (SA516 GR 60 or SA-537 GR B) provided that ultrasonic examination is added to the basic test requirements of the respective ASME II material specifications
Cooler tubing	SA213, Type 304
Stainless steel reducers and pipe caps for bottom embedments	SA403, Type 304
Stainless steel couplings, plugs and fittings for leak chase	A182, Type 304
Carbon steel couplings, plugs and fittings for leak chase	A105 Grade I or II
Equipment hatch bolts	SA193, GR B7
Equipment hatch nuts	SA-194-2H, SA-194-7
Equipment hatch, emergency air lock attachment, and personnel air lock gasket	Type "W" Neoprene, 40 ±5 Shore A Deg or Parker silicone rubber compound S-595-5 or S-595-65±5 with radiation certification or EPDM compound No. E401 40 ±5 durometer
Integral Forgings (Pen No. 39, 40, 41, 52, 73, 74, 75, 76, 77, 78)	SA508, Gr 1, Code Case 1332-6, NDTT requirements, Charpy V-Notch of -10°F and DWT per ASTM E208 of +10°F. Test temp for DWT to be +20°F
Integral forgings (Pen. No. 28, 51)	SA282, Type 304

NOTES

(1) Materials listed in this table may have been replaced with materials of equivalent design characteristics. The term equivalent is described in UFSAR Section 1.12, "Equivalent Materials".
(2) Specifications are applicable as discussed in Section 3.8.

TABLE 3.8-9

LOAD COMBINATIONS FOR LINER PLATE AND ANCHORS

Category	Load Combinations	Stress Allowables (Per ASME III Nomenclature)
Emergency	D + Pd + Td + SSE	Pm + Pb + Q<3 Sm
Test	D + 1.15 Pd	Pm <0.9 Sy Pm + Pb <1.35 Sy
Anticipated (Cyclic)	150 c of Po 600 c of T 150 c of 1/2 SSE	Use method of paragraph NB-3222.4 (d) or (e)
Severe Operational	D + Pmin + 1/2 SSE D + Pmin + 1/2 SSE D + Pmin + Tmin + 1/2 SSE	Pm < Sm Pm + Pb < 1.5 Sm Pm + Pb + Q <3 Sm
	Load Combinations for Liner Plate Anchors	
Emergency	D + Pd + Td	$(P/P')^2 + (S/S')^2 < 1$
Severe Operational	D + Pmin + Tmin	$(P/P')^2 + (S/S')^2 < 1$

where:

D = Dead load effect of reinforced concrete structure acting on the liner plus dead load of the liner.

Pd = Maximum design pressure (pressure resulting from design basis accident plus safety margin).

Td = Load due to thermal expansion resulting when the liner is exposed to the design temperature.

SSE = Stresses in the liner derived from applying the effect of the safe shutdown earthquake.

1/2 SSE = Stresses in the liner derived from applying the effect of the 1/2 safe shutdown earthquake (which is equivalent to one OBE). The anticipated number of cycles, one hundred fifty (150) cycles of 1/2 SSE are an assumed number of earthquake cycles based on a 60 year span.

Po = Differential pressure between operating and atmospheric pressure. The anticipated number of cycles, one hundred fifty (150) cycles are assumed on the basis of 2.5 refueling cycles per year on a 60-year span. The design limit for operating cycles is 1000 cycles.

TABLE 3.8-9 (Cont)

T	=	Load due to thermal expansion resulting when the liner is exposed to the differential temperature between operating and seasonal refueling temperatures. Six hundred cycles are assumed on the basis of 10 such variations per year on a 60 year span. The design limit for operating temperature variations is 4000 cycles.
Pmin	=	Minimum pressure resulting during operation of the containment.
Tmin	=	Load due to thermal expansion resulting when the liner is exposed to the minimum pressure.
Sy	=	Yield strength of the material.
Sm	=	Basic allowable stress from ASME III.
P,S	=	Applied loads on anchors - tension and shear.
P’,S’	=	Allowable anchor loads - tension and shear.

TABLE 3.8-10

ASTM A615 REINFORCING STEEL
GRADE 50, STONE & WEBSTER
SPECIAL CHEMISTRY
CHEMICAL AND MECHANICAL PROPERTIES

Properties	Values
Carbon (percent by weight)	0.35 maximum
Manganese (percent by weight)	1.25 maximum
Silicon (percent by weight)	0.15 - 0.25
Phosphorus (percent by weight)	0.05 maximum
Sulfur (percent by weight)	0.05 maximum
Tensile strength (psi)	70,000 minimum
Yield strength (psi)	50,000 minimum
Elongation in 8 inches (percent)	13 minimum

NOTE:

*The permissible variation in chemical properties for check analysis is provided in ASTM A29, Table 1.

TABLE 3.8-11

CONTAINMENT INTERNAL STRUCTURE
CONCRETE STRUCTURES LOAD COMBINATIONS COMPARISON

Load Combination	SRP Section 3.8.3* Factors												
	D	L	To	Ro	E	E′	Pa	Ta	Ra	Yr	Yj	Ym	
1	1.4	1.7											
2	1.4	1.7			1.9								
1b	1.05	1.275	1.275	1.275									
2b	1.05	1.275	1.275	1.275	1.425								
3	1.0	1.0	1.0	1.0		1.0							
4	1.0	1.0					1.5	1.0	1.0				
5	1.0	1.0			1.25		1.25	1.0	1.0	1.0	1.0	1.0	
6	1.0	1.0				1.0		1.0	1.0	1.0	1.0	1.0	1.0

	BVPS-2 Load												
	D	L	To	Ro	E	E′	Pa	Ta	Ra	Yr	Yj	Ym	Cl
1	1.05												1.275
2	1.4	1.7			1.9								
3	0.9				1.4								
4	1.0	1.0				1.0							
5	0.9					1.0							
6	1.0	1.0					1.5	1.0	1.0				
7	1.0	1.0			1.25		1.25	1.0	1.0	1.0	1.0	1.0	
8	1.0	1.0				1.0	1.0	1.0	1.0	1.0	1.0	1.0	

NOTE:

*U.S. Nuclear Regulatory Commission 1975.

TABLE 3.8-12

SEISMIC CATEGORY I STRUCTURES AND EQUIPMENT
FOR OTHER THAN THE CONTAINMENT STRUCTURE

Figure Description	Figure Number
Plot Plan	3.8-8
Plant Arrangement at Plan El 735'-6"	3.8-25
Plant Arrangement at Plan El 752'-6"	3.8-26
Plant Arrangement at Plan El 760'-6"	3.8-27
Plant Arrangement at Plan El 774'-6"	3.8-28
Plant Arrangement Part Plan	3.8-29
Auxiliary Building Arrangement* Plant El 710'-6" and 718'-6"	3.8-30
Auxiliary Building Arrangement* Plan El 735'-6"	3.8-31
Auxiliary Building Arrangement* Plan El 755'-6"	3.8-32
Auxiliary Building Arrangement* Plan El 773'-6"	3.8-33
Auxiliary Building Arrangement* Sections 1-1 and 2-2	3.8-34
Auxiliary Building Arrangement* Sections 3-3 and 4-4	3.8-35
Auxiliary Building Arrangement* Section 5-5, Plan El 710'-6"	3.8-36
Safeguards Area Plan El 769'-0" and 758'-0"	3.8-1
Safeguards Area Plan El 741'-0"	3.8-2
Safeguards Area Plan El 718'-6"	3.8-3
Safeguards Area Section 1-1	3.8-4

TABLE 3.8-12 (Cont)

Figure Description	Figure Number
Main Steam Valve and Cable Vault Area** Plan El 773’-6"	3.8-1
Main Steam Valve and Cable Vault Area** Plan El 735’-6"	3.8-2
Main Steam Valve and Cable Vault Area** Plan El 718’-6"	3.8-3
Main Steam Valve and Cable Vault Area** Sections 2-2 and 5-5	3.8-6
Main Steam and Cable Vault Area** Plan El 755’-6", 798’-0", 808’-6"	3.8-37
Fuel and Decontamination Building Arrangement - Sheet 1	9.1-1
Fuel and Decontamination Building Arrangement - Sheet 2	9.1-2
Control Building Plan El 735’-6"	3.8-40
Control Building Plan El 707’-6"	3.8-41
Control Building Section 103 and 106	3.8-42
Diesel Generator Building Arrangement	3.8-43
Service Building*** Plan El 745’-6" and 730’-6"	3.8-44
Service Building*** Plan El 760’-6" and 780’-6"	3.8-45
Service Building*** Sections 1-1 and 2-2	3.8-46
Electrical Cable Tunnel	3.8-47
Service Water Valve Pit	3.8-48

TABLE 3.8-12 (Cont)

Figure Description	Figure Number
Primary Demineralized Water Tank Enclosure	3.8-1 3.8-2
Pipe Trench for BVPS-1 and BVPS-2 Crosstie Piping - Sheet 1 (non-Seismic Category I)	3.8-48
Pipe Trench for BVPS-1 and BVPS-2 Crosstie Piping - Sheet 2 (non-Seismic Category I)	3.8-49
Emergency Out Fall Structure	3.8-50
Equipment Hatch Platform	3.8-1
Waste Handling Building Arrangement (non-Seismic Category I)	3.8-51
Condensate Polishing Building (non-Seismic Category I) Plan El 722'-6" and 735'-6"	3.8-52
Condensate Polishing Building (non-Seismic Category I) Plan El 752'-6" and 774'-6"	3.8-53
Condensate Polishing Building (non-Seismic Category I) Sections 1-1, 2-2, 3-3, 4-4, 5-5	3.8-54
Condensate Polishing Building (non-Seismic Category I) Plan El 762'-6" and 794'-6", Section 6-6	3.8-55
Refueling Water Storage Tank and Chemical Addition Tank	3.8-1 3.8-2
Gaseous Waste Storage Tanks (non-Seismic Category I)	3.8-56

NOTES:

*The auxiliary building steel structure above el 773'-6" is Seismic Category I but does not provide tornado protection.
**The steel frame roof portion of the main steam valve and cable vault area is non-Seismic Category I.
***The service building above el 780'-6" is non-Seismic Category I.

TABLE 3.8-13

CRITERIA FOR STABILITY FACTORS AGAINST SLIDING AND OVERTURNING

Loading Combination	Overturning	Stability Sliding	Flotation
D + H + E	1.5	1.5	
D + H + W	1.5	1.5	
D + H + E′	1.1	1.1	
D + H + W_t	1.1	1.1	
D + F			1.1

NOTES:

D = Dead loads or their related internal movements and forces, including permanent equipment loads and hydrostatic loads.

H = Lateral earth pressure.

E = Loads generated by the operating basis earthquake.

W = Loads generated by the design wind.

E′ = Loads generated by the safe shutdown earthquake.

W_t = Loads generated by the design tornado.

F = Buoyant force of probable maximum flood.

Removed in Accordance with RIS 2015-17

FIGURE 3.8-1

REACTOR CONTAINMENT MACHINE LOCATION, PLAN E. 767'-10"

BEAVER VALLEY POWER STATION - UNIT 2
UPDATED FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 3.8-2
REACTOR CONTAINMENT MACHINE
LOCATIONS, PLAN EL. 738'-10"
BEAVER VALLEY POWER STATION - UNIT 2
UPDATED FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 3.8-3
REACTOR CONTAINMENT MACHINE
LOCATIONS, PLAN EL. 718'-6"
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 3.8-4

REACTOR CONTAINMENT MACHINE LOCATION PLAN EL. 692'-11"

BEAVER VALLEY POWER STATION-UNIT 2
UPDATED FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 3.8-5

REACTOR CONTAINMENT MACHINE LOCATIONS, SECTIONS 1-1, 6-6 & 10-10

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FIGURE 3.8-6

REACTOR CONTAINMENT MACHINE LOCATIONS, SECTIONS 2-2, 5-5 & 9-9

BEAVER VALLEY POWER STATION - UNIT 2
UPDATED FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

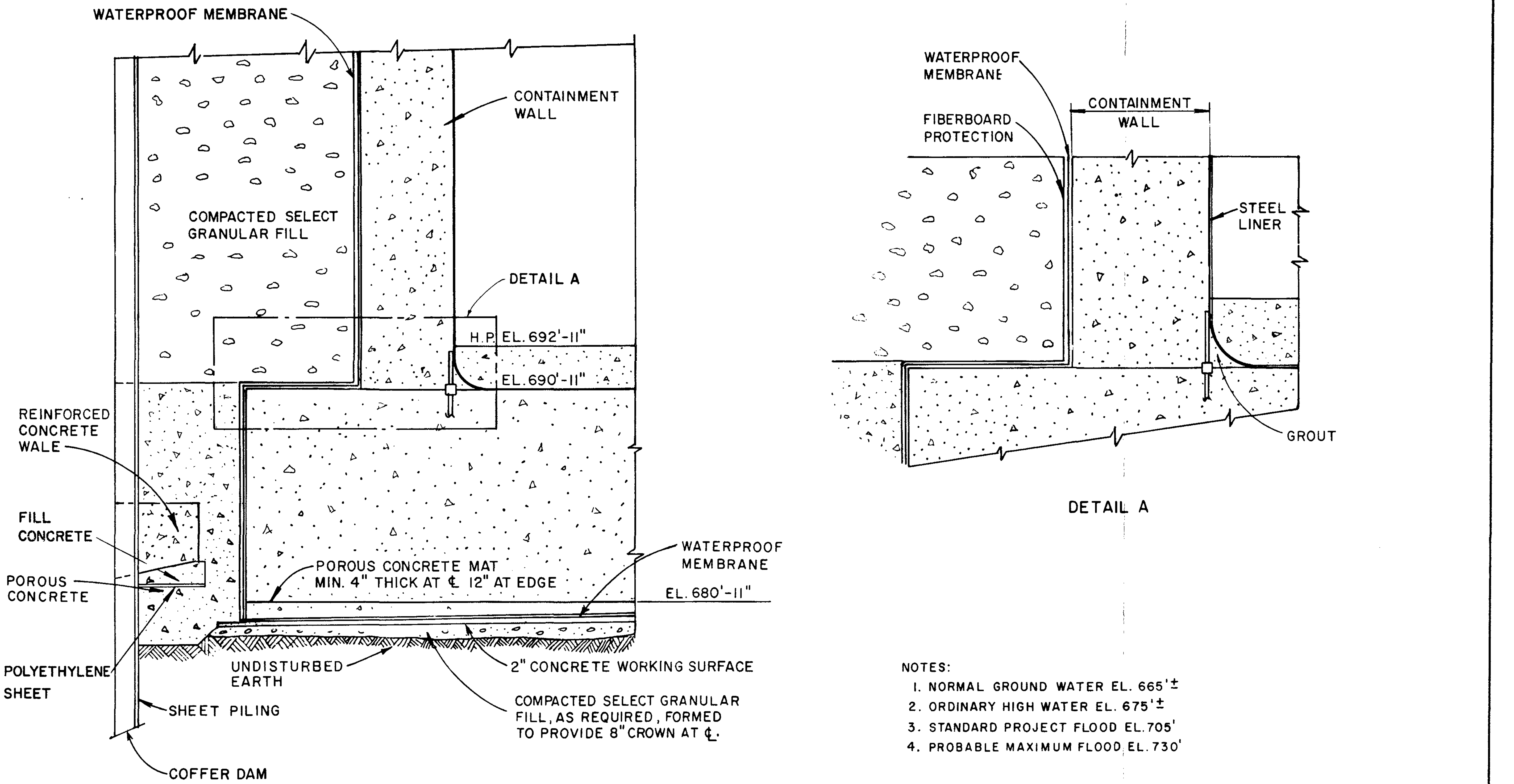
FIGURE 3.8-7

REACTOR CONTAINMENT MACHINE LOCATIONS, SECTIONS 3-3 & 4-4

BEAVER VALLEY POWER STATION - UNIT 2
UPDATED FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 3.8-8
PLOT PLAN
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT



NOTES:

1. NORMAL GROUND WATER EL. 665'±
2. ORDINARY HIGH WATER EL. 675'±
3. STANDARD PROJECT FLOOD EL. 705'
4. PROBABLE MAXIMUM FLOOD EL. 730'

FIGURE 3.8-9
REACTOR CONTAINMENT WATERPROOFING
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

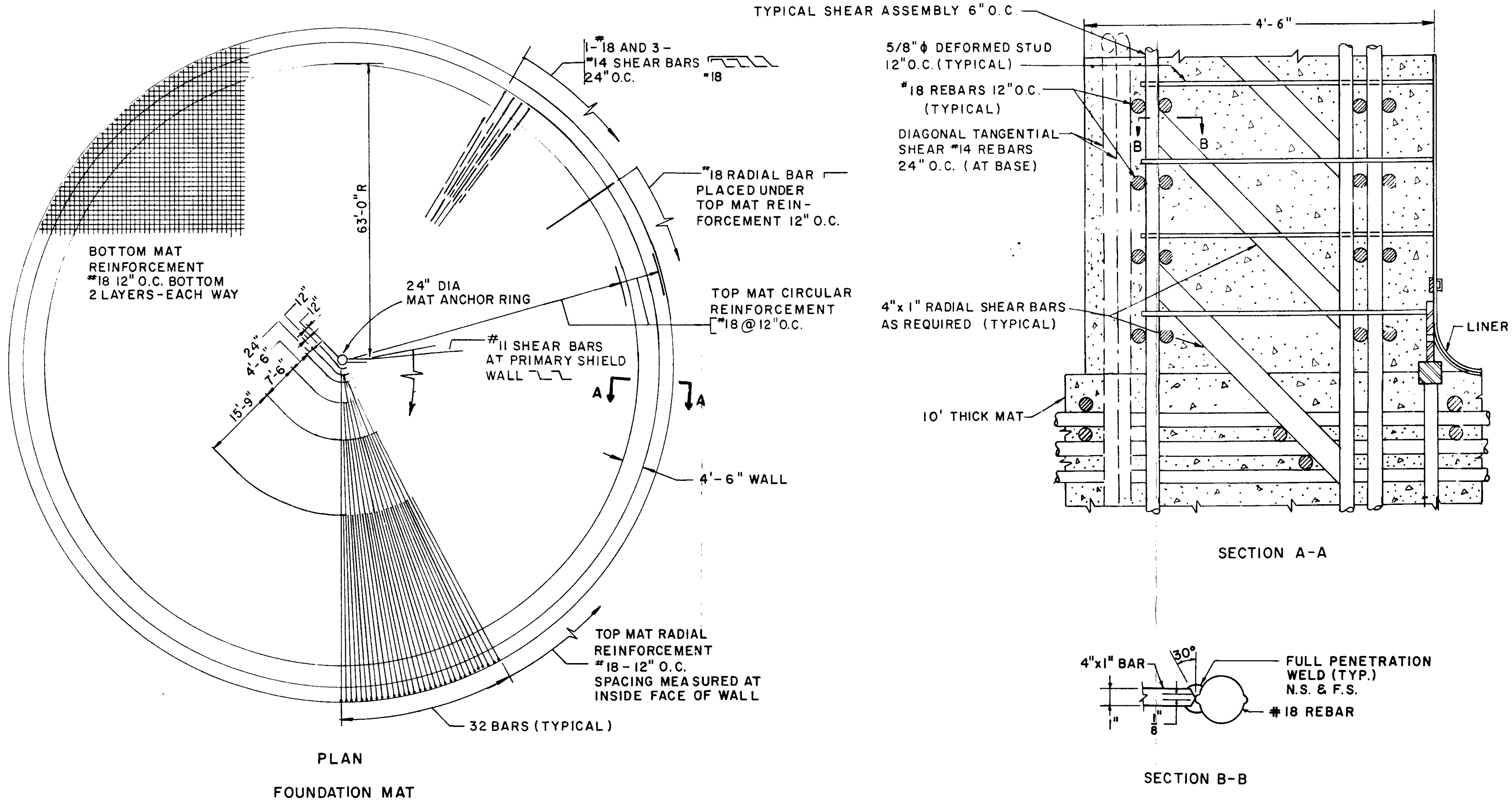


FIGURE 3.8-10
FOUNDATION MAT & WALL BASE
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

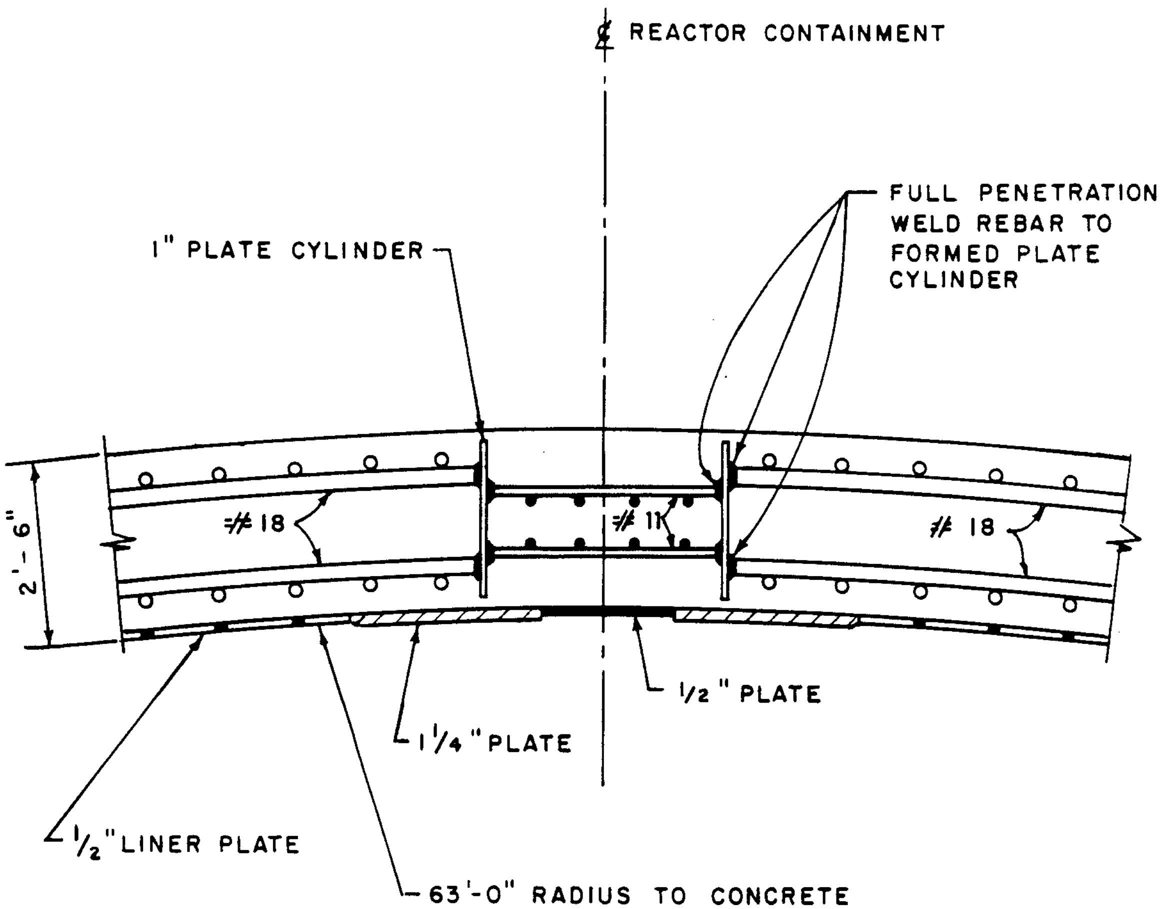


FIGURE 3.8-11
CONCENTRIC RING AT APEX OF DOME
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

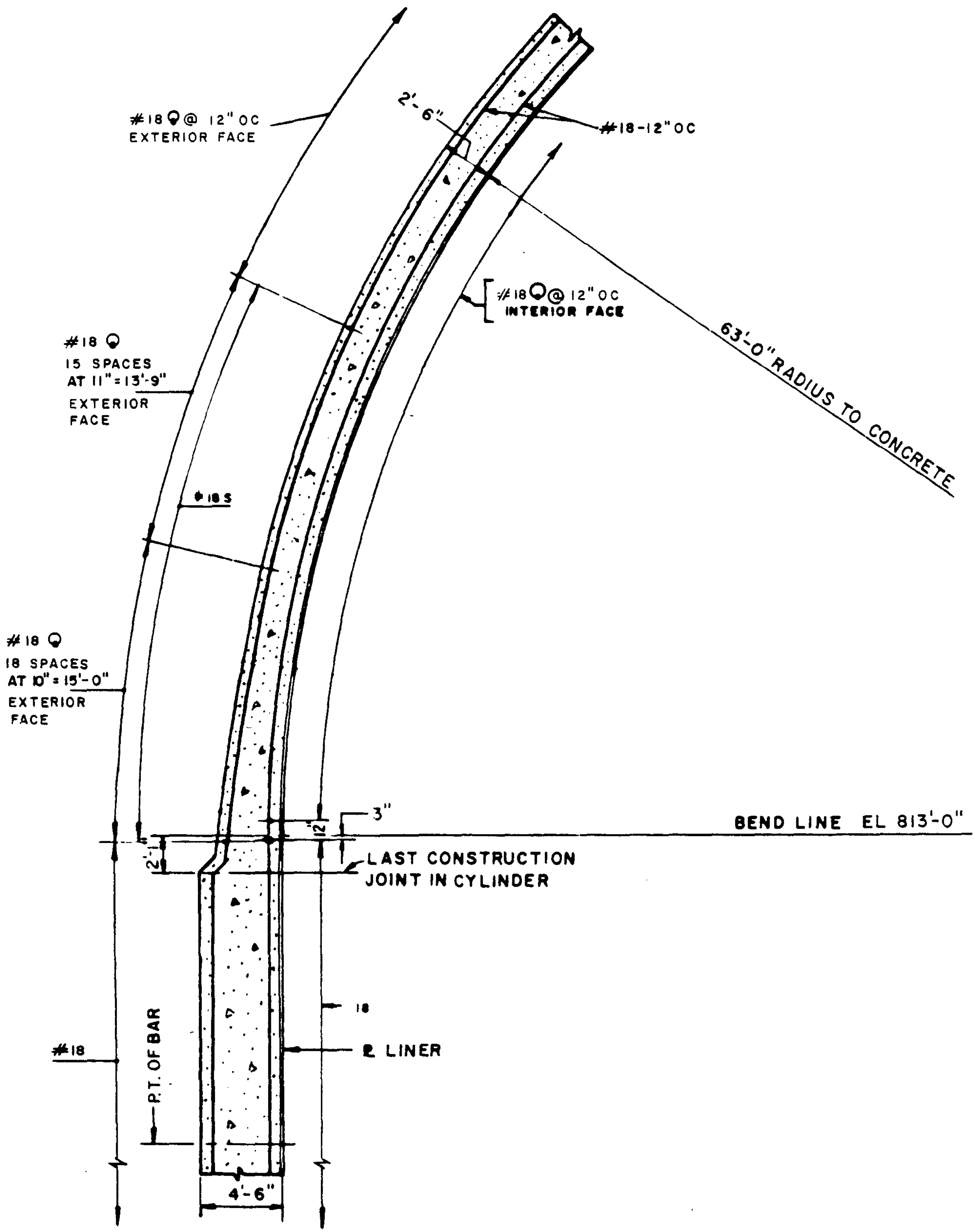
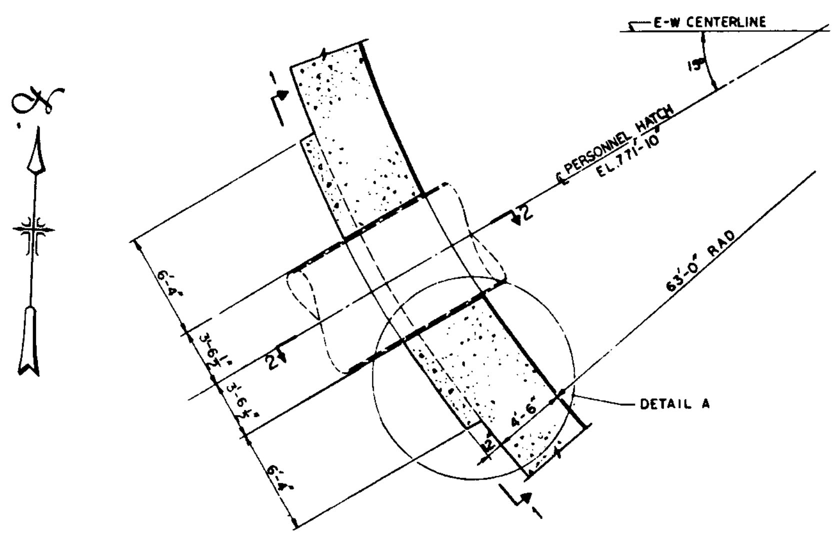


FIGURE 3.8-12

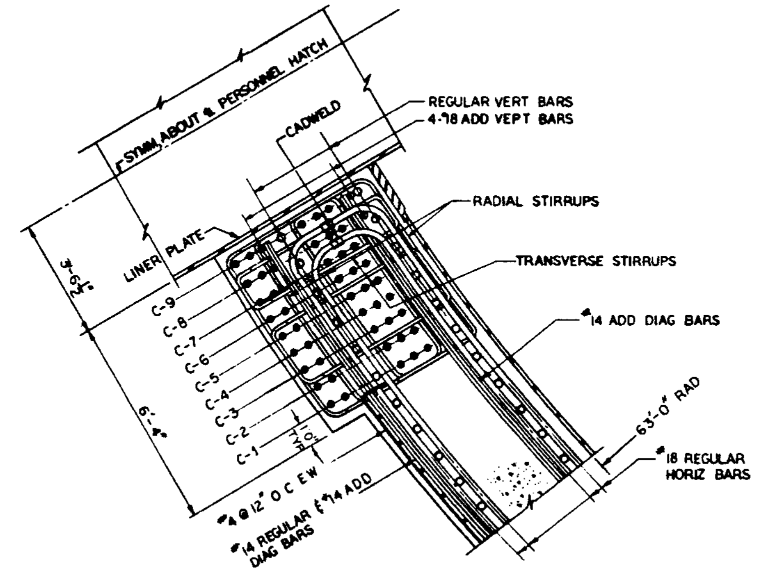
TYPICAL DETAIL OF DOME— CYLINDER JUNCTION

BEAVER VALLEY POWER STATION-UNIT 2

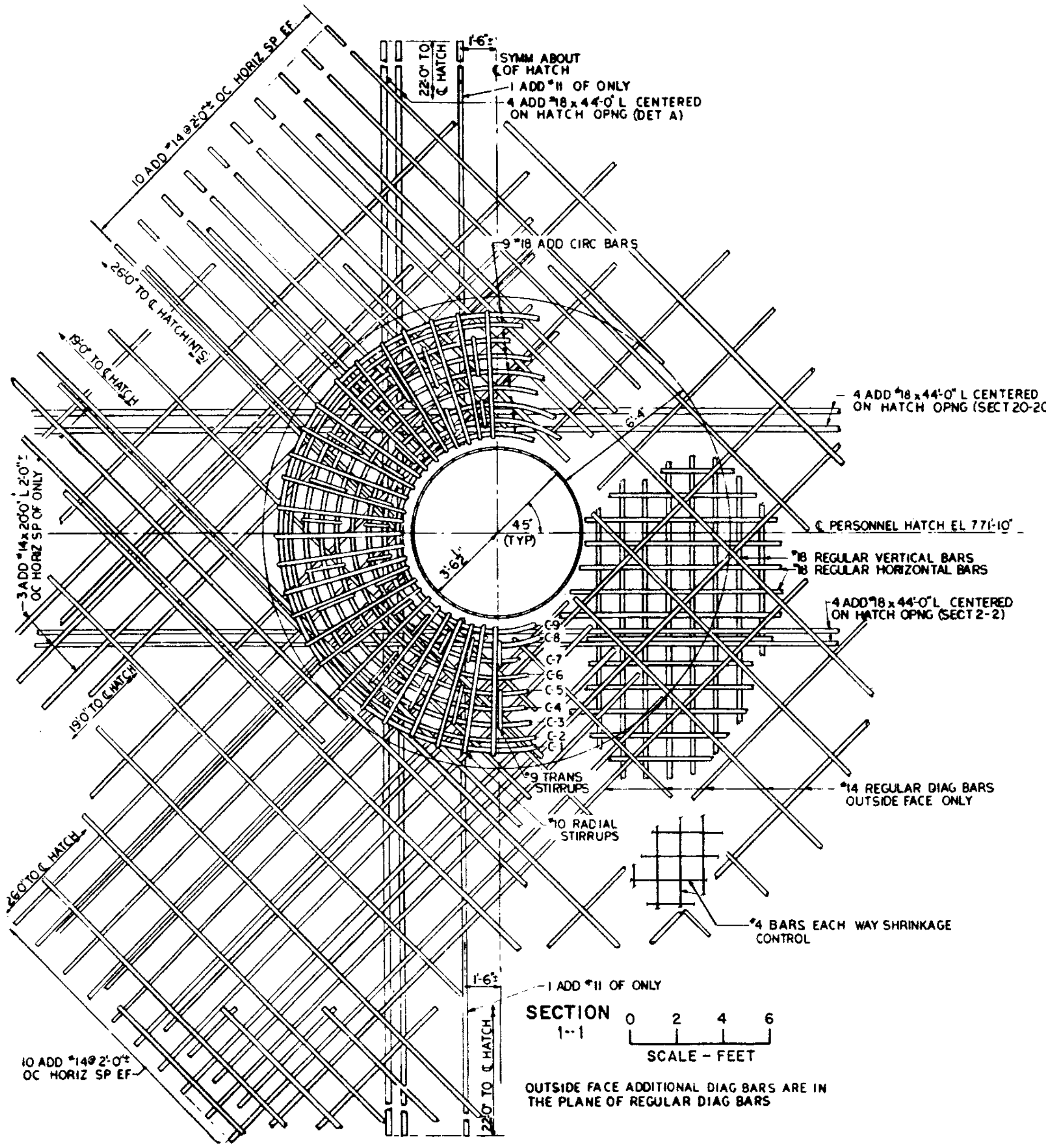
FINAL SAFETY ANALYSIS REPORT



PLAN AT EL 771'-10"

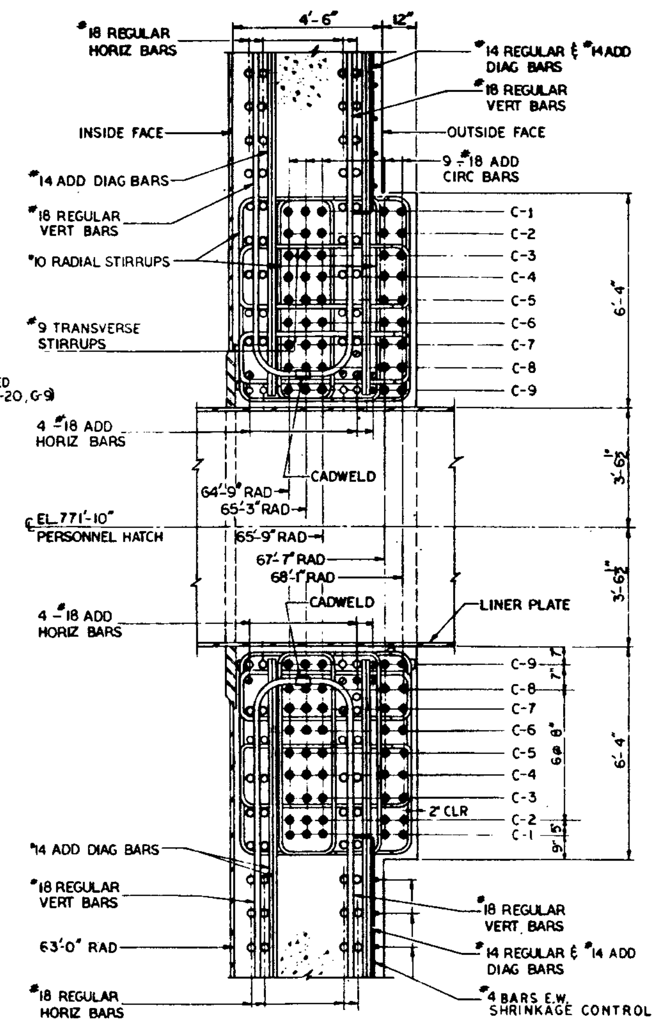


DETAIL A

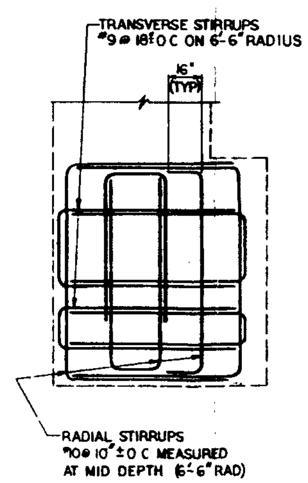


SECTION 1-1

OUTSIDE FACE ADDITIONAL DIAG BARS ARE IN THE PLANE OF REGULAR DIAG BARS



SECTION 2-2



STIRRUPS DETAIL

LEGEND:

- ● CIRCULAR REBARS WITH RADIUS FROM ℄ OF HATCH AND ℄ OF REACTOR CONTAINMENT
- ○ REGULAR REBARS
- ⊘ ADDITIONAL HORIZONTAL OR VERTICAL REBARS

FIGURE 3.8-13
PERSONNEL HATCH REINFORCING DETAILS
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

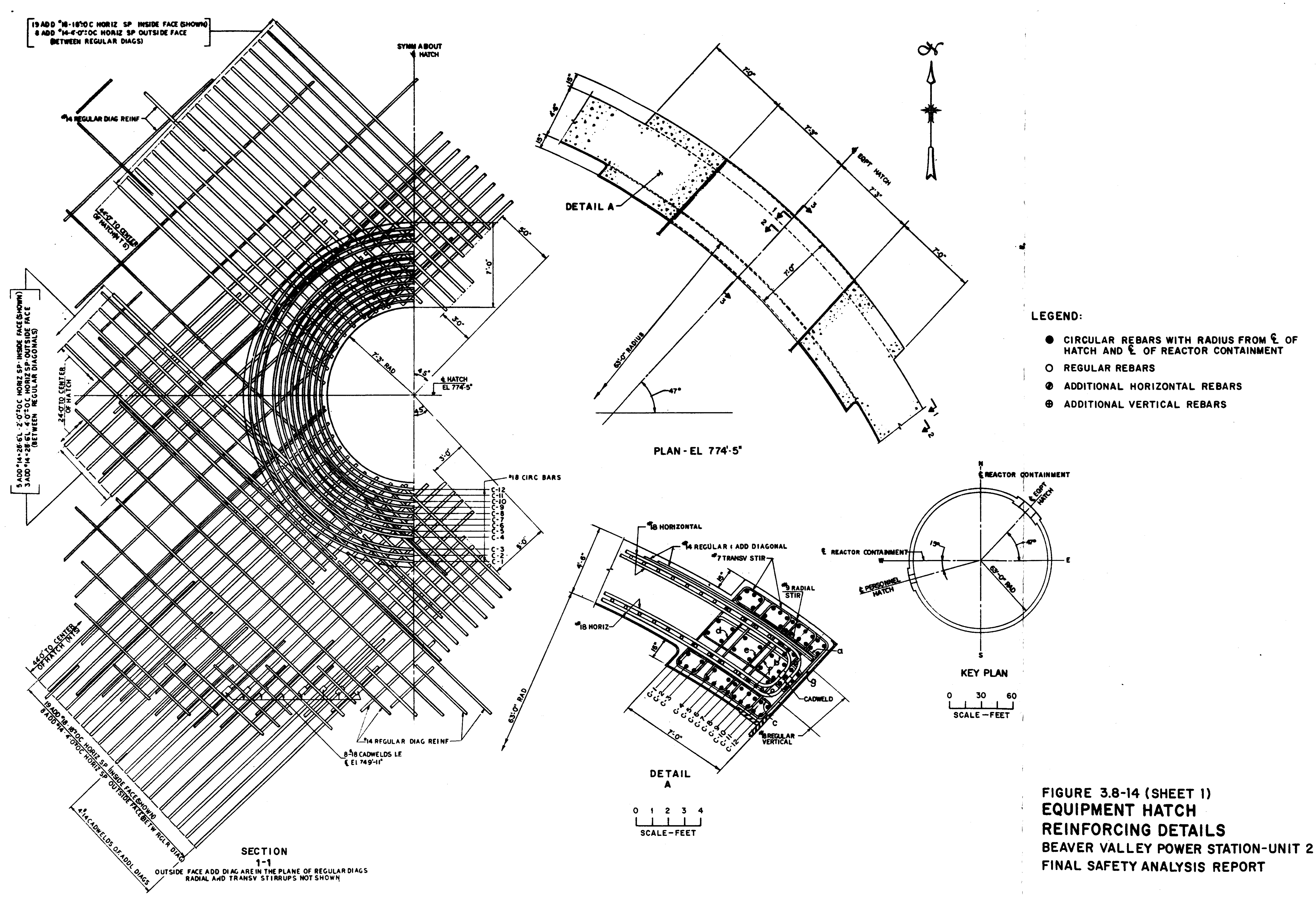


FIGURE 3.8-14 (SHEET 1)

EQUIPMENT HATCH REINFORCING DETAILS

BEAVER VALLEY POWER STATION-UNIT 2

FINAL SAFETY ANALYSIS REPORT

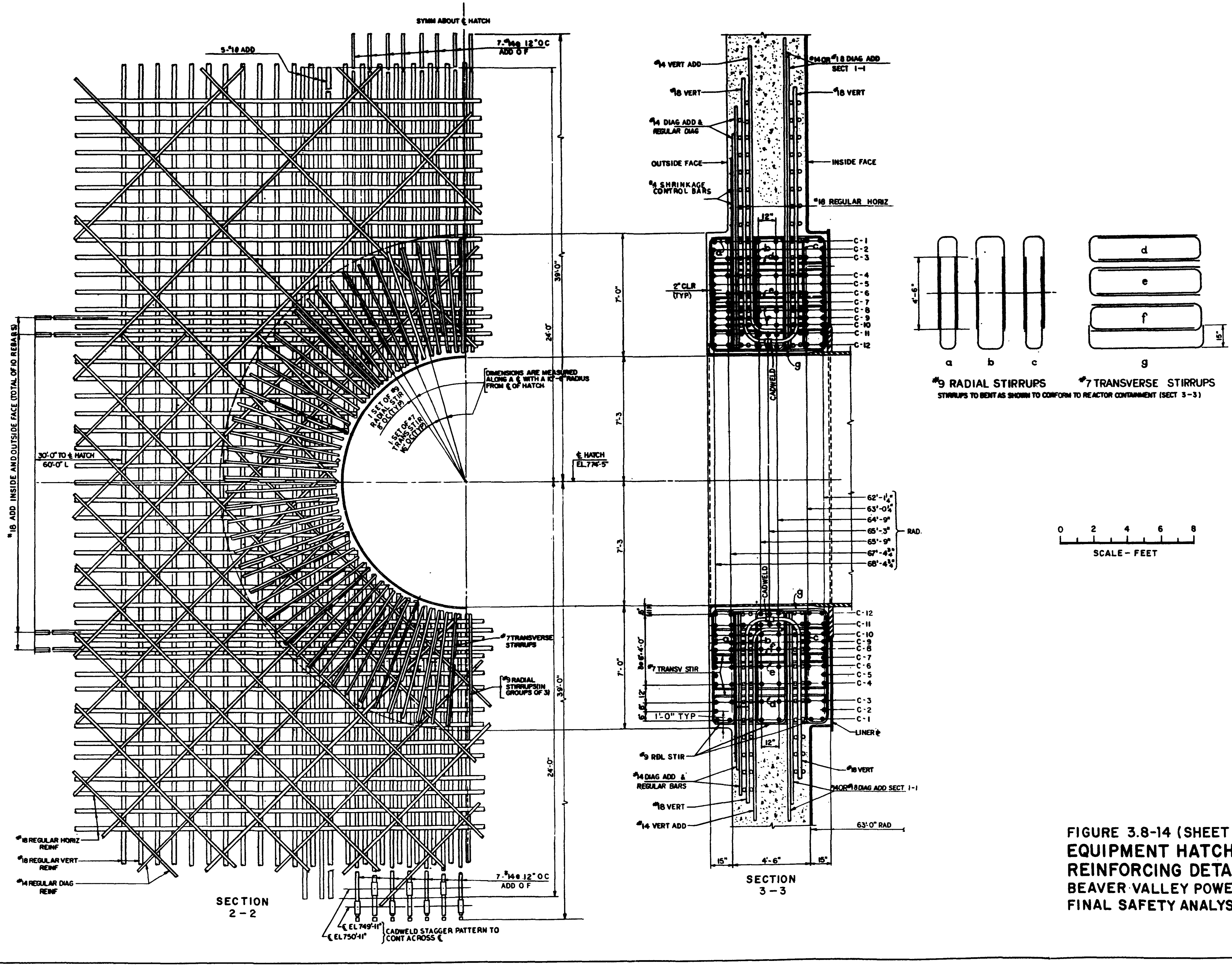
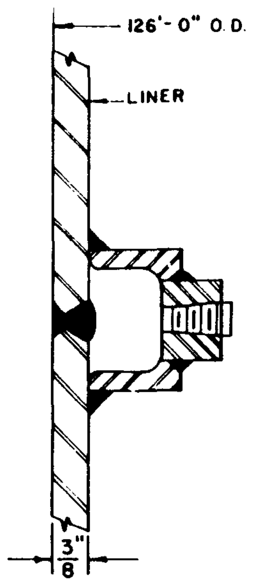
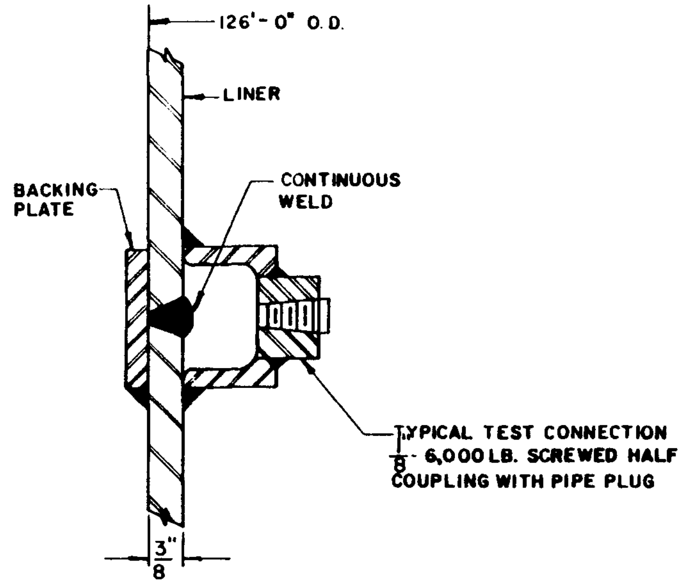


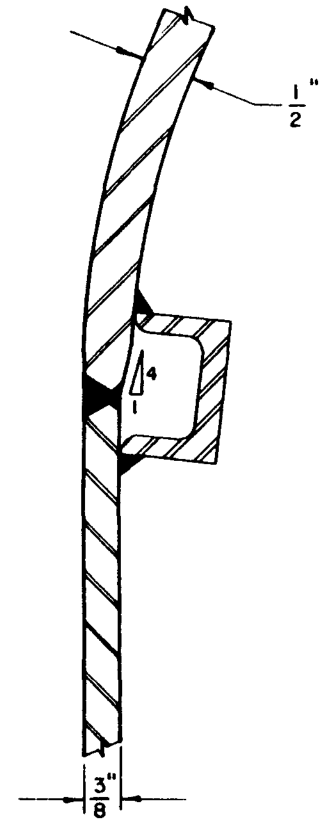
FIGURE 3.8-14 (SHEET 2)
EQUIPMENT HATCH
REINFORCING DETAILS
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT



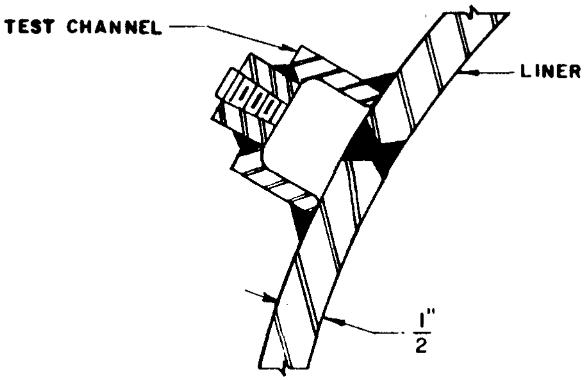
TYPICAL WALL JOINT
WITHOUT BACKING PLATE



TYPICAL WALL JOINT
WITH BACKING PLATE



WALL TO DOME JOINT



TYPICAL DOME JOINT

FIGURE 3.8-15
TYPICAL LINER DETAILS
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

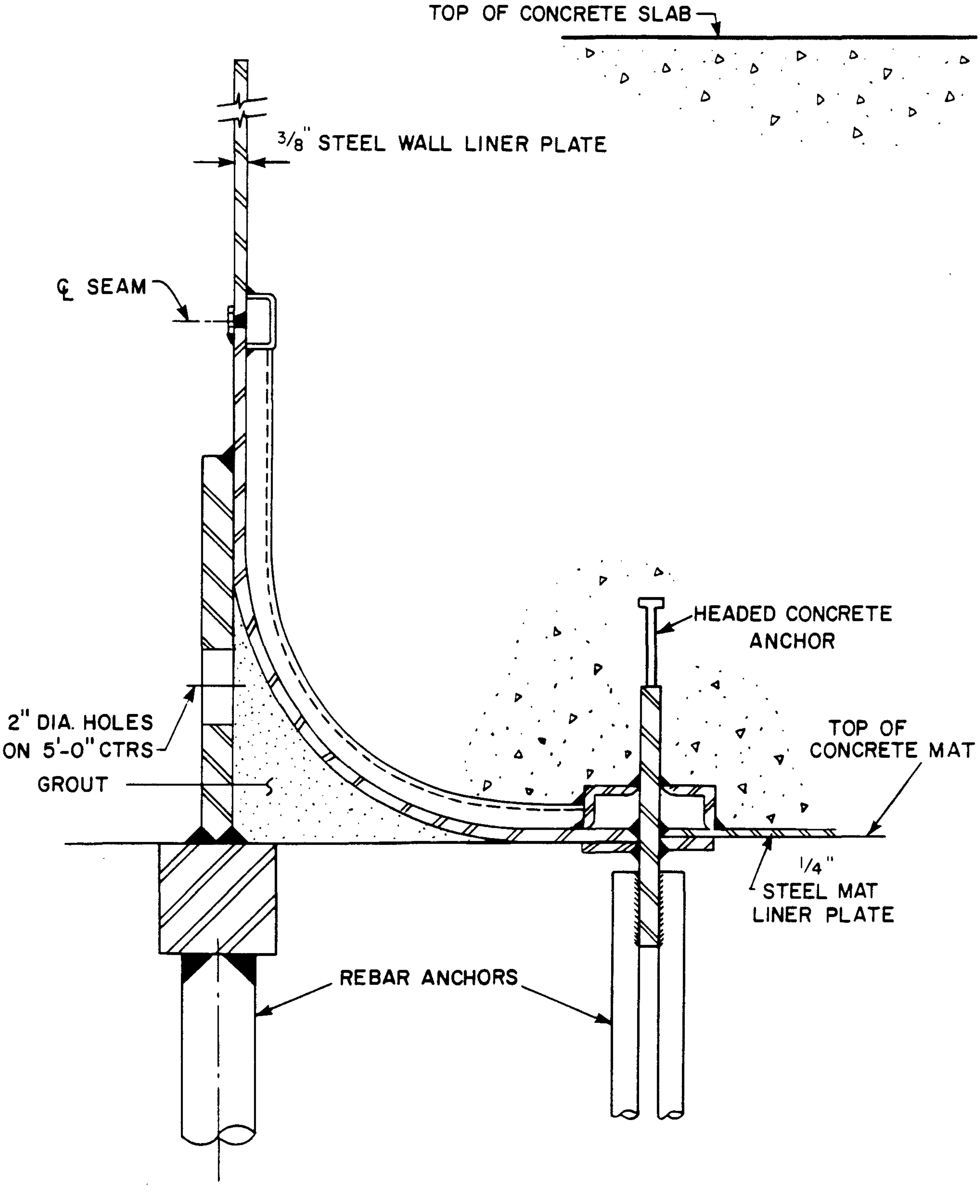
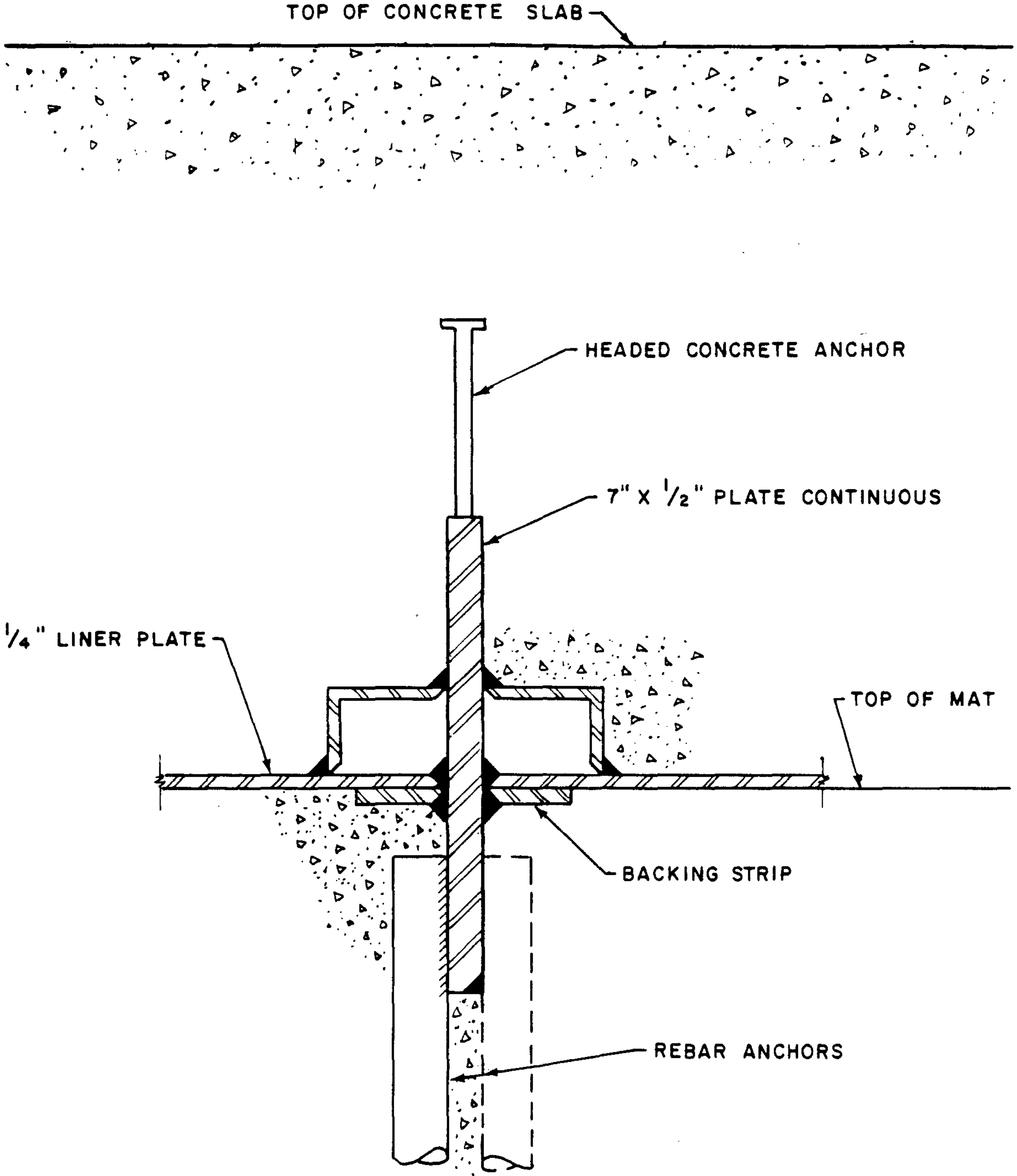
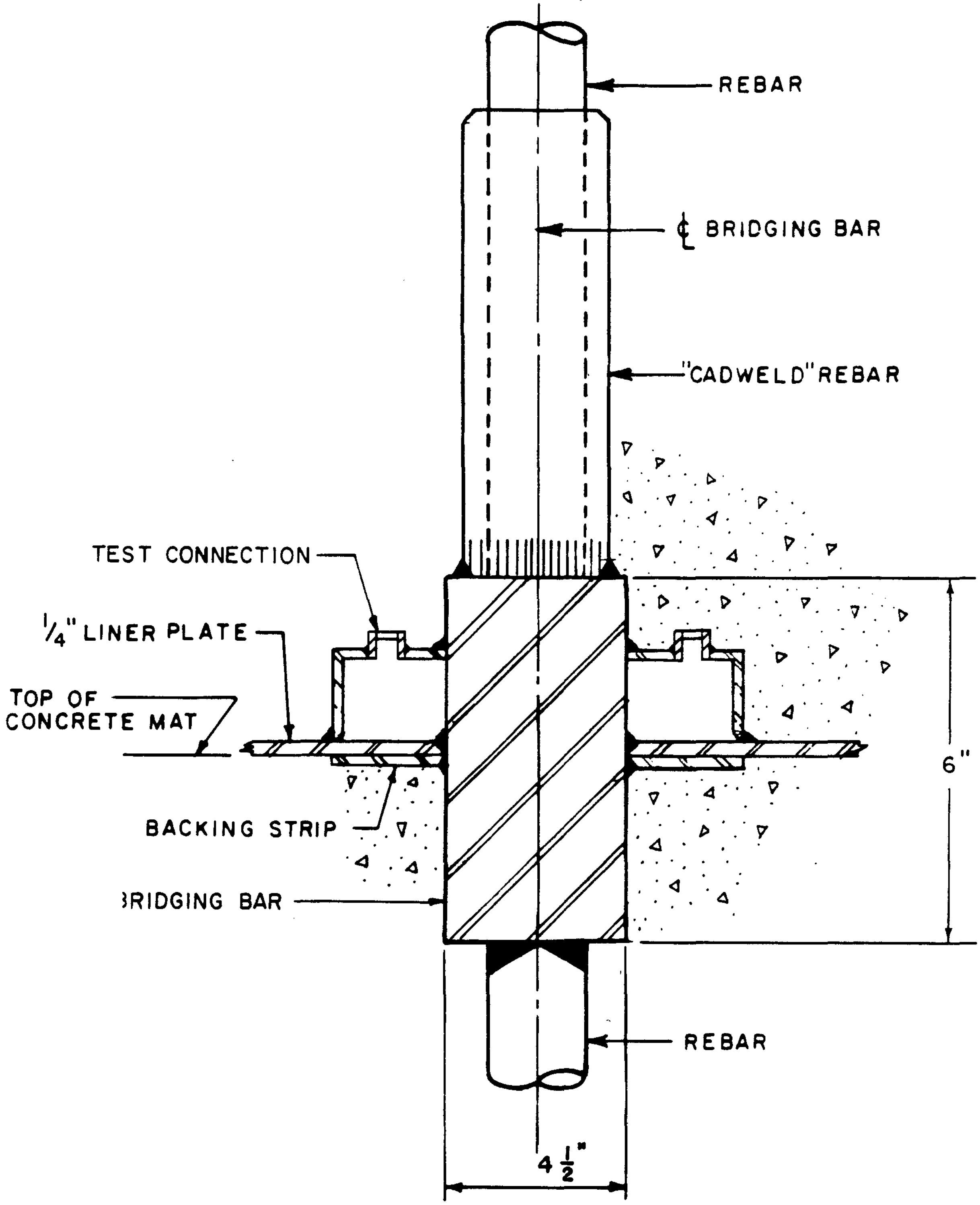


FIGURE 3.8-16
WALL AND MAT JOINT
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT



TYPICAL SECTION BRIDGING BAR USED TO ANCHOR CONCRETE SLAB TO CONTAINMENT MAT THROUGH MAT LINER

FIGURE 3.8-17
SECTION – TYPICAL
BOTTOM LINER BRIDGING PLATE
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT



TYPICAL SECTION THROUGH BRIDGING BAR USED TO PROVIDE MAIN REINFORCING STEEL CONTINUITY THROUGH MAT LINER

FIGURE 3.8-18
SECTION - TYPICAL BRIDGING BAR
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

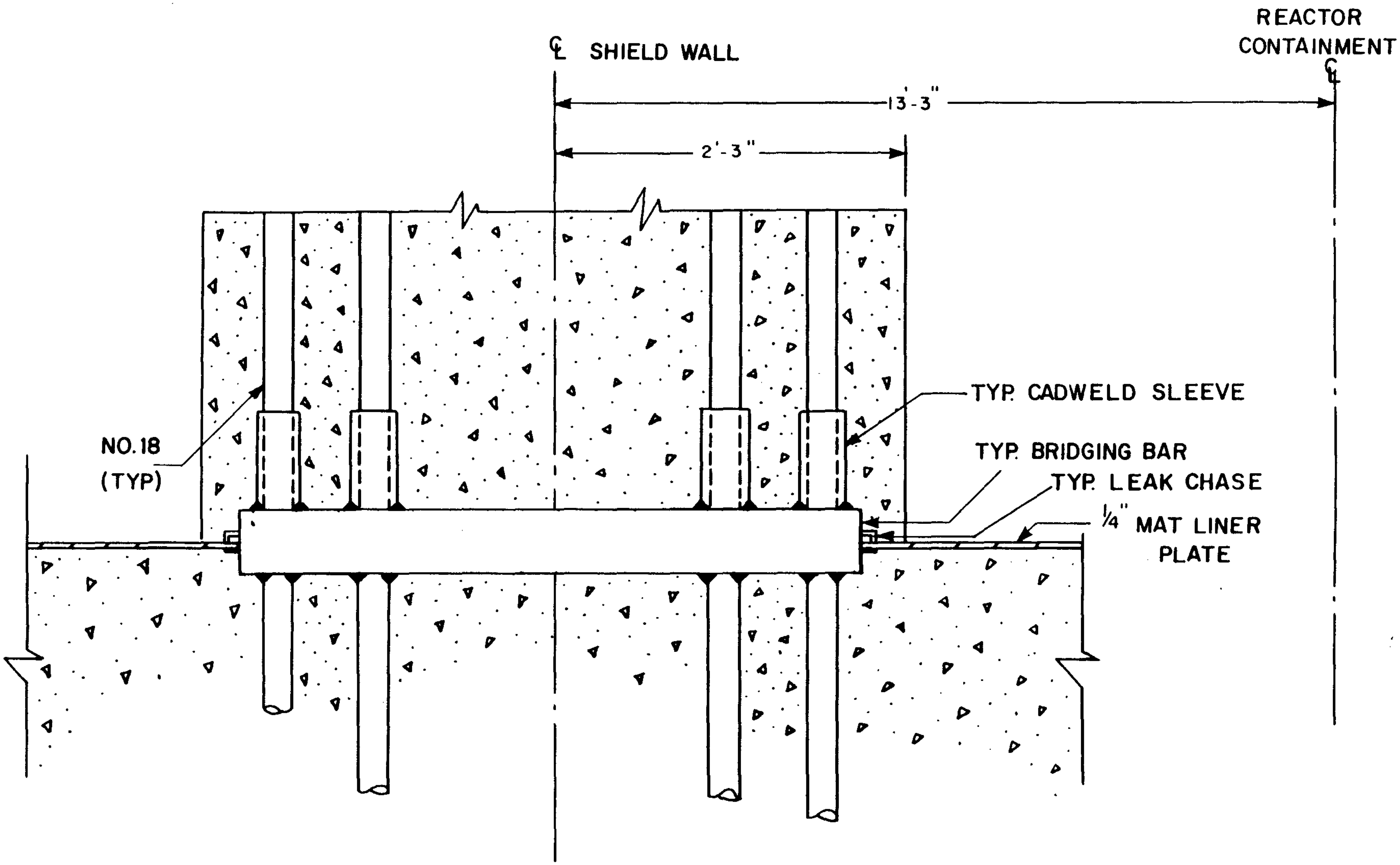


FIGURE 3.8-19
SECTION-SHIELD WALL BASE
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

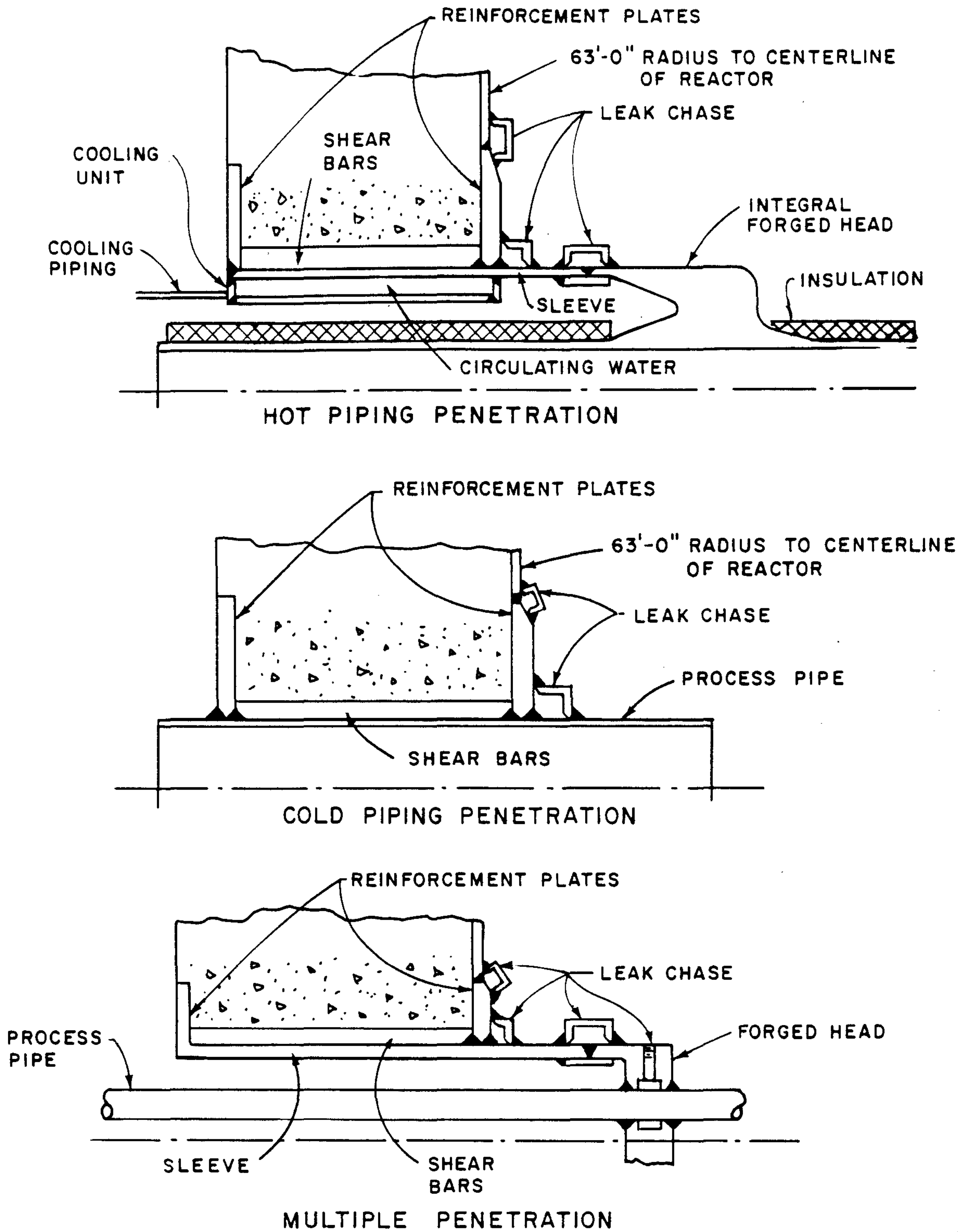
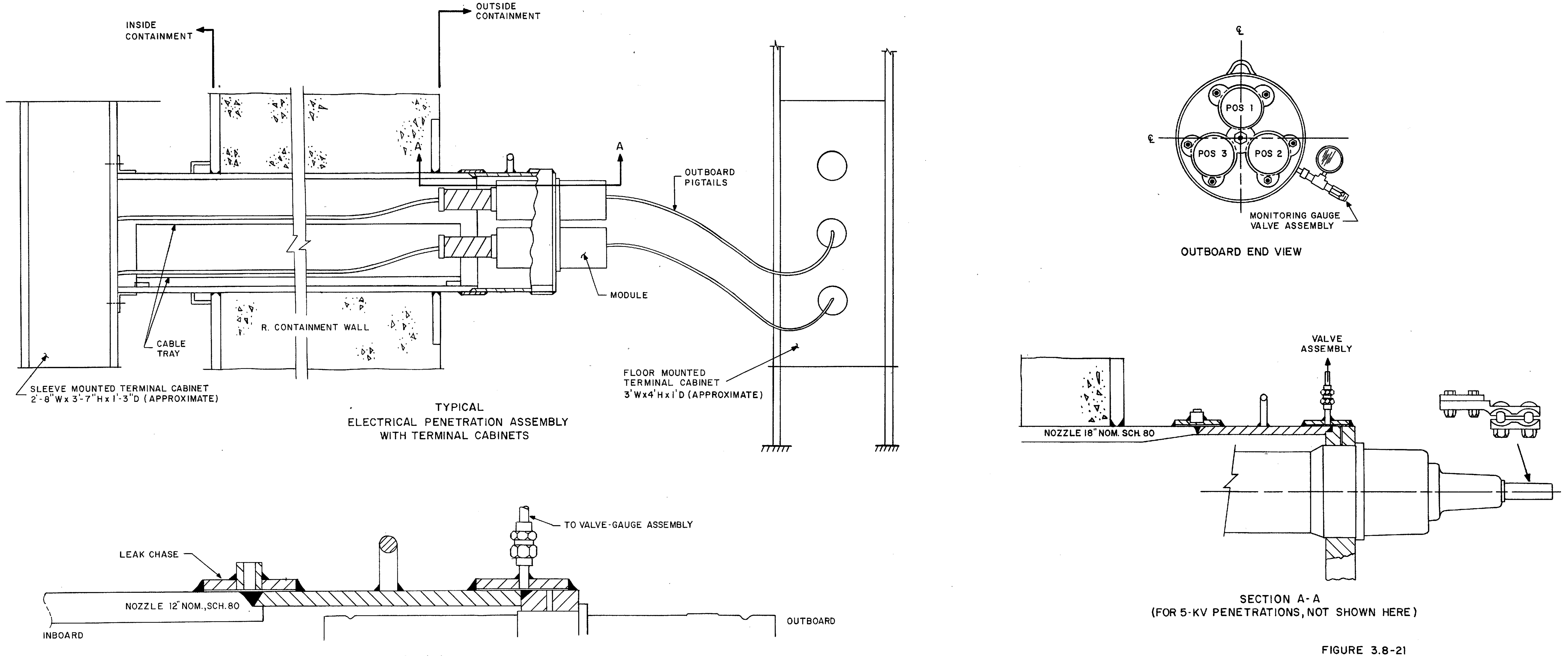


FIGURE 3.8-20
TYPICAL PIPING PENETRATIONS
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT



TYPICAL
ELECTRICAL PENETRATION ASSEMBLY
WITH TERMINAL CABINETS

OUTBOARD END VIEW

SECTION A-A
(FOR 600-V PENETRATIONS, SHOWN ABOVE)

SECTION A-A
(FOR 5-KV PENETRATIONS, NOT SHOWN HERE)

FIGURE 3.8-21
TYPICAL ELECTRICAL PENETRATION
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

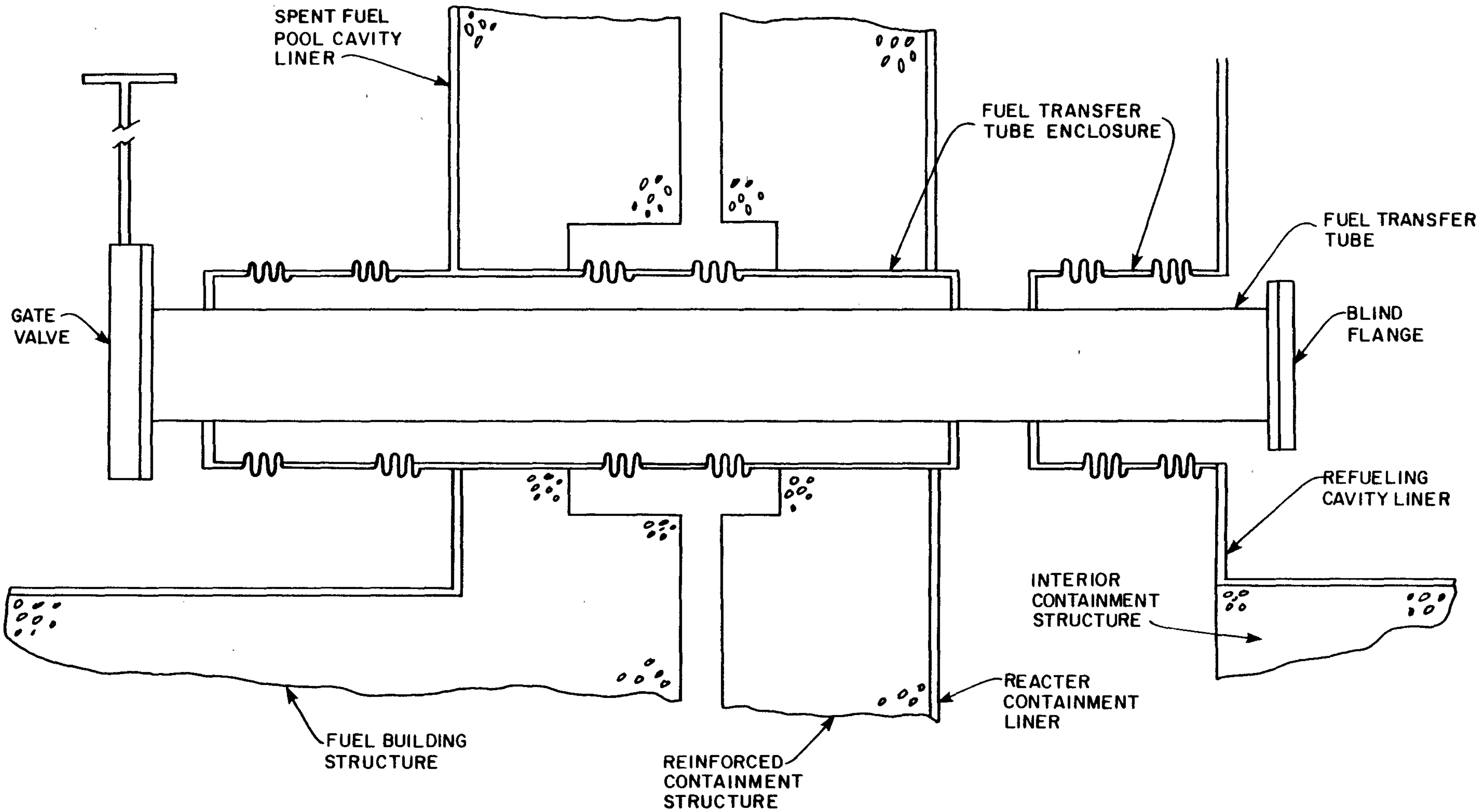


FIGURE 3.8-24
FUEL TRANSFER TUBE ENCLOSURE
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.8 - 25

PLANT ARRANGEMENT AT
PLAN EL. 735'-6"

BEAVER VALLEY POWER STATION - UNIT 2
UPDATED FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 3.8-26
PLANT ARRANGEMENT AT PLAN EL. 752'-6"
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 3.8-27
PLANT ARRANGEMENT AT PLAN EL.760'-6"
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 3.8-28
PLANT ARRANGEMENT AT PLAN EL. 774'-6"
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 3.8-29
PLANT ARRANGEMENT PART PLAN
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 3.8-30
AUXILIARY BUILDING ARRANGEMENT
PLAN EL. 710'-6" AND 718'-6"
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 3.8-31
AUXILIARY BUILDING ARRANGEMENT
PLAN EL. 735'-6"
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 3.8-32
AUXILIARY BUILDING ARRANGEMENT
PLAN EL. 755'-6"
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 3.8-33
AUXILIARY BUILDING ARANGEMENT
PLAN EL. 773'-6"
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 3.8-34
AUXILIARY BUILDING ARRANGEMENTS[1]
SECTIONS 1-1 AND 2-2
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 3.8-35
AUXILIARY BUILDING
ARRANGEMENT SECTIONS
3-3 AND 4-4
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 3.8-36
AUXILIARY BUILDING ARRANGEMENT
SECTION 5-5, PLAN EL.710'-6"
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 3.8-37
MAIN STEAM & CABLE VAULT AREA
PLAN EL. 755'-6", 798'-0", 808'-6"
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 3.8-40

CONTROL BUILDING
PLAN EL. 735'-6"

BEAVER VALLEY POWER STATION - UNIT 2
UPDATED FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 3.8-41
CONTROL BUILDING
PLAN EL. 707'-6"
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

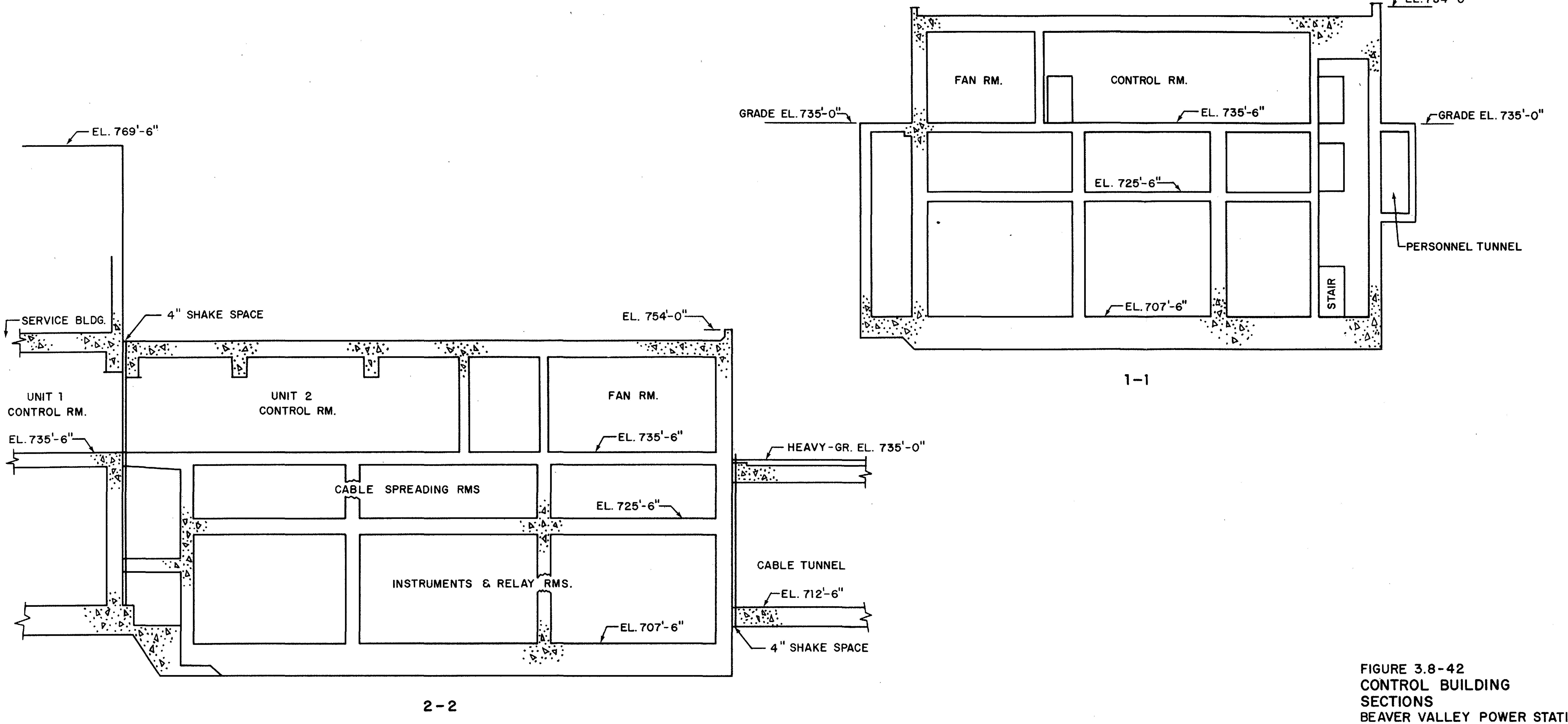


FIGURE 3.8-42
CONTROL BUILDING
SECTIONS
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 3.8-43
DIESEL GENERATOR
BUILDING-ARRANGEMENT
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 3.8-44

SERVICE BUILDING
PLAN EL 745'-6" AND 730'-6"
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 3.8-45
SERVICE BUILDING
PLAN EL 760'-6" AND 780'-6"
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

Removed in Accordance with RIS 2015-17

FIGURE 3.8-46
SERVICE BUILDING
SECTIONS 1-1 AND 2-2
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

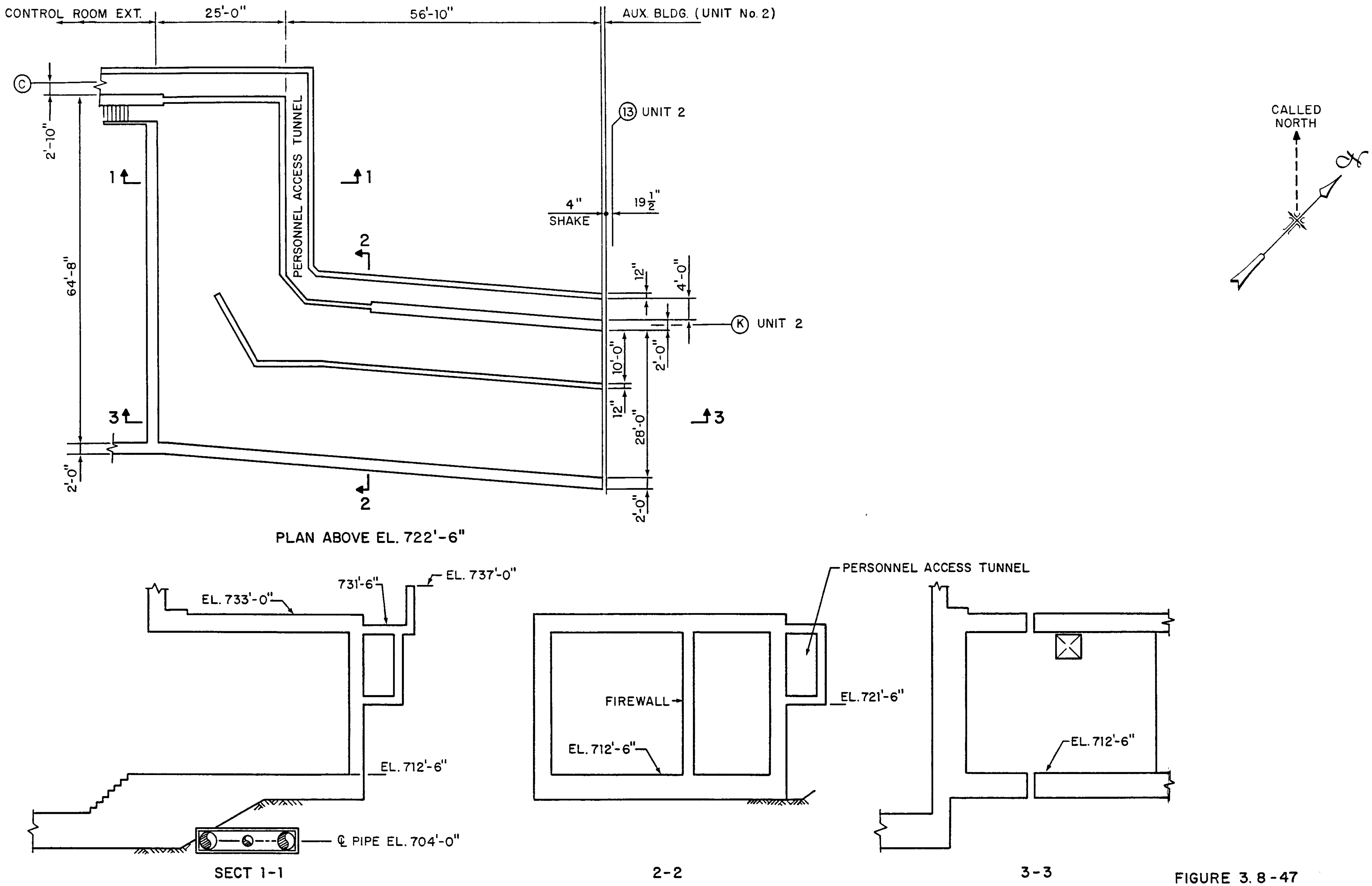


FIGURE 3.8-47
ELECTRICAL CABLE TUNNEL
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

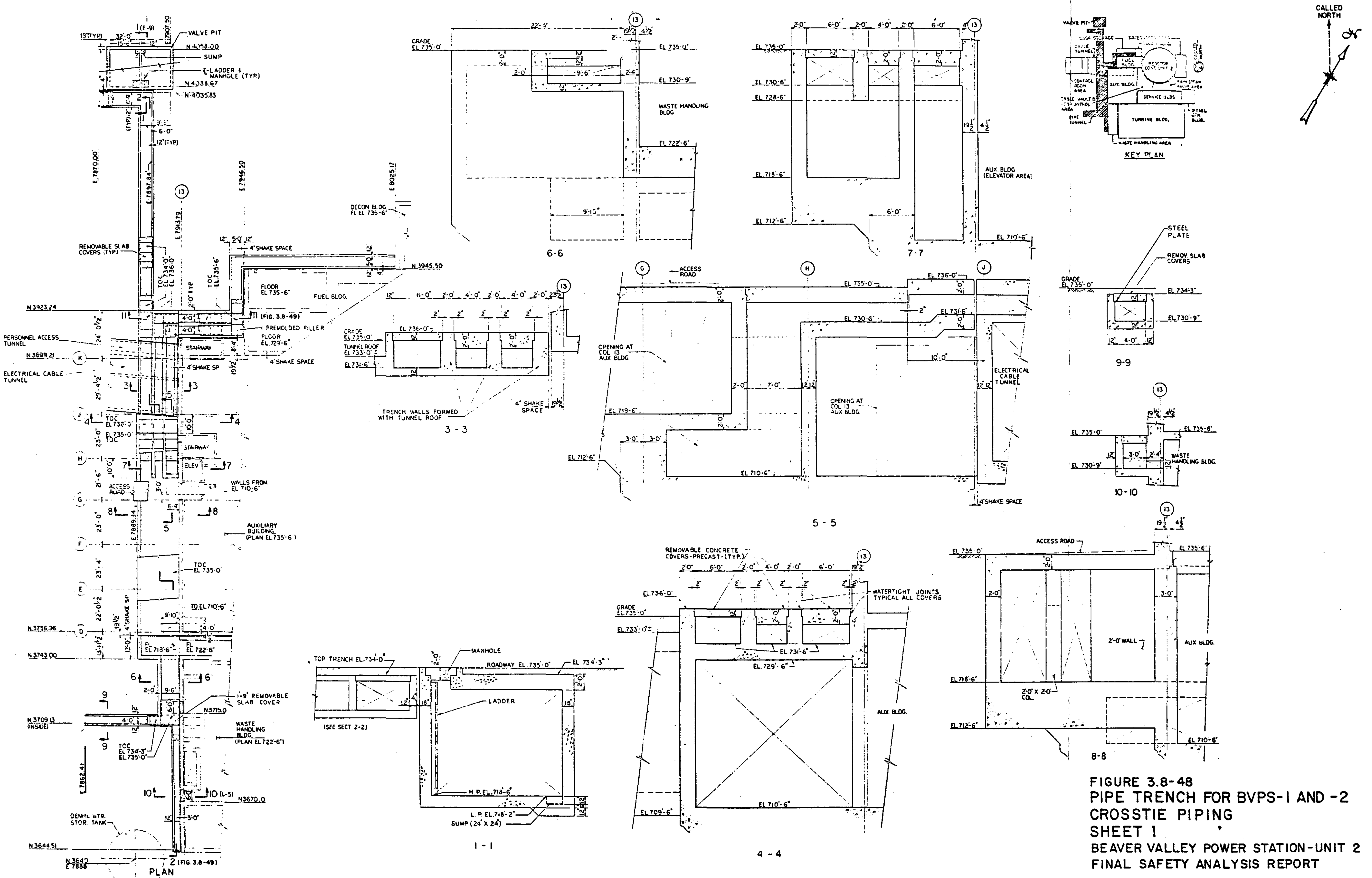
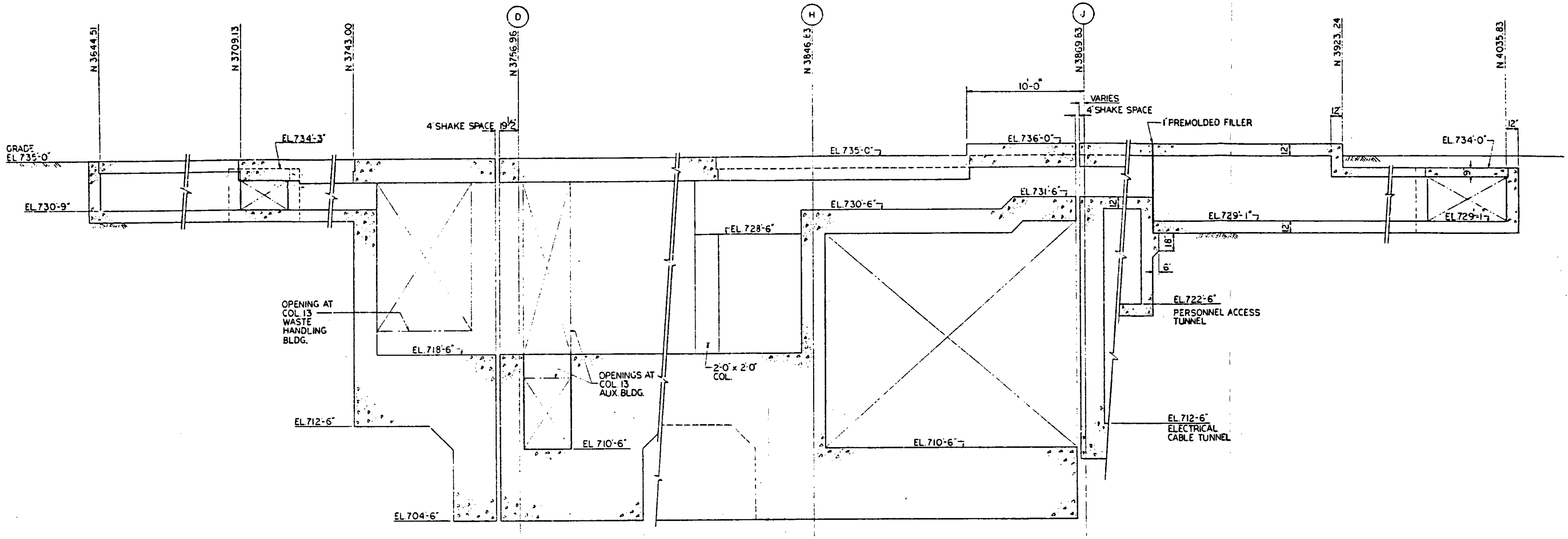


FIGURE 3.8-48
PIPE TRENCH FOR BVPS-1 AND -2
CROSSTIE PIPING
SHEET 1
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT



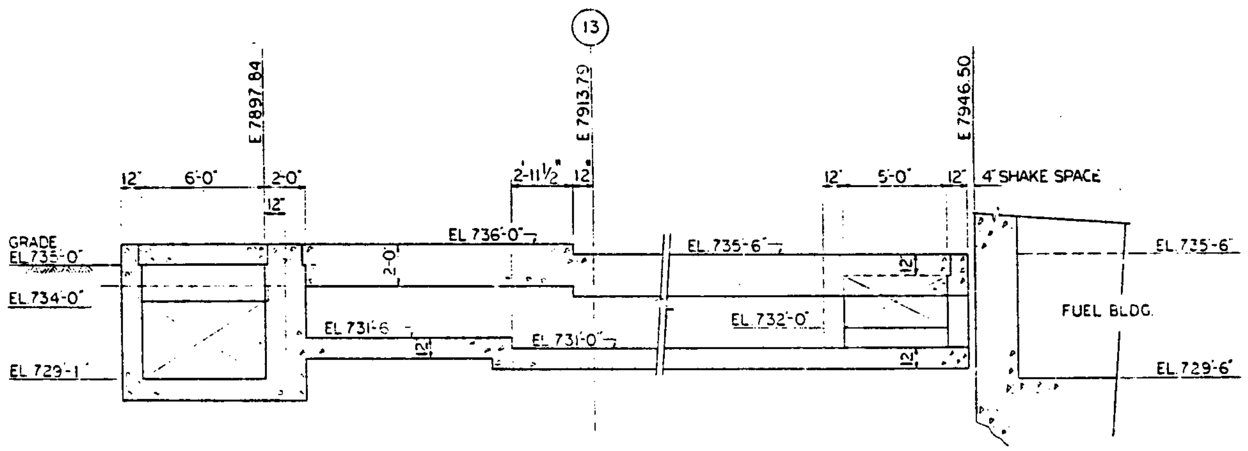


FIGURE 3.8-49
PIPE TRENCH FOR BVPS-1 AND-2
CROSSTIE PIPING
SHEET 2
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

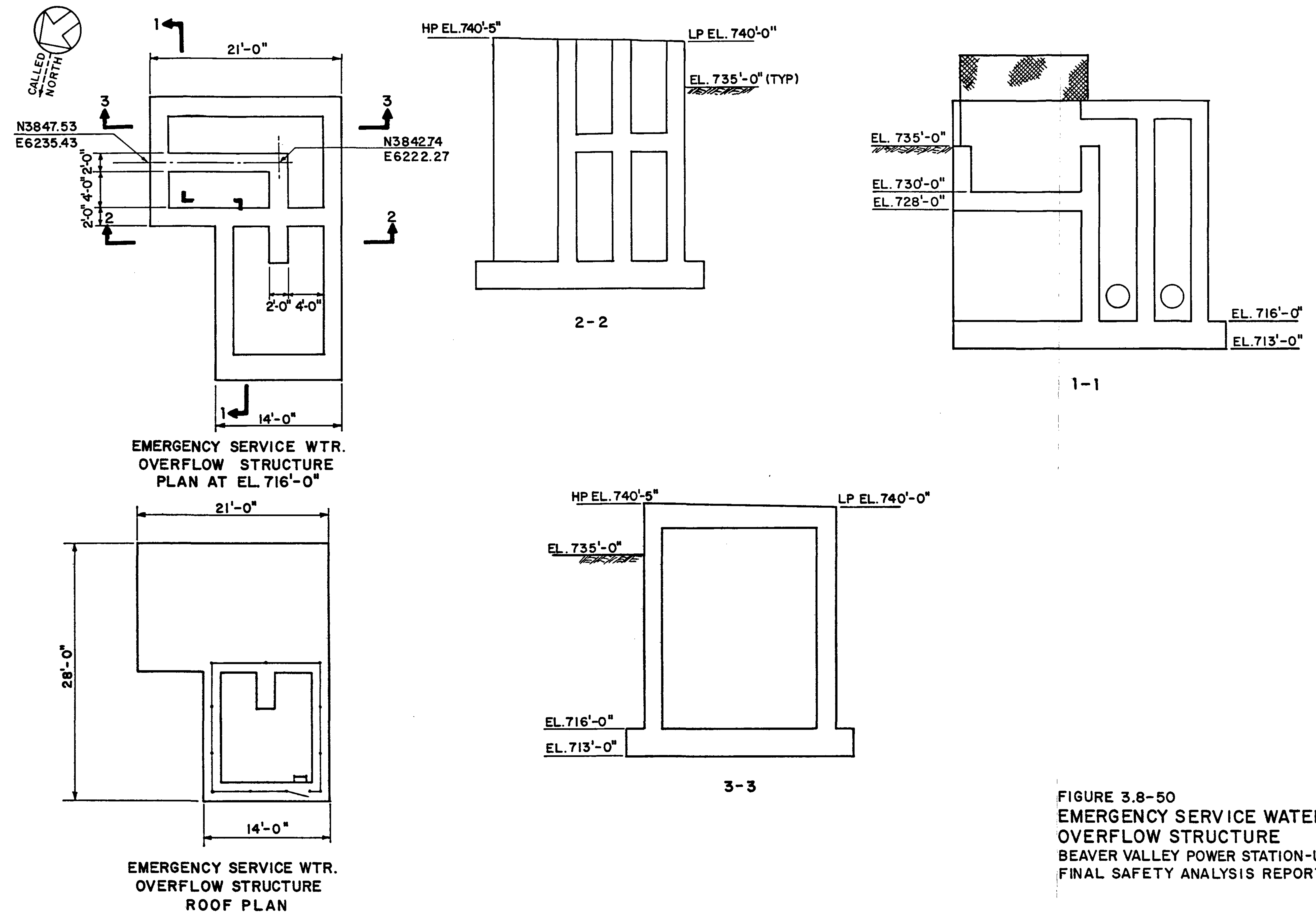


FIGURE 3.8-50
EMERGENCY SERVICE WATER
OVERFLOW STRUCTURE
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.8-51
WASTE HANDLING BUILDING
ARRANGEMENT
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.8-52
CONDENSATE POLISHING BUILDING
PLAN EL. 722'-6" AND 735'-6
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.8-53
CONDENSATE POLISHING BUILDING
PLAN EL. 752'- 6" AND 774'-6"
BEAVER VALLEY POWER STATION- UNIT 2
UPDATED FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.8-54
CONDENSATE POLISHING BUILDING
SECTIONS 1-1, 2-2, 3-3, 4-4, 5-5
BEAVER VALLEY POWER STATION-UNIT 2
UPDATED FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.8-55
CONDENSATE POLISHING BUILDING
(NON-CATEGORY I)
PLAN EL. 762'-6" AND 794'-6",
SECTION 6-6
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.8-56
GASEOUS WASTE STORAGE TANK
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

3.9 MECHANICAL SYSTEMS AND COMPONENTS

Sections whose identification numbers include the letter B contain material within balance-of-plant (BOP) scope, while sections whose identification numbers include the letter N contain material within the nuclear steam supply system (NSSS) scope.

3.9B.1 Special Topics for Mechanical Components

3.9B.1.1 Design Transients

The design of the reactor coolant system (RCS), RCS component supports, and reactor internals considers the following five operating conditions, defined in Section III of the ASME Boiler and Pressure Vessel Code.

1. Normal Conditions

   Any condition in the course of start-up, operation in the design power range, hot standby, and system shutdown (other than upset, emergency, faulted, or testing conditions).

2. Upset Conditions (Incidents of Moderate Frequency)

   Any deviations from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment.

   The upset conditions include:

   a. Transients which result from any single operator error or control malfunction,

   b. Transients caused by a fault in a system component requiring its isolation from the system,

   c. Transients due to loss of load or power,

   d. Abnormal incidents not resulting in a forced outage,

   e. Forced outages for which the corrective action does not include any repair of mechanical damage.

   The estimated duration of an upset condition is included in the design specifications for each component. The durations are based on the worst case for each component of the upset conditions listed in Section 3.9N.1. In this manner, each component is designed for the most severe upset transient experienced.

3. Emergency Conditions (Infrequent Incidents)

   Those deviations from normal conditions which require shutdown for correction of the conditions or repair of damage in the system. The conditions have a low probability of occurrence, but are included to assure that no gross loss of structural integrity results as a concomitant effect of any damage developed in the system. The total number of postulated occurrences for such events shall not cause more than 25 stress cycles having an S value greater than that for 10^6 cycles from the applicable fatigue design curves of the ASME Code, Section III. Beaver Valley Power Station - Unit 2 (BVPS-2) does not have any transients classified as emergency.

4. Faulted Conditions (Limiting Faults)

   Those combinations of conditions associated with extremely-low-probability, postulated events whose consequences involve the integrity and operability of the nuclear energy system. Such considerations require compliance with safety criteria specified by jurisdictional authorities.

5. Test Conditions

   Testing conditions are those tests in addition to the 10 hydrostatic or pneumatic tests permitted by NB-6222 and NB-6322 (ASME Section III), including leak tests or subsequent hydrostatic tests.

To provide the necessary high degree of integrity for the equipment in the NSSS, the transient conditions selected for equipment fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from various operating conditions in the plant. The transients selected are representative of operating conditions which are considered to occur during plant operation. They are of sufficient severity and/or frequency to be significant to component cyclic behavior. The transients selected may be regarded as conservative representations of transients which, when used as a basis for component fatigue evaluation, provide confidence that the component has been adequately designed for the 40-year service life of the plant. For operation beyond the original 40-year service life, the 40-year design cycles were evaluated against 60-year projected operational cycles and were determined to be bounding for the period of extended operation, except in certain specific cases described in the UFSAR License Renewal Supplement (Chapter 19, section 19.2.3.1.1, 19.2.3.1.2, and 19.2.3.1.3). The Metal Fatigue of Reactor Coolant Pressure Boundary Program (section 19.1.27) will ensure that components do not exceed their fatigue design bases.

The transients used in the design and fatigue analysis of all ASME III Class 1 components and their supports in the BOP scope are defined in the Piping Design Specification.

3.9B.1.2 Computer Programs Used In Analysis

Lists of computer programs that are used in the design of Seismic Category I components and piping systems within the BOP scope are provided in Appendices 3A.2 and 3A.3, respectively. Also included in these sections are brief descriptions of each program, the extent of its application, and program verifications which demonstrate the applicability and validity of each program.

3.9B.1.3 Experimental Stress Analysis

No experimental stress analysis has been employed in lieu of analytical methods for the design of BOP equipment, components, and piping systems.

3.9B.1.4 Considerations for the Evaluation of the Faulted Condition

The elastic analysis techniques described in Section 3.7B.3.1.1 are utilized in the qualification of Seismic Category I ASME Code and non-code equipment within BOP scope. Stress limits utilized for the faulted plant condition are as outlined in Section 3.9B.2.2. The design conditions and stress limits defined are applicable for an elastic system (and equipment) analysis. Inelastic analyses have not been employed. Should inelastic analysis be required, detailed design bases, demonstrating maintenance of function and/or structural integrity, will be established prior to implementation.

The loading conditions considered in the analysis and the procedure used for the modeling and analytical methods for the reactor coolant loop and supports are provided in Section 5.4.14.

The procedure used for modeling of piping systems, and analytical methods employed for the pipe stress analysis, are provided in Section 3.7B.3.

3.9B.2 Dynamic Testing and Analysis

3.9B.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects Testing

A preoperational vibration, thermal expansion (in discrete temperature step increments), and dynamic effects testing program will be conducted during start-up testing on: 1) ASME Code Class 1, 2, and 3 piping systems within BOP scope; 2) other high energy piping systems inside Seismic Category I structures; 3) high energy portions of systems whose failure could reduce the functioning of any Seismic Category I plant feature to an unacceptable safety level; and 4) Seismic Category I portions of moderate energy piping systems located outside containment. The purpose of the tests is to confirm that these piping systems, restraints, components, and supports have been designed adequately to withstand the flow induced dynamic loadings under operational transient and steady-state conditions anticipated

during service, and to confirm that normal thermal motion is not restrained.

The preoperational tests will be performed to verify, as nearly as possible, the performance of the systems under actual operating conditions. Where required, simulated signals or inputs are used to demonstrate the full operating range of the systems that are used during normal operation, and verify and calibrate as close as possible to actual operating conditions. Systems that are not used during normal plant operation but must be in a state of readiness to perform safety functions, are checked under various modes and test conditions prior to initial BVPS-2 start-up. Whenever practical, these tests are performed under the conditions expected when the systems would be required to function. When these conditions cannot be attained or appropriately simulated at the time of the test, the system is tested to the extent practical under the given conditions, with additional testing completed at a time when appropriate conditions are attained.

A list of the systems and the types of tests being conducted is contained in Table 3.9B-1. The different flow modes of operation and transients to which each system will be subjected during the tests are contained in Chapter 14. The test titles, test prerequisites, test objectives, and summary of testing are also described in Chapter 14.

3.9B.2.1.1 Piping Vibration and Dynamic Effects Testing

For each system defined in items Table 3.9B-1, all flow modes of operation that the systems are subjected to during the tests will be visually observed, where accessible. In addition, selected systems that were stress analyzed for fluid flow instabilities will have instrumented measurements at selected locations for the specific flow modes analyzed.

The measured results will be compared to the analytically predicted values and determined acceptable if they are equal to or less than the predicted values. If the measured values exceed predicted values, acceptability will depend on the remaining margins in the analytically predicted stress levels and the magnitude of the measured loads or displacements. Since both of these parameters are variables dependent on specific locations in the systems, acceptance criteria will be established in the form of a load or displacement limit for each individual location. These load/displacement limits will be documented in the appropriate system test procedures after completion of the stress analysis and prior to preoperational testing for vibration and/or dynamic effects. If the measured loads/displacements exceed the acceptable limits, the analysis will be reviewed to explain the anomaly, an evaluation of the effects on the piping system will be performed, and the total results and evaluation documented to ensure that piping and pipe support criteria still satisfy the applicable code requirements.

Instrumented measurements will also be conducted (as needed) for other systems and conditions. For ASME Code Class 1, 2, and 3 piping systems, design and supervision of the tests, definition of acceptance criteria, evaluations of tests results, and the making of any changes in the piping system necessary to ensure that the piping is adequately designed and supported, are performed as required by Section III of the ASME code.

The selected locations in the piping systems at which visual inspections and measurements (as needed) will be performed and isometrics depicting the locations of these points will be contained in the appropriate test procedures. If vibrations are observed which from visual examination appear to be excessive in the opinion of experienced engineers who will supervise, conduct, and witness the various tests, then either: 1) an instrumented test program will be conducted and the system reanalyzed (or compared to existing analysis) to demonstrate that the observed levels do not cause ASME code stress and fatigue limits to be exceeded; 2) the cause of the vibration will be eliminated; or 3) a corrective support system will be designed and installed and the effect of the modification will be incorporated in the pipe stress analysis. Special attention will be paid to piping sections between supports; connections to instruments, vents, and drains; other free end connections; and motor or diaphragm operators for valves. Instrumented measurement of vibration is generally limited to those systems subject to potential fluid transient loads. The selection of monitoring points will be based upon calculated results and accessibility.

3.9B.2.1.2 Thermal Expansion Testing

Locations for monitoring thermal expansion are chosen based on 1) expected large movements, 2) areas with tight clearances, and 3) certain snubber locations to verify unrestricted motion. Piping snubbers that are monitored during the test program are listed in Table 3.9B-3.

The purpose of the thermal expansion monitoring program is to verify that the piping systems are free from interferences and unexpected restraints on thermal expansion, and that the pipe supports are functioning as intended. During hot functional testing, the piping systems will be raised to their operating temperatures in discrete temperature step increments. Displacement instrumentation and visual observations will be utilized to monitor the thermal expansions at selected temperatures and predetermined locations which will be contained in the appropriate test process. The types of instruments to be used for thermal expansion measurements, requirements that instruments be calibrated, installation details, etc, will also be contained in the specific test procedures for each system.

If the measured thermal motion is not as predicted, system heatup will be terminated and the system will be examined to verify that thermal expansion is not being restricted by any type of interference, and the pipe support system will be examined to verify

that all support components are functioning properly or to locate points of binding of restraints. If improper function of supports are found, adjustments or other corrective actions will be made to eliminate the unacceptable condition. If binding of restraints is found, the restraints will be adjusted to eliminate the unacceptable condition or reanalyzed to verify that the existing condition is acceptable.

If the piping and support system are found to be functioning properly, but thermal expansion movements vary from the predicted values by more than the acceptable tolerance, then the analysis will be reviewed to explain the anomaly, an evaluation of the effects on stress and loads will be performed, and the total results and evaluation documented to ensure that piping and pipe support criteria still satisfy the applicable code requirements. The acceptance tolerance for monitoring thermal expansions will vary for individual locations and/or systems and will be primarily dependent upon the calculated stress levels. The tolerance will be established as a percentage allowable deviation from the expected movements for each individual location. These tolerances will be established and documented after completion of the stress analysis and prior to commencement of hot functional testing. The specific acceptance tolerances will be contained in the appropriate system test procedures.

3.9B.2.2 Seismic Qualification Testing of Safety-Related Mechanical Equipment

The methods and procedures used in the design and qualification of Seismic Category I mechanical equipment within BOP scope are outlined in Sections 3.7B.3.1.1, 3.9B.3, and 3.10B. Loading combinations include operating, as well as earthquake loading, for consideration by testing and/or analytical methods.

Safety-related mechanical equipment (Seismic Category I), not covered by the ASME Boiler and Pressure Vessel Code, are seismically qualified in accordance with the procedures of Section 3.7B.3.1.1. Non ASME Code equipment typically includes diesel generators, fans, coolers, and emergency ventilation equipment. Cranes are seismically qualified in accordance with criteria that preclude the possibility of the crane being dislodged by a seismic disturbance.

Complete results of qualification for safety-related electrical and mechanical equipment were made available at the NRC's Seismic Qualification Review Team (SQRT) and Pump and Valve Operability Review Team (PVORT) Audits performed at the site in accordance with NUREG/CR 3137 “Seismic and Dynamic Qualification of Safety-Related Electrical and Mechanical Equipment."

Except as noted, if codes are used in the design of a component, the guidelines generally require the addition of operating loads to the operating basis earthquake (OBE) load, with no increase in code

allowable stress. If no codes are used, the stress level under the combined loading is limited to 75 percent of the minimum yield strength of the material in accordance with the ASTM specification. The general criteria for analysis of the safe shutdown earthquake (SSE), pipe rupture (if applicable), and operating loads, require that deformation of components be allowed only with no loss of safety function. Stresses under combined loadings are generally limited to the smaller of 100 percent of the minimum yield strength, or 70 percent of the minimum ultimate tensile strength, of the material

(at temperature) in accordance with the ASTM, or equivalent, specification for the material.

3.9B.2.3 Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady State Conditions

Refer to Section 3.9N.2.3.

3.9B.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Refer to Section 3.9N.2.4.

3.9B.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Condition

Refer to Section 3.9N.2.5.

3.9B.2.6 Correlations of Reactor Internals Vibration Tests With the Analytical Results

Refer to Section 3.9N.2.6.

3.9B.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9B.3.1 Loading Combinations, System Operating Transients, and Stress Limits

All ASME III Class 1 piping is designed and analyzed for the following loading conditions to the requirements of the ASME Code Section III, Subsection NB and/or applicable code cases as indicated in the ASME Code Baseline Document. All other Class 1 components are in the NSSS scope and are identified under Section 3.9N.

1. Design Condition
2. Normal Condition
3. Upset Condition
4. Emergency Condition
5. Faulted Condition
6. Test Condition

For ASME III Class 1 piping, the loading combinations and the allowable stress limits are defined in Table 3.9B-5. A comparison of this table with Regulatory Guide 1.48 is given in Table 3.9B-6. The system operating transients used in the design and fatigue analysis of all ASME III Class 1 piping within the BOP scope are defined in the Piping Design Specification. The transients used in the design and fatigue analysis of the RCS piping are also listed in Table 3.9N-1 and described in Section 3.9N. In accordance with ASME Section III, faulted system thermal transients are not included in the fatigue evaluation. BVPS-2 does not consider any emergency thermal transients in fatigue evaluation. The analysis techniques described in Section 3.7B.3.1.2 are utilized in implementing these criteria.

3.9B.3.1.2 ASME III Class 2 and 3 Components

ASME III Class 2 and 3 piping is designed and analyzed to the requirements of the ASME Code Section III, Subsections NC and ND, respectively, and/or applicable code cases as indicated in the ASME Code Baseline Document. Loading combinations and allowable stress limits are defined in Tables 3.9B-8 and 3.9B-9. Analytical requirements for buried piping are addressed in Section 3.7B.3.12.

Table 3.9B-7 provides loading conditions and stress limits for ASME Code Class 2 and 3 components (excluding piping) of Seismic Category I fluid systems,

which are constructed in accordance with ASME III Subsections NC and ND. These conditions generally relate to ASME III, Code Class 1 requirements, and include combinations as follows:

1. Design Condition I

   Includes the specified design loads (temperature, pressure, etc) plus OBE loads.

2. Design Condition II

   Includes the specified design loads (as previously mentioned), SSE loads, and pipe rupture loads (if applicable).

The design loading combinations are analogous to either the Code Class 1 normal or upset conditions for Design Condition I, or to the faulted (or emergency, if applicable) condition for Design Condition II. The stress limits for these design conditions are presented in Table 3.9B-7. Since design temperature and pressure exceed those associated with upset, emergency, and faulted conditions, satisfaction of primary stress limits is ensured.

These requirements which, when implemented, were a supplement to the requirements of ASME III, Subsections NC and ND, are intended to be consistent with the present code format and philosophy.

The stress limits and design conditions presented in Table 3.9B-7 are intended to ensure that no gross deformation of the component occurs. A comparison of Table 3.9B-7 with Regulatory Guide 1.48 is given in Table 3.9B-10. These limits are applicable for an elastic system (and component) analysis. It is not currently anticipated that inelastic deformation will be allowed on any ASME III Code Class 2 and 3 components. If, and when, inelastic analysis is contemplated, detailed design bases, demonstrating maintenance of either function and/or structural integrity, will be proposed prior to implementation. The analysis techniques of Section 3.7B.3.1.1 are utilized in implementing these criteria. The results of qualification analysis and/or tests for ASME III Class 2 and 3 components were made available at the NRC's Seismic Qualification Review Team (SQRT) and Pump and Valve Operability Review Team (PVORT) Audits performed at the site in accordance with NUREG/CR 3137, "Seismic and Dynamic Qualification of Safety-Related Electrical and Mechanical Equipment."

3.9B.3.2 Pump and Valve Operability Assurance

The ASME Boiler and Pressure Vessel Code, Section III, is used to ensure Pressure Boundary Integrity of Nuclear Power Plant Components. It should be noted that the documents ANSI/ASME QP-1, QP-2, QP-4 (formerly N551.1, N551.2, N551.4 respectively), N41.6, and B16.41 referenced by the NRC are used for guidance, and are in draft form pending endorsement by the NRC. The BVPS-2 SQRT/PVORT Qualification Programs address to the maximum practical extent the intent of these guidelines during the preparation of component design specifications and test procedures related to the qualification of ASME Code Class 2 and 3 active pump assemblies and power-operated active valve assemblies. These active pump and valve assemblies are qualified to the specific requirements delineated in this section. A list of all BOP active pumps and valves is presented in Tables 3.9B-18 and 3.9B-19.

Pumps and valves installed in Seismic Category I piping systems are designed in accordance with the requirements of ASME III, NB, NC, and ND. Inactive pumps and valves are designed for the loading combinations of Sections 3.9B.3.1 and 3.9B.3.2, and for the stress limits indicated in Table 3.9B-7.

The equipment types to be covered by the pump and valve operability assurance program are active pumps and valves and will be listed in the PVORT master list of safety-related equipment. This listing will identify the methods, codes, and standards used and the design criteria considered for qualification.

Active components are those that must perform mechanical motions during the course of accomplishing their safety functions.

Inactive components are those for which mechanical movement does not occur in order for the component to accomplish its intended safety function.

Qualification of components is based on an integrated program approach by conformance to design specifications, acceptance testing, and analyses in accordance with appropriate regulatory guide, codes, and standards. Implementation is verified by supporting documents that are audited for compliance and qualification acceptance towards assurance of active pumps and valves operability.

Sections 3.9B.3.2.1 and 3.9B.3.2.2 describe various testing methods performed in accordance with Standard Review Plan Section 3.10 for active pumps and valves. Preoperational testing of pumps such as hydrostatic and various performance tests are performed in the manufacturer’s shop as well as in the installed condition. This testing is in addition to qualification by analysis to faulted (SSE) loads including other operating and dynamic loads associated with the faulted condition.

Safety-related active valves are likewise subjected to a series of stringent tests prior to service to ensure functionability of these valves. These preoperational tests and analyses assure the functional reliability of the pump and valve assemblies for their intended safety system service and performance.

Operability of active pumps and valves is assured by satisfying the requirements of various programs. Safety-related valves are qualified by prototype testing and analysis, and safety-related active pumps by analysis with suitable stress limits and nozzle loads. The content of these programs is detailed as follows.

3.9B.3.2.1 Pump Operability Assurance Program

Active pumps are qualified for operability by being subjected to rigid tests both prior to and after installation in BVPS-2. These tests include:

1. Hydrostatic tests of ASME Section III requirements.

2. Performance tests, while the pump is operated with flow, to determine total developed head, minimum and maximum head, net positive suction head requirements, and other pump/motor parameters.

3. Operability qualification of pump motors for the environmental conditions over the installed life (aging, radiation, accident environmental simulation, etc) according to IEEE Standard 323-1971 or IEEE Standard 382-1972.

Also monitored during these operation tests are bearing temperatures and vibration levels, which are shown to be below appropriate limits specified to the manufacturer for design of each active pump.

After the pump is installed in BVPS-2, it undergoes performance testing and the required periodic in-service inspection and operation as applicable. These tests demonstrate reliability of the pump for the design life of BVPS-2.

In addition to these tests, the safety-related active pumps are qualified for operability during an SSE condition by ensuring that:

1. The pump is not damaged during the seismic event.

2. The pump continues operating when subjected to the SSE loads.

The pump manufacturer is required to show that the pump operates normally when subjected to the maximum applicable amplified seismic (floor) accelerations, attached piping nozzle loads, and dynamic system loads associated with the faulted operating condition. Analysis and/or testing procedures are utilized in accordance with

those outlined in Section 3.7B.3.1.1. Natural frequency calculations are performed in order to determine maximum seismic accelerations based on applicable amplified (floor) response spectra.

In order to avoid damage during the faulted condition, the stresses caused by the combination of normal operating loads, SSE, and dynamic system loads are limited as indicated in Table 3.9B-7. The average membrane stress (σ_m) for the faulted conditions loads is maintained at 1.2 S, or approximately 0.75 σ_y (σ_y = yield stress) and the maximum stress in local fibers (σ_m + bending stress σ_b) is limited to 1.8 S. In addition, the pump stresses caused by the maximum seismic nozzle loads are limited to the stresses outlined in Table 3.9B-7. The maximum seismic nozzle loads are also considered in an analysis of the pump supports to assure that an unacceptable level of misalignment cannot occur. A static shaft deflection analysis of the rotor is performed with horizontal and vertical accelerations based on floor response levels. The deflections determined from the static shaft analysis are compared to the allowable rotor clearances. The nature of seismic disturbances dictates that the maximum contact (if it occurs) is of short duration.

Performance of these analyses with the conservative loads stated and, with the restrictive stress limits of Table 3.9B-7 as allowables, assures that critical parts of the pump are not damaged during the short duration of the faulted condition, and that the reliability of the pump for post-faulted condition operation is not impaired by the seismic event.

In addition to the post-faulted condition operation, it is necessary to assure that the pump functions throughout the SSE. The pump/motor combination is designed to rotate at a constant speed under all conditions unless the rotor becomes completely seized (no rotation). Typically, the rotor can be seized 5 full seconds before a circuit breaker trips the pump, to prevent damage to the motor. However, the high rotary inertia in the operating pump rotor and the nature of the random, short-duration loading characteristics of the seismic event prevent the rotor from becoming seized. In actuality, the seismic loadings cause only a slight increase, if any, in the torque (motor current) necessary to drive the pump at the constant design speed. Therefore, the pump does not shut down during the SSE, and operates at the design speed despite the SSE loads.

To complete the seismic qualification procedures, the pump motor is independently qualified for operation during the maximum seismic event. Any auxiliary equipment which is vital to the operation of the pump, or pump motor, and which is not qualified for operation during the pump analysis or motor qualifications, is also separately qualified for operation at the accelerations occurring at its mounting. The pump motor and vital auxiliary equipment are qualified by meeting the requirements of IEEE Standard 344-1971. If the testing option is chosen, sinusoidal or sine-beat testing is justified by satisfying one or more of the following requirements to demonstrate that the multi-frequency response is negligible or the input is of sufficient magnitude to conservatively account for this effect:

1. The equipment response is basically due to one mode.

2. The sinusoidal or sine-beat response spectra envelop the floor response spectra in the region of significant response.

3. The floor response spectra consist of one dominant mode and have a peak at this frequency.

The degree of coupling in the equipment, in general, determines if a single or multi-axis test is required. Multi-axis testing is required if there is considerable cross coupling. If coupling is very light, then single axis testing is justified. If the degree of coupling can be determined, then single axis testing can be used with the input sufficiently increased to include the effect of coupling on the response of the equipment.

From previous arguments, the safety-related pump/motor assemblies are not damaged and continue operating under SSE loading, and, therefore, perform their intended functions. These proposed requirements take into account the complex characteristics of the pump and are sufficient to demonstrate and assure the seismic operability of the active pumps.

The functional ability of active pumps after a faulted condition is assured since only normal operating loads and steady-state nozzle loads exist. Since it is demonstrated that the pumps are not damaged during the faulted condition, the post-faulted condition operating loads are identical to the normal BVPS-2 operating loads. This is assured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions, and post-faulted conditions, are limited by the magnitudes of the normal condition nozzle loads. The post-faulted condition ability of the pumps to function under these applied loads is proven during the normal operating BVPS-2 conditions for active pumps.

3.9B.3.2.2 Valve Operability Assurance Program

Safety-related active valves must perform their mechanical motion either coincidentally with a postulated event or after a postulated event has occurred. Assurance that these valves will operate when required must be supplied. Qualification tests accompanied by analyses are conducted for all active valves.

Valves without significant extended structure are proven seismically adequate by analysis of piping seismic adequacy. For valves with operators having significant extended structures, and if these structures are essential to maintaining pressure integrity, analysis is performed based upon static forces resulting from equivalent earthquake accelerations acting at the centers of gravity of the extended masses. For active valves, this requirement for analysis is

extended to the mechanical (nonpressure boundary) components of valve top-works to ensure operability.

The safety-related valves are subjected to a series of stringent tests prior to service and during BVPS-2 life. Prior to installation, the following tests are performed:

1. Shell hydrostatic test to ASME III requirements.

2. Backseat (if applicable) and main seat leakage tests.

3. Disc hydrostatic test.

4. Functional tests to verify that the valve will open and close within the specified time limits when subjected to the design differential pressure.

5. Operability qualification of motor operators for the environmental conditions over the installed life (aging, radiation, accident environmental simulation, etc) according to IEEE Standard 323-1971 or IEEE Standard 382-1972.

Cold hydro qualification tests, hot functional qualification tests, periodic in-service inspections (ISI), and periodic in-service operation are performed in situ to verify and assure the functional ability of the valve. These tests guarantee reliability of the valve for the design life of BVPS-2. The valves are designed using either stress analyses or the pressure containing minimum wall thickness requirements. On all active valves, an analysis of the extended structure is also performed for static equivalent seismic SSE loads supplied at the center of gravity of the extended structure. The maximum stress limits allowed in these analyses show structural integrity. The limits that are used for Class 2 and 3 active valves are shown in Table 3.9B-7.

In addition to these tests and analyses, representative valves of each design type are tested for verification of operability during a simulated seismic event by demonstrating operational capabilities within the specified limits. The testing procedures are described as follows.

The valve is mounted in a manner which conservatively represents a typical valve installation. The valve includes the operator and all appurtenances normally attached to the valve in service.

The operability of the valve during an SSE is demonstrated by satisfying the following criteria:

1. Except as stated by the following, active valves are designed to have a first natural frequency greater than 33 Hz. This may be shown by suitable test or analysis.

2. The actuator and yoke of the valve system are statically loaded an amount greater than that determined by an analysis, as representing SSE accelerations applied at the center of gravity of the operator alone in the direction of the weakest axis of the yoke. The design pressure of the valve is simultaneously applied to the valve during the static deflection tests.

3. The valve is then operated while in the deflected position (from the normal operating mode to the faulted mode). The valve must perform its safety-related function within the specified operating time limits.

4. Motor operators and other electrical appurtenances necessary for operation are qualified as operable during an SSE by appropriate IEEE Seismic Qualification Standards, such as IEEE 382-1972 and IEEE 344-1971, prior to their installation on the valve.

The accelerations used for static valve qualification are 3.0 g horizontal and 3.0 g vertical. The piping designer maintains the motor operator accelerations to these levels with an adequate margin of safety.

If the frequency of the valve, by test or analysis, is less than 33 Hz, the valve system is analyzed to determine the equivalent acceleration applied during the static test. The analysis provides the amplification of the input acceleration considering the natural frequency of the valve and the frequency content of the applicable plant floor response spectra. The adjusted acceleration is then used in the static analysis and valve operability is assured by the methods outlined in steps 2 through 4 stated previously, using the modified acceleration input.

This testing program applies only to valves with overhanging structures (the motor operator). The testing is conducted on a representative number of valves. Valves from each of the primary safety-related design types (for example, motor-operated gate valve) are tested. Valve sizes which cover the range of sizes in service are qualified by the tests and the results are used to qualify all valves within the intermediate range of sizes. Stress and deformation analysis are used to support the interpolation.

Valves which are safety-related, but can be classified as not having an overhanging structure, such as check valves and safety-relief valves, are considered separately.

Check valves are characteristically simple in design and their operation is not affected by seismic accelerations or the maximum applied nozzle loads. The check valve design is compact and there

are no extended structures or masses whose motion could cause distortions which could restrict operation of the valve. The nozzle loads due to maximum seismic excitation do not affect the functional ability of the valve, since the valve disc is designed to be isolated from the casing wall. The clearance supplied by the design around the disc prevents the disc from becoming bound or restricted due to any casing distortions caused by nozzle loads. Therefore, the design of these valves is such that once the structural integrity of the valve is assured using standard design or analysis methods, the ability of the valve to operate is ensured by the design features. In addition to these design considerations, the valve also undergoes the following tests and analysis:

1. Stress analysis including the SSE loads,

2. In-shop hydrostatic test,

3. In-shop seat leakage test, and

4. Periodic in situ valve exercising and inspection to ensure the functional ability of the valve.

Safety and relief valves are subjected to tests and analyses similar to check valves, that is, stress and deformation analyses for SSE loads, in-shop hydrostatic and seat leakage tests, and periodic in situ valve inspection. In addition, a static load equivalent to the SSE is applied to the top of the bonnet, and the pressure is increased until the valve mechanism is activated. Successful actuation within design requirements ensures its overpressurization safety capabilities during a seismic event.

Using the methods described, all the safety-related valves in the system are qualified for operability during a seismic event. These methods conservatively simulate the seismic event and ensure that the active valves perform their safety-related function when necessary.

Alternative valve operability testing, such as dynamic vibration testing, is allowed if it is shown to adequately ensure the faulted condition functional ability of the valve system. The results of qualification analysis and/or tests for active pumps and valves were made available at the NRC's Seismic Qualification Review Team (SQRT) and Pump and Valve Operability Review Team (PVORT) Audits performed at the site in accordance with NUREG/CR3137 “Seismic and Dynamic Qualification of Safety-Related Electrical and Mechanical Equipment.”

3.9B.3.3 Design and Installation of Pressure Relief Devices

Pressure relieving devices, consisting of safety and relief valves, are provided in ASME III Code Class 1, 2 and 3 piping systems to prevent overpressurization. The design of these valves takes into consideration the reaction force when the valve opens.

The design criteria for all pressure relieving devices are in accordance with the rules of ASME Boiler and Pressure Vessel Code Section III. In addition, relief and safety valve discharge events are considered as occasional loads and classified as upset, emergency, or faulted, depending upon the expected frequency of occurrence, with the exception that safety valve discharge events in the RCS are classified as either emergency or faulted only. These loads are combined with other service loadings and maintained within the appropriate stress limits as indicated in Section 3.9.3.1.

3.9B.3.3.1 Open Relief System

An open relief system is one where the fluid is discharged directly to atmosphere or to a vent pipe that is structurally uncoupled from the safety valve, and the discharge point is located within 4 discharge pipe diameters from the centerline of the valve inlet and within 6 discharge pipe diameters from the centerline of the valve discharge. Systems not meeting these dimensional criteria and/or where the effluent is carried to a discharge pipe connected directly to the safety valve are considered as closed discharge systems.

The total steady state load for an open system discharge is expressed as the sum of pressure and momentum forces as follows:

$$F = P_eA + (W/g_c)\ V_e$$

where:

F	=	reaction force (lbf)
P_e	=	static gage pressure at exit point (psig)
A	=	exit flow area (in^2)
W	=	mass flow rate (relieving capacity stamped on the valve X 1.11, adjusted for units to be compatible if necessary)(lbm/sec)
V_e	=	exit fluid velocity (ft/sec)
g_c	=	gravitational constant (32.2 lbm-ft/lbf-sec^2)

The response of the piping system due to this applied load is determined either by time-history dynamic analysis or by a static analysis. Both types of analysis are performed using the NUPIPE-SW program, described in Appendix 3A.

If time-history analysis is employed, the forcing function applied at the point of discharge is approximated by a linear forcing function with the force ranging from zero to the calculated reaction force over the period of time ranging from zero to the effective valve opening time "t"; after the time "t", the reaction force remains steady until the conclusion of acceptable time-history integration.

If a static analysis method is employed, a dynamic load factor (DLF) is applied to the reaction force to ensure the consideration of the dynamic effects of the suddenly applied load. A conservative DLF of

conservative DLF of 2.0 is assumed in most instances. However, a smaller value of DLF may be determined by relating the reaction force rise time to the valve assembly natural period and utilizing the appropriate figure from Biggs (1964).

In this type of analysis, the calculated maximum discharge force multiplied by the appropriate DLF is applied to the discharge point in a direction opposite to the flow.

Where more than one valve is mounted on a common header, multiple valve discharge cases are considered. Simultaneous discharge of various possible valve combinations are considered consistent with system design and operating procedures to establish the design basis.

3.9B.3.3.2 Closed Relief System

A closed relief system may be either a system which discharges into a closed vessel or an open discharge system with longer discharge pipes than allowed for an open system. Of particular concern in closed relief systems are the large forces that may occur on piping that contains water seals (slug flow), two-phase flow, or if there is a water column in the discharge piping.

To establish the forcing functions necessary to perform a structural analysis of the piping, thermal/hydrodynamic models of the piping system are constructed. These models consist of one-dimensional representation of the piping system divided into reservoirs, pumps, valves, lengths of piped segments, branch connections, and other special piping components. Effects such as flow restrictions and frictional resistance are considered. The time dependent pressure, temperature, density, velocity, and momentum are computed. Unbalanced segment forces are then obtained as function of time.

The forcing functions are then applied to the structural model and system responses are determined by performing a time-history dynamic analysis using the NUPIPE-SW program or other commercial NUPIPE program.

Stone & Webster Engineering Corporation programs used to determine fluid forcing functions are STEHAM, WATHAM, and WATSLUG. Appendix 3A provides a description of each code.

As an alternative to using computer programs to generate fluid transient forcing functions, conservative hand calculations are performed to develop bounding pipe segment forces. These forces then may be analyzed statically using an appropriate DLF, or dynamically to obtain piping structural responses.

3.9B.3.4 Component Supports

3.9B.3.4.1 Equipment Component Supports

In general, equipment component supports are not considered to be within the jurisdiction of ASME III since Subsection NF was not in effect at the time most equipment component support systems were purchased. However, ASME criteria was used extensively as guidance in terms of both design and fabrication. Equipment component supports for the RCS, which were considered to be on the level of ASME III, Class 1, are described in Section 5.4.14. All remaining supports, which were considered to be within a classification of ASME III, Class 2 and 3, are described in Section 3.9B.2. A small number of ASME III Class 2 and 3 components have been supplied for which the support structures are according to ASME III, NF Requirements, due to the procurement date of the component(s).

3.9B.3.4.1.1 Loads, Load Combinations, and Stress Limits

Loads, load combinations, and stress limits for equipment component supports are identified in Table 3.9B-16.

3.9B.3.4.2 Piping Component Supports

3.9B.3.4.2.1 Loading Combinations, System Operating Transients, and Stress Limits

Loading combinations, system operating transients, and stress limits for piping component supports fall into two categories:

a. That part of the support which transmits loads from the pressure retaining boundary to the building structure (welded attachment) is considered locally as an extension of the piping and falls under the jurisdiction of ASME III outlined in Sections 3.9B.3.1.1 and 3.9B.3.1.2.

b. That part of the support other than the pressure boundary interface is evaluated as a piping component support structure as outlined in Section 3.9B.3.4.2.2.

3.9B.3.4.2.2 Piping Component Support Structures

a. Piping component support structures are defined as that part of the support structure other than the pressure boundary interface. This includes supplemental material such as plate, structural steel, structural bolts, etc. It also includes pre-fabricated vendor components such as struts, snubbers, spring hangers, concrete anchor bolts, etc.

b. The load combinations used for the analysis of piping component support structures are combined in a conservative manner consistent with the requirements of ANSI B31.7-1969,

Division 1-720 and 1-721, and ANSI B31.1.0-1967, Paragraphs 120 and 121. [USA Standard Code for Pressure Piping, Nuclear Power Piping ANSI B31.7-1969 and USA Standard Code for Pressure Piping, Power Piping ANSI B31.1.0-1967.] The load combinations and allowable stresses are shown in Tables 3.9B-14 and 3.9B-15.

c. Piping component support structure design loads are arrived at by the summation of the resultant forces and moments for sustained loads, occasional loads (including building settlement and hydrotest loads), dynamic loads, and thermal expansion loads. The summation of these loads produce a maximum positive design load and a maximum negative design load. (Earthquake and other cyclic loads are combined by SRSS).

d. Stress levels on supplemental material generated by the design loads are compared to the allowable stresses consistent with those in the AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings - 1969, the ANSI (B31.1) Code, and Tables 3.9B-14 and 3.9B-15.

e. Prefabricated vendor components are qualified by comparing the support design loads to the rated loads of the component(s) as specified by the vendor(s).

3.9B.3.4.2.3 Standard Prefabricated Piping (Hanger) Supports

The design of all hanger supports conforms to the requirements of the Manufacturers Standardization Society Standard Practices SP58 and SP69, and the requirements of ASME III (use of code cases are in accordance with Regulatory Guides 1.84 and 1.85 unless otherwise stated herein).

3.9B.3.4.2.4 Snubbers

1. Mechanical Properties

   Mechanical and hydraulic snubbers are utilized to limit piping movements resulting from dynamic loadings while permitting normal thermal expansion. Snubbers are not provided for use as vibration arrestors.

2. Structural Analysis and System Evaluation

   a. The entire piping system is mathematically modeled for complete structural analysis. In the mathematical model for dynamic loading, the snubbers are modeled as struts with spring stiffness dependent on snubber size. The analysis determines the forces and moments acting on each component and the forces acting on the snubbers due to

all the dynamic loading conditions defined in the piping design specification. The design load on snubbers includes those loads caused by seismic forces (OBE, SSE, seismic anchor movements), and reaction forces caused by relief valve discharge, turbine stop valve closure, etc.

b. The flexibility of support components is considered in the support designs. The assigned stiffness values are adequate for determining pipe stress and pipe support loads by computerized pipe analysis (see Appendix 3A for descriptions of computer programs used).

c. Snubber end fitting clearance and lost motion are minimized in the design and fabrication of the piping component supports.

3. Design Specifications

All snubbers are designed, manufactured, tested, inspected, and handled in strict accordance with the applicable design specifications. Each design specification addresses the following:

a. Functional requirements

b. Operating environment

c. Applicable codes and standards

d. Materials and fluid standards

e. Environmental, structural, and performance design verification tests

f. Production unit functional verification tests and certification

g. Packaging, shipping, handling, and storage requirements

h. Attachment and installation provisions

i. Quality assurance and assembly quality control procedures

4. Installation and Operability Verification

a. All snubbers are installed as detailed on the pipe support drawings and are designed and set to accommodate normal thermal expansion of the pipe. The manufacturer's installation procedure is packaged with each snubber as it is sent to the job site.

b. The installed cold position of a snubber takes into account the predicted thermal displacements of the pipe obtained from the stress analysis.

c. After inspection of the pipe and installation of the supports, the snubber is adjusted to the cold position indicated on the support drawing.

d. Snubbers are designed to be accessible for removal and installation as required for functional testing and periodic plant maintenance.

5. Use of Additional Snubbers

The effects of snubber installation during or after plant construction, which may not have been included in the

original design analysis, are incorporated in the pipe stress analysis which takes into consideration any effects the additional snubbers would have on the piping system. The stress analysis is then reviewed, and an evaluation of the effects on stress and loads is performed and documented to ensure that the piping and piping supports satisfy applicable code requirements.

6. Inspection and Testing

   a. Prior to plant start-up, snubbers are inspected to verify operability and proper installation and cold position setting. During preoperational and initial start-up testing, thermal and vibratory movement, clearance adequacy, and hot position setting are measured and/or visually observed as specified in Section 3.9B.2. The acceptance criteria and procedure for measuring these movements are included in the appropriate system test procedures. Corrective action is taken to resolve problems observed during system testing.

   b. The plant maintenance program will establish a procedure for the periodic inspection and functional testing of snubbers to verify operability.

   c. After erection, inspections are made to assure adequate clearance and support during the hot and cold position.

7. Classification and Identification of Safety-Related Snubbers

   a. All mechanical and hydraulic snubbers utilized on safety-related piping components are designed and procured to the requirements of the applicable pipe support design specification. Snubber manufacturers fabricate snubber units in accordance with the ASME B&PV Code, Section III, Subsection NF. Fabrication to NF requirements only provides added conservatism and exceeds BVPS-2 design commitments; however, the snubber units are not considered as NF components. It shall not be construed to mean that any NF requirements are being invoked for the design, analysis, testing, maintenance, inspection, surveillance, etc. for any portion of the snubber assemblies. The entire snubber assembly (that is, pipe attachment, snubber unit, and supporting structure) is considered a non-NF component.

   b. Table 3.9B-13 classifies and identifies snubbers as to their system, component, number, type, and supplier.

3.9B.3.4.3 Duct Supports

The analytical methods for the design of Seismic Category 1 duct supports are the same as those methods described for equipment and components in Section 3.7B.3.1.1. The loading combinations and allowable stress limits are the same as for pipe supports shown in Table 3.9B-14. However, loadings due to relative building displacements and thermal expansion of the duct are not considered because of the existence of flexible joints in the ducts at building shake spaces and between the supports.

3.9B.4 Refer to Section 3.9N.

3.9B.5 Refer to Section 3.9N.

3.9B.6 Inservice Testing of Pumps and Valves

The inservice test program includes baseline preservice and periodic inservice testing to ensure that all applicable pumps and valves remain in a state of operational readiness throughout the life of the plant. The inservice test program is based on the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (OM Code). A listing of all pumps and valves requiring inservice testing can be found in the BVPS-2 Inservice Testing (IST) Program for Pumps and Valves.

3.9B.6.1 Inservice Testing of Pumps

The inservice test program applies to certain safety-related ASME Class 1, 2 or 3 centrifugal and positive displacement pumps that are provided with an emergency power source, and are required for shutting down a reactor to the cold shutdown condition, maintaining the cold shutdown condition, or mitigating the consequences of an accident at BVPS-2. The required pump parameters and test frequencies are discussed in the BVPS-2 Inservice Testing (IST) Program for Pumps and Valves.

3.9B.6.2 Inservice Testing of Valves

The inservice test program applies to certain ASME Class 1, 2 or 3 valves that are required to perform a specific function in shutting down a reactor to the cold shutdown condition, maintaining the cold shutdown condition, or mitigating the consequences of an accident at BVPS-2. The valve test requirements and test frequencies are discussed in the BVPS-2 Inservice Testing (IST) Program for Pumps and Valves.

3.9N MECHANICAL SYSTEMS AND COMPONENTS

3.9N.1 Special Topics for Mechanical Components

3.9N.1.1 Design Transients

The following five operating conditions defined in ASME Section III are considered in the design of the RCS components and reactor internals:

1. Normal conditions

   Any condition in the course of start-up, operation in the design power range, hot standby, and system shutdown, other than upset, emergency, faulted, or testing conditions.

2. Upset conditions (incidents of moderate frequency)

   Any deviations from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The upset conditions include those transients which result from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, and transients due to loss of load or power. Upset conditions include any abnormal incidents not resulting in a forced outage, and also forced outages for which the corrective action does not include any repair of mechanical damage. The estimated duration of an upset condition is included in the design specifications.

3. Emergency conditions (infrequent incidents)

   Those deviations from normal conditions which require shutdown for correction of the conditions or repair of damage in the system. These conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system. The total number of postulated occurrences for such events shall not cause more than 25 stress cycles having an S value greater than that for 10^6 cycles from the applicable fatigue design curves of the ASME Code, Section III. Beaver Valley Power Station - Unit 2 does not have any transients classified as "emergency", except as described in Section 3.9N.5.2.

4. Faulted conditions (limiting faults)

   Those combinations of conditions associated with extremely low probability postulated events, whose consequences are such that the integrity and operability of the nuclear energy system may be impaired to the extent that consideration of public health and safety is involved. Such considerations require compliance with safety criteria as may be specified by jurisdictional authorities.

5. Test conditions

   Test conditions are those tests in addition to the hydrostatic or pneumatic tests permitted by the ASME Code, Section III, including leak tests or subsequent hydrostatic tests.

To provide the necessary high degree of integrity for the equipment in the RCS, the transient conditions selected for equipment fatigue elevation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from various operating conditions in the plant. To a large extent, the specific transient operating conditions to be considered for equipment fatigue analyses are based upon engineering judgment and experience. The transients selected are representative of operating conditions which prudently should be considered to occur during plant operation and are sufficiently severe or frequent to be of possible significance to component cyclic behavior. The transients selected may be regarded as a conservative representation of transients which, when used as a basis for component fatigue evaluation, provide confidence that the component is appropriate for its application over the design life of BVPS-2.

The following design conditions are given in the equipment specifications for RCS components. The design transients and the number of cycles of each that is used for fatigue evaluations are

shown in Table 3.9N-1. In accordance with the ASME Section III, faulted conditions are not included in fatigue evaluations.

Normal Conditions

The following primary system transients are considered normal conditions:

1. Heatup and cooldown at 100°F/hr (maximum)

   The design heatup and cooldown transients are conservatively represented by continuous operations performed at a uniform temperature rate of 100°F/hr maximum. (These operations can take place at lower rates approaching the minimum of 0°F/hr.

   For these cases, the heatup occurs from ambient (assumed to be 120°F) to the no-load temperature and pressure condition and the cooldown represents the reverse situation. In actual practice, the rate of temperature change of 100°F/hr will not be attained because of other limitations such as:

   a. Slower initial heatup rates when using pump energy only.

   b. Interruptions in the heatup and cooldown cycles due to such factors as drawing a pressurizer steam bubble, rod withdrawal, sampling, water chemistry, and gas adjustments.

2. Unit loading and unloading at five percent of full power per minute

   The unit loading and unloading cases are conservatively represented by a continuous and uniform ramp power change of five percent per minute and between 0 percent load and full load. This load swing is the maximum possible consistent with operation of the reactor control system. The reactor temperature will vary with load as prescribed by the reactor control system.

3. Step load increase and decrease of ten percent of full power

   The $\pm$ 10 percent step change in load demand is a transient which is assumed to be a change in turbine control valve opening due to disturbances in the electrical network into which the plant output is tied. The reactor control system

is designed to restore plant equilibrium without reactor trip following a ± 10 percent step change in turbine load demand initiated from nuclear plant equilibrium conditions in the range between 15 percent and 100 percent full load. In effect, during load change conditions, the reactor control system or the operator attempts to match turbine and reactor outputs in such a manner that peak reactor coolant temperature is minimized and reactor coolant temperature is restored to its programmed set point at a sufficiently slow rate to prevent excessive pressurizer pressure decrease.

Following a step decrease in turbine load, the secondary side steam pressure and temperature initially increase since the decrease in nuclear power lags behind the step decrease in turbine load. During the same increment of time, the RCS average temperature and pressurizer pressure also initially increase. Because of the power mismatch between the turbine and reactor and the increase in reactor coolant temperature, the control system automatically inserts the control rods to reduce core power. With the load decrease, the reactor coolant temperature will ultimately be reduced from its peak value to a value below its initial equilibrium value at the inception of the transient. The reactor coolant average temperature set point change is made as a function of turbine generator load as determined by first stage turbine pressure measurement. The pressurizer pressure will also decrease from its peak pressure value and follow the reactor coolant's decreasing temperature trend. At some point during the decreasing pressure transient, the saturated water in the pressurizer begins to flash, which reduces the rate of pressure decrease. Subsequently, the pressurizer heaters come on to restore the plant pressure to its normal value.

Following a step increase in turbine load, the reverse situation occurs; that is, the secondary side steam pressure and temperature initially decrease, and the reactor coolant average temperature and pressure also initially decrease. The operator may withdraw the control rods to increase core power. The decreasing pressure transient is reversed by actuation of the pressurizer heaters and eventually the system pressure is restored to its normal value. Alternatively an operator may decrease turbine load as described by administrative controls. The reactor coolant average temperature will be raised to a value above its initial equilibrium value at the beginning of the transients.

4. Large step load decrease with steam dump

This transient applies to a step decrease in turbine load from full power, of such magnitude that the resultant rapid increase in reactor coolant average temperature and

secondary side steam pressure and temperature will automatically initiate a secondary side steam dump that will prevent both reactor trip and lifting of steam generator safety valves. Thus, when a unit is designed to accept a step decrease of 95 percent from full power (complete loss of outside load but retaining the plant auxiliary load of 5 percent), the steam dump system provides the heat sink to accept 85 percent of the turbine load. The remaining 10 percent of the total step change is assumed by the reactor control system (control rods). If a steam dump system was not provided to cope with this transient, there would be such a strong mismatch between turbine steam demand and reactor thermal output that a reactor trip and lifting of steam generator safety valves would occur.

5. Steady-state fluctuations

The reactor coolant average temperature, for purpose of design, is assumed to increase and decrease a maximum of 6°F in one minute. The temperature changes are assumed to be around the programmed value of T_{avg} (T_{avg} $\pm$ 3°F). The corresponding reactor coolant average pressure is assumed to vary accordingly (2,250 $\pm$ 25 psia).

6. Refueling

At the beginning of the refueling operation, the RCS is assumed to have been cooled down to 140°F. The vessel head is removed, and the refueling canal is filled. This is done by transferring water from the refueling water storage tank, which is outdoors and conservatively assumed to be at 32°F, (piping maintained at 45°F), into the loops by means of the low head safety injection pumps. The refueling water flows directly into the reactor vessel via the cold legs.

<u>Upset Conditions</u>

The following primary system transients are considered upset conditions:

1. Loss of load (without immediate reactor trip)

This transient applies to a step decrease in turbine load from full power (turbine trip) without immediately initiating a reactor trip, and represents the most severe pressure transient on the RCS under upset conditions. The reactor eventually trips as a consequence of a high pressurizer level trip initiated by the reactor protection system. Since redundant means of tripping the reactor are provided as part of the reactor protection system (RPS), transients of this nature are not expected but are included to ensure a conservative design.

2. Loss of power

   This transient applies to a blackout situation involving the loss of all outside electrical power to the station, followed by reactor and turbine trips. Under these circumstances, the reactor coolant pumps (RCPs) are de-energized, and following coastdown of the RCPs, natural circulation in the system develops into an equilibrium condition. This condition permits removal of core residual heat through the steam generators, which at this time are receiving feedwater from the auxiliary feedwater system. Steam is removed for reactor cooldown through steam generator safety valves provided for this purpose.

3. Loss of flow

   This transient applies to a partial loss of flow from full power, in which an RCP is tripped out of service as the result of a loss of power to that pump. The consequences of such an accident are a reactor and turbine trip, on low reactor coolant flow, followed by automatic opening of the steam dump system, and flow reversal in the affected loop. The flow reversal causes reactor coolant at cold leg temperature to pass through the steam generator and be cooled still further. This cooler water then flows through the hot leg piping and enters the reactor vessel outlet nozzles. The net result of the flow reversal is a sizeable reduction in the hot leg coolant temperature of the affected loop.

4. Reactor trip from full power

   A reactor trip from full power may occur from a variety of causes resulting in temperature and pressure transients in the RCS and in the secondary side of the steam generator. This is the result of continued heat transfer from the reactor coolant in the steam generator. The transient continues until the reactor coolant and steam generator secondary side temperatures are in equilibrium at zero power conditions. A continued supply of feedwater and controlled dumping of steam will remove the core residual heat and prevent the steam generator safety valves from lifting. The reactor coolant temperature and pressure undergo a rapid decrease from full power values as the RPS causes the control rods to move into the core.

5. Inadvertent safety injection actuation

   A spurious safety injection signal results in an immediate reactor trip followed by actuation of the high head safety injection (HHSI)/charging pumps. These pumps deliver borated water from the refueling water storage tank to the RCS cold legs. The initial portion of this transient is similar to the reactor trip from full power with no cooldown. Controlled steam dump and auxiliary feedwater flow after trip removes core residual heat. Reactor coolant temperature and pressure decrease as the control rods move into the core.

Later in the transient, the injected water causes the RCS pressure to increase and the primary and secondary temperatures to decrease gradually. The transient continues until the operator stops the HHSI/charging pumps. It is assumed that the plant is then returned to no load conditions, with pressure and temperature changes controlled within normal limits.

6. Operating basis earthquake

   The mechanical stresses resulting from the OBE are considered on a component basis. Fatigue analysis, where required by the codes, is performed by the supplier as part of the stress analysis report. The earthquake loads are a part of the mechanical loading conditions specified in the equipment specifications. The origin of their determination is separate and distinct from those transients resulting from fluid pressure and temperature. They are, however, considered in the design analysis.

7. Reactor Coolant System Cold Overpressurization

   RCS cold overpressurization occurs during startup and shutdown conditions at low temperature, with or without existence of a steam bubble in the pressurizer and is especially severe when the reactor coolant system is in a water-solid configuration. The event is inadvertent, and usually generated by any one of a variety of malfunctions or operator errors. All events which have occurred to date may be categorized as belonging to either events resulting in the addition of mass (mass input transient), or events resulting in the addition of heat (heat input transient). All these possible transients are represented by composite "umbrella" design transients, referred to here as RCS cold overpressurization.

8. Inadvertent Auxiliary Spray

   The inadvertent pressurizer auxiliary spray transient will occur if the auxiliary spray valve is opened inadvertently during normal operation of the unit. This will introduce cold water into the pressurizer with a very sharp pressure decrease as a result.

   The temperature of the auxiliary spray water is dependent upon the performance of the regenerative heat exchanger. The most conservative case is when the letdown stream is shut off and the charging fluid enters the pressurizer unheated. Therefore, for design purposes, the temperature of the spray water is assumed to be 100°F. The spray flow is assumed to be 200 gpm. It is further assumed that the auxiliary spray will, if actuated, continue for five minutes until it is shut off.

   The pressure decreases rapidly to the low pressure reactor trip setpoint. At this pressure, the pressurizer low pressure reactor trip is assumed to be actuated; this accentuates the pressure decrease until the pressure is

finally limited to the hot leg saturation pressure. In five minutes, the spray is stopped, and all the pressurizer heaters return the pressure to 2250 psia.

Emergency Conditions

No transient is classified as an emergency condition.

Faulted Conditions

The following primary system transients are considered faulted conditions:

1. Reactor coolant pipe break (large LOCA)

   Following rupture of a reactor coolant pipe resulting in a large loss-of-coolant, the primary system pressure decreases, causing the primary system temperature to decrease. Because of the rapid blowdown of coolant from the system and the comparatively large heat capacity of the metal sections of the components, it is likely that the metal will still be at or near the operating temperature by the end of blowdown. It is conservatively assumed that the emergency core cooling system (ECCS) is actuated to introduce water at a minimum temperature of 45°F into the RCS. The safety injection signal will also result in reactor and turbine trips.

   This postulated break was eliminated by the application of "leak-before-break" technology for excluding from the design basis the dynamic effects of postulated pipe ruptures in primary coolant piping as allowed by GDC-4 (dated December 17, 1986).

   In addition, postulated breaks for the pressurizer surge line as well as portions of the RCS, RHRS, and SIS have been eliminated by application of "leak-before-break" technology.

   Protection criteria against dynamic effects associated with pipe breaks is covered in Section 3.6.

2. Large main steam line break (MSLB)

   This transient is based on the complete severance of the largest steam line. The following conservative assumptions were made:

   a. The reactor is initially in a hot, zero-power condition.

   b. The MSLB results in immediate reactor trip and ECCS actuation.

   c. A large shutdown margin, coupled with no feedback or decay heat, prevents heat generation during the transient.

   d. The ECCS operates at design capacity and repressurizes the RCS within a relatively short time.

The preceding conditions will result in the most severe temperature and pressure variations which the primary system will encounter during a steam break accident.

3. Steam generator tube rupture

   This accident postulates the DER of a steam generator tube resulting in a decrease in pressurizer level and RCS pressure. Reactor trip will occur due to low pressurizer pressure or overtemperature ΔT and a safety injection signal is also initiated. Safety injection actuation automatically isolates the main feedwater lines by closing the main feedwater isolation valves and trips the main feed pumps. When this accident occurs, some of the reactor coolant blows into the affected steam generator causing the shell side level to rise. Shortly after the break, the primary system pressure is reduced to below the secondary safety valve setting. Subsequent recovery procedures call for isolation of the steam line leading from the affected steam generator. This accident will result in a transient which is no more severe than that associated with a reactor trip from full power accompanied by RCS cooldown and depressurization. Therefore, it requires no special treatment insofar as fatigue evaluation is concerned.

4. Safe shutdown earthquake (SSE)

   The mechanical stresses due to the vibratory motion of the SSE are considered on a component basis.

Test Conditions

The following primary system transients under test conditions are discussed:

1. Turbine roll test

   This transient is imposed upon the plant during the hot functional test period for turbine cycle checkout. Reactor coolant pump power will be used to heat the reactor coolant to operating temperature (no-load conditions) and the steam generated will be used to perform a turbine roll test.

2. Primary side hydrostatic test

   The pressure tests include both shop and field hydrostatic tests which occur as a result of component or system testing. This hydrotest is performed at a water temperature, which is compatible with reactor vessel material ductility requirements, and a test pressure of 3,107 psig (1.25 times design pressure). In this test, the RCS is pressurized to 3,107 psig coincident with steam generator secondary side pressure of 0 psig.

3. Secondary side hydrostatic test

   The secondary side of the steam generator is pressurized to 1.25 design pressure with a minimum water temperature of 120°F coincident with the primary side of 0 psig.

4. Primary side leakage test

   Subsequent to each time the primary system has been opened, a leakage test will be performed. During this test, the primary system pressure is, for design purposes, raised to 2,500 psia, with the system temperature above the minimum temperature imposed by reactor vessel material ductility requirements, while the system is checked for leaks. In actual practice, the primary system is pressurized to less than 2,500 psia, as measured at the pressurizer, to prevent the pressurizer safety valves from lifting during the test.

   During this leakage test, the secondary side of the steam generator must be pressurized so that the pressure differential across the tubesheet does not exceed 1,600 psi.

Supplemental Transients

1. Pressurizer Insurge

   Surge line flow from the Hot Leg to the Pressurizer can cause rapid cooling of the Pressurizer nozzle and lower shell. This transient is most severe during plant startup and shutdown operation since there is a larger temperature difference (ΔT) from the Pressurizer liquid space to the hot leg temperature. Detailed analysis using various increments of ΔT has shown acceptable results. Confirmation of the analytical assumptions is required as a Fatigue Management activity.

2. Selected CVCS Transients

   Interruption of letdown flow stops the preheating of charging flow via the regenerative heat exchanger. This in turn can cause cooling of the charging nozzle. Interruption of charging flow, or excess letdown flow can affect the relative temperatures of the regenerative or letdown heat exchangers. These components are designed for specific temperature ranges and numbers of occurrences. Monitoring of these transients is required as a Fatigue Management activity.

3. Auxiliary Feedwater Injections

   These transients can occur when the plant is at hot standby or no-load conditions. It is assumed that the low steam generation rate is made up by intermittent (slug) feeding of auxiliary feedwater into the steam generator.

   Feedwater additions required during plant heatup and cooldown operations are also assumed to be covered by the Auxiliary Feedwater Injections transient.

Monitoring of these transients is required as a Fatigue Management activity.

4. RHR Actuation

The full design transient for the RHR system is a change from 70°F to 350°F as RCS flow is initiated through the RHR piping, pumps and heat exchangers. This occurs with each plant shutdown. At lower RCS temperatures, the RHR pumps are occasionally stopped and restarted to aid in RCS cleanup activities. Monitoring of these intermediate transients is required as a Fatigue Management activity.

3.9N.1.2 Computer Programs Used in Analysis

The following computer programs have been used in dynamic and static analyses to determine mechanical loads, stresses, and deformations of Seismic Category I components and equipment:

1. ANSYS

ANSYS is a general purpose finite element analysis program with structural and heat transfer capabilities. ANSYS is used as-needed for engineering analysis. ANSYS is a recognized program in the public domain.

2. MULTIFLEX

The Multiflex program is an engineering tool for calculation of hydraulic loads on the reactor internals structure and the pressure vessel during rapid thermal hydraulic transients caused by an imposed driving force on the system. The hydraulic loads are computed in this code by taking into account the hydraulic-structural interactions. It is constructed by incorporating structural models into the thermal-hydraulic code, BLODWN-2A. This code is described by Takeuchi (et al 1976 and 1982.)

3. FORCE

The FORCE-2 computer code determines the vertical hydraulic loads on the reactor vessel internals during blowdown, by utilizing a detailed geometric description of the vessel components and transient pressures and mass velocities computed by the MULTIFLEX code. The FORCE-2 code is applicable for all pressure and mass velocity transients of a postulated LOCA. This code is described by Takeuchi (et al 1976), Appendix B.

4. WECAN

Finite element structural analysis program used to perform analyses of structural systems under the excitation of general forcing functions.

3.9N.1.3 Experimental Stress Analysis

No experimental stress analysis methods are used for Seismic Category I system or components. However, Westinghouse makes extensive use of measured results from prototype plants and various scale model tests, as described in Section 3.9N.2.

3.9N.1.4 Considerations for the Evaluation of the Faulted Condition

3.9N.1.4.1 Loading Conditions

The structural evaluations performed on the RCS components consider the loadings specified as shown in Table 3.9N-2. These loads result from thermal expansion, pressure, weight, OBE, SSE, design basis LOCA, and plant operational thermal and pressure transients.

3.9N.1.4.2 Analysis of Primary Components

Equipment which serves as part of the pressure boundary in the reactor coolant loop includes the steam generators, the RCPs, the pressurizer, and the reactor vessel. This equipment is American Nuclear Society (ANS) Safety Class 1 and the pressure boundary meets the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

The results of the reactor coolant loop analysis are used to determine the loads acting on the equipment nozzles and the support/component interface locations. These loads are supplied for all loading conditions on an "umbrella" load basis. That is, on the basis of previous plant analyses, a set of loads is determined which should be larger than those seen in any single plant analysis. The umbrella loads represent a conservative means of allowing detailed component analysis prior to the completion of the system analysis. Upon completion of the system analysis, conformance is demonstrated between the actual plant loads and the loads used in the analyses of the components. Any deviation where the actual load is larger than the umbrella load is handled by an individualized analysis.

Seismic analyses are performed individually for the RCP, the pressurizer, and the steam generator. Detailed and complex dynamic models are used for the dynamic analyses. The response spectra corresponding to the building elevation at the highest component/building attachment elevation is used for the component analysis. The reactor pressure vessel is qualified by static stress analysis based on loads that have been derived from dynamic analysis.

The pressure boundary portions of Class 1 valves in the RCS are designed and analyzed according to the requirements of NB-3500 of ASME III. These valves are identified in Section 3.9N.3.2.

Valves in sample lines connected to the RCS are not considered to be ANS Safety Class 1 nor ASME Class 1. This is because the nozzles where the line connects to the primary system piping contains a 3/8-inch orifice. This orifice restricts the flow such that a loss through a severance of one of these lines can be made up by normal charging flow.

3.9N.1.4.3 Reactor Vessel Loss-of-Coolant Accident Analysis

The LOCA analysis which is performed for the reactor vessel includes asymmetric pressure distributions on the internals and on the vessel exterior walls.

A detailed 3-D dynamic model of the reactor vessel and internals is prepared, which includes the stiffnesses of the reactor vessel supports and the attached piping. Hydraulic forces are developed in the internals for breaks in the 4-inch (Schedule 160) branch line attached to the cold leg piping and the 3-inch (Schedule 160) branch line attached to the hot leg piping; these forces are characterized by time-dependent forcing functions on the vessel and core barrel. In the derivation of these forcing functions, the fluid-structure (or hydroelastic) interaction in the downcomer region between the barrel and the vessel is taken into account. As a result of the pipe break, loop mechanical loads are also applied to the vessel.

The loads from these sources, the internals hydraulic forces, and the loop mechanical forces, are applied simultaneously in a nonlinear elastic dynamic time history analysis on the model of the vessel, supports, and internals using the WECAN computer code. The results of this analysis are the dynamic loads on the reactor vessel supports and vessel time history displacements. The support loads and vessel displacements are then used in the evaluation of the reactor pressure vessel supports and reactor coolant loops.

3.9N.1.4.4 Stress Criteria for ASME Class 1 Components

All ASME Class 1 components are designed and analyzed for the design, normal, upset, and emergency conditions to the rules and requirements of the ASME Code Section III (Table 3.9N-3). The design analysis, or test methods and associated stress or load allowable limits that will be used in evaluation of faulted conditions, are those defined in Appendix F of the ASME Code. The supplementary options that follow (in elastic components/elastic system analysis) are methods used in addition to the guidelines provided in Appendix F of the code.

Elastic system analysis and component inelastic analysis is an acceptable method of evaluation for faulted conditions if primary stress limits for components are taken as greater of 0.70 S_u or S_y + 1/3 (S_u - S_y) for membrane stress, and greater of 0.70 S_{ut} or S_y + 1/3 (S_{ut} - S_y) for membrane-plus-bending stress, where material properties are taken at appropriate temperature.

Inelastic component analysis was used for the RCP support feet. The pump casing with the pump support feet is shown on Figure 3.9N-1.

The pump foot was analyzed for a set of umbrella loads which are greater than the loads expected in any plant. The umbrella loads are calculated for the faulted condition and each of the maximas of the six load components, F_x, F_y,...,M_z, are assumed to occur simultaneously. For example, the maximum F_x is chosen by surveying many past plants. This is applied simultaneously with the maximum F_y, F_z,...,M_z, all determined similarly. The actual plant loads are calculated and compared to the umbrella loads. Conformance indicates adequacy of the

component for the specific plant application. The actual loads are, in themselves, also conservative since the maximum for each of the six load components is determined and assumed to act concurrently with the others. For the LOCA condition, the dynamic time history analyses show that the maximum values of the six load components do not act concurrently. The seismic event, although evaluated by response spectra analysis, is also dynamic and it is unlikely that the load component maximums at the foot will coincide.

The casing foot was analyzed by means of a three-dimensional stress analysis using the ANSYS computer code, (DeSalvo and Swanson 1972). Since the foot is similar to a beam type structure, the average stress across the section is very low. The primary bending stresses are, therefore, the controlling features. The resulting faulted condition stress intensity was compared with the membrane-plus-bending criteria of 0.7 S_{ut}. The true ultimate stress, S_{ut}, is determined from the engineering ultimate stress (the engineering stress at the point of maximum load) by assuming constancy of volume. Using this assumption, the true ultimate stress is given by:

$$S_{ut} = S_u (1 + \varepsilon)$$

where ε is the engineering strain corresponding to the point of maximum load.

The actual plant loads calculated in the dynamic RCS analysis are less than the umbrella loads used in the analysis of the pump foot. Therefore, the analysis demonstrates that the pump foot is adequately designed to withstand the most severe faulted condition loads.

3.9N.2 Dynamic Testing and Analysis

3.9N.2.1 Preoperational Vibration and Dynamic Effects Testing on Piping

Evaluation of preoperational piping vibrational and dynamics effects will be conducted for the reactor coolant loop/supports system during start up functional testing of BVPS-2. The purpose of this evaluation will be to demonstrate that the primary coolant system is within acceptable limits. Particular attention will be provided at these locations where the vibration is expected to be the most severe for the particular transient being studied.

It should be noted, the layout, size, etc, of the reactor coolant loop and surge line piping used in BVPS 2 is very similar to that employed in Westinghouse plants now in operation. The operating experience that has been obtained from these plants indicates that the reactor coolant loop and surge line piping are adequately designed and supported to minimize vibration. In addition, vibration levels of the RCP, which is the only mechanical component that could cause vibration of the reactor coolant loop and surge line piping, are measured and held to the limits given in Section 5.4.1. Thus, excessive vibration of the reactor coolant loop and surge line piping will not be present.

3.9N.2.2 Seismic Qualification Testing of Safety-Related Mechanical Equipment

Westinghouse utilizes analysis, testing, or a combination of test and analysis to seismically qualify equipment. Testing is the preferred method; however, analysis is utilized when one of the following conditions is satisfied:

1. The equipment is too large or the external loads, connecting elements, or appurtenances cannot be simulated with a shaker table test.

2. The only requirement that must be satisfied relative to the safety of the plant is the maintenance of structural integrity (mechanical equipment only).

3. The component represents a simple linear system or nonlinearities can be conservatively accounted for in the analysis.

The operability of Seismic Category I mechanical equipment must be demonstrated if the equipment is determined to be active; that is, mechanical operation is relied on to perform a safety function. The operability of active Class 2 and Class 3 pumps, active Class 1, Class 2, or Class 3 valves, and their respective drives, operators, and vital auxiliary equipment is shown by satisfying the criteria given in Section 3.9N.3.2. There is no other active mechanical equipment.

Other Seismic Category I equipment within the Nuclear Steam Supply System (NSSS) scope is shown to have structural integrity during all plant conditions in one of the following manners: 1) by analysis satisfying the stress criteria applicable to the particular piece of equipment, or 2) by test showing that the equipment retains its structural integrity under the simulated test environment.

A list of Seismic Category I equipment is provided in Table 3.2-1.

3.9N.2.3 Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

The vibration characteristics and behavior due to flow induced excitation are very complex and not readily ascertained by analytical means alone. Reactor components are excited by the flowing coolant which causes oscillatory pressures on the surfaces. The integration of these pressures over the applied area should provide the forcing functions to be used in the dynamic analysis of the structures. In view of the complexity of the geometries and the random character of the pressure oscillations, a closed form solution of the vibratory problem by integration of the differential equation of motion is not always practical and realistic. The determination of the forcing functions as a direct correlation of pressure oscillations cannot be

practically performed independently of the dynamic characteristics of the structure. The main objective is to establish the characteristics of the forcing functions that essentially determine the response of the structures. By studying the dynamic properties of the structure from previous analytical and experimental work, the characteristics of the forcing function can be deduced. These studies indicate that the most important forcing functions are flow turbulence and pump-related excitation. The relevance of such excitations depends on many factors such as type and location of component and flow conditions. The effects of these forcing functions have been studied from test runs on models, prototype plants, and in component tests (Westinghouse 1973, 1975; Bloyd and Singleton 1975; Bloyd et al 1976).

The H.B. Robinson No. 2 plant has been established as the prototype design for three-loop plant internals. H.B. Robinson was instrumented and tested during hot functional testing. Beaver Valley Power Station - Unit 2 is similar to H.B. Robinson, and the only significant differences are the modifications resulting from the use of 17 by 17 fuel and the replacement of the annular thermal shield with neutron shielding panels.

The original test and analysis of the three-loop configuration are presented in Bloyd and Singleton (1975). These results are augmented by Westinghouse (1973; 1975) and Bloyd (et al 1976) to address the effects of successive hardware modifications, which are discussed in the following paragraphs.

The only structural changes in the internals resulting from the design change from the 15 by 15 to the 17 by 17 fuel assembly are the guide tube and control rod drive line. The new 17 by 17 guide tubes are stronger and more rigid; hence, they are less susceptible to flow-induced vibration. The fuel assembly itself is relatively unchanged in mass and spring rate, and thus no significant deviation is expected from the 15 by 15 fuel assembly vibration characteristics.

The vibration levels due to core barrel excitation for Trojan and BVPS-2, both having neutron shielding pads, are expected to be similar. The Trojan plant was instrumented and tested during hot functional testing. Results from Trojan (Bloyd 1976), as well as scale model test results (Westinghouse 1973), show that core barrel vibration of plants with neutron shielding pads is less than that of plants with thermal shields. The Trojan results verify the adequacy of the neutron pad core barrel and 17 by 17 guide tube designs and provide plant data applicable to BVPS-2.

3.9N.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Because the BVPS-2 reactor internals design configuration is well characterized, as was discussed in Section 3.9N.2.3, it is not

considered necessary to conduct instrumented tests on the BVPS-2 plant hardware. The requirements of Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program," will be met by conducting the confirmatory preoperational examination for integrity in accordance with paragraph C.2.3, “Inspection Program.” This examination will include the points on Figure 3.9N-2 summarized as follows:

1. All major load bearing elements of the reactor internals relied upon to retain the core structure in place.

2. The lateral, vertical, and torsional restraints provided within the vessel.

3. Those locking and bolting devices whose failures could adversely affect the structural integrity of the internals.

4. Those other locations on the reactor internals components that are similar to those which were examined on the prototype H.B. Robinson No. 2 design.

The inside of the vessel is inspected before and after the hot functional test, with all the internals removed, to verify that no loose parts or foreign materials are in evidence.

A particularly close inspection will be made on the following items or areas, using a 5x or 10x magnifying glass where applicable. The locations of these areas are shown on Figure 3.9N-2.

1. Lower Internals

    a. Upper barrel to flange girth weld.

    b. Upper barrel to lower barrel girth weld.

    c. Upper core plate aligning pin. Examine bearing surfaces for shadow marks, burnishing, buffing, or scoring. Inspect welds for integrity.

    d. Irradiation specimen guide screw locking devices and dowel pins. Check for lockweld integrity.

    e. Baffle assembly locking devices. Check for lockweld integrity.

    f. Lower barrel to core support girth weld.

    g. Neutron shield pad screw locking devices and dowel pin lockwelds. Examine the interface surfaces for evidence of tightness. Check for lockweld integrity.

    h. Radial support key welds.

i. Insert screw locking devices. Examine soundness of lockwelds.

j. Core support columns and instrumentation guides. Check all the joints for tightness and soundness of the locking devices.

k. Secondary core support assembling welds.

l. Lower radial support keys and inserts, examine bearing surfaces for shadow marks, burnishing, buffing, or scoring. Check the integrity of the lockwelds. These members supply the radial and torsional constraint of the internals at the bottom relative to the reactor vessel while permitting axial growth between the two. Subsequent to the hot functional testing, the bearing surfaces of the key and keyway will show burnishing, buffing, or shadowing marks that would indicate pressure loading and relative motion between these parts. Minor scoring of engaging surfaces is also possible and acceptable.

m. Gaps at baffle joints. Check the gaps at baffle to baffle joints.

2. Upper Internals

a. Thermocouple conduits, clamps, and couplings.

b. Guide tube, support column, and thermocouple column assembly locking devices.

c. Support column and thermocouple conduit assembly clamp welds.

d. Upper core plate alignment inserts. Examine bearing surfaces for shadow marks, burnishing, buffing, or scoring. Check the locking devices for integrity of lockwelds.

e. Thermocouple conduit gusset and clamp welds.

f. Thermocouple end plugs. Check for tightness.

g. Guide tube enclosure and card welds.

Acceptance standards are the same as required in the shop by the original design drawings and specifications.

During the hot functional test, the internals will be subjected to a total operating time at greater than normal full flow conditions (three pumps operating) of at least 240 hours. This provides a

cyclic loading of approximately 10^7 cycles on the main structural elements of the internals. In addition, there will be some operating time with only one or two pumps operating.

When there are no signs of abnormal wear, no harmful vibrations detected, or no apparent structural changes taking place, as determined by pre- and post-hot functional inspections, the three-loop core support structures are considered to be structurally sound and adequate for operations.

3.9N.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

The response of reactor internals components due to an excitation produced by complete severance of a branch line pipe is analyzed. Assuming a pipe break occurs in a very short period of time of 1 millisecond, the rapid drop of pressure at the break produces a disturbance which propagates along the primary loop and excites the internal structures.

The LOCA breaks that were considered consist of the 4-inch line break (4-inch schedule 160) in the cold leg and the 3-inch line break (3-inch schedule 160) in the hot leg. The LOCA hydraulic forcing functions (horizontal and vertical forces) that were used in the analyses were generated using MULTIFLEX 3.0 computer code described by Takeuchi (et. al 1982).

Mathematical Model of the Reactor Pressure Vessel (RPV) System

The mathematical model of the RPV system is a three-dimensional nonlinear finite element model which represents dynamic characteristics of the reactor vessel/internals/fuel in the six geometric degrees of freedom. The RPV system model was developed using the WECAN computer code (Westinghouse Electric Computer Analysis). The WECAN finite element model consists of three concentric structural sub-models connected by nonlinear impact elements and stiffness matrices. The first sub-model represents the reactor vessel shell and associated components. The reactor vessel is restrained by reactor vessel supports and by the attached primary coolant piping. The reactor vessel support system is represented by stiffness matrices.

The second sub-model represents the reactor core barrel assembly, lower support plate, tie plates, and secondary core support components. This sub-model is physically located inside the first, and is connected to it by a stiffness matrix at the internals support ledge. Core barrel to vessel shell impact is represented by nonlinear elements at the core barrel flange, core barrel nozzle, and lower radial support locations.

The third and innermost sub-model represents the upper support plate, guide tubes, support columns, upper and lower core plates, and the fuel. This sub-model includes the specific properties of the Westinghouse 17x17 Robust Fuel Assembly with Intermediate Flow Mixing devices (IFMs). The third sub-model is connected to the first and second by stiffness matrices and nonlinear elements.

The WECAN computer code, which is used to determine the response of the reactor vessel and its internals, is a general purpose finite element code. In the finite element approach, the structure is divided into a finite number of members or elements. The inertia and stiffness matrices, as well as the force array, are first calculated for each element in the local coordinates. Employing appropriate transformation, the element global matrices and arrays are then computed. Finally, the global element matrices and arrays are assembled into the global structural matrices and arrays, and used for dynamic solution of the differential equation of motion for the structure:

$$[M]\ \{\ddot{U}\} + [D]\ \{\dot{U}\} + [K]\ \{U\} = \{F\} \qquad \text{(Equation 1)}$$

where $[M]$ = Global inertia matrix

$[D]$ = Global damping matrix

$[K]$ = Global stiffness matrix

$\{\ddot{U}\}$ = Acceleration array

$\{\dot{U}\}$ = Velocity array

$\{U\}$ = Displacement array

$\{F\}$ = Force array, including impact, thrust forces, hydraulic forces, constraints and weight.

WECAN solves equation (1) using the nonlinear modal superposition theory. An initial computer run is made to calculate the eigenvalues (frequencies) and eigenvectors (mode shapes) for the mathematical model. This information is stored, and is used in a subsequent computer run which solves equation (1). The first time step performs a static solution of equation (1) to determine the initial displacements of the structure due to deadweight and normal operating hydraulic forces. After the initial time step, WECAN calculates the dynamic solution of equation (1). Time history nodal displacements and impact forces are stored for post-processing.

The following typical discrete elements from the WECAN finite element library are used to represent the reactor vessel and internals components:

- Three-dimensional elastic pipe
- Three-dimensional mass with rotary inertia
- Three-dimensional beam
- Three-dimensional linear spring
- Concentric impact element
- Linear impact element
- 6 x 6 stiffness matrix
- 18 Card stiffness matrix
- 18 Card mass matrix
- Three-dimensional friction element

Analytical Methods

The RPV system finite element model as described above was used to perform the LOCA analysis. Following a postulated LOCA pipe rupture, forces are imposed on the reactor vessel and its internals. These forces result from the release of the pressurized primary system coolant. The release of pressurized coolant results in traveling depressurization waves in the primary system. These depressurization waves are characterized by a wavefront with low pressure on one side and high pressure on the other. The wavefront translates and reflects throughout the primary system until the system is completely depressurized. The rapid depressurization results in transient hydraulic loads on the mechanical equipment of the system.

The LOCA loads applied to the reactor pressure vessel system consist of (a) reactor internal hydraulic loads (vertical and horizontal), and (b) reactor coolant loop mechanical loads. All the loads are calculated individually and combined in a time-history manner.

RPV Internal Hydraulic Loads

Depressurization waves propagate from the postulated break location into the reactor vessel through either a hot leg or a cold leg nozzle.

After a postulated break in the cold leg, the depressurization path for waves entering the reactor vessel is through the nozzle into the region between the core barrel and reactor vessel. This region is called the down-comer annulus. The initial waves propagate up, around, and down the down-comer annulus, then up through the region circumferentially enclosed by the core barrel; that is, the fuel region.

The region of the down-comer annulus close to the break depressurizes rapidly but, because of restricted flow areas and finite wave speed (approximately 3,000 feet per second), the opposite side of the core barrel remains at a high pressure. This results in a net horizontal force on the core barrel and reactor pressure vessel. As the depressurization wave propagates around the down-comer annulus and up through the core, the barrel differential pressure reduces, and similarly, the resulting hydraulic forces drop.

In the case of a postulated break in the hot leg, the waves follow a dissimilar depressurization path, passing through the outlet nozzle and directly into the upper internals region, depressurizing the core and entering the down-comer annulus from the bottom exit of the core barrel. Thus, after a break in the hot leg, the down-comer annulus would be depressurized with very little difference in pressure across the outside diameter of the core barrel.

A hot leg break produces less horizontal force because the depressurization wave travels directly to the inside of the core barrel (so that the down-comer annulus is not directly involved) and internal differential pressures are not as large as for a cold leg break. Since the differential pressure is less for a hot leg break, the horizontal force applied to the core barrel is less for a hot leg break than for a cold leg break. For breaks in both the hot leg and cold leg, the depressurization waves would continue to propagate by reflection and translation through the reactor vessel and loops.

The MULTIFLEX computer code described by Takeuchi (et al 1976 & 1982) calculates the hydraulic transients within the entire primary coolant system. It considers subcooled, transition, and two-phase (saturated) blowdown regimes. The MULTIFLEX program employs the method of characteristics to solve the conservation laws, and assumes one-dimensionality of flow and homogeneity of the liquid-vapor mixture.

The MULTIFLEX code considers a coupled fluid-structure interaction by accounting for the deflection of constraining boundaries, which are represented by separate spring-mass oscillator systems. A beam model of the core support barrel has been developed from the structural properties of the core barrel; in this model, the cylindrical barrel is vertically divided into various segments and the pressure as well as the wall motions are projected onto the plane parallel to the broken inlet nozzle. The spatial pressure variation at each time step is transformed into 10 horizontal forces, which act on the 10 mass points of the beam model. Each flexible wall is bounded on either side by a hydraulic flow path. The motion of the flexible walls is determined by solving the global equations of motion for the masses representing the forced vibration of an undamped beam.

Reactor Coolant Loop Mechanical Loads

The reactor coolant loop mechanical loads are applied to the RPV nozzles by the primary coolant loop piping. The loop mechanical loads result from the release of normal operating forces present in the pipe prior to the separation as well as transient hydraulic forces in the reactor coolant system. The magnitudes of the loop release forces are determined by performing a reactor coolant loop analysis for normal operating loads (pressure, thermal, and deadweight). The loads existing in the pipe at the postulated break location are calculated and are "released" at the initiation of the LOCA transient by application of the loads to the broken piping ends. These forces are applied with a ramp time of 1 millisecond because of the assumed instantaneous break opening time. For breaks in the branch lines the force applied at the reactor vessel would be insignificant. The restraints on the main coolant piping would eliminate any force to the reactor vessel caused by a break in the branch line.

Results of the Analysis

The severity of a postulated break in a reactor vessel is related to three factors: the distance from the reactor vessel to the break location, the break opening area, and the break opening time. The nature of the decompression following a LOCA, as controlled by the internals structural configuration previously discussed, results in larger reactor internal hydraulic forces for pipe breaks in the cold leg than in the hot leg (for breaks of similar area and distance from the RPV). Pipe breaks farther away from the reactor vessel are less severe because the pressure wave attenuates as it propagates toward the reactor vessel. The LOCA hydraulic and mechanical loads described in the previous sections were applied to the WECAN model of the reactor pressure vessel system.

The results of LOCA analysis include time history displacements and nonlinear impact forces for all major components. The time history displacements of upper core plate, lower core plate and core barrel at the upper core plate elevation are provided as input for the reactor core evaluations. The impact forces calculated at the vessel-internals interfaces are used to evaluate the structural integrity of the reactor vessel and its internals. Using appropriate post-processors, component linear forces are also calculated.

Seismic Evaluation

The non-linear dynamic seismic analysis of the reactor pressure vessel system uses the reactor pressure vessel system described above and the synthesized time history accelerations. The only difference between the seismic and LOCA model is that, in the seismic model, fluid-solid interactions are represented by hydrodynamic mass matrices in the downcomer region (between the core barrel and reactor vessel). On the other hand, in LOCA analysis, the fluid-solid interactions are accounted for through the hydraulic forcing functions generated by the MULTIFLEX Code.

Seismic Results

The results of system seismic analysis include time history displacements and impact forces for all major components. The time history displacements of upper core plate, lower core plate and core barrel at the upper core plate elevation are provided as input for the reactor core evaluations. The impact forces calculated at the vessel-internals interfaces are used to evaluate the structural integrity of the reactor vessel and its internals.

Components Subjected to Transverse Excitation

Various reactor internals components are subjected to transverse excitation during blowdown. Specifically, the barrel, guide tubes, and upper support columns are analyzed to determine their response to this excitation.

Core Barrel - For the hydraulic analysis of the pressure transients during hot leg blowdown, the maximum pressure drop across the barrel is a uniform radial compressive impulse.

The barrel is then analyzed for dynamic buckling using the following conservative assumptions:

1. The effect of the fluid environment is neglected.

2. The shell is treated as simply supported.

During cold leg blowdown, the upper barrel is subjected to a non-axisymmetric expansion radial impulse, which changes as the rare faction wave propagates both around the barrel and down the outer flow annulus between vessel and barrel.

Thus, the core barrel is subjected to non-axisymmetric pressure differentials across the barrel wall. These pressure differentials vary in circumferential as well as in axial direction of the core barrel.

The total response of the core barrel during cold leg break of blowdown consists of its response in shell modes of vibrations, as well as its response in the beam mode of vibration. The analysis of the core barrel response to cold leg blowdown is performed as follows:

1. The core barrel is analyzed as a shell with two variable sections to model the support flange and core barrel. The core barrel response due to shell modes such as m=1, n=0, 2, 3, 4, ... is determined.

2. The barrel with the core and thermal shielding pads is analyzed as a beam; that is, beam mode m=1 and n=1, fixed at the top and elastically supported at the lower radial support, and the dynamic response is obtained.

The total core barrel response is determined from the algebraic sum of the two responses described in steps 1 and 2.

Guide Tubes - The guide tubes in closest proximity to the outlet nozzle of the ruptured loop are the most severely loaded during a blowdown. The transverse guide tube forces decrease with increasing distance from the ruptured nozzle location.

All of the guide tubes are designed to maintain the function of the control rods for a break size of 144 in^2 and smaller. No credit for the function of the control rods is assumed for break size areas above 144 in^2. However, the design of the guide tube will permit control rod operation in all but four control rod positions, which is sufficient to maintain the core in a subcritical configuration, for break sizes up to a double-ended hot leg break. This double-ended hot leg break imposes the limiting lateral guide tube loading.

Upper Support Columns - Upper support columns located close to the broken nozzle during hot leg break will be subjected to transverse loads due to crossflow. The loads applied to the columns are computed with a method similar to the one used for the guide tubes; that is, by taking into consideration the increase in flow across the column during the accident. The columns are studied as beams with variable sections and the resulting stresses are obtained using the reduced section modulus and appropriate stress risers for the various sections.

The stresses due to the SSE (vertical and horizontal components) are combined with the blowdown stresses in order to obtain principal stresses and deflection.

All reactor internals components were found to be within acceptable stress and deflection limits for both hot leg and cold leg LOCAS, each occurring simultaneously with the SSE.

Both static and dynamic stress intensities are within acceptable limits. In addition, the cumulative fatigue usage factor is also within the allowable usage factor of unity.

The stresses due to the SSE (vertical and horizontal components) were combined in an unfavorable manner with the blowdown stresses in order to obtain the largest stress and deflection.

These results indicate that the maximum deflections and stress in the critical structures are below the established allowable limits. For the transverse excitation, it is shown that the upper barrel does not buckle during a hot leg break and that it has an allowable stress distribution during a cold leg break.

Even though control rod insertion is not required for BVPS-2 shutdown, this analysis shows that the deflection of most of the guide tubes is within the limits established experimentally to assure control rod insertion. For the guide tubes deflected above, the no loss of function limit, it must be assumed that the rods will not drop. However, the core will still shut down due to the negative reactivity insertion in the form of core voiding. Shutdown will be aided by the great majority of rods that do drop. Seismic deflections of the guide tubes are generally negligible by comparison with the no loss of function limit.

3.9N.2.6 Correlation of Reactor Internals Vibration Tests with the Analytical Results

As stated in Section 3.9N.2.3, it is not considered necessary to conduct instrumented tests of the BVPS-2 reactor vessel internals. Adequacy of these internals will be verified by use of the Sequoyah and Trojan results (Bloyd et al 1976). Westinghouse (1973) and Bloyd and Singleton (1975) describe predicted vibration behavior based on studies performed prior to BVPS-2 tests. These studies, which utilize analytical models, scale model tests results, component tests, and results of previous BVPS-2 tests, are used to characterize the forcing functions and establish component structural characteristics so that the flow-induced vibratory behavior and response levels for BVPS-2 are estimated. These estimates are then compared to values deduced from BVPS-2 test data obtained from the Sequoyah and Trojan (Bloyd et al 1976) internals vibration measurement programs.

3.9N.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

The ASME Code Class components are constructed in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section III.

A detailed discussion of ASME Code Class 1 components is provided in Section 3.9N.1.

Design loading conditions for core support structures are given in Table 3.9N-13 and Table 3.9N-14. Loading conditions are discussed in Section 3.9.5.2.

In general for reactor internals components and for core support structures, the criteria for acceptability in regard to mechanical integrity analyses are that adequate core cooling and core shutdown must be assured. This implies that the deformation of the reactor internals must be sufficiently small so that the geometry remains substantially intact. Consequently, the limitations established on the internals are concerned principally with the maximum allowable deflections and stability of the parts, in addition to a stress criterion to assure integrity of the components.

For the LOCA plus the SSE condition, deflections of critical internal structures are limited. In a hypothesized downward vertical displacement of the internals, energy absorbing devices limit the displacement after contacting the vessel bottom head, ensuring that the geometry of the core remains intact.

The following mechanical functional performance criteria apply:

1. Following the design basis accident (DBA), the functional criterion to be met for the reactor internals is that BVPS-2 shall be shut down end cooled in an orderly fashion so that fuel cladding temperature is kept within specified limits. This criterion implies that the deformation of critical components must be kept sufficiently small to allow core cooling.

2. For large breaks, the reduction in water density greatly reduces the reactivity of the core, thereby shutting down the core whether the rods are tripped or not. The subsequent refilling of the core by the ECCS uses borated water to maintain the core in a subcritical state. Therefore, the main requirement is to assure effectiveness of the ECCS. Insertion of the control rods, although not needed, gives further assurance of ability to shut BVPS-2 down and keep it in a safe shutdown condition.

3. The inward upper barrel deflections are controlled to ensure that there is no contacting of the nearest rod cluster control guide tube. The outward upper barrel deflections are controlled in order to maintain an adequate annulus for the coolant between the vessel inner diameter and core barrel outer diameter.

4. The rod cluster control guide tube deflections are limited to ensure operability of the control rods.

5. To ensure column loading of rod cluster control guide tubes, the upper core plate deflection is limited.

Methods of analysis and testing for core support structures are discussed in Sections 3.9N.1.3, 3.9N.1.4.1, 3.9N.2.3, 3.9N.2.5, and 3.9N.2.6. Stress limits and deformation criteria are given in Sections 4.2.2.4 and 4.2.2.5.

3.9N.3.1 Loading Combinations, Design Transients, and Stress Limits

Design pressure, temperature, and other loading conditions that provide the bases for design of fluid systems Code Class 2 and 3 components are presented in the sections which describe the systems.

3.9N.3.1.1 Design Loading Combinations

The design loading combinations for ASME Code Class 2 and 3 components and supports are given in Table 3.9N-4. The design loading combinations are categorized with respect to normal, upset, emergency, and faulted conditions. Design of component supports is discussed in Section 3.9N.3.4.

3.9N.3.1.2 Design Stress Limits

The design stress limits established for the components are sufficiently low to assure that violation of the pressure retaining boundary will not occur. These limits, for each of the loading conditions, are component oriented and are presented in Tables 3.9N-5, 3.9N-6, 3.9N-7 and 3.9N-8.

3.9N.3.2 Pump and Valve Operability Assurance

3.9N.3.2.1 Pump and Valve Qualification for Operability Program

Mechanical equipment classified as safety-related must be capable of performing its function under postulated plant conditions. Equipment with faulted condition function requirements include active pumps and valves in fluid systems important to safety. (Active components are those whose operability is relied upon to perform a safety function (as well as reactor shutdown function) during the transients or events considered in the respective operating condition categories. Seismic analysis is presented in Section 3.7N and discusses all safety-related mechanical equipment. A list of all NSSS active pumps and valves is presented in Tables 3.9N-9 and 3.9N-10.

All active pumps are qualified for operability by being subjected to rigid tests both prior to and after installation in the plant. The in-shop tests include: 1) hydrostatic tests of pressure-retaining parts to 150 percent of the design pressure; 2) seal leakage tests at the same pressure used in the hydrostatic tests; and 3) performance tests, while the pump is operated with flow, to determine total developed head, minimum and maximum head, net positive suction head requirements, and other pump/motor parameters. Also monitored during these operating tests are bearing temperatures and vibration levels. Bearing temperature limits are determined, by the manufacturer, based

on the bearing material, clearances, oil type, and rotational speed. After the pump is installed in the plant, it undergoes performance testing and the required periodic in-service inspection and operation. "These tests will demonstrate the continuing operability of the pump compared to the initial in-plant operating capability, as required to conform with ASME OM Code requirements."

In addition to these tests, the safety-related active pumps are qualified for operability during SSE condition by assuring that: 1) the pump will not be damaged during the seismic event; and 2) the pump will continue operating despite the SSE loads.

The pump manufacturer is required to show that the pump will operate normally when subjected to the maximum seismic accelerations and maximum faulted plant condition nozzle loads. It is required that test or analysis be used to show that the lowest natural frequency of the pump is greater than 33 Hz. The pump, when having a natural frequency above 33 Hz, is considered essentially rigid. This frequency is considered sufficiently high to avoid problems with amplification between the component and structure for all seismic areas. A static shaft deflection analysis of the rotor is performed with the conservative SSE accelerations of 3 g in the horizontal directions and 2 g vertical acting simultaneously. The deflections determined from the static shaft analysis are compared to the allowable rotor clearances. The nature of seismic disturbances dictates that the maximum contact (if it occurs) will be of short duration. In order to avoid damage during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE, and dynamic system loads are limited to the material elastic limit, as indicated in Table 3.9N-7. In addition, the pump casing stresses caused by the maximum faulted nozzle loads will be limited to the stresses outlined in Table 3.9N-7. The changes in operating rotor clearances caused by casing distortions due to these nozzle loads will be considered in an analysis of the pump supports. Furthermore, the calculated misalignment will be shown to be less than that misalignment which could cause pump misoperation. The stresses in the supports will be below those in Table 3.9N-7; therefore, the support distortion will be elastic and of short duration (equal to the duration of the seismic event).

Performing these analyses with the conservative loads stated and with the restrictive stress limits of Table 3.9N-7 as allowable, assures that critical parts of the pump will not be damaged during the faulted condition, and the reliability of the pump for post-faulted condition operation will not be impaired by the seismic event.

To complete the seismic qualification procedures, the pump motor is independently qualified for operation during the maximum seismic event. Any auxiliary equipment which is identified to be vital to the operation of the pump or pump motor, and which is not qualified for operation during the pump analysis or motor qualifications, is also separately qualified for operation at the accelerations it would

see at its mounting. The pump motor is qualified by meeting the requirements of IEEE Standard 344-1971, as described in Section 3.10N.

This program gives the required assurance that the safety-related pump/motor assemblies will not be damaged, will continue operating under SSE loadings, and will perform their intended functions. These requirements take into account the complex characteristics of the pump and are sufficient to demonstrate and assure the seismic operability of the active pumps.

Since the pump is not damaged during the faulted condition, the functional ability of active pumps after the faulted condition is assured, since only normal operating loads and steady-state nozzle loads exist. The post-faulted condition operating loads will be identical to the normal plant operating loads. This is assured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions and post-faulted conditions are limited by the magnitudes of the normal condition nozzle loads. The post-faulted condition ability of the pumps to function under these applied loads is proven during the normal operating plant conditions for active pumps.

Safety-related valves are subjected to a series of stringent tests prior to service and during the plant life. Prior to installation, the following tests are performed: shell hydrostatic test to ASME Section III requirements, backseat and main seat leakage tests, disc hydrostatic test, and operational tests to verify that the valve will open and close. Qualification of motor operators for environmental conditions is discussed in Sections 3.11 and Table 1.8-1 and Regulatory Guide 1.73. Cold hydro tests, hot functional qualification test, periodic in-service inspections, and periodic in-service operation are performed in situ to verify and assure the functional ability of the valve. These tests guarantee reliability of the valve for the design life of the plant. On active valves, an analysis of the extended structure is performed for static equivalent seismic SSE loads applied at the center of gravity of the extended structure. The maximum stress limits for active Class 2 and 3 valves are shown in Table 3.9N-3.

In addition to these tests and analyses, representative valves of each design type are tested for verification of operability during a simulated seismic event by demonstrating operational capabilities within the specified limits. The testing procedures are described as follows:

The valve is mounted in a manner which conservatively represents typical valve installations. The valve includes the operator pilot solenoid valves and limit switches normally attached to the valve in-service.

The faulted condition nozzle loads are considered in the test in either of two ways: 1) loads equivalent to the faulted condition nozzle loads are simultaneously applied to the valve through its mounting during the test described subsequently or 2) by analysis, the nozzle loads are shown not to affect the operability of the valve.

The operability of the valve during SSE is demonstrated by satisfying the following criteria:

1. All the active valves are designed to have a first natural frequency which is greater than 33 Hz.

2. The actuator and yoke of the valve system are statically deflected an amount equal to the deflection caused by the faulted condition accelerations applied at the center of gravity of the operator alone in the direction of the weakest axis of the yoke. The design pressure of the valve will be simultaneously applied to the valve during the static deflection tests.

3. The valve is cycled while in the deflected position. The time required to open or close the valve in the deflected position will be compared to similar data taken in the undeflected condition to evaluate the significance of any change.

4. Motor operators, external limit switches, and pilot solenoid valves necessary for operation are qualified by IEEE Standard 344-1971, as described in Section 3.10N.

The accelerations which are used for the static valve qualification shall be equivalent, as justified by analysis, to 3 g in the horizontal direction, and 2 g in the vertical. The piping designer must maintain the accelerations to these levels.

This testing program applies to valves with extended structures. The testing is conducted on a representative number of valves. Valves from each of the primary safety-related design types are tested. Valve sizes which cover the range of sizes in-service are qualified by the tests and the results are used to qualify all valves within the intermediate range of sizes.

Valves which are safety-related but can be classified as not having an extended structure, such as check valves and safety valves, are considered separately.

Check valves are characteristically simple in design and their operation will not be affected by seismic accelerations, or the maximum applied nozzle loads. The check valve design is compact and there are no extended structures of masses whose motion could cause distortions which could restrict operation of the valve. The nozzle

loads due to maximum seismic excitation will not affect the functional ability of the valve, since the valve disc is designed to be isolated from the body wall. The clearance supplied by the design around the disc will prevent the disc from becoming bound or restricted due to any body distortions caused by nozzle loads. Therefore, the design of these valves is such that once the structural integrity of the valve is assured using standard methods, the ability of the valve to operate is assured by the design features. The valve will also undergo: 1) in-shop hydrostatic tests; 2) in-shop seat leakage test; and 3) periodic in situ valve exercising and inspection to assure the functional ability of the valve.

The pressurizer safety valves are qualified by the following procedures (these valves are also subjected to tests and analysis similar to check valves): stress and deformation analyses for SSE loads, in-shop hydrostatic and seat leakage tests, and periodic in situ valve inspection. In addition to these tests, a static load equivalent to the SSE is applied at the top of the bonnet, and the pressure is increased until the valve mechanism actuates. Successful actuation within the design requirements of the valve assures its overpressurization safety capabilities during a seismic event.

Using the methods described, all the safety-related valves in the systems are qualified for operability during a seismic event. These methods conservatively simulate the seismic event and ensure that the active valves will perform their safety-related function when necessary.

This testing program for valves is conservative. Alternate valve operability testing, such as dynamic vibration testing, will be allowed if it is shown to adequately assure the faulted condition functional ability of the valve system.

3.9N.3.2.2 Pump Motor and Valve Electric Motor Operator Qualification

Active pump motors (and vital pump appurtenances) and active valve electric motor operators (and limit switches and pilot solenoid valves), are seismically qualified by meeting the requirements of IEEE Standard 344-1971, as described in Section 3.10N.

3.9N.3.3 Design and Installation Details in Mounting of Pressure Relief Devices

This is discussed in 3.9B.3.3.

3.9N.3.4 Component Supports

Westinghouse has supplied supports only for those Class 2 and 3 components also supplied by Westinghouse to which the supports are attached. The loads and loading combinations of the supports are the same as those of the supported component. These loads and combinations are given in Table 3.9N-4.

The Class 2 and 3 auxiliary equipment supplied by Westinghouse are grouped into two general categories. One group consists of tanks and heat exchangers. The other group is auxiliary pumps. Design criteria for the supports for these components are discussed below.

3.9N.3.4.1 Tanks and Heat Exchangers

The supports for auxiliary tanks and heat exchangers are of two types: linear and, for the most part, plate and shell type supports. The supports meet either the requirements of Subsection NF of the ASME Code or the requirements of the AISC Code, depending on the procurement date of the component. Components procured prior to the inclusion of Subsection NF into the ASME Code were designed to the AISC Code requirements. A listing of the tanks and heat exchangers and the codes to which the respective supports were designed is identified in the ASME Code Baseline Document.

3.9N.3.4.2 Auxiliary Pumps

The supports for Class 2 and 3 auxiliary pumps are plate and shell and, for the most part, linear-type supports. The supports for all Class 2 and 3 pumps supplied by Westinghouse are designed by the pump manufacturer to pressure boundary stress limits, (ASME III, Subsection NC/ND, as applicable), with the exception of the boric acid transfer pumps, the supports for which are designed to the limits of the AISC Code. A listing of the Class 2 and 3 auxiliary pumps and the ASME III Code edition/addenda to which they were purchased is identified in the ASME Code Baseline Document.

3.9N.4 Control Rod Drive Systems

3.9N.4.1 Descriptive Information of Control Rod Drive System

Control Rod Drive Mechanism

Control rod drive mechanisms (CRDMS) are located on the dome of the reactor vessel. They are coupled to rod control clusters which have absorber material over the entire length of the control rods. The CRDM is shown on Figure 3.9N-3 and schematically on Figure 3.9N-4.

The primary function of the CRDM is to insert or withdraw rod cluster control assemblies (RCCAs) within the core to control average core temperature and to shut down the reactor.

The CRDM is a magnetically-operated jack, which is an arrangement of three electromagnets energized in a controlled sequence to insert or withdraw RCCAs in the reactor core in discrete steps. Rapid insertion of the RCCAs occurs when electrical power is interrupted.

The CRDM consists of four separate subassemblies. They are the pressure vessel, coil stack assembly, latch assembly, and the drive rod assembly.

1. The pressure vessel includes a latch housing and rod travel housing which are connected by a threaded, seal welded, maintenance joint which facilitates access to the latch assembly. The threaded closure at the top of the rod travel housing contains a vent plug and utilizes a canopy seal weld for pressure integrity.

   The latch housing is the lower portion of the vessel and contains the latch assembly. The rod travel housing is the upper portion of the vessel and provides space for the drive rod during its upward movement as the control rods are withdrawn from the core.

2. The coil stack assembly includes the coil housings, an electrical conduit and connector, and three operating coils: 1) the stationary gripper coil, 2) the moveable gripper coil, and 3) the lift coil.

   The coil stack assembly is a separate unit which is installed on the drive mechanism by sliding it over the outside of the latch housing. It rests on the base of the latch housing without mechanical attachment.

   Energizing the operating coils causes movement of the pole pieces and latches in the latch assembly.

3. The latch assembly includes the guide tube, stationary pole pieces, moveable pole pieces, and two sets of latches: 1) the moveable gripper latches, and 2) the stationary gripper latches.

   The latches engage grooves in the drive rod assembly. The moveable gripper latches are moved up or down in 5/8-inch steps by the lift pole to raise or lower the drive rod. The stationary gripper latches hold the drive rod assembly while the moveable gripper latches are repositioned for the next 5/8-inch step.

4. The drive rod assembly includes a flexible coupling, a drive rod, a disconnect button, a disconnect rod, and a locking button.

   The drive rod has a 5/8-inch groove pitch which receives the latches during holding or moving of the drive rod. The flexible coupling is attached to the drive rod and provides the means for coupling to the RCCA.

   The disconnect button, disconnect rod, and locking button provide positive locking of the coupling to the RCCA and permit remote uncoupling of the drive rod from the rod cluster control.

The CRDM is a trip design. Tripping can occur during any part of the power cycle sequencing if electrical power to the coils is interrupted.

The CRDM is threaded and seal welded on a housing on top of the reactor vessel. The drive rod which is positioned in the mechanism is coupled to the RCCA directly below.

The mechanism is capable of raising or lowering a 360-pound load, (which includes the drive rod weight) at a rate of 45 in/min. Withdrawal of the RCCA is accomplished by magnetic forces while insertion is by gravity.

The mechanism internals are designed to operate in 650°F reactor coolant. The pressure vessel is designed to contain reactor coolant at 650°F and 2,500 psia. The three operating coils are designed to operate with a 40-year life at 392°F with forced air cooling required to maintain a temperature at or below 392°F.

The CRDM shown schematically on Figure 3.9N-4 withdraws and inserts an RCCA as shaped electrical pulses are received by the operating coils. An ON or OFF sequence, repeated by silicon controlled rectifiers in the power programmer, causes either withdrawal or insertion of the control rod. Position of the control rod is measured by 42 discrete coils mounted on the position indicator assembly surrounding the rod travel housing. Each coil magnetically senses the entry and presence of the top of the ferromagnetic drive rod assembly as it moves through the coil center line.

During plant operation the stationary gripper coil of the drive mechanism holds the RCCA in a static position until a stepping sequence is initiated at which time the moveable gripper coil and lift coil is energized sequentially.

<u>Rod Cluster Control Assembly Withdrawal</u>

The RCCA is withdrawn by repetition of the following sequence of events (Figure 3.9N-4):

1. Moveable gripper coil (B) - ON

   The latch locking plunger raises and swings the moveable gripper latches into the drive rod assembly groove. A 1/16-inch axial clearance exists between the latch teeth and the drive rod.

2. Stationary gripper coil (A) - OFF

   The force of gravity, acting upon the drive rod assembly and attached control rod, causes the stationary gripper latches and plunger to move downward 1/16 inch until the load of the drive rod assembly and attached control rod is transferred to the moveable gripper latches. The plunger continues to move downward and swings the stationary gripper latches out of the drive rod assembly groove.

3. Lift coil (C) - ON

   The 5/8-inch gap between the moveable gripper pole and the lift pole closes and the drive rod assembly raises one step length (5/8 inch).

4. Stationary gripper coil (A) - ON

   The plunger raises and closes the gap below the stationary gripper pole. The three links, pinned to the plunger, swing, and the stationary gripper, latches into a drive rod assembly groove. The latches contact the drive rod assembly and lift it (and the attached control rod) 1/16 inch. The 1/16-inch vertical drive rod assembly movement transfers the drive rod assembly load from the moveable gripper latches to the stationary gripper latches.

5. Moveable gripper coil (B) - OFF

   The latch locking plunger separates from the moveable gripper pole under the force of a spring and gravity. Three links, pinned to the plunger, swing the three moveable gripper latches out of the drive rod assembly groove.

6. Lift coil (C) - OFF

   The gap between the moveable gripper pole and lift pole opens. The moveable gripper latches drop 5/8 inch to a position adjacent to a drive rod assembly groove.

7. Repeat step 1

   The sequence described in Items 1 through 6 is termed as one step or one cycle. The RCCA moves 5/8 inch for each step or cycle. The sequence is repeated at a rate of up to

72 steps/min and the drive rod assembly (which has a 5/8-inch groove pitch) is raised 72 grooves/min. The RCCA is thus withdrawn at a rate up to 45 in/min.

Rod Cluster Control Assembly Insertion

The sequence for RCCA insertion is similar to that for control rod withdrawal, except the timing of lift coil (C) ON and OFF is changed to permit lowering the control assembly.

1. Lift coil (C) - ON

   The 5/8-inch gap between the moveable gripper and lift pole closes. The moveable gripper latches are raised to a position adjacent to a drive rod assembly groove.

2. Moveable gripper coil (B) - ON

   The latch locking plunger raises and swings the moveable gripper latches into a drive rod assembly groove. A l/16-inch axial clearance exists between the latch teeth and the drive rod assembly.

3. Stationary gripper coil (A) - OFF

   The force gravity, acting upon the drive rod assembly and attached RCCA, causes the stationary gripper latches and plunger to move downward 1/16 inch until the load of the drive rod assembly and attached RCCA is transferred to the moveable gripper latches. The plunger continues to move downward and swings the stationary gripper latches out of the drive rod assembly groove.

4. Lift coil (C) - OFF

   The force of gravity and spring force separates the moveable gripper pole from the lift pole and the drive rod assembly and attached rod cluster control drop down 5/8 inch.

5. Stationary gripper (A) - ON

   The plunger raises and closes the gap below the stationary gripper pole. The three links, pinned to the plunger, swing the three stationary gripper latches into a drive rod assembly groove. The latches contact the drive rod assembly and lift it (and the attached control rod) 1/16 inch. The l/16-inch vertical drive rod assembly movement transfers the drive rod assembly load from the moveable gripper latches to the stationary gripper latches.

6. Moveable gripper coil (B) - OFF

The latch locking plunger separates from the moveable gripper pole under the force of a spring and gravity. Three links, pinned to the plunger, swing the three moveable gripper latches out of the drive rod assembly groove.

7. Repeat step 1

   The sequence is repeated, as for RCCA withdrawal, up to 72 times/min which gives an insertion rate of 45 in/min.

Holding and Tripping of the Control Rods

During most of the plant operating time, the CRDMs hold the RCCAs withdrawn from the core in a static position. In the holding mode, only one coil, the stationary gripper coil (A), is energized on each mechanism. The drive rod assembly and attached RCCAs hang suspended from the three latches.

If power to the stationary gripper coil is cut off, the combined weight of the drive rod assembly and the RCCA and the stationary gripper return spring is sufficient to move latches out of the drive rod assembly groove. The control rod falls by gravity into the core. The trip occurs as the magnetic field, holding the stationary gripper plunger half against the stationary gripper pole, collapses and the stationary gripper plunger half is forced down by the stationary gripper return spring and weight acting upon the latches. After the RCCA is released by the mechanism, it falls freely until the control rods enter the dashpot section of the thimble tubes in the fuel assembly.

3.9N.4.2 Applicable Control Rod Drive System Design Specifications

For those components in the control rod drive system (CRDS), comprising portions of the reactor coolant pressure boundary, conformance with the General Design Criteria 15, 30, 31, and 32, and Section 50.55a of 10 CFR 50 is discussed in Sections 3.1 and 5.2. Conformance with the regulatory guides pertaining to materials suitability is described in Sections 4.5 and 5.2.3.

Design Bases

The design of the reactor control components takes into consideration temperature effects, thermal clearances, and stresses on structural members resulting from normal and accident conditions.

Design Stresses

The CRDS is designed to withstand stresses originating from various operating conditions as summarized in Table 5.2-1.

Allowable Stresses

For normal operating conditions, Section III of the ASME Boiler and Pressure Code is used. All pressure boundary components are analyzed as Class I components under Article NB-3000.

Dynamic Analysis

The cyclic stresses due to dynamic loads and deflections are combined with the stresses imposed by loads from component weights, hydraulic forces, and thermal gradients for the determination of the total stresses of the CRDS.

Control Rod Drive Mechanisms

The CRDMs pressure housings are Class 1 components designed to meet the stress requirements for normal operating conditions of Section III of the ASME Boiler and Pressure Vessel Code. Both static and alternating stress intensities are considered. The stresses originating from the required design transients are included in the analysis.

A dynamic seismic analysis is required on the CRDMs when a seismic disturbance has been postulated to confirm the ability of the pressure housing to meet ASME Code, Section III, allowable stresses, and to confirm its ability to trip when subjected to the seismic disturbance.

Control Rod Drive Mechanism Operational Requirements

The basic operational requirements for the CRDMs are:

1. 5/8-inch step,

2. 144-inch travel (nominal value),

3. 360-lb maximum dynamic load,

4. Step in or out at 45 in/min (72 steps/min),

5. Electrical power interruption shall initiate release of drive rod assembly,

6. Trip delay time of less than 150 milliseconds - Free fall of drive rod assembly shall begin less than 150 milliseconds after power interruption no matter what holding or stepping action is being executed with any load and coolant temperature of 100°F to 550°F, and

7. Forty-year design life with normal refurbishment,

3.9N.4.3 Design Loads, Stress Limits, and Allowable Deformations

3.9N.4.3.1 Pressure Vessel

The pressure retaining components are analyzed for loads corresponding to normal, upset, and faulted conditions. The analysis performed depends on the mode of operation under consideration.

The scope of the analysis requires many different techniques and methods, both static and dynamic.

Some of the loads that are considered on each component where applicable are as follows:

1. Control rod trip (equivalent static load),
2. Differential pressure,
3. Spring preloads,
4. Coolant flow forces (static),
5. Temperature gradients,
6. Differences in thermal expansion,
    a. Due to temperature differences
    b. Due to expansion of different materials,
7. Interference between components,
8. Vibration (mechanically or hydraulically induced),
9. All operational transients listed in Table 3.9N-1,
10. Pump overspeed,
11. Seismic loads (operation basis earthquake and design basis earthquake), and
12. Blowdown forces (due to cold and hot leg break).

The main objective of the analysis is to satisfy allowable stress limits, given in NB-3200 and NA Appendix F, to assure an adequate design margin, and to establish deformation limits which are concerned primarily with the functioning of the components. The stress limits are established not only to assure that peak stresses will not reach unacceptable values, but also limit the amplitude of the oscillatory stress component in consideration of fatigue characteristics of the materials. Standard methods of strength of materials are used to establish the stresses and deflections of these

components. The dynamic behavior of the reactivity control components has been studied using experimental test data and experience from operating reactors.

3.9N.4.3.2 Drive Rod Assembly

All postulated failures of the drive rod assemblies either by fracture or uncoupling lead to a reduction in reactivity. If the drive rod assembly fractures at any elevation below the gripper latches, that portion remaining coupled falls with, and is guided by the RCCA. This always results in reactivity decrease for control rods. Such an occurrence on a part length rod could result in reactivity increase but is not considered an excessive reactivity increase.

3.9N.4.3.3 Latch Assembly and Coil Stack Assembly

With respect to the CRDM as a whole, critical clearances are present in the following areas:

1. Latch assembly (diametral clearances),

2. Latch arm-drive rod clearances,

3. Coil stack assembly-thermal clearances, and

4. Coil fit in coil housing.

The following description defines clearances that are designed to provide reliable operation in the CRDM in these four critical areas. These clearances have been proven by life tests and actual field performance at operating plants.

<u>Latch Assembly - Thermal Clearances</u>

The magnetic jack has several locations where parts made of Type 410 stainless steel fit over parts made from Type 304 stainless steel. Differential thermal expansion is therefore important. Minimum clearance of these parts at 68°F is 0.011 inch. At the maximum design temperature of 650°F, minimum clearance is 0.0045 inch and at the maximum expected operating temperature of 550°F is 0.0057 inch.

<u>Latch Arm - Drive Rod Clearances</u>

The CRDM incorporates a load transfer action. The moveable or stationary gripper latch is not under load during engagement due to load transfer action.

Figure 3.9N-5 shows latch clearance variation with the drive rod as a result of minimum and maximum temperatures. Figure 3.9N-6 shows clearance variations over the design temperature range.

Coil Stack Assembly - Thermal Clearances

The assembly clearance of the coil stack assembly over the latch housing was selected so that the assembly could be removed under all anticipated conditions of thermal expansion.

At 70°F the inside diameter of the coil stack is 7.308/7.298 inches. The outside diameter of the latch housing is 7.260/7.270 inches.

Thermal expansion of the mechanism due to operating temperature of the CRDM results in the minimum inside diameter of the coil stack being 7.310 inches at 222°F and the maximum latch housing diameter being 7.302 inches at 532°F.

Under the extreme tolerance conditions listed previously, it is necessary to allow time for a 70°F coil housing to heat during a replacement operation.

Four coil stack assemblies were removed from four hot CRDMs mounted on 11.035-inch centers on a 550°F test loop, allowed to cool, and then placed without incident as a test to prove the preceding.

Coil Fit in Coil Housing

Control rod drive mechanism and coil housing clearances are selected so that coil heat up results in a close to tight fit. This is done to facilitate thermal transfer and coil cooling in a hot CRDM.

3.9N.4.4 Control Rod Drive System Performance Assurance Program

Evaluation of Material's Adequacy

The ability of the pressure housing components to perform throughout the design lifetime as defined in the equipment specification is confirmed by the stress analysis report required by the ASME Boiler and Pressure Vessel Code, Section III.

Internal components subjected to wear will withstand a minimum of 3,000,000 steps without refurbishment, as confirmed by life tests (Westinghouse 1977). Latch assembly inspection is recommended after approximately 2.5 by 10^6 steps have been accumulated on a CRDM.

To confirm the mechanical adequacy of the fuel assembly, the CRDM and the RCCA, functional test programs have been conducted on a full scale 12-foot control rod. The 12-foot prototype assembly was tested under simulated conditions of reactor temperature, pressure, and flow for approximately 1,000 hours. The prototype mechanism accumulated about 3,000,000 steps and 600 trips. At the end of the test the CRDM was still operating satisfactorily. A correlation was developed to predict the amplitude of the flow-excited vibration of individual fuel rods and fuel assemblies. Inspection of the drive line components did not reveal significant fretting.

These tests include verification that the trip time of the CRDM meets the design requirement of 2.2 seconds from start of RCCA motion to dashpot entry. Trip times are to be confirmed for each CRDM prior to initial reactor operation, and at periodic intervals after initial reactor operation, as required by the proposed Technical Specification.

There are no significant differences between the prototype CRDMs and the production units. Design materials, tolerances, and fabrication techniques are the same.

These tests have been reported by Bohm and LaFaille (1971).

If an RCCA cannot be moved by its mechanism, adjustments in the boron concentration ensure that adequate shutdown margin would be achieved following a trip. Thus, inability to move one RCCA can be tolerated. More than one inoperable RCCA could be tolerated but would impose additional demands on the plant operator.

In order to demonstrate proper operation of the CRDM and to ensure acceptable core power distributions during operation, RCCA partial-movement checks are performed on the RCCAs (Technical Specifications, Chapter 16). In addition, periodic drop tests of the full length RCCAs are performed at each refueling shutdown, to demonstrate continued ability to meet trip time requirements and to ensure core subcriticality after reactor trip. During these tests the acceptable drop time of each assembly is not greater than 2.2 seconds at full flow and operating temperature, from the beginning of motion to dashpot entry.

Actual operating experience indicates excellent performance of the CRDM both with respect to tripping action and stepping.

All units are production tested prior to shipment to confirm ability of the mechanism to meet design specification-operational requirements.

Each CRDM undergoes a production test according to the following summarization:

Test	Acceptance Criteria
Cold (ambient) hydrostatic	ASME Section III
Confirm step length and load transfer (stationary gripper to moveable gripper or moveable gripper to stationary gripper)	Step Length 5/8 $\pm$ 0.015 inch axial movement Load Transfer 0.047 inch nominal axial

	movement
Cold (ambient) performance test at design load - 5 full travel excursions	Operating Speed 45 cm/min Trip Delay Free fall of drive rod to begin within 150 milliseconds

3.9N.5 Reactor Vessel Internals

3.9N.5.1 Design Arrangements

The reactor vessel internals are described as follows:

The components of the reactor internals are divided into three parts: 1) the lower core support structure (including the entire core barrel and neutron shield pad assembly); 2) the upper support structure; and 3) the incore instrumentation support structure. The reactor internals support the core, maintain fuel alignment, limit fuel assembly movement, maintain alignment between fuel assemblies and CRDMs, direct coolant flow past the fuel elements, direct coolant flow to the pressure vessel head, provide gamma and neutron shielding, and guides for the incore instrumentation. The coolant flows from the vessel inlet nozzles down the annulus between the core barrel and the vessel wall and then into a plenum at the bottom of the vessel. It then reverses and flows up through the core support and through the lower core plate. The lower core plate is sized to provide the desired inlet flow distribution to the core. After passing through the core, the coolant enters the region of the upper support structure and then flows radially to the core barrel outlet nozzles and directly through the vessel outlet nozzles. A small portion of the coolant flows between the baffle plates and the core barrel to provide additional cooling of the barrel. Similarly, a small amount of the entering flow is directed into the vessel head plenum and exits through the vessel outlet nozzles.

Lower Core Support Structure

The major component and support member of the reactor internals is the lower core support structure, shown on Figure 3.9N-7. This support structure assembly consists of the core barrel, the core baffle, the lower core plate and support columns, the neutron shield pads, and the core support which is welded to the core barrel. All the major material for this structure is Type 304 stainless steel or equivalent. The lower core support structure is supported at its upper flange from the ledge in the reactor vessel head flange, and its lower end is restrained in its transverse movement by a radial support system attached to the vessel wall. Within the core barrel are an axial baffle and a lower core plate, both of which are attached to the core

barrel wall and form the enclosure periphery of the assembled core. The lower core support structure and principally the core barrel serve to provide passageways and control for the coolant flow. The lower core plate is positioned at the bottom level of the core below the baffle plates and provides support and orientation for the fuel assemblies.

The lower core plate is a member through which the necessary flow distribution holes for each fuel assembly are machined. Fuel assembly locating pins (two for each assembly) are also inserted into this plate. Columns are placed between this plate and the core support of the core barrel in order to provide stiffness and to transmit the core load to the core support. Adequate coolant distribution is obtained through the use of the lower core plate and core support.

The neutron shield pad assembly consists of four pads that are bolted and pinned to the outside of the core barrel. These pads are constructed of Type 304 stainless steel or equivalent and are approximately 48 inches wide by 148 inches long by 2.8 inches thick. The pads are located azimuthally to provide the required degree of vessel protection. Specimen guides, in which material surveillance samples can be inserted and irradiated during reactor operation, are attached to the pads. The samples are held in the guide by a preloaded spring device at the top and bottom to prevent sample movement. Additional details of the neutron shield pads and irradiation specimen holders are given by DeSalvo and Swanson (1972).

Vertical downward loads from weight, fuel assembly preload, control rod dynamic loading, hydraulic loads, and earthquake acceleration are carried by the lower core plate partially into the lower core plate support flange on the core barrel shell and partially through the lower support columns to the core support, and thence through the core barrel shell to the core barrel flange supported by the vessel head flange. Transverse loads from earthquake acceleration, coolant cross flow, and vibration are carried by the core barrel shell and distributed between the lower radial support to the vessel wall, and to the vessel flange. Transverse loads of the fuel assemblies are transmitted to the core barrel shell by direct connection of the lower core plate to the barrel wall and by upper core plate alignment pins which are welded into the core barrel.

The main radial support system of the lower end of the core barrel is accomplished by "key" and “keyway” joints to the reactor vessel wall. At equally spaced points around the circumference, an Inconel clevis block is welded to the vessel inner diameter. Inconel inserts are bolted to these clevis blocks to provide “keyway” geometry. Opposite each of these is a "key" which is attached to the internals. At assembly, as the internals are lowered into the vessel, the keys engage the keyways in the axial direction. With this design, the internals are provided with a support at the furthest extremity, and may be viewed as a beam supported at the top and bottom.

Radial and axial expansions of the core barrel are accommodated, but transverse movement of the core barrel is restricted by this design. With this system, cyclic stresses in the core support structures are within the ASME Section III limits. In the event of an abnormal downward vertical displacement of the internals following a hypothetical failure, energy absorbing devices limit the displacement after contacting the vessel bottom head. The load is then transferred through the energy absorbing devices of the internals to the vessel.

The bottom of the energy absorbers is contoured similar to the reactor vessel bottom head inside geometry. Assuming a downward vertical displacement, the potential energy of the system is absorbed mostly by the strain energy of the energy absorbing devices.

Upper Core Support Assembly

The upper core support assembly, shown on Figures 3.9N-8 and 3.9N-9, consists of the top support plate assembly and the upper core plate between which are contained support columns and guide tube assemblies. The support columns establish the spacing between the top support plate assembly and the upper core plate and are fastened at top and bottom to these plates.

The support columns transmit the mechanical function of providing a passageway for the thermocouple. The guide tube assemblies sheath and guide the control rod drive shafts and control rods. They are fastened to the top support plate and are restrained by pins in the upper core plate for proper orientation and support. Additional guidance for the control rod drive shafts is provided by the upper guide tube which is attached to the upper support plate and guide tube.

The upper core support assembly is positioned in its proper orientation with respect to the lower support structure by flat-sided pins pressed into the core barrel, which in turn engage in slots in the upper core plate. At an elevation in the core barrel where the upper core plate is positioned, the flat-sided pins are located at angular positions of 90° from each other. Four slots are milled into the core plate at the same positions. As the upper support structure is lowered into the main internals, the slots in the plate engage the flat-sided pins in the axial direction. Lateral displacement of the plate and of the upper support assembly is restricted by this design. Fuel assembly locating pins protrude from the bottom of the upper core plate and engage the fuel assemblies as the upper assembly is lowered into place. Proper alignment of the lower core support structure, the upper core support assembly, the fuel assemblies, and control rods are thereby assured by this system of locating pins and guidance arrangement. The upper core support assembly is restrained from any axial movement by a large circumferential spring which rests between the upper barrel flange and the upper core support assembly, and is compressed by the reactor vessel head flange.

Vertical loads from weight, earthquake acceleration, hydraulic loads, and fuel assembly preload are transmitted through the upper core plate via the support columns to the top support plate assembly and then the reactor vessel head. Transverse loads from coolant cross flow, earthquake acceleration, and possible vibrations are distributed by the support columns to the top support plate and upper core plate. The top support plate is particularly stiff to minimize deflection.

Incore Instrumentation Support Structures

The incore instrumentation support structures consist of an upper system to convey and support thermocouples penetrating the vessel through the head, and a lower system to convey and support flux thimbles penetrating the vessel through the bottom. (Figure 7.7-8 shows the basic flux-mapping system.)

The upper system utilizes the reactor vessel head penetrations. Instrumentation port columns are slip-connected to inline columns that are in turn fastened to the upper support plate. These port columns protrude through the head penetrations. The thermocouples are carried through these port columns and the upper support plate at positions above their readout locations. The thermocouple conduits are supported from the columns of the upper core support system. The thermocouple conduits are sealed stainless steel tubes.

In addition to the upper incore instrumentation, there are reactor vessel bottom port columns which carry the retractable, cold worked stainless steel flux thimbles that are pushed upward into the reactor core. Conduits extend from the bottom of the reactor vessel down through the concrete shield area and up to a thimble seal line. The minimum bend radii are about 144 inches and the trailing ends of the thimbles (at seal line) are extracted approximately 15 feet during refueling of the reactor in order to avoid interference within the core. The thimbles are closed at the leading ends and serve as the pressure barrier between the reactor pressurized water and the containment atmosphere.

Mechanical seals between the retractable thimbles and conduits are provided at the seal line. During normal operation, the retractable thimbles are stationary and move only during refueling or for maintenance, at which time a space of approximately 15 feet above the seal line is cleared for the retraction operation.

The incore instrumentation support structure is designed for adequate support of instrumentation during reactor operation, and is rugged enough to resist damage or distortion under the conditions imposed by handling during the refueling sequence.

These are the only conditions which affect the incore instrumentation support structure. Reactor vessel surveillance specimen capsules are covered in Section 5.3.1.6. All the necessary details with regard

to irradiation surveillance, including a cross section of the reactor showing the capsule identity and location, are included in the FSAR.

3.9N.5.2 Design Loading Conditions

Normal and Upset

The normal and upset loading conditions that provide the basis for the design of the reactor internals are:

1. Fuel and reactor internals weight,
2. Fuel and core component spring forces including spring preloading forces,
3. Differential pressure and coolant flow forces,
4. Temperature gradients,
5. Vibratory loads including OBE seismic,
6. The normal and upset operational thermal transients listed in Table 3.9N-1,
7. Control rod trip (equivalent static load),
8. Loads due to loop(s) out-of-service, and
9. Loss of load pump overspeed.

Emergency

The emergency loading conditions that provide the basis for the design of the reactor internals are:

1. Small LOCA,
2. Small steam break, and
3. Complete loss of flow.

Faulted

The faulted conditions that provide the basis for the design of the reactor internals are:

1. Large LOCA, and
2. Safe shutdown earthquake.

The main objective of the design analysis is to satisfy allowable stress limits, to assure an adequate margin, and to establish

deformation limits which are concerned primarily with the functioning of the components. The stress limits are established not only to assure that peak stresses will not reach unacceptable values, but also limit the amplitude of the oscillatory stress component in consideration of fatigue characteristics of the materials. Both low and high cycle fatigue stresses are considered when the allowable amplitude of oscillation is established. Dynamic analysis on the reactor internals is provided in Section 3.9N.2.

As part of the evaluation of design loading conditions, extensive testing and inspection are performed from the initial selection of raw materials up to and including component installation and plant operation. Among these tests and inspections are those performed during component fabrication, plant construction, start-up and checkout, and during plant operation.

3.9N.5.3 Design Loading Categories

The combination of design loadings fit into the normal, upset, or faulted conditions as defined in the ASME Code, Section III.

Loads and deflections imposed on components due to shock and vibration are determined analytically and experimentally in both scaled models and operating reactors. The cyclic stresses due to these dynamic loads and deflections are combined with the stresses imposed by loads from component weight, hydraulic forces, and thermal gradients for the determination of the total stresses of the internals.

The reactor internals are designed to withstand stresses originating from various operating conditions as summarized in Table 3.9N-1.

The scope of the stress analysis problem is very large, requiring many different techniques and methods, both static and dynamic. The analysis performed depends on the mode of operation under consideration.

Allowable Deflections

For normal operating conditions, downward vertical deflection of the lower core support plate is negligible. For the LOCA plus the SSE conditions, the deflection criteria of critical internal structures are the limiting values given in Table 3.9N-11. The corresponding no loss of function limits are included in Table 3.9N-11 for comparison purposes with the allowed criteria.

The criteria for the core drop accident are based upon analyses which have to determine the total downward displacement of the internal structures following a hypothesized core drop resulting from the loss of the normal core barrel supports. The initial clearance between the secondary core support structures and the reactor vessel lower head in the hot condition is approximately

1/2 inch. An additional displacement of approximately 3/4 inch would occur due to strain of the energy absorbing devices of the secondary core support; thus, the total drop distance is about 1 1/4 inches which is insufficient to permit the tips of the RCCA to come out of the guide thimble in the fuel assemblies.

Specifically, the secondary core support is a device which will never be used except during a hypothetical accident of the core support (core barrel, barrel flange, etc). There are four supports in each reactor. This device limits the fall of the core and absorbs much of the energy of the fall which otherwise would be imparted to the vessel. The energy of the fall is calculated assuming a complete and instantaneous failure of the primary core support and is absorbed during the plastic deformation of the controlled volume of stainless steel, loaded in tension.

Design loading categories are further discussed in Section 3.9N.1.

3.9N.5.4 Design Bases

The design bases for the mechanical design of the reactor vessel internals components are as follows:

1. The reactor internals in conjunction with the fuel assemblies direct reactor coolant through the core to achieve acceptable flow distribution and to restrict bypass flow so that the heat transfer performance requirements are met for all modes of operation. In addition, required cooling for the pressure vessel head shall be provided so that the temperature differences between the vessel flange and head do not result in leakage from the flange during reactor operation.

2. In addition to neutron shielding provided by the reactor coolant, the reactor internals are designed to limit the exposure of the pressure vessel in order to maintain the required ductility of the material for all modes of operation.

3. Provisions shall be made for installing incore instrumentation useful for the plant operation, and providing the vessel material test specimens required for a pressure vessel irradiation surveillance program.

4. The core internals are designed to withstand mechanical loads arising from an OBE, SSE, end pipe ruptures and meet the requirements of Item 5.

5. The reactor shall have mechanical provisions which are sufficient to adequately support the core and internals and

to assure that the core is intact with acceptable heat transfer geometry following transients arising from abnormal operating conditions.

6. Following the DBA, BVPS-2 shall be capable of being shut down and cooled in an orderly fashion so that fuel cladding temperatures are kept within specified limits. This implies that the deformation of certain critical reactor internals must be kept sufficiently small to allow core cooling.

The functional limitations for the core structures during the design basis accident are shown in Table 3.9N-11. To ensure no column loading of rod cluster control guide tubes, the upper core plate deflection shall not exceed the value shown in Table 3.9N-11.

Details of the dynamic analyses, input forcing functions, and response loadings are presented in Section 3.9B.2.

The following identifies the basis for the design stress and deflection criteria:

Allowable Stresses

For normal operating conditions Section III of the ASME Boiler and Pressure Vessel Code is used as a basis for evaluating acceptability of calculating stresses. Both static and alternating stress intensities are considered. It should be noted that the allowable stresses in Section III of the ASME Code are based on unirradiated material properties. The strength of Type 304 stainless steel used for internals is not changed when exposed to an irradiation level of less than 1×10^{21} neutron per sq cm and increases when exposed to high levels; thus, it is considered that use of the allowable stresses in Section III is appropriate and conservative for irradiated internal structures.

The allowable stress limits during the DBA used for the core support structures are defined in the 1974 edition of the ASME Code for Core Support Structure, Subsection NG, and the Criteria for Faulted Conditions. Stress limits for reactor vessel internal structures are presented in Table 3.9N-12. The design and construction of the BVPS-2 core support structures conforms to the requirements of Subsection NG, except that the core support structures are not Code stamped and a plant specific stress report has not been written. This is because procurement of the BVPS-2 core support structures predated the inclusion of Subsection NG into the ASME Code.

3.9N.6 References for Section 3.9

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Tables for Section 3.9

TABLE 3.9B-1

SYSTEMS AND TYPES OF TEST CONDUCTED

System Designation	System Name	Preoperational Tests* Vibration	Thermal Expansion	Dynamic Effects
ASS	Auxiliary Steam	NR	V	NR
BDG	Steam Generator Blowdown	NR	V&I	V
BRS	Boron Recovery	NR	V	NR
CCP	Component Cooling	V	NR	NR
CHS	Chemical and Volume Control	V	V&I	V
CDS	Chilled Water	V	NR	NR
CND	Condensate Demineralizer	V	NR	NR
CNM	Condensate	V	V	NR
EDG EDA EDS EDF	Diesel Generator Auxiliary Systems	V	V	NR
ESS	Extraction Steam	V	V	V
FNC	Fuel Pool Cooling	V	NR	NR
FWE	Auxiliary Feedwater	V	NR	NR
FWS	Feedwater	V	V&I	V&I
GSS	Turbine Gland Steam Seal	NR	V	NR
HDH	Feedwater Heater High Pressure Drains	V	V	NR
HDL	Feedwater Heater Low Pressure Drains	V	V	NR
MSS	Main Steam	V	V&I	V&I
QSS	Quench Spray	V	NR	V
RCS	Reactor Coolant	V	V&I	V

TABLE 3.9B-1 (Cont)

System Designation	System Name	Preoperational Tests* Vibration	Thermal Expansion	Dynamic Effects
RCS(PSRV)	Pressurizer Safety and Relief Valve Piping	NR	V&I	V&I
RHS	Residual Heat Removal	V	V&I	NR
RSS	Recirculation Spray	V	NR	V
SDS	Steam Drains	NR	V	NR
SIS	Safety Injection	V	NR	NR
SWS	Service Water	V	NR	V

*Type of test is as follows:
V = Visual
I = Instrumented
NR = Not Required

TABLE 3.9B-3

PIPING SNUBBERS MONITORED DURING PREOPERATIONAL TESTING

SNUBBER MARK NUMBER	**PIPE LINE NUMBER**	**SNUBBER MARK NUMBER**	**PIPE LINE NUMBER**	**SNUBBER MARK NUMBER**	**PIPE LINE NUMBER**
2BDG-PSSP007	2BDG-025-011-2	2CHS-PSSP-011	2CHS-002-098-1	2DGS-PSSP044B	2DGS-004-239-4
2BDG-PSSP055X	2BDG-025-010-2	2CHS-PSSP012	2CHS-002-098-1	2DGS-PSSP045A	2DGS-004-239-4
2BDG-PSSP325Y	2BDG-003-222-4	2CHS-PSSP014X	2CHS-002-098-1	2DGS-PSSP045B	2DGS-004-239-4
2BDG-PSSP326Y	2BDG-003-223-4	2CHS-PSSP015A	2CHS-002-098-1	2DGS-PSSP879	2DGS-002-001-1
2BDG-PSSP852	2BDG-002-070-2	2CHS-PSSP015X	2CHS-003-126-1	2EDG-PSSP027A	2EDG-022-001-3
2BDG-PSSP866	2BDG-002-066-2	2CHS-PSSP016	2CHS-002-096-1	2EDG-PSSP027B	2EDG-022-001-3
2BDG-PSSP867	2BDG-022-066-2	2CHS-PSSP016X	2CHS-002-142-1	2EDG-PSSP029A	2EDG-038-014-3
2BDG-PSSP868	2BDG-002-066-2	2CHS-PSSP017	2CHS-002-096-1	2EDG-PSSP029B	2EDG-038-014-3
2BDG-PSSP874	2BDG-002-066-2	2CHS-PSSP017X	2CHS-002-140-1	2EDG-PSSP030Y	2EDG-038-014-3
2BDG-PSSP876	2BDG-002-066-2	2CHS-PSSP024	2CHS-002-096-1	2EDG-PSSP031A	2EDG-022-002-3
2BDG-PSSP878	2BDG-002-066-2	2CHS-PSSP024X	2CHS-002-097-1	2EDG-PSSP031B	2EDG-022-002-3
2BDG-PSSP884	2BDG-002-066-2	2CHS-PSSP025	2CHS-002-096-1	2EDG-PSSP032A	2EDG-022-012-3
2BDG-PSSP887	2BDG-002-020-2	2CHS-PSSP025X	2CHS-002-097-1	2EDG-PSSP032B	2EDG-O22-012-3
2BDG-PSSP895	2BDG-002-068-2	2CHS-PSSP660X	2CHS-002-141-1	2EDG-PSSP033A	2EDG-022-013-3
2BDG-PSSP907	2BDG-002-068-2	2CHS-PSSP661X	2CHS-002-097-1	2EDG-PSSP033B	2EDG-022-013-3
2BDG-PSSP927	2BDG-003-261-2	2CHS-PSSP662X	2CHS-002-097-1	2EDG-PSSP033Y	2EDG-038-003-3
2BDG-PSSP945	2BDG-002-070-2	2CHS-PSSP663X	2CHS-002-097-1	2EDG-PSSP035A	2EDG-038-003-3
2BDG-PSSP947	2BDG-002-066-2	2CHS-PSSP664X	2CHS-003-120-2	2EDG-PSSP035B	2EDG-038-003-3
2BDG-PSSP951	2BDG-002-070-2	2CHS-PSSP667X	2CHS-002-098-1	2EDG-PSSP037A	2EDG-038-014-3
2BDG-PSSP953	2BDG-002-070-2	2CHS-PSSP668X	2CHS-002-098-1	2EDG-PSSP037B	2EDG-038-014-3
2BRS-PSSP091Y	2BRS-002-005-3	2CHS-PSSP669X	2CHS-002-096-1	2EDG-PSSP042B	2EDG-038-014-3
2CHS-PSSP001	2CHS-002-096-1	2CHS-PSSP673X	2CHS-002-001-1	2EDG-PSSP042B	2EDG-038-014-3
2CHS-PSSP002	2CHS-002-096-1	2CHS-PSSP684	2CHS-002-094-2	2FWS-PSSP001	2FWS-016-012-2
2CHS-PSSP003	2CHS-002-096-1	2CHS-PSSP685C	2CHS-002-002-2	2FWS-PSSP002A	2FWS-016-012-2
2CHS-PSSP005	2CHS-002-097-1	2CHS-PSSP783	2CHS-750-129-1	2FWS-PSSP002B	2FWS-016-012-2
2CHS-PSSP006	2CHS-003-126-1	2DGS-PSSP023	2DGS-002-001-1	2FWS-PSSP003A	2FWS-016-012-2
2CHS-PSSP007	2CHS-003-345-2	2DGS-PSSP037	2DGS-002-001-1	2FWS-PSSP003B	2FWS-016-012-2
2CHS-PSSP008	2CHS-003-345-2	2DGS-PSSP043A	2DGS-004-239-4	2FWS-PSSP005	2FWS-016-017-2
2CHS-PSSP009	2CHS-003-345-2	2DGS-PSSP043B	2DGS-004-239-4	2FWS-PSSP006	2FWS-016-022-2
2CHS-PSSP010	2CHS-002-098-1	2DGS-PSSP044A	2DGS-004-239-4	2FWS-PSSP007	2FWS-016-022-2

TABLE 3.9B-3 (CONTINUED)

SNUBBER MARK NUMBER	PIPE LINE NUMBER	SNUBBER MARK NUMBER	PIPE LINE NUMBER	SNUBBER MARK NUMBER	PIPE LINE NUMBER
2FWS-PSSP008	2FWS-016-022-2	2MSS-PSSP131B	2MSS-032-002-4	2RCS-PSSP009A	2RCS-006-156-4
2FWS-PSSP012	2FWS-016-022-2	2MSS-PSSP132A	2MSS-032-002-4	2RCS-PSSP009B	2RCS-006-156-4
2FWS-PSSP016	2FWS-016-022-2	2MSS-PSSP132B	2MSS-032-002-4	2RCS-PSSP010A	2RCS-006-106-4
2FWS-PSSP036	2FWS-016-016-4	2MSS-PSSP144	2MSS-032-003-4	2RCS-PSSP010B	2RCS-006-106-4
2FWS-PSSP039	2FWS-016-016-4	2MSS-PSSP149	2MSS-032-003-4	2RCS-PSSP011	2RCS-002-033-1
2FWS-PSSP060	2FWS-016-032-4	2MSS-PSSP151A	2MSS-032-003-4	2RCS-PSSP011X	2RCS-006-106-4
2MSS-PSSP001	2MSS-032-035-2	2MSS-PSSP151B	2MSS-032-003-4	2RCS-PSSP012	2RCS-002-053-1
2MSS-PSSP002A	2MSS-032-035-2	2MSS-PSSP164	2MSS-038-004-4	2RCS-PSSP012A	2RCS-006-106-4
2MSS-PSSP002B	2MSS-032-035-2	2MSS-PSSP165	2MSS-038-004-4	2RCS-PSSP012B	2RCS-006-106-4
2MSS-PSSP003A	2MSS-032-035-2	2MSS-PSSP168	2MSS-038-004-4	2RCS-PSSP013A	2RCS-002-056-1
2MSS-PSSP003B	2MSS-032-035-2	2MSS-PSSP363A	2MSS-028-067-4	2RCS-PSSP013B	2RCS-002-056-1
2MSS-PSSP005	2MSS-032-035-2	2MSS-PSSP363B	2MSS-028-007-4	2RCS-PSSP015X	2RCS-006-105-4
2MSS-PSSP006	2MSS-032-035-2	2MSS-PSSP364A	2MSS-028-007-4	2RCS-PSSP016	2RCS-002-013-1
2MSS-PSSP007	2MSS-032-039-2	2MSS-PSSP364B	2MSS-028-007-4	2RCS-PSSP016X	2RCS-006-105-4
2MSS-PSSP008A	2MSS-032-039-2	2MSS-PSSP456	2MSS-046-003-3	2RCS-PSSP017A	2RCS-002-016-1
2MSS-PSSP008B	2MSS-032-039-2	2MSS-PSSP476	2MSS-010-200-3	2RCS-PSSP017B	2RCS-002-016-1
2MSS-PSSP009	2MSS-032-043-2	2RCS-PSSP001A	2RCS-014-084-1	2RCS-PSSP017X	2RCS-006-105-4
2MSS-PSSP011A	2MSS-032-043-2	2RCS-PSSP001X	2RCS-004-081-1	2RCS-PSSP018X	2RCS-006-104-4
2MSS-PSSP011B	2MSS-032-043-2	2RCS-PSSP002	2RCS-014-084-1	2RCS-PSSP019	2RCS-003-015-1
2MSS-PSSP012	2MSS-032-043-2	2RCS-PSSP002A	2RCS-004-080-1	2RCS-PSSP-019X	2RCS-006-104-4
2MSS-PSSP103	2MSS-032-001-4	2RCS-PSSP003	2RCS-008-061-1	2RCS-PSSP020X	2RCS-006-104-4
2MSS-PSSP107	2MSS-032-001-4	2RCS-PSSP003X	2RCS-006-215-1	2RCS-PSSP021X	2RCS-006-104-4
2MSS-PSSP110	2MSS-032-001-4	2RCS-PSSP004	2RCS-008-021-1	2RCS-PSSP022	2RCS-002-025-1
2MSS-PSSP111A	2MSS-032-001-4	2RCS-PSSP004X	2RCS-006-215-1	2RCS-PSSP022X	2RCS-003-110-1
2MSS-PSSP111B	2MSS-032-001-4	2RCS-PSSP005	2RCS-008-021-1	2RCS-PSSP022Y	2RCS-003-110-1
2MSS-PSSP112A	2MSS-032-001-4	2RCS-PSSP006	2RCS-008-041-1	2RCS-PSSP023X	2RCS-003-110-1
2MSS-PSSP112B	2MSS-032-001-4	2RCS-PSSP006A	2RCS-006-107-1	2RCS-PSSP026	2RCS-002-045-1
2MSS-PSSP124	2MSS-032-002-4	2RCS-PSSP007	2RCS-008-041-1	2RCS-PSSP026A	2RCS-006-111-4
2MSS-PSSP130	2MSS-032-002-4	2RCS-PSSP007X	2RCS-006-107-1	2RCS-PSSP026B	2RCS-006-111-4
2MSS-PSSP131A	2MSS-032-002-4	2RCS-PSSP008X	2RCS-150-082-1	2RCS-PSSP027	2RCS-002-045-1

TABLE 3.9B-3 (CONTINUED)

SNUBBER MARK NUMBER	PIPE LINE NUMBER	SNUBBER MARK NUMBER	PIPE LINE NUMBER	SNUBBER MARK NUMBER	PIPE LINE NUMBER
2RCS-PSSP028	2RCS-002-025-1	2RCS-PSSP664X	2RCS-002-031-1	2RHS-PSSP008A	2RHS-010-016-2
2RCS-PSSP029	2RCS-002-025-1	2RCS-PSSP667X	2RCS-002-039-1	2RHS-PSSP008B	2RHS-010-016-2
2RCS-PSSP-029X	2RCS-006-111-4	2RCS-PSSP668X	2RCS-002-039-1	2RHS-PSSP009	2RHS-010-016-2
2RCS-PSSP030	2RCS-002-045-1	2RCS-PSSP669X	2RCS-002-051-1	2RHS-PSSP010A	2RHS-010-006-2
2RCS-PSSP030X	2RCS-003-108-1	2RCS-PSSP673X	2RCS-002-059-1	2RHS-PSSP010B	2RHS-012-006-2
2RCS-PSSP031	2RCS-002-045-1	2RCS-PSSP674X	2RCS-002-059-1	2RHS-PSSP011A	2RHS-012-018-2
2RCS-PSSP031X	2RCS-003-108-1	2RCS-PSSP882	2RCS-012-112-4	2RHS-PSSP011B	2RHS-012-018-2
2RCS-PSSP034	2RCS-002-065-1	2RCS-PSSP883A	2RCS-012-112-4	2RHS-PSSP012X	2RHS-006-015-4
2RCS-PSSP035	2RCS-002-065-1	2RCS-PSSP883B	2RCS-012-112-4	2RHS-PSSP013X	2RHS-006-027-4
2RCS-PSSP035X	2RCS-012-112-4	2RCS-PSSP884A	2RCS-012-112-4	2RHS-PSSP014A	2RHS-010-005-2
2RCS-PSSP036	2RCS-002-067-1	2RCS-PSSP884B	2RCS-012-112-4	2RHS-PSSP014B	2RHS-010-005-2
2RCS-PSSP037A	2RCS-150-082-1	2RCS-PSSP885	2RCS-012-112-4	2RHS-PSSP-015A	2RHS-010-004-2
2RCS-PSSP037B	2RCS-150-082-1	2RCS-PSSP887A	2RCS-006-106-4	2RHS-PSSP015B	2RHS-010-004-2
2RCS-PSSP038	2RCS-150-082-1	2RCS-PSSP887B	2RCS-006-106-4	2RHS-PSSP501X	2RHS-010-010-2
2RCS-PSSP038X	2RCS-002-065-1	2RCS-PSSP890	2RCS-006-111-4	2RHS-PSSP515X	2RHS-012-006-2
2RCS-PSSP039A	2RCS-004-173-1	2RCS-PSSP891A	2RCS-006-111-4	2RHS-PSSP518X	2RHS-012-056-1
2RCS-PSSP039B	2RCS-004-173-1	2RCS-PSSP891B	2RCS-006-111-4	2RHS-PSSP520X	2RHS-012-001-1
2RCS-PSSP118A	2RCS-150-082-1	2RCS-PSSP893	2RCS-006-011-4	2RHS-PSSP521X	2RHS-012-001-1
2RCS-PSSP118B	2RCS-150-082-1	2RCS-PSSP894	2RCS-006-242-4	2RHS-PSSP522X	2RHS-012-001-1
2RCS-PSSP653	2RCS-002-045-1	2RCS-PSSP896	2RCS-006-242-4	2RHS-PSSP524X	2RHS-012-018-2
2RCS-PSSP655X	2RCS-003-055-1	2RCS-PSSP897	2RCS-006-156-4	2RHS-PSSP525X	2RHS-012-018-2
2RCS-PSSP656X	2RCS-003-055-1	2RCS-PSSP898	2RCS-003-109-1	2RHS-PSSP526X	2RHS-012-018-2
2RCS-PSSP657X	2RCS-003-015-1	2RCS-PSSP906	2RCS-006-113-4	2RHS-PSSP527X	2RHS-012-018-2
2RCS-PSSP658A	2RCS-002-036-1	2RCS-PSSP910	2RCS-760-260-2	2RHS-PSSP531A	2RHS-010-010-2
2RCS-PSSP658B	2RCS-002-036-1	2RHS-PSSP001	2RHS-010-005-2	2RHS-PSSP531B	2RHS-010-010-2
2RCS-PSSP659	2RCS-003-035-1	2RHS-PSSP002	2RHS-010-005-2	2RHS-PSSP818	2RHS-006-015-4
2RCS-PSSP660X	2RCS-003-035-1	2RHS-PSSP003	2RHS-010-010-2	2RHS-PSSP821	2RHS-006-015-4
2RCS-PSSP661A	2RCS-002-019-1	2RHS-PSSP003X	2RHS-010-016-2	2SIS-PRR815	2SIS-012-288-1
2RCS-PSSP661B	2RCS-002-019-1	2RHS-PSSP005	2RHS-006-015-4	2SIS-PRR817	2SIS-012-071-1
2RCS-PSSP663X	2RCS-002-011-1	2RHS-PSSP007	2RHS-010-004-2	2SIS-PRR824	2SIS-012-069-1

TABLE 3.9B-3 (CONTINUED)

SNUBBER MARK NUMBER	PIPE LINE NUMBER	SNUBBER MARK NUMBER	PIPE LINE NUMBER
2SIS-PRR826	2SIS-012-069-1	2SVS-PSSP016B	2SVS-010-173-2
2SIS-PSSP002	2SIS-012-067-1	2SVS-PSSP020A	2SVS-004-040-2
2SIS-PSSP003	2SIS-012-067-1	2SVS-PSSP020B	2SVS-004-040-2
2SIS-PSSP004	2SIS-012-071-1	2SVS-PSSP024A	2SVS-010-175-2
2SIS-PSSP005	2SIS-012-071-1	2SVS-PSSP024B	2SVS-010-175-2
2SIS-PSSP006	2SIS-012-071-1	2SVS-PSSP028A	2SVS-004-044-2
2SIS-PSSP007	2SIS-012-069-1	2SVS-PSSP028B	2SVS-004-044-2
2SIS-PSSP018X	2SIS-012-069-1	2SVS-PSSP084Y	2SVS-004-042-2
2SIS-PSSP201A	2SIS-006-269-1	2SVS-PSSP652A	2SVS-010-178-1
2SIS-PSSP201B	2SIS-006-269-1	2SVS-PSSP652B	2SVS-010-178-2
2SIS-PSSP202X	2SIS-006-269-1	2SVS-PSSP656A	2SVS-008-012-2
2SIS-PSSP203X	2SIS-006-269-1	2SVS-PSSP656B	2SVS-008-012-2
2SIS-PSSP204	2SIS-006-268-1	2SVS-PSSP658A	2SVS-008-012-2
2SIS-PSSP205	2SIS-006-266-1	2SVS-PSSP658B	2SVS-008-012-2
2SIS-PSSP206	2SIS-006-266-1	2SVS-PSSP659	2SVS-008-012-2
2SIS-PSSP207	2SIS-006-268-1	2SVS-PSSP663A	2SVS-006-015-2
2SIS-PSSP208X	2SIS-006-270-1	2SVS-PSSP665	2SVS-008-012-2
2SIS-PSSP209A	2SIS-006-271-1		
2SIS-PSSP209B	2SIS-006-271-1		
2SIS-PSSP210	2SIS-006-271-1		
2SIS-PSSP211	2SIS-006-267-1		
2SIS-PSSP212	2SIS-006-267-1		
2SIS-PSSP213	2SIS-006-270-1		
2SIS-PSSP609	2SIS-012-289-1		
2SIS-PSSP609	2SIS-012-289-1		
2SIS-PSSP609	2SIS-012-289-1		
2SIS-PSSP610	2SIS-012-067-1		
2SVS-PSSP007A	2SVS-010-171-2		
2SVS-PSSP007B	2SVS-010-171-2		
2SVS-PSSP012A	2SVS-004-036-2		
2SVS-PSSP012B	2SVS-004-036-2		
2SVS-PSSP016A	2SVS-020-173-2		

TABLE 3.9B-5

LOAD COMBINATIONS FOR ASME III CLASS 1 PIPING [3,5]

Plant Design or Operating Condition	NB 3600 Equations	Load (Moment Combination) [1,4]	Design or Service Stress Limits [2]
Design	9	$P_d + D + E + H$	1.5 Sm [5]
Normal/Upset	10	$P_{max} + T + R + H + E + A + L + R''$	3.0 Sm
	11	$P_{max} + T + R + H + E + A + L + R''$	---
	12	$T + R + R''$	3.0 Sm
	13	$P_{max} + D + E + H + L$	3.0 Sm
	14	$P_{max} + T + R + H + E + A + L + R''$	$\Sigma\ \mu \leq 1.0$
Emergency	9	$P_E + D + H$	2.25 Sm
Faulted	9	$P_F + D + H + E' + Y'$	3.0 Sm
Test	--	P_h	0.9 S_y
	--	$P_h + D$	1.35 S_y

NOTES:

1. For definitions of each load, refer to Table 3.9B-11.
2. Sm is the design stress intensity, S_y is the tabulated yield strength, and μ is the usage factor as defined in the ASME III code or applicable code cases.
3. Code Class 1 piping 1 inch NPS and less is analyzed in accordance with Tables 3.9B-8 and 3.9B-9.
4. The methods used for various load combinations are described in Table 3.9B-17.
5. For piping analyzed in accordance with ASME Section III 1989 Edition, design or service stress limits is 1.8 Sm for Service Level "B" (upset condition)

TABLE 3.9B-6

COMPARISON OF ASME III (2) CLASS 1 PIPING REQUIREMENTS
REGULATORY GUIDE 1.48 VS TABLE 3.9B-5

Component(1)	Normal or Upset(3) + OBE	Emergency	Normal + Faulted + SSE	Regulatory Position	Comparison With Regulatory Position
Pipe	NB-3654	NB-3655	NB-3656	C.1	Agree

NOTES: (from Regulatory Guide 1.48, Pages 1.48-7 and -8)

1. Applies to all components that are relied upon to cope with the effects of specified BVPS-2 conditions.

2. Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code including 1972 Winter Addenda thereto.

3. Identification of the specific transients or events to be considered under each plant condition will be addressed in a future regulatory guide.

TABLE 3.9B-7

STRESS LIMITS FOR ASME SECTION III CLASS 2 AND 3 COMPONENTS (ELASTIC ANALYSIS)

Component and Design Condition	ASME III Code Class	Primary Stress Limits [1] Membrane (Pm)	Membrane Plus Bending (P + P)
Pressure Vessels			
I	2(NC3300)	or 1.1 S	1.65 S
II	3(ND3300)	2.0 S	2.40 S
I[2]	2(NC3200)	1.1 S	1.65 S
II[3]		2.0 S	2.40 S
Pumps[4,5], Inactive			
I	2(NC3400)	or 1.1 S	1.65 S
II	3(ND3400)	2.0 S	2.40 S
Pumps[4,5], Active			
I	2(NC3400)	or 1.0 S	1.50 S
II	3(ND3400)	1.2 S	1.80 S
Valves[5,6]			
I	2(NC3500)	or 1.1 S	1.65 S
II	3(ND3500)	2.0 S	2.40 S
Tanks[5], (Steel)			
I	2(NC38-3900)	or 1.1 S	1.65 S
II	3(ND38-3900)	2.0 S	2.40 S

NOTES:

1. S - Allowable stress values at design temperature from ASME Section III, Appendix I, as allowed by class.

   S - Design stress intensity values at design temperature from ASME Section III, Appendix I, as allowed by class.

2. Fatigue analysis may be required with operating conditions, reference paragraph NC-3219 and Appendix XIV of ASME Section III, Subsection NC.

TABLE 3.9B-7 (Cont)

NOTES: (Cont)

3. When a complete analysis is performed in accordance with NC 3211.1(c), the faulted stress limits of Appendix F shall apply.

4. In accordance with NC-3400 and ND-3400, any design method which has been demonstrated to be satisfactory for the specified design conditions may be used.

5. Stress limits of ASME Section III, Subsection NF, are used for the design of supports as applicable.

6. The standard or alternative design rules of NC-3500 and ND-3500 may be used in conjunction with the stress limits specified.

   Valve nozzle (piping load) stress analysis is not required when both the following conditions are satisfied by calculation:

   a. Section modulus and area at the plane normal to the flow passage through the region at the valve body crotch is at least 110 percent of that for the piping connected (or joined) to the valve body inlet and outlet nozzles; and,

   b. Code allowable stress, S, for valve body material, is equal to or greater than the code allowable stress, S, of connected piping material. If the valve body material allowable stress is less than that of the connected piping, the valve section modulus and area, as calculated in (a) previously, shall be multiplied by the ratio of the allowable stress for the pipe divided by the allowable stress of the valve.

   If unable to comply with these requirements, the design by analysis procedure of NB-3545.2 is an acceptable alternative method.

   Casting quality factor of 1.0 shall be used.

   Design requirements listed in this table are not applicable to valve discs, stems, cast rings, or other parts of valves which are contained within the confines of the body and bonnet.

TABLE 3.9B-8

LOAD COMBINATIONS FOR ASME III CLASS 2 AND 3 PIPING
EXCEPT QUENCH SPRAY, RECIRCULATION SPRAY, AND SAFETY INJECTION SYSTEMS(2,4,7)

Plant Design or Operating Condition	NC 3600 Equations	Load (Moment Combination)(1)	Design or Service Stress Limits(5)
Design	8	$P_d + D$	S_h
Normal/Upset	9	$P_{max} + D + E + H$	$1.2\ S_h$(7)
	9	$P_{max} + D + W$	$1.2\ S_h$
	10(3)	T + R + A + R″	S_A
	10	S	$3\ S_c$
	11(3)	P_d + D + T + R + A + R″	$S_h + S_A$
Emergency	9	$P_E + D + H$	$1.8\ S_h$(7)
Faulted	9	P_F + D + E′ + H + Y′	$2.4\ S_h$(7)
Test	8	$P_h + D$	$1.2\ S_h$(6)
	--	P_h	$0.9\ S_y$

NOTES:

1. For definitions of each load, refer to Table 3.9B-11.
2. Table 3.9B-9 illustrates loading combinations for QSS, RSS and SIS.
3. Loads due to OBE anchor movements (A) are shown in Equation 10 and 11 above, but may alternately be considered in Equation 9 only.
4. The methods used for various load combinations are described in Table 3.9B-17.
5. S_h, S_c, S_A, and S_y are allowable stress limits as defined in the ASME III Code or applicable code cases.
6. An allowable of 1.35 S_y may be used provided the appropriate ASME III Class 1 stress indices are used.
7. For piping analyzed in accordance with ASME Section III 1989 Edition, design or service stress limits for NC 3600 equations 9 normal/upset condition is 1.8 S_h, emergency condition is 2.25 S_h and faulted condition is 3.0 S_h.

TABLE 3.9B-9

LOAD COMBINATIONS FOR ASME III CLASS 2 AND 3 PIPING ON QUENCH SPRAY, RECIRCULATION SPRAY AND SAFETY INJECTION SYSTEMS [1,4]

Plant Design or Operating Condition	NC 3600 Equations	Load (Moment Combination) [2]	Design or Service Stress Limits [6]
Design	8	$P_d + D$	S_h
Normal/Upset	9 [5]	$P_{max} + D + E + H$	$1.2\ S_h$
	10 [3]	$T + R + A + R''$	S_A
	10	S	$3\ S_c$
	11 [3]	$P_d + D + T + R + A + R''$	$S_h + S_A$
Emergency	9	$P_e + D + H$	$1.8\ S_h$
Faulted	9	$P_f + D + E' + H + Y'$	$1.8\ S_h$
	10 [3]	$T + R' + A' + X$	S_A
	11 [3]	$P_f + D + T + R' + A' + X$	$S_h + S_A$
Test	8	$P_{h + D}$	$1.2\ S_h$ [7]
	--	P_h	$0.9\ S_y$

NOTES:

1. Class 1 portions of the safety injection system are analyzed in accordance with Table 3.9B-5.
2. For definitions of each load, refer to Table 3.9B-11.
3. Loads due to seismic anchor movements (A,A′) are shown in Equations 10 and 11, but may alternately be considered in Equation 9 only.
4. The methods used for various load combinations are described in Table 3.9B-17.
5. Earthquake loads need not be considered concurrently with flow transient loads resulting from RSS system testing during plant shutdown and refueling. Since there are only 5 OBES postulated over the life of the plant and the RSS startup flow transient loads last less than 10 minutes per year, the probability of these events occurring simultaneously is extremely low.
6. S_h, S_c, S_A, and S_y are allowable stress limits as defined in the ASME III code or applicable code cases.
7. An allowable of 1.35 S_y may be used provided the appropriate ASME III Class 1 stress indices are used.

TABLE 3.9B-10

COMPARISON OF CLASS 2 AND 3 REQUIREMENTS - REGULATORY GUIDE 1.48 VS. TABLES 3.9B-7, 3.9B-8, AND 3.9B-9

Components	Regulatory Guide 1.48 Loading Combinations	Regulatory Guide 1.48 Design Limits			Reg. Position	Tables 3.9B-7, 3.9B-8, and 3.9B-9 Loading Combinations	Tables 3.9B-7, 3.9B-8, and 3.9B-9 Design Limits			Comparison
Pressure Vessels [9]		P_m	P_m (or P_L) + P_b				P_m	P_m (or P_L) + P_b		Acceptable alternate,
Classes 2 and 3	Normal	1.1 S	1.65 S		C.6.a	Design + OBE	1.1 S	1.65S		Code Case 1607,
ASME Section	Upset + OBE	1.1 S	1.65 S		C.6.a					Regulatory Guide 1.84
III NC/ND-3300	Emergency	1.1 S	1.65 S		C.6.a	Design + SSE	2.0 S	2.40S		
	Faulted + SSE	1.5 S	2.25 S		C.6.b					
Pressure Vessels		P_m	P_m (or P_L) + P_b	Q			P_m	P_m (or P_L) + P_b	Q	Reflects proposed Code
Class 2	Normal			NB-3223	C.7.a	Normal)		As applicable		revision
ASME Section	Upset + OBE			NB-3223	C.7.a	Upset)		See Add-100 (b)		
III NC-3200	Emergency		NB 3224		C.7.b	Emergency)				
	Faulted + SSE		NB 3225		C.7.b	Faulted)				
Part AD						Design + OBE	1.1 Sm	1.65 Sm		
						Design + SSE	2.0 Sm	2.40 Sm		
Piping [10]	Normal)					Normal)				Acceptable alternate,
Classes 2 and 3	Upset + OBE)	NC-3611.1(b)(4)(b)(1)	$1.2S_h$		C.8.a	Upset + OBE)	NC-3611.1(b)(4)(b)(1)	$1.2S_h$		Code Case 1606, Regulatory Guide 1.84
	Emergency)					Emergency)	NC-3611.1(b)(4)(b)(2)	$1.8S_h$		
	Faulted + SSE	NC-3611.1(b)(4)(b)(2)	$1.8S_h$		C.8.b	Faulted + SSE		$2.4S_h$		
Pumps		P_m	P_m (or P_L) + P_b				P_m	P_m (or P_L) + P_b		Acceptable alternate,
Classes 2 and 3	Normal	1.1 S	1.65 S			Design + OBE	1.1 S	1.65 S		Code Case 1636,
Inactive	Upset + OBE	1.1 S	1.65 S		C.9.a					Regulatory Guide 1.84
	Emergency	1.1 S	1.65 S			Design + SSE	2.0 S	2.40 S		
	Faulted + SSE	1.2 S	1.8 S		C.9.b					
Active [11]		P_m	P_m (or P_L) + P_b				P_m	P_m (or P_L) + P_b		Acceptable alternate program
	Normal)									
	Upset + OBE)	S	1.5S		C.10.a	Design + OBE	1.0 S	1.5 S		
	Emergency)									
	Faulted + SSE					Design + SSE	1.2 S	1.8 S		

TABLE 3.9B-10 (Cont)

Components	Regulatory Guide 1.48 Loading Combinations	Design Limits	Reg. Position	Tables 3.9B-7, 3.9B-8, and 3.9B-9 Loading Combinations	Design Limits P_m	P_m (or P_L) + P_b	Comparison
Valves Classes 2 and 3 Inactive	Normal)			Design + OBE	1.1 S	1.65 S	Acceptable alternate, Code Case 1635, Regulatory Guide 1.84
	Upset + OBE)	1.1 Pr	C.11.a				
	Emergency)			Design + SSE	2.0 S	2.4 S	
	Faulted + SSE	1.2 Pr	C.11.b		including	≤ Pr	
Active	All Conditions	1.0 Pr	C.12.a		Same as for Inactive		Acceptable alternate

NOTES ( ) Indicates note number from Regulatory Guide 1.48 as reprinted below.

(9) Division 1 of Section VIII of ASME Boiler and Pressure Vessel Code does not provide rules for design analysis. If detailed analysis is performed, Division 1 vessels should meet, as a minimum, equations a and b below, which are applicable to Regulatory Positions 6.a and 6.b, respectively.

a. $m\sigma < 1.1\, s > \dfrac{\sigma m + \sigma b}{1.5}$

b. $m\sigma < 1.5\, S > \dfrac{\sigma m + \sigma b}{1.5}$

where:

m = primary membrane stress
b = primary bending stress
S = allowable stress value as specified in Appendix I Section III of ASME Boiler and Pressure Vessel Code

(10) For loadings designated in Regulatory Position 8.a(2). Only equation 9 of NC-3651 need be met.

(11) In addition to compliance with the design limits specified, assurance of operability under all design loading combinations should be provided by any appropriate combination of the following suggested measures:

a. In situ testing (for example, preoperational testing after the component is installed in the plant).
b. Full-scale prototype testing.
c. Reduced-scale prototype testing.
d. Detailed stress and deformation analyses (includes experimental stress and deformation analyses).

In the performance of tests or analyses to demonstrate operability, the structural interaction of the entire assembly (for example, valve-operator and pump-motor assembly) should be considered. If superposition of test results for other than the combined loading condition is proposed, the applicability of such a procedure should be demonstrated. The design limits for nonactive pumps and valves may be used for the applicable loading combinations if appropriate analyses and/or testing confirms that operability is not impaired when designed to these limits.

TABLE 3.9B-11

DEFINITIONS OF LOADINGS APPLICABLE TO PIPING SYSTEMS

D	-	Sustained mechanical loads, including deadweight of piping, components, contents, and insulation (includes snow loads when applicable.
P_d	-	Design pressure
P_{max}	-	Maximum internal pressure occuring in the normal and upset plant operating conditions. For Class 1 piping only, P_{max} shall designate the "Range of Operating Pressures" for normal, upset, and test conditions.
P_E	-	The greater of P_{max} or internal pressure occuring during the emergency plant operating conditions ($P_E \leq 1.5\ P_d$)
P_F	-	The greater of P_{max} or internal pressure occuring during the faulted plant operating condition ($P_F \leq 2.0\ P_d$)
P_h	-	Hydrostatic pressure during hydrotest including an additional 75 psi to account for potential static head pressures plus an assumed 6 percent maximum overpressure. (Where P_h is evaluated with other sustained loads, the longitudinal pressure stress shall be used. Where P_h is evaluated alone, the circumferential pressure stress shall be used.)
T	-	Loads due to thermal expansion of the system in response to average fluid temperature.
R	-	Loads induced in the piping due to the thermal growth of equipment and/or structures to which the piping is connected as a result of the BVPS-2 normal and upset plant conditions.
R′	-	Loads induced in the piping due to thermal growth of equipment and/or structures to which the piping is connected as a result of BVPS-2 faulted plant conditions.
R"	-	Loads induced in the piping due to pressure response of the containment during a plant test condition, including any thermal expansion effects occurring during the test condition.
E	-	Inertia effects of the OBE.
E′	-	Inertia effects of the SSE.

TABLE 3.9B-11 (Continued)

A - Loads induced in the piping due to response of the connected equipment and/or civil structures to the OBE (commonly referred to as OBE anchor movements), including ground motion and building rocking motion between independent structures (orbital motion).

A′ - Loads induced in the piping due to response of the connected equipment and/or civil structures to the SSE (commonly referred to as SSE anchor movements), including ground motion and building rocking motion between independent structures (orbital motion).

S - Loads induced due to building settlement effects.

H - Loads resulting from occasional loads other than seismic. Examples of these loads would be: water hammer, steam hammer, opening and closing of safety relief valves, etc.

L - Local stress effects in piping and/or piping components due to sudden changes in fluid temperature. These loads are commonly referred to as thermal transient effects.

X - Loads induced in the piping due to pressure/temperature response (growth) of the containment during a BVPS-2 faulted plant condition.

Y′ - Effects of pipe striking pipe (pipe whip) or effects of blowdown of an adjacent system (jet impingement loads), as defined for the BVPS 2 faulted plant condition.

W - Wind loads.

TABLE 3.9B-13

SAFETY-RELATED PIPE SUPPORT SNUBBERS*

System	Piping Component Line No.	No. of Snubbers	Type	Supplier
Stm. Gen. Blowdown	2BDG-002-66-2	1	Mechanical	Pacific Scientific
	2BDG-002-66-2	2	Hydraulic	Lisega
	2BDG-002-70-2	1	Hydraulic	Lisega
	2BDG-003-261-2	1	Mechanical	Pacific Scientific
Boron Recovery	2BRS-002-005-3	1	Hydraulic	Lisega
Component Cooling	2CCP-020-464-3	2	Mechanical	Pacific Scientific
	2CCP-020-473-3	1	Mechanical	Pacific Scientific
Chemical and Volume Control	2CHS-002-001-1	1	Hydraulic	Lisega
	2CHS-002-002-2	1	Mechanical	Pacific Scientific
	2CHS-004-076-2	1	Hydraulic	Lisega
	2CHS-002-096-1	2	Hydraulic	Lisega
	2CHS-002-097-1	1	Hydraulic	Lisega
	2CHS-003-106-2	2	Mechanical	Pacific Scientific
	2CHS-003-126-1	2	Hydraulic	Lisega
	2CHS-002-140-1	1	Hydraulic	Lisega
	2CHS-002-141-1	1	Hydraulic	Lisega
	2CHS-002-142-1	1	Hydraulic	Lisega
Drains (Hydrogen-ated)	2DGS-002-1-1	1	Hydraulic	Lisega
Emergency Diesel Combustion	2EDG-038-003-3	1	Mechanical	Pacific Scientific
Air and Exhaust Gases	2EDG-038-014-3	5	Mechanical	Pacific Scientific
Emergency Feedwater	2FWE-004-117-2	2	Hydraulic	Lisega
	2FWE-006-151-3	2	Hydraulic	Lisega
Feedwater	2FWS-016-11-4**	1	Hydraulic	Lisega
	2FWS-016-012-2	6	Hydraulic	Lisega
	2FWS-016-016-4**	1	Hydraulic	Lisega
	2FWS-016-016-4**	1	Hydraulic	Lisega
	2FWS-016-017-2	1	Hydraulic	Lisega
	2FWS-016-022-2	1	Hydraulic	Lisega
	2FWS-016-032-4**	1	Hydraulic	Lisega

TABLE 3.9B-13 (Cont)

System	Piping Component Line No.	No. of Snubbers	Type	Supplier
Main Steam	2MSS-032-001-4**	4	Hydraulic	Lisega
	2MSS-032-001-4**	5	Mechanical	Pacific Scientific
	2MSS-032-002-4**	2	Hydraulic	Lisega
	2MSS-032-002-4**	1	Hydraulic	Paul-Munroe
	2MSS-032-002-4**	5	Mechanical	Pacific Scientific
	2MSS-032-003-4**	2	Hydraulic	Lisega
	2MSS-032-003-4**	6	Mechanical	Pacific Scientific
	2MSS-038-004-4**	3	Mechanical	Pacific Scientific
	2MSS-032-035-2	2	Mechanical	Pacific Scientific
	2MSS-032-035-2	5	Hydraulic	Lisega
	2MSS-032-039-2	3	Hydraulic	Lisega
	2MSS-032-043-2	2	Mechanical	Pacific Scientific
	2MSS-032-043-2	2	Hydraulic	Lisega
	2MSS-003-046-3	1	Mechanical	Pacific Scientific
	2MSS-010-200-3	1	Mechanical	Pacific Scientific
Quench Spray	2QSS-010-003-2	1	Hydraulic	Paul-Munroe
	2QSS-010-003-2	1	Mechanical	Pacific Scientific
	2QSS-010-004-2	2	Mechanical	Pacific Scientific
	2QSS-006-243-2	2	Mechanical	Pacific Scientific
Reactor Coolant	2RCS-002-025-1	2	Hydraulic	Lisega
	2RCS-002-045-1	3	Hydraulic	Lisega
	2RCS-002-65-1	1	Hydraulic	Lisega
	2RCS-004-081-1	1	Hydraulic	Lisega
	2RCS-006-104-4**	4	Mechanical	Pacific Scientific
	2RCS-006-105-4**	5	Mechanical	Pacific Scientific
	2RCS-006-106-4**	7	Mechanical	Pacific Scientific
	2RCS-006-107-1	2	Mechanical	Pacific Scientific
	2RCS-003-108-1	2	Mechanical	Pacific Scientific
	2RCS-003-109-1	1	Mechanical	Pacific Scientific
	2RCS-003-110-1	3	Mechanical	Pacific Scientific
	2RCS-006-111-4**	9	Mechanical	Pacific Scientific
	2RCS-012-112-4**	7	Mechanical	Pacific Scientific
	2RCS-006-113-4**	1	Mechanical	Pacific Scientific
	2RCS-006-156-4	3	Mechanical	Pacific Scientific
	2RCS-006-242-4**	2	Mechanical	Pacific Scientific

TABLE 3.9B-13 (Cont)

System	Piping Component Line No.	No. of Snubbers	Type	Supplier
Residual Heat Removal	2RHS-012-001-1	2	Mechanical	Pacific Scientific
	2RHS-010-004-2	1	Mechanical	Pacific Scientific
	2RHS-010-010-2	1	Mechanical	Pacific Scientific
	2RHS-006-015-4**	1	Hydraulic	Lisega
	2RHS-010-016-2	2	Mechanical	Pacific Scientific
	2RHS-006-027-4**	1	Mechanical	Pacific Scientific
Recirculation Spray	2RSS-012-001-2	2	Hydraulic	Paul-Munroe
	2RSS-012-002-2	2	Hydraulic	Lisega
	2RSS-012-009-2	2	Hydraulic	Paul-Munroe
	2RSS-012-010-2	2	Hydraulic	Paul-Munroe
	2RSS-012-011-2	1	Hydraulic	Paul-Munroe
	2RSS-004-013-2	2	Hydraulic	Lisega
	2RSS-004-016-2	2	Hydraulic	Lisega
	2RSS-004-024-2	1	Hydraulic	Lisega
Safety Injection	2SIS-014-001-2	2	Mechanical	Pacific Scientific
	2SIS-014-002-2	1	Mechanical	Pacific Scientific
	2SIS-010-003-2	3	Mechanical	Pacific Scientific
	2SIS-010-004-2	1	Mechanical	Pacific Scientific
	2SIS-008-006-2	1	Mechanical	Pacific Scientific
	2SIS-010-048-2	1	Mechanical	Pacific Scientific
	2SIS-004-098-2	2	Mechanical	Pacific Scientific
	2SIS-006-164-2	2	Mechanical	Pacific Scientific
	2SIS-010-245-2	1	Hydraulic	Lisega
	2SIS-006-270-1	1	Mechanical	Pacific Scientific
	2SIS-006-271-1	2	Mechanical	Pacific Scientific

TABLE 3.9B-13 (Cont)

System	Piping Component Line No.	No. of Snubbers	Type	Supplier
Service Water	2SWS-024-066-3	1	Mechanical	Pacific Scientific
	2SWS-016-072-3	3	Mechanical	Pacific Scientific
	2SWS-016-073-3	2	Mechanical	Pacific Scientific
	2SWS-016-074-3	4	Mechanical	Pacific Scientific
	2SWS-016-075-3	5	Mechanical	Pacific Scientific
	2SWS-024-960-3	2	Mechanical	Pacific Scientific

NOTES:

* None of the snubbers listed are used as vibration arrestors or dual-purpose restraints.

** Snubbers on NNS piping that are within a QA Category I stress analysis boundary.

TABLE 3.9B-14

LOAD COMBINATIONS FOR PIPE SUPPORTS EXCEPT QSS, RSS, AND SIS(3,4,5)

Plant Operating Condition	Load Combinations (1,6)	Allowable Tensile Stress (2,7)
Normal/Upset	D + T + R + R″ + S(8)	0.6 S_y
	D + E + H + T + R + A + W + S (9)	0.8 S_y
Emergency	D + H	0.8 S_y
Faulted	D + E′ + H + Y′	0.8 S_y

NOTES:

1. For definition of terms, see Table 3.9B-11.

2. Buckling criterion for pipe supports is in accordance with the AISC Specification.

3. Generally, an enveloped design load is used, thus producing a conservative load combination. The above load combination and limits may be used when specific loading methods are needed.

4. Refer to Table 39B-15 for allowable tensile stress values for QSS, RSS, and SIS systems.

5. QSS, RSS, and SIS systems correspond to:

   QSS - Quench spray system
   RSS - Recirculation spray system
   SIS - Safety injection system

6. For instrumentation tubing that is normally dead-ended (i.e., no flow) the thermal loads are determined using a temperature gradient based on the maximum temperature of the source line.

7. The above allowables are the basic tensile stress allowables and include a 1/3 increase for dynamic type loads. All other requirements of the AISC Specification related to member stresses are satisfied.

8. During containment pressure test, only system thermal conditions that occur during the test are considered.

9. Wind loads (W) are not considered acting concurrently with OBE inertia effects (E) and OBE anchor movements (A).

TABLE 3.9B-15

LOAD COMBINATIONS FOR PIPE SUPPORTS FOR QSS, RSS, AND SIS[3,4]

Plant Operating Condition	Load Combinations[1,5]	Allowable Tensile Stress[2,6]
Normal/Upset	D + T + R + R″ + S[7]	0.6 S_y
	D + E + H + T + R + A + W + S[8,9]	0.8 S_y
Emergency	D + H	0.8 S_y
Faulted	D + E′ + H + Y′ + T + R′ + A′ + X	0.95 S_y[10]

NOTES:

1. For definition of terms, see Table 3.9B-11.
2. Buckling criterion for pipe supports is in accordance with the AISC Specification.
3. Generally, an enveloped design load is used, thus producing a conservative load combination. The above load combination and limits may be used when specific loading methods are needed.
4. QSS, RSS, and SIS Systems correspond to:

   QSS - Quench Spray System
   RSS - Recirculation Spray System
   SIS - Safety Injection System

5. For instrumentation tubing that is normally dead-ended (i.e., no flow) the thermal loads are determined using a temperature gradient based on the maximum temperature of the source line.
6. The above allowables are the basic tensile stress allowables and include a 1/3 increase for dynamic type loads. All other requirements of the AISC Specification related to member stresses are satisfied.
7. During containment pressure test, only system thermal conditions that occur during the test are considered.
8. Wind loads (W) are not considered acting concurrently with OBE inertia effects (E) and OBE anchor movements (A).
9. For the RSS system, earthquake loads need not be considered concurrently with the flow transients loads resulting from RSS systems testing during plant shutdown and refueling.

TABLE 3.9B-15 (Cont)

10. The higher allowable shown for the faulted condition ($.95S_y$) is used because of the inclusion of secondary type loads.

TABLE 3.9B-16

LOADS, LOAD COMBINATIONS, AND STRESS LIMITS FOR
S&W DESIGNED ASME III, CLASS 2 AND 3 EQUIPMENT SUPPORTS

Plant Design or Operating Condition	Loads and Loading Combinations	Stress Limits	Reference Source
Normal	Deadweight of Component and Supports Temperature Pressure Mechanical (Piping) Loads***	Structural Members Tension and Bending (F_t) = $0.6S_y$ Shear (F_y) = $0.4S_y$ Bolts (Either above or:) Tension (F_t) = S /2 Shear (F_v) = $0.62S_u/3$ Welds	ASME III Subsection NF Subart: NF-3100 NF-3230 Article XVII-2000 Table NF-3292.1-1 (Above used as a guide)
Upset	Normal and OBE	Same as normal	Same as normal
Emergency	Not applicable*	--	--
Faulted	Normal and SSE****	Structural Members Lesser of: 1.2 (S_y/F_t) or .7 (S_u/F_t)** Bolts 0.7 $S_u/F_t < S_y$	ASME III Subsection NF Subart: NF-3230 Appendix F Subart: F-1370 (Above used as a guide)

NOTES:

* As stated in Section 3.9B.1-1.

** Limits used only when faulted stresses exceed normal/upset allowables (conservative). Not to exceed S_y.
S_y is specified minimum material yield strengh at temperature.
S_u is specified minimum material ultimate strength at temperature.
For bolting materials 0.7 S_u is less than S_y.

*** Includes thermal expansion and another point motion loads.

**** Pipe rupture loads (pipe whip or jet impingement) which might affect component supports are evaluated or avoided by barriers or restraints.

TABLE 3.9B-17
METHODS OF LOAD COMBINATIONS

1. The “Range of Moments” concept is used for the following loads: thermal expansion loads (T), thermal anchor displacement loads (R, R’), loads due to containment temperature and/or pressure response (R”, X), and loads due to thermal transient effects (L).

2. Seismic anchor displacement loads (A, A’) and sustained mechanical loads such as deadweight (D) are combined with other occasional loads by absolute summation.

3. Occasional loads (H) are combined with seismic loads (E, E’) by the SRSS method. Where multiple occasional loads (H) exist concurrently, they are combined with each other by the SRSS method, except as described in Item 6 below.

4. Pipe whip and jet impingement loads (Y’) are combined with other dynamic loads by the SRSS method.

5. (Deleted)

6. Steady state blowdown loads (H) resulting from system operation or tests (e.g., steady-state safety valve discharge) are combined with other occasional loads (H) by absolute summation, by SRSS, or may be considered separately, depending on the time phasing between these events.

7. Wind loads (W) are combined with other applicable loads by absolute summation.

TABLE 3.9B-18
ACTIVE PUMPS (BOP)[1]

Mark No.	Type	Class	Function	Normal Mode	Post-LOCA Mode
2CCP*P21A	Centrifugal	3	Component Cooling Water	On	Off/On
2CCP*P21B	Centrifugal	3	Component Cooling Water	On	Off/On
2CCP*P21C	Centrifugal	3	Component Cooling Water	On	Off/On
2EGF*P21A	Vertical	3	Emergency Diesel Generator Fuel Oil	Off	On/Off
2EGF*P21B	Vertical	3	Emergency Diesel Generator Fuel Oil	Off	On/Off
2EGF*P21C	Vertical	3	Emergency Diesel Generator Fuel Oil	Off	On/Off
2EGF*P21D	Vertical	3	Emergency Diesel Generator Fuel Oil	Off	On/Off
2FNC*P21A	Centrifugal	3	Circulate Fuel Pool Borated Water	On	Off/On
2FNC*P21B	Centrifugal	3	Circulate Fuel Pool Borated Water	On	Off/On
2FWE*P22	Centrifugal	3	Aux. Feedwater to Steam Gen.	Off	On
2FWE*P23A	Centrifugal	3	Aux. Feedwater to Steam Gen.	Off	On
2FWE*P23B	Centrifugal	3	Aux. Feedwater to Steam Gen.	Off	On
2QSS*P21A	Centrifugal	2	Quench Spray to Containment	Off	On
2QSS*P21B	Centrifugal	2	Quench Spray to Containment	Off	On

TABLE 3.9B-18 (Cont)

Mark No.	Type	Class	Function	Normal Mode	Post-LOCA Mode
2RSS*P21A	Deep Draft Vert.	2	Containment Water Recirculation	Off	On
2RSS*P21B	Deep Draft Vert.	2	Containment Water Recirculation	Off	On
2RSS*P21C	Deep Draft Vert.	2	Containment Water Recirculation	Off	On
2RSS*P21D	Deep Draft Vert.	2	Containment Water Recirculation	Off	On
2SWS*P21A	Deep Draft Vert.	3	Service Water to Components	On	On
2SWS*P21B	Deep Draft Vert.	3	Service Water to Components	On	On
2SWS*P21C	Deep Draft Vert.	3	Service Water to Components	On	On
2SWS*P25A	Centrifugal	3	Control Room Refrigerator Condenser Recirculation	On	On/Off
2SWS*P25B	Centrifugal	3	Control Room Refrigerator Condenser Recirculation	On	On/Off

NOTE: 1) This pump listing is for design applicability of "Active" components. For test requirements of those pumps contained in the BVPS-2 IST Program, see the BVPS-2 Inservice Testing Program for Pumps and Valves (ASME OM Code).

TABLE 3.9B-19

ACTIVE VALVES (BOP)[2]

Mark No.	Type	Class	Actuated By	Function	Normal Position[1]
2ASS*AOV130A	Globe	3	Air	HELB Isolation	O
2ASS*AOV130B	Globe	3	Air	HELB Isolation	O
2BDG*AOV100A1	Globe	2	Air	Steam Gen. Blowdown Isolation	O
2BDG*AOV100B1	Globe	2	Air	Steam Gen. Blowdown Isolation	O
2BDG*AOV100C1	Globe	2	Air	Steam Gen. Blowdown Isolation	O
2BDG*AOV101A1	Globe	2	Air	Steam Gen. Blowdown Isolation	O
2BDG*AOV101A2	Globe	2	Air	Steam Gen. Blowdown Isolation	O
2BDG*AOV101B1	Globe	2	Air	Steam Gen. Blowdown Isolation	O
2BDG*AOV101B2	Globe	2	Air	Steam Gen. Blowdown Isolation	O
2BDG*AOV101C1	Globe	2	Air	Steam Gen. Blowdown Isolation	O
2BDG*AOV101C2	Globe	2	Air	Steam Gen. Blowdown Isolation	O
2BDG*AOV102A1	Globe	2	Air	Sample Line Isolation	O
2BDG*AOV102A2	Globe	2	Air	Sample Line Isolation	O
2BDG*AOV102B1	Globe	2	Air	Sample Line Isolation	O
2BDG*AOV102B2	Globe	2	Air	Sample Line Isolation	O
2BDG*AOV102C1	Globe	2	Air	Sample Line Isolation	O
2BDG*AOV102C2	Globe	2	Air	Sample Line Isolation	O
2CCP*AOV107A	Globe	3	Air	Thermal Barrier Rupture Isolation	O
2CCP*AOV107B	Globe	3	Air	Thermal Barrier Rupture Isolation	O
2CCP*AOV107C	Globe	3	Air	Thermal Barrier Rupture Isolation	O
2CCP*AOV171	Globe	3	Air	SC-3/NNS Isolation	O
2CCP*AOV172	Globe	3	Air	SC-3/NNS Isolation	O
2CCP*AOV173	Globe	3	Air	SC-3/NNS Isolation	O
2CCP*AOV174	Globe	3	Air	SC-3/NNS Isolation	O
2CCP*DCV100-1	Globe	3	E/H	CCP Pump Min. Flow	M
2CCP*DCV100-2	Globe	3	E/H	CCP Pump Min. Flow	M
2CCP*DCV101A	Butterfly	3	Air	CCP Heat Exchanger Supply	M
2CCP*DCV101B	Butterfly	3	Air	CCP Heat Exchanger Supply	M
2CCP*DCV101C	Butterfly	3	Air	CCP Heat Exchanger Supply	M
2CCP*MOV112A	Butterfly	3	Motor	RHS Supply	C
2CCP*MOV112B	Butterfly	3	Motor	RHS Supply	C

TABLE 3.9B-19 (Cont)

Mark No.	Type	Class	Actuated By	Function	Normal Position[1]
2CCP*MOV118	Ball	3	Motor	SC-3/NNS Break	O
2CCP*MOV119	Ball	3	Motor	SC-3/NNS Break	O
2CCP*MOV120	Ball	3	Motor	SC-3/NNS Break	O
2CCP*MOV150-1	Butterfly	2	Motor	Containment Isolation	O
2CCP*MOV150-2	Butterfly	2	Motor	Containment Isolation	O
2CCP*MOV151-1	Butterfly	2	Motor	Containment Isolation	O
2CCP*MOV151-2	Butterfly	2	Motor	Containment Isolation	O
2CCP*MOV156-1	Butterfly	2	Motor	Containment Isolation	O
2CCP*MOV156-2	Butterfly	2	Motor	Containment Isolation	O
2CCP*MOV157-1	Butterfly	2	Motor	Containment Isolation	O
2CCP*MOV157-2	Butterfly	2	Motor	Containment Isolation	O
2CCP*MOV175-1	Butterfly	3	Motor	SC-3/NNS Isolation	O
2CCP*MOV175-2	Butterfly	3	Motor	SC-3/NNS Isolation	O
2CCP*MOV176-1	Butterfly	3	Motor	SC-3/NNS Isolation	O
2CCP*MOV176-2	Butterfly	3	Motor	SC-3/NNS Isolation	O
2CCP*MOV177-1	Butterfly	3	Motor	SC-3/NNS Isolation	O
2CCP*MOV177-2	Butterfly	3	Motor	SC-3/NNS Isolation	O
2CCP*MOV178-1	Butterfly	3	Motor	SC-3/NNS Isolation	O
2CCP*MOV178-2	Butterfly	3	Motor	SC-3/NNS Isolation	O
2CCP*RV102	Relief	2	Self	Containment Isolation	-
2CCP*RV103	Relief	2	Self	Containment Isolation	-
2CCP*RV104	Relief	2	Self	Containment Isolation	-
2CCP*RV105	Relief	2	Self	Containment Isolation	-
2CCP*324	Butterfly	3	Manual	Train Separation	O
2CCP*325	Butterfly	3	Manual	Train Separation	O
2CCP*5	Check	3	Self	Pump Discharge Check Valve	-
2CCP*6	Check	3	Self	Pump Discharge Check Valve	-
2CCP*4	Check	3	Self	Pump Discharge Check Valve	-
2CCP*27B	Butterfly	3	Manual	Train Separation	O
2CCP*27A	Butterfly	3	Manual	Train Separation	O
2CCP*354	Butterfly	3	Manual	Train Separation	O
2CCP*355	Butterfly	3	Manual	Train Separation	O

TABLE 3.9B-19 (Cont)

Mark No.	Type	Class	Actuated By	Function	Normal Position[1]
2CCP*321	Gate	3	Manual	Train Separation	O
2CCP*322	Gate	3	Manual	Train Separation	O
2CCP*289	Check	3	Self	Thermal Barrier Rupture Isolation	-
2CCP*291	Check	3	Self	Thermal Barrier Rupture Isolation	-
2CCP*290	Check	3	Self	Thermal Barrier Rupture Isolation	-
2CCP*352	Check	3	Self	SC-3/NNS Break Check Valve	-
2CCP*323	Gate	3	Manual	Train Separation	O
2CCP*326	Gate	3	Manual	Train Separation	O
2CHS*RV160	Relief	2	Self	Penetration Overpressure	-
2CHS*RV203	Relief	2	Self	Regenerative Heat Exchanger Relief	-
2CHS*RV260A	Relief	2	Self	Penetration Overpressure	-
2CHS*RV260B	Relief	2	Self	Penetration Overpressure	-
2CHS*RV260C	Relief	2	Self	Penetration Overpressure	-
2CHS*RV8144	Relief	2	Self	Regenerative Heat Exchanger Relief	-
2CHS*SOV206	Globe	2	Solenoid	A.H. Emerg. Boration	C
2CHS*474	Check	2	Weight	Containment Isolation	-
2CHS*476	Check	2	Weight	Containment Isolation	-
2CHS*475	Check	2	Weight	Containment Isolation	-
2CHS*473	Check	2	Weight	Containment Isolation	-
2CHS*472	Check	2	Weight	Containment Isolation	-
2CHS*31	Check	2	Weight	Containment Isolation	-
2CHS*FCV113B	Globe	2	Air	Normal Boration	O
2CHS*136	Check	2	Self	Alt. Emerg. Boration	-
2CVS*SOV102	Globe	2	Solenoid	Cont. Rad. Monitor Discharge	O
2CVS*SOV151A	Globe	2	Solenoid	Cont. Vacuum Suction	O
2CVS*SOV151B	Globe	2	Solenoid	Cont. Vacuum Suction	O
2CVS*SOV152A	Globe	2	Solenoid	Cont. Vacuum Suction	O
2CVS*SOV152B	Globe	2	Solenoid	Cont. Vacuum Suction	O
2CVS*SOV153A	Globe	2	Solenoid	Cont. Vacuum Suction	O
2CVS*SOV153B	Globe	2	Solenoid	Cont. Vacuum Suction	O
2CVS*93	Check	2	Self	Cont. Atm. Rad. Monitor Discharge	-

TABLE 3.9B-19 (Cont)

Mark No.	Type	Class	Actuated By	Function	Normal Position[1]
2DAS*AOV100A	Globe	2	Air	Cont. Sump Pump Discharge	C
2DAS*AOV100B	Globe	2	Air	Cont. Sump Pump Discharge	O
2DGS*AOV108A	Globe	2	Air	Prim. Drains Transfer Pump Discharge	C
2DGS*AOV108B	Globe	2	Air	Prim. Drains Transfer Pump Discharge	O
2DAS*RV110	Relief	2	Self	Containment Isol. Thermal Relief	-
2DGS*RV115	Relief	2	Self	Containment Isol. Thermal Relief	-
2EGA*100	Check	3	Self	D.G. Air Tank Inlet	-
2EGA*101	Check	3	Self	D.G. Air Tank Inlet	-
2EGA*118	Check	3	Excess Flow	SC-3/NNS Isolation Flow	-
2EGA*119	Check	3	Excess Flow	SC-3/NNS Isolation Flow	-
2EGA*130	Check	3	Self	D.G. Air Tank Inlet	-
2EGA*131	Check	3	Self	D.G. Air Tank Inlet	-
2EGA*155	Check	3	Excess Flow	SC-3/NNS Isolation Flow	-
2EGA*156	Check	3	Excess Flow	SC-3/NNS Isolation Flow	-
2EGF*7	Check	3	Self	DG Fuel Oil Transfer Pump Discharge	-
2EGF*9	Check	3	Self	DG Fuel Oil Transfer Pump Discharge	-
2EGF*8	Check	3	Self	DG Fuel Oil Transfer Pump Discharge	-
2EGF*10	Check	3	Self	DG Fuel Oil Transfer Pump Discharge	-
2FNC*109	Check	3	Self	Fuel Pool Coolant Pump Discharge	-
2FNC*108	Check	3	Self	Fuel Pool Coolant Pump Discharge	-
2FPW*AOV205	Globe	2	Air	Fire Protect/Contain. Isolation	C
2FPW*AOV206	Globe	2	Air	Fire Protect/Contain. Isolation	C
2FPW*382	Check	2	Weight	Fire Protect/Contain. Isolation	-
2FPW*388	Check	2	Weight	Fire Protect/Contain. Isolation	-
2FPW*761	Check	2	Weight	Fire Protect/Contain. Isolation	-
2FPW*753	Check	2	Weight	Fire Protect/Contain. Isolation	-
2FWE*FCV122	ARC	3	Flow	Aux. Feedwater Flow Control	-
2FWE*FCV123A	ARC	3	Flow	Aux. Feedwater Flow Control	-
2FWE*FCV123B	ARC	3	Flow	Aux. Feedwater Flow Control	-
2FWE*HCV100A	Globe	2	E/H	Aux. Feedwater Flow Control	O

TABLE 3.9B-19 (Cont)

Mark No.	Type	Class	Actuated By	Function	Normal Position[1]
2FWE*HCV100B	Globe	2	E/H	Aux. Feedwater Flow Control	O
2FWE*HCV100C	Globe	2	E/H	Aux. Feedwater Flow Control	O
2FWE*HCV100D	Globe	2	E/H	Aux. Feedwater Flow Control	O
2FWE*HCV100E	Globe	2	E/H	Aux. Feedwater Flow Control	O
2FWE*HCV100F	Globe	2	E/H	Aux. Feedwater Flow Control	O
2FWE*RV101	Relief	3	Self	Aux. Feedwater Relief	-
2FWE*SOV100A	Globe	3	Solenoid	Prim.PH.Demin.Wtr.Stor.Tank Isol.	O
2FWE*SOV100B	Globe	3	Solenoid	Prim.PH.Demin.Wtr.Stor.Tank Isol.	O
2FWE*92	Butterfly	3	Manual	P.P.D.W.S.T. Switchover to S.W.	C
2FWE*95	Butterfly	3	Manual	P.P.D.W.S.T. Switchover to S.W.	O
2FWE*90	Butterfly	3	Manual	P.P.D.W.S.T. Switchover to S.W.	C
2FWE*93	Butterfly	3	Manual	P.P.D.W.S.T. Switchover to S.W.	O
2FWE*94	Butterfly	3	Manual	P.P.D.W.S.T. Switchover to S.W.	O
2FWE*91	Butterfly	3	Manual	P.P.D.W.S.T. Switchover to S.W.	C
2FWE*42A	Check	2	Self	Aux. Feedwater	-
2FWE*42B	Check	2	Self	Aux. Feedwater	-
2FWE*43A	Check	2	Self	Aux. Feedwater	-
2FWE*43B	Check	2	Self	Aux. Feedwater	-
2FWE*44A	Check	2	Self	Aux. Feedwater	-
2FWE*44B	Check	2	Self	Aux. Feedwater	-
2FWE*99	Check	2	Self	Aux. Feedwater	-
2FWE*100	Check	2	Self	Aux. Feedwater	-
2FWE*101	Check	2	Self	Aux. Feedwater	-
2FWS*FCV478	Globe	2	Air	Feedwater Control/Isolation	M
2FWS*FCV479	Globe	2	Air	Feedwater Control/Isolation	C
2FWS*FCV488	Globe	2	Air	Feedwater Control/Isolation	M
2FWS*FCV489	Globe	2	Air	Feedwater Control/Isolation	C
2FWS*FCV498	Globe	2	Air	Feedwater Control/Isolation	M
2FWS*FCV499	Globe	2	Air	Feedwater Control/Isolation	C
2FWS*HYV157A	Gate	2	E/H	Feedwater Isolation	O
2FWS*HYV157B	Gate	2	E/H	Feedwater Isolation	O
2FWS*HYV157C	Gate	2	E/H	Feedwater Isolation	O

TABLE 3.9B-19 (Cont)

Mark No.	Type	Class	Actuated By	Function	Normal Position[1]
2FWS*28	Check	2	Self	Feedwater Isolation	-
2FWS*29	Check	2	Self	Feedwater Isolation	-
2FWS*30	Check	2	Self	Feedwater Isolation	-
2GNS*SOV853A	Globe	2	Solenoid	Accumulator Vent	C
2GNS*SOV853B	Globe	2	Solenoid	Accumulator Vent	C
2GNS*SOV853C	Globe	2	Solenoid	Accumulator Vent	C
2GNS*SOV853D	Globe	2	Solenoid	Accumulator Vent	C
2GNS*SOV853E	Globe	2	Solenoid	Accumulator Vent	C
2GNS*SOV853F	Globe	2	Solenoid	Accumulator Vent	C
2GNS*SOV854A	Globe	2	Solenoid	Accumulator Vent	C
2GNS*SOV854B	Globe	2	Solenoid	Accumulator Vent	C
2HCS*MOV112A	Ball	2	Motor	RBNR Inlet Isolation	C
2HCS*MOV112B	Ball	2	Motor	RBNR Inlet Isolation	C
2HCS*MOV116	Ball	2	Motor	Containment Isolation	C
2HCS*MOV117	Ball	2	Motor	Containment Isolation	C
2HCS*MOV120A	Plug	2	Motor	RBNR Outlet Isolation	C
2HCS*MOV120B	Plug	2	Motor	RBNR Outlet Isolation	C
2HCS*SOV114A	Globe	2	Solenoid	Containment Isolation	C
2HCS*SOV114B	Globe	2	Solenoid	Containment Isolation	C
2HCS*SOV115A	Globe	2	Solenoid	Containment Isolation	C
2HCS*SOV115B	Globe	2	Solenoid	Containment Isolation	C
2HCS*SOV133A	Globe	2	Solenoid	Containment Isolation	C
2HCS*SOV133B	Globe	2	Solenoid	Containment Isolation	C
2HCS*SOV134A	Globe	2	Solenoid	Containment Isolation	C
2HCS*SOV134B	Globe	2	Solenoid	Containment Isolation	C
2HCS*SOV135A	Globe	2	Solenoid	Containment Isolation	C
2HCS*SOV135B	Globe	2	Solenoid	Containment Isolation	C
2HCS*SOV136A	Globe	2	Solenoid	Containment Isolation	C
2HCS*SOV136B	Globe	2	Solenoid	Containment Isolation	C
2HCS*111	Ball	2	Manual	Containment Isolation Backup	C
2HCS*110	Ball	2	Manual	Containment Isolation Backup	C
2HVC*MOD201A	Butterfly	3	Motor	Isolation-Outdoor Air Norm. Supply	O

TABLE 3.9B-19 (Cont)

Mark No.	Type	Class	Actuated By	Function	Normal Position[1]
2HVC*MOD201B	Butterfly	3	Motor	Isolation-Outdoor Air Norm. Supply	O
2HVC*MOD201C	Butterfly	3	Motor	Isolation-Exhaust to Outdoor	C
2HVC*MOD201D	Butterfly	3	Motor	Isolation-Exhaust to Outdoor	C
2HVC*MOD204A	Butterfly	3	Motor	Isolation-Outdoor Emerg. Supply	C
2HVC*MOD204B	Butterfly	3	Motor	Isolation-Outdoor Emerg. Supply	C
2HVC*SOV201A	Globe	3	Solenoid	Modulate Refrig. Supply to DX Coils 2HVC*ACU201A,B	M
2HVC*SOV201B	Globe	3	Solenoid	Modulate Refrig. Supply to DX Coils 2HVC*ACU201A,B	M
2HVR*MOD23A[3]	Butterfly	2	Motor	Isolation-Containment Purge Supply	C
2HVR*MOD23B[3]	Butterfly	2	Motor	Isolation Containment Purge Supply	C
2HVR*MOD25A[3]	Butterfly	2	Motor	Isolation-Containment Purge Exhaust	C
2HVR*MOD25B[3]	Butterfly	2	Motor	Isolation-Containment Purge Exhaust	C
2IAC*MOV130	Plug	2	Motor	Containment Isolation	O
2IAC*MOV133	Plug	2	Motor	Containment Isolation	O
2IAC*MOV134	Plug	2	Motor	Containment Isolation	O
2IAC*22	Check	2	Weight	Containment Isolation	-
2LMS*SOV950	Globe	2	Solenoid	Cont. Atm. Pressure to Indicator	O
2LMS*SOV951	Globe	2	Solenoid	Cont. Atm. Pressure to Indicator	O
2LMS*SOV952	Globe	2	Solenoid	Cont. Atm. Pressure to Indicator	O
2LMS*SOV953	Globe	2	Solenoid	Cont. Atm. Pressure to Indicator	O
2MSS*AOV102A	Globe	2	Air	Main Steam Warmup	C
2MSS*AOV102B	Globe	2	Air	Main Steam Warmup	C
2MSS*AOV102C	Globe	2	Air	Main Steam Warmup	C
2MSS*AOV101A	Globe	2	Air	Main Steam Isolation	O
2MSS*AOV101B	Globe	2	Air	Main Steam Isolation	O
2MSS*AOV101C	Globe	2	Air	Main Steam Isolation	O
2MSS*SOV105A	Globe	2	Solenoid	Steam to Terry Turbine	C
2MSS*SOV105B	Globe	2	Solenoid	Steam to Terry Turbine	C
2MSS*SOV105C	Globe	2	Solenoid	Steam to Terry Turbine	C
2MSS*SOV105D	Globe	2	Solenoid	Steam to Terry Turbine	C
2MSS*SOV105E	Globe	2	Solenoid	Steam to Terry Turbine	C

TABLE 3.9B-19 (Cont)

Mark No.	Type	Class	Actuated By	Function	Normal Position[1]
2MSS*SOV105F	Globe	2	Solenoid	Steam to Terry Turbine	C
2MSS*SOV120	Globe	2	Solenoid	Main Steam Radiation Monitor Isolation	C
2MSS*SV101A	Relief	2	Self	Main Steam Safety	-
2MSS*SV101B	Relief	2	Self	Main Steam Safety	-
2MSS*SV101C	Relief	2	Self	Main Steam Safety	-
2MSS*SV102A	Relief	2	Self	Main Steam Safety	-
2MSS*SV102B	Relief	2	Self	Main Steam Safety	-
2MSS*SV102C	Relief	2	Self	Main Steam Safety	-
2MSS*SV103A	Relief	2	Self	Main Steam Safety	-
2MSS*SV103B	Relief	2	Self	Main Steam Safety	-
2MSS*SV103C	Relief	2	Self	Main Steam Safety	-
2MSS*SV104A	Relief	2	Self	Main Steam Safety	-
2MSS*SV104B	Relief	2	Self	Main Steam Safety	-
2MSS*SV104C	Relief	2	Self	Main Steam Safety	-
2MSS*SV105A	Relief	2	Self	Main Steam Safety	-
2MSS*SV105B	Relief	2	Self	Main Steam Safety	-
2MSS*SV105C	Relief	2	Self	Main Steam Safety	-
2MSS*19	Check	3	Self	Steam Supply to Terry Turbine	-
2MSS*20	Check	3	Self	Steam Supply to Terry Turbine	-
2MSS*18	Check	3	Self	Steam Supply to Terry Turbine	-
2MSS*196	Check	3	Self	Steam Supply to Terry Turbine	-
2MSS*199	Check	3	Self	Steam Supply to Terry Turbine	-
2MSS*352	Check	3	Self	Steam Supply to Terry Turbine	-
2PAS*SOV103	Globe	2	Solenoid	SC-2/NNS Isolation	C
2PAS*SOV105A1	Globe	2	Solenoid	Containment Isolation	C
2PAS*SOV105A2	Globe	2	Solenoid	Containment Isolation	C
2PAS*SOV106	Globe	3	Solenoid	Gas Return Isolation	C
2PAS*SOV107	Globe	2	Solenoid	2RSS*P21A Sample Isolation	C

TABLE 3.9B-19 (Cont)

Mark No.	Type	Class	Actuated By	Function	Normal Position[1]
2PAS*SOV108	Globe	2	Solenoid	2RSS*P21B Sample Isolation	C
2PAS*SOV113	Globe	3	Solenoid	SWS/PAS Isolation	C
2PAS*SOV114	Globe	3	Solenoid	SWS/PAS Isolation	C
2QSS*AOV120A	Globe	2	Air	Refuel Wtr Clg Pump Suct.Isolation	O
2QSS*AOV120B	Globe	2	Air	Refuel Wtr Clg Pump Suct.Isolation	O
2QSS*MOV100A	Gate	2	Motor	QSS Pump Suction	O
2QSS*MOV100B	Gate	2	Motor	QSS Pump Suction	O
2QSS*MOV101A	Gate	2	Motor	QSS Pump Discharge	O
2QSS*MOV101B	Gate	2	Motor	QSS Pump Discharge	O
2QSS*RV101A	Relief	2	Self	Relief for 2QSS*MOV101A	-
2QSS*RV101B	Relief	2	Self	Relief for 2QSS*MOV101B	-
2QSS*SOV101B	Globe	2	Solenoid	QSS Chem. Inj. Header Isolation	O
2QSS*SOV102B	Globe	2	Solenoid	QSS Chem. Inj. Header Isolation	O
2QSS*4	Check	2	Weight	Containment Isolation	-
2QSS*3	Check	2	Weight	Containment Isolation	-
2QSS*304	Check	2	Self	Chem. Inj. Pump Discharge	-
2RCS*RV100	Relief	2	Self	PRT Relief	-
2RCS*68	Check	2	Weight	PRT N_2 Supply	-
2RCS*72	Check	2	Weight	PRT Water Supply	-
2RHS*FCV605A	Butterfly	2	Air	RHR Heat Exchanger Bypass	M
2RHS*FCV605B	Butterfly	2	Air	RHR Heat Exchanger Bypass	M
2RHS*HCV758A	Butterfly	2	Air	RHR Heat Exchanger Isolation	M

TABLE 3.9B-19 (Cont)

Mark No.	Type	Class	Actuated By	Function	Normal Position[1]
2RHS*HCV758B	Butterfly	2	Air	RHB Heat Exchanger Isolation	M
2RHS*RV721A	Relief	2	Self	RHS Pump Suction Relief	-
2RHS*RV721B	Relief	2	Self	RHS Pump Suction Relief	-
2RHS*RV100	Relief	2	Self	Cont. Penet. X-24 Thermal Relief	-
2RSS*MOV154C	Gate	2	Motor	RSS Pump Mini Flow Recirc.	C
2RSS*MOV154D	Gate	2	Motor	RSS Pump Mini Flow Recirc.	C
2RSS*MOV155A	Butterfly	2	Motor	RSS Pump Suction	O
2RSS*MOV155B	Butterfly	2	Motor	RSS Pump Suction	O
2RSS*MOV155C	Butterfly	2	Motor	RSS Pump Suction	O
2RSS*MOV155D	Butterfly	2	Motor	RSS Pump Suction	O
2RSS*MOV156A	Gate	2	Motor	RSS Pump Discharge	O
2RSS*MOV156B	Gate	2	Motor	RSS Pump Discharge	O
2RSS*MOV156C	Gate	2	Motor	RSS Pump Discharge	O
2RSS*MOV156D	Gate	2	Motor	RSS Pump Discharge	O
2RSS*RV156A	Relief	2	Self	Thermal Relief for 2RSS*MOV156A	-
2RSS*RV156B	Relief	2	Self	Thermal Relief for 2RSS*MOV156B	-
2RSS*RV156C	Relief	2	Self	Thermal Relief for 2RSS*MOV156C	-
2RSS*RV156D	Relief	2	Self	Thermal Relief for 2RSS*MOV156D	-
2RSS*29	Check	2	Weight	Containment Isolation	-
2RSS*31	Check	2	Weight	Containment Isolation	-
2RSS*32	Check	2	Weight	Containment Isolation	-
2RSS*30	Check	2	Weight	Containment Isolation	-
2SDS*AOV111A1	Globe	2	Air	Containment Isolation (Steam Drains)	O
2SDS*AOV111A2	Globe	2	Air	Containment Isolation (Steam Drains)	O
2SDS*AOV111B1	Globe	2	Air	Containment Isolation (Steam Drains)	O
2SDS*AOV111B2	Globe	2	Air	Containment Isolation (Steam Drains)	O
2SDS*AOV111C1	Globe	2	Air	Containment Isolation (Steam Drains)	O
2SDS*AOV111C2	Globe	2	Air	Containment Isolation (Steam Drains)	O
2SDS*AOV129A	Globe	2	Air	Containment Isolation (RHR Drain)	O
2SDS*AOV129B	Globe	2	Air	Containment Isolation (RHR Drain)	O
2SIS*RV130	Relief	2	Self	Containment Isolation	-
2SIS*RV175	Relief	2	Self	Safety Accumulator Test Relief	-

TABLE 3.9B-19 (Cont)

Mark No.	Type	Class	Actuated By	Function	Normal Position[1]
2SIS*130	Check	2	Weight	Containment Isolation	-
2SIS*132	Check	2	Weight	Containment Isolation	-
2SIS*133	Check	2	Weight	Containment Isolation	-
2SIS*83	Check	2	Weight	Containment Isolation	-
2SIS*84	Check	2	Weight	Containment Isolation	-
2SIS*94	Check	2	Weight	Containment Isolation	-
2SIS*95	Check	2	Weight	Containment Isolation	-
2SIS*42	Check	2	Weight	SIS Fill Containment Isolation	-
2SIS*895	Check	2	Self	SIS Pump Discharge	-
2SIS*894	Check	2	Self	SIS Pump Discharge	-
2SSR*AOV100A1	Globe	2	Air	Containment Isolation	O
2SSR*AOV100A2	Globe	2	Air	Containment Isolation	O
2SSR*AOV102A1	Globe	2	Air	Containment Isolation	C
2SSR*AOV102A2	Globe	2	Air	Containment Isolation	C
2SSR*AOV109A1	Globe	2	Air	Containment Isolation	O
2SSR*AOV109A2	Globe	2	Air	Containment Isolation	O
2SSR*AOV112A1	Globe	2	Air	Containment Isolation	O
2SSR*AOV112A2	Globe	2	Air	Containment Isolation	O
2SSR*AOV117A	Globe	2	Air	Steam Generator Isolation	O
2SSR*AOV117B	Globe	2	Air	Steam Generator Isolation	O
2SSR*AOV117C	Globe	2	Air	Steam Generator Isolation	O
2SSR*AOV116A	Globe	2	Air	Charging Pump Discharge Sample	C
2SSR*AOV116B	Globe	2	Air	Charging Pump Discharge Sample	C
2SSR*AOV116C	Globe	2	Air	Charging Pump Discharge Sample	C
2SSR*AOV118A	Globe	2	Air	Letdown Flow Sample Valve	C
2SSR*AOV118B	Globe	2	Air	Volume Control Tank Gas Space Sample	C
2SSR*AOV118C	Globe	2	Air	Volume Control Tank Liquid Sample	C
2SSR*AOV118D	Globe	2	Air	Reactor Coolant Filter Influent Sample	C

TABLE 3.9B-19 (Cont)

Mark No.	Type	Class	Actuated By	Function	Normal Position[1]
2SSR*RV117	Relief	2	Self	Containment Penetration Relief	-
2SSR*RV118	Relief	2	Self	Containment Penetration Relief	-
2SSR*RV119	Relief	2	Self	Containment Penetration Relief	-
2SSR*RV120	Relief	2	Self	Containment Penetration Relief	-
2SSR*RV121	Relief	2	Self	Containment Penetration Relief	-
2SSR*RV122	Relief	2	Self	Containment Penetration Relief	-
2SSR*SOV128A1	Globe	2	Solenoid	Containment Isolation	C
2SSR*SOV128A2	Globe	2	Solenoid	Containment Isolation	C
2SSR*SOV129A1	Globe	2	Solenoid	Containment Isolation	C
2SSR*SOV129A2	Globe	2	Solenoid	Containment Isolation	C
2SSR*SOV129B	Globe	2	Solenoid	Sample Pump Isolation	C
2SSR*SOV130A1	Globe	2	Solenoid	Containment Isolation	O
2SSR*SOV130A2	Globe	2	Solenoid	Containment Isolation	O
2SVS*HCV104	Globe	2	E/H	Residual Heat Removal	C
2SVS*PCV101A	Globe	2	E/H	Atmosphere Dump	C
2SVS*PCV101B	Globe	2	E/H	Atmosphere Dump	C
2SVS*PCV101C	Globe	2	E/H	Atmosphere Dump	C
2SVS*80	Check	2	Self	Residual Heat Release Check Valves	-
2SVS*81	Check	2	Self	Residual Heat Release Check Valves	-
2SVS*82	Check	2	Self	Residual Heat Release Check Valves	-
2SWE*MOV116A	Butterfly	3	Motor	Barge Accident SC-3/NNS Isolation	C
2SWE*MOV116B	Butterfly	3	Motor	Barge Accident SC-3/NNS Isolation	C
2SWM*MOV562	Plug	3	Motor	Chlorination SC-3/NNS Isolation	O
2SWM*MOV563	Plug	3	Motor	Chlorination SC-3/NNS Isolation	O
2SWM*MOV564	Plug	3	Motor	Chlorination SC-3/NNS Isolation	O
2SWM*MOV565	Plug	3	Motor	Chlorination SC-3/NNS Isolation	O
2SWS*AOV114	Butterfly	3	Air	SC-3/NNS Isolation	O
2SWS*AOV118A	Globe	3	Air	Unit 1 Seal Wtr. Isolation	O
2SWS*AOV118B	Globe	3	Air	Unit 1 Seal Wtr. Isolation	O
2SWS*MOV102A	Butterfly	3	Motor	SWS Pump Discharge Isolation	O
2SWS*MOV102B	Butterfly	3	Motor	SWS Pump Discharge Isolation	O
2SWS*MOV102C1	Butterfly	3	Motor	SWS Pump Discharge Isolation	C

TABLE 3.9B-19 (Cont)

Mark No.	Type	Class	Actuated By	Function	Normal Position[1]
2SWS*MOV102C2	Butterfly	3	Motor	SWS Pump Discharge Isolation	C
2SWS*MOV103A	Butterfly	3	Motor	RSS Supply	C
2SWS*MOV103B	Butterfly	3	Motor	RSS Supply	C
2SWS*MOV104A	Gate	3	Motor	RSS Supply	O
2SWS*MOV104B	Gate	3	Motor	RSS Supply	O
2SWS*MOV104C	Gate	3	Motor	RSS Supply	O
2SWS*MOV104D	Gate	3	Motor	RSS Supply	O
2SWS*MOV105A	Butterfly	3	Motor	RSS Supply	O
2SWS*MOV105B	Butterfly	3	Motor	RSS Supply	O
2SWS*MOV105C	Butterfly	3	Motor	RSS Supply	O
2SWS*MOV105D	Butterfly	3	Motor	RSS Supply	O
2SWS*MOV106A	Butterfly	3	Motor	CCP/CCS-Contmt Cooling Coil Supply	O
2SWS*MOV106B	Butterfly	3	Motor	CCP/CCS-Contmt Cooling Coil Supply	O
2SWS*MOV107A	Butterfly	3	Motor	CCS SC-3/NNS Isolation	O
2SWS*MOV107B	Butterfly	3	Motor	CCS SC-3/NNS Isolation	O
2SWS*MOV107C	Butterfly	3	Motor	CCS SC-3/NNS Isolation	O
2SWS*MOV107D	Butterfly	3	Motor	CCS SC-3/NNS Isolation	O
2SWS*MOV113A	Gate	3	Motor	Diesel Supply	C
2SWS*MOV113B	Gate	3	Motor	Diesel Supply	C
2SWS*MOV113C	Gate	3	Motor	Diesel Supply	C
2SWS*MOV113D	Gate	3	Motor	Diesel Supply	C
2SWS*MOV148A	Butterfly	3	Motor	Rod Control AC Supply	C
2SWS*MOV148B	Butterfly	3	Motor	Rod Control AC Supply	C
2SWS*MOV152-1	Butterfly	2	Motor	Containment Isolation	O
2SWS*MOV152-2	Butterfly	2	Motor	Containment Isolation	O
2SWS*MOV155-1	Butterfly	2	Motor	Containment Isolation	O
2SWS*MOV155-2	Butterfly	2	Motor	Containment Isolation	O
2SWS*MOV162	Butterfly	3	Motor	SC-3/NNS Isolation	O

TABLE 3.9B-19 (Cont)

Mark No.	Type	Class	Actuated By	Function	Normal Position[1]
2SWS*MOV163	Butterfly	3	Motor	SC-3/NNS Isolation	C
2SWS*MOV164	Butterfly	3	Motor	SC-3/NNS Isolation	C
2SWS*MOV165	Butterfly	3	Motor	SC-3/NNS Isolation	O
2SWS*RV152	Relief	2	Self	Containment Isolation	-
2SWS*RV153	Relief	2	Self	Containment Isolation	-
2SWS*RV154	Relief	2	Self	Containment Isolation	-
2SWS*RV155	Relief	2	Self	Containment Isolation	-
2SWS*SOV130A	Globe	3	Solenoid	Unit 2 Seal Wtr Supply	O
2SWS*SOV130B	Globe	3	Solenoid	Unit 2 Seal Wtr Supply	O
2SWS*59	Check	3	Self	SWS Pump Discharge Check	-
2SWS*58	Check	3	Self	SWS Pump Discharge Check	-
2SWS*57	Check	3	Self	SWS Pump Discharge Check	-
2SWS*162	Gate	3	Manual	"C" Charging Pump Supply Isolation	C
2SWS*165	Gate	3	Manual	"C" Charging Pump Supply Isolation	C
2SWS*111	Check	3	Self	Diesel Check Valve	-
2SWS*110	Check	3	Self	Diesel Check Valve	-
2SWS*113	Check	3	Self	Diesel Check Valve	-
2SWS*112	Check	3	Self	Diesel Check Valve	-
2SWS*488	Check	3	Self	SWS Pump Vacuum Breaker	-
2SWS*487	Check	3	Self	SWS Pump Vacuum Breaker	-
2SWS*486	Check	3	Self	SWS Pump Vacuum Breaker	-
2SWS*40	Butterfly	3	Manual	SC-3/NNS Isolation	O
2SWS*41	Butterfly	3	Manual	SC-3/NNS Isolation	O
2SWS*104	Butterfly	3	Manual	RSS Heat Exchangers	C
2SWS*107	Check	3	Self	Header Check (Barge Acc Loop Cycle)	-

TABLE 3.9B-19 (Cont)

Mark No.	Type	Class	Actuated By	Function	Normal Position[1]
2SWS*106	Check	3	Self	Header Check(Barge Acc Loop Cycle)	-
2SWS*122	Gate	3	Manual	Fuel Pool Backup	C
2SWS*124	Gate	3	Manual	Fuel Pool Backup	C
2SWS*353	Check	3	Self	SC-3/NNS Isolation	-
2SWS*1084	Check	3	Self	Control Room Cooling	-
2SWS*1085	Check	3	Self	Control Room Cooling	-
2VRS*AOV109A1	Globe	2	Air	Hydrogenated Gas Vent	O
2VRS*AOV109A2	Globe	2	Air	Hydrogenated Gas Vent	O

TABLE 3.9B-19 (Cont)

NOTES:

1) Normal Positions

O - Open
C - Closed
M - Modulating

2) This valve listing is for design applicability of "Active" components. For test requirements of those valves contained in the BVPS-2 IST Program, see the BVPS-2 Inservice Testing Program for Pumps and Valves (ASME OM Code).

3) 2HVR*MOD23A,B; 25A,B. These dampers are locked closed during normal operation and in the "active" mode only during cold shutdown and refueling.

TABLE 3.9N-1

SUMMARY OF REACTOR COOLANT SYSTEM DESIGN TRANSIENTS

Normal Conditions	Occurrences
Heatup and cooldown at 100°F/hr (pressurizer cooldown 200°F/hr)	200 (each)(2)
Unit loading and unloading at 5% of full power/min	13,200 (each)
Step load increase and decrease of 10% of full power	2,000 (each)
Large step load decrease with steam dump	200
Steady-state fluctuations	infinite
Refueling	80
Upset Conditions	
Loss of load (without immediate reactor trip)	80
Loss of power	40
Loss of flow	80
Reactor trip from full power	400
Operating basis earthquake	
Westinghouse components (20 earthquakes of 20 cycles each)	400
Piping components (5 earthquakes of 10 cycles each)	50
Inadvertent safety injection actuation	60
RCS cold overpressurization	10(2)
Inadvertent auxiliary spray	10(2)
Faulted Conditions*	**Occurrences**
Reactor coolant pipe break (large loss of coolant accident)	1
Large steam line break	1
Steam generator tube rupture	1 (included in upset condition "reactor trip from full power," above)
Safe shutdown earthquake	1

TABLE 3.9N-1 (Cont)

Test Conditions	Occurrences
Turbine roll test	10
Primary side hydrostatic test	5
Secondary side hydrostatic test	5
Primary side leakage test	50
Supplemental Transients	**Occurrences**
Pressurizer Insurge	Varies (2)(3)
Selected CVCS Transients	
Isolation of Letdown Flow (Mode 4)	400 (2)
Isolation of Charging Flow (Mode 1, 2, 3)	500 (2)
Isolation of Charging Flow (Mode 4)	2000 (2)
Auxiliary Feedwater Injections	2000 (2)
RHR Actuation	200 (2)

NOTE:

* In accordance with the ASME Boiler and Pressure Vessel Code, Section III, faulted conditions are not included in fatigue evaluation.

(2) Critical transient where the predicted count may approach the design count during the period of extended operation. Occurrences of this transient must be counted in accordance with ITS 5.5.3.

(3) The design occurrences depend on specific attributes and severity of the actual transient. This is monitored as part of the Fatigue Management Program.

TABLE 3.9N-2

LOADING COMBINATIONS FOR ASME CLASS 1 COMPONENTS

Condition Classification	Loading Combination
Normal	Normal condition transients, temperature, pressure deadweight
Upset	Upset condition transients, temperature, pressure deadweight operating basis earthquake
Faulted	Faulted condition transients, temperature, pressure, deadweight, safe shutdown earthquake or (safe shutdown earthquake and pipe rupture loads)

TABLE 3.9N-3

ALLOWABLE STRESSES FOR ASME SECTION III CLASS 1 COMPONENTS*

Operating Condition Classification	Vessels/Tanks	Pumps	Valves
Normal	ASME Section III	ASME Section III	ASME Section III
Upset	ASME Section III	ASME Section III	ASME Section III
Faulted	Section 3.9N.1.4.4	Section 3.9N.1.4.4	**

NOTES:

* A test of the components may be performed in lieu of analysis.
** Class 1 Valve Faulted Condition Criteria

Active	Inactive
a) Calculate P_m from paragraph NB3545.1 with Internal Pressure $P_s = 1.25P_s$ $P_m \leq 1.5\ S_m$	a) Calculate P_m from paragraph NB3545.1 with Internal Pressure $P_s = 1.50P_s$ $P_m \leq 2.4\ S_m$ or $0.7\ S_u$
b) Calculate S_n from paragraph NB3545.2 with $Cp = 1.5$ $P_s = 1.25P_s$ $Q_t2 = 0$ $P_{ed} = 1.3x$ value of P_{ed} from equations of 3545.2(b)(1) $S_n \leq 3S_m$	b) Calculate S_n from paragraph NB3545.2 with $Cp = 1.5$ $P_s = 1.50P_s$ $Q_t2 = 0$ $P_{ed} = 1.3x$ value of P_{ed} from equations of NB3545.2(b)(1) $S_n \leq 3S_m$

P_e, P_m, P_b, Q_t, C_p, S_n and S_m as defined by Section III, ASME Code

TABLE 3.9N-4

DESIGN LOADING COMBINATIONS FOR ASME CODE CLASS 2 AND 3 COMPONENTS AND SUPPORTS

Condition Classification	Loading Combination
Design	Design pressure, temperature, deadweight
Normal	Normal condition pressure deadweight, temperature, nozzle loads*
Upset	Upset condition pressure, deadweight, temperature, OBE, nozzle loads*
Faulted	Faulted condition pressure, temperature, deadweight, SSE, nozzle loads*

NOTE:

* Nozzle loads are those loads associated with the particular plant operating conditions for the component under consideration.

TABLE 3.9N-5

STRESS CRITERIA FOR SAFETY RELATED
ASME CLASS 2 AND CLASS 3 TANKS

Condition	Stress Limits
Design and normal	The vessel shall conform to the requirements of ASME Section VIII, Division 1
Upset	$\sigma_m \leq 1.1$ S $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65$ S
Faulted	$\sigma_m \leq 2.0$ S $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 2.4$ S

TABLE 3.9N-6

STRESS CRITERIA FOR ASME CODE CLASS 2 AND CLASS 3
INACTIVE PUMPS

Condition	Stress Limits
Design and normal	ASME Section III, Subsection NC-3400 (or ND-3400)
Upset	$\sigma_m \leq 1.1$ S $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65$ S
Faulted	$\sigma_m \leq 2.0$ S $(\sigma_m \text{ or } \sigma_L) + \sigma_m \leq 2.4$ S

TABLE 3.9N-7

DESIGN CRITERIA FOR ACTIVE PUMPS

Condition	Stress Limits
Design and normal	ASME Section III, Subsection NC-3400 and ND-3400
Upset	$\sigma_m \leq 1.0$ S $\sigma_m + \sigma_b \leq 1.5$ S
Faulted	$\sigma_m \leq 1.2$ S $\sigma_m + \sigma_b \leq 1.8$ S

TABLE 3.9N-8

STRESS CRITERIA FOR SAFETY-RELATED ASME CODE CLASS 2 AND CLASS 3 VALVES

Condition	Stress Limits (1-5)	P_{max} (6)
Design and Normal	Valve bodies shall conform to the requirements of ASME Section III, NC-3,500 (or ND-3,500)	
Upset	$\sigma_m \leq 1.1$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.65$ S	1.1
Faulted	$\sigma_m \leq 2.0$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 2.4$ S	1.5

NOTES:

1. Valve nozzle (piping load) stress analysis is not required when both the following conditions are satisfied by calculation: 1) the section modulus and area of a plane, normal to the flow, through the region defined as the valve body crotch is at least 10 percent greater than the piping connected (or joined) to the valve body inlet and outlet nozzles; and 2) code allowable stress, S, for valve body material is equal to or greater than the code allowable stress, S, of the connecting piping material. If the valve body material allowable stress is less than that of the connected piping, the valve section modulus and area as calculated in the preceding Item 1 previously mentioned shall be multiplied by the ratio $(\sigma S_{pipe}/S_{valve})$. If unable to comply with the requirement, the design by analysis procedure of NB-3545.2 is an acceptable alternate method.

2. Casting quality factor of 1.0 shall be used.

3. These stress limits are applicable to the pressure retaining boundary, and include the effects of loads transmitted by the extended structures, when applicable.

4. Design requirements listed in this table are not applicable to valve discs, stems, seat rings, or other parts of valves which are contained within the confines of the body and bonnet, or otherwise not part of the pressure boundary.

5. These rules do not apply to Class 2 and 3 safety relief valves. Safety relief valves will be designed in accordance with ASME Section III requirements.

TABLE 3.9N-8 (Cont)

6. The maximum pressure resulting from upset, or faulted conditions, shall not exceed the tabulated factors listed under P_{max} times the design pressure or the rated pressure at the applicable operating condition temperature. If the pressure rating limits are met at the operating conditions, the stress limits in Table 3.9N-4 are considered to be satisfied.

TABLE 3.9N-9

ACTIVE PUMPS[1]

Pump	Item No.	System	ANS Safety Class	Normal Mode	Post-LOCA Mode	Function
Centrifugal charging pumps #1, #2 or #3	APCH	CVCS	2	ON/OFF	ON	High head safety injection, also boration for safe shutdown
Boric acid transfer pumps #1 or #2	APBA	CVCS	3	ON/OFF	OFF	Boration for safe shutdown
Low head safety injection pumps #1 or #2	APLH	SIS	2	OFF*	ON	Low head safety injection

NOTE:

* Low head pumps operate during refueling to fill the reactor cavity from the RWST.

1) This pump listing is for Design Applicability of "Active" Components. For test requirements of those pumps contained in the BVPS-2 IST Program, see the BVPS-2 Inservice Testing Program for Pumps and Valves (ASME OM Code).

TABLE 3.9N-10

ACTIVE VALVES (W SCOPE)

Mark No.	Vendor Mark No.	System	Actuated By	Type/ Safety Class	Normal Position	Function
2RCS*MOV535,536,537	8000A/B/C	RCS	Motor	Gate/1	O	Block valve
2RCS*RV551A,B,C	8010A/B/C	RCS	Self	Safety/1	-	RCS pressure protection
2RCS*AOV519	8028	RCS	Air	Diaphragm/2	C	Containment isolation
2RCS*AOV101	8033	RCS	Air	Diaphragm/2	C	Containment isolation
2RCS*SOV200A,B	8035A/B	RCS	Solenoid	Globe/1	C	Head vent isolation
2RCS*SOV201A,B	8038A/B	RCS	Solenoid	Globe/1	C	Head vent isolation
2RCS*HCV250A,B	HCV-443A/B	RCS	Solenoid	Globe/2	C	Head vent discharge
2RCS*PCV455C,D	PCV-455C/D	RCS	Solenoid	Globe/1	C	PORV - pzr relief valve
2RCS*PCV456	PCV-456	RCS	Solenoid	Globe/1	C	PORV - pzr relief valve

TABLE 3.9N-10 (Cont)

Mark No.	Vendor Mark No.	System	Actuated By	Type/ Safety Class	Normal Position	Function
2CHS*MOV381	8100	CVCS	Motor	Gate/2	O	Containment isolation-seal return
2CHS*MOV350	8104	CVCS	Motor	Globe/2	C	BATP discharge isolation
2CHS*MOV289	8107	CVCS	Motor	Gate/2	O	Charging/Containment isolation
2CHS*MOV378	8112	CVCS	Motor	Gate/2	O	Containment isolation-seal return
2CHS*MOV308A-C	8113A/B/C	CVCS	Motor	Globe/2	O	Containment isolation-seal injection
2CHS*MOV8130A/B	8130A/B	CVCS	Motor	Gate/2	O	CCP suction isolation
2CHS*MOV8131A/B	8131A/B	CVCS	Motor	Gate/2	O	CCP suction isolation
2CHS*MOV8132A/B	8132A/B	CVCS	Motor	Gate/2	O	CCP discharge isolation
2CHS*MOV8133A/B	8133A/B	CVCS	Motor	Gate/2	O	CCP discharge isolation
2CHS*MOV310	8146	CVCS	Motor	Gate/2	O	Charging isolation
2CHS*AOV200A/C	8149A/C	CVCS	Air	Globe/2	C	Letdown/containment isolation
2CHS*AOV200B	8149B	CVCS	Air	Globe/2	O	Letdown/containment isolation
2CHS*AOV204	8152	CVCS	Air	Globe/2	O	Containment isolation
2CHS*MOV201	8153	CVCS	Motor	Globe/1	C	RCPB
2CHS*75,76	8314A/B	CVCS	Self	Check/3	-	BATP discharge

TABLE 3.9N-10 (Cont)

Mark No.	Vendor Mark No.	System	Actuated By	Type/ Safety Class	Normal Position	Function
2CHS*474,475,476	8367A/B/C	CVCS	Weight	Check/1	-	RCPB (seal injection)
2CHS*188,189,190	8368A/B/C	CVCS	Self	Check/1	-	RCPB (seal injection)
2CHS*784,785	8377A/B	CVCS	Self	Check/1	-	RCPB (press. spray)
2CHS*870,871	8378A/B	CVCS	Self	Check/1	-	RCPB (norm. charging)
2CHS*31	8381 (Spare)	CVCS	Weight	Check/2	-	Containment isolation
2CHS*18	8440	CVCS	Self	Check/2	-	VCT to CCP suction
2CHS*136	8442	CVCS	Self	Check/2	-	BATP to CCP suction
2CHS*152,153,154	8480A/B/C	CVCS	Self	Check/2	-	CCP miniflow
2CHS*22,23,24	8481A/B/C	CVCS	Self	Check/2	-	CCP discharge
2CHS*LCV115B/D	LCV-115B/D	CVCS	Motor	Gate/2	C	RWST to CCP suction
2CHS*LCV115C/E	LCV-115C/E	CVCS	Motor	Gate/2	O	VCT to CCP suction
2CHS*LCV460A/B	LCV-460A/B	CVCS	Air	Globe/1	O	Letdown stop valve

TABLE 3.9N-10 (Cont)

Mark No.	Vendor Mark No.	System	Actuated By	Type/ Safety Class	Normal Position	Function
2CHS*FCV113A	--	CVCS	Air	Globe/3	C	Train A boration
2RHS*MOV701A,B	8701A/B	RHRS	Motor	Gate/1	C	RHR suction isolation
2RHS*MOV702A,B	8702A/B	RHRS	Motor	Gate/1	C	RHR suction isolation
2RHS*MOV720A,B	8703A/B	RHRS	Motor	Gate/1	C	RHR discharge isolation
2RHS*3,4	8704A/B	RHRS	Self	Check/2	-	RHR pump discharge

TABLE 3.9N-10 (Cont)

Mark No.	Vendor Mark No.	System	Actuated By	Type/ Safety Class	Normal Position	Function
2SIS*MOV840	8800	SIS	Motor	Globe/2	C	Containment isolation
2SIS*MOV867C/D	8801A/B	SIS	Motor	Gate/2	C	ECCS injection flowpath
2SIS*MOV867A/B	8803A/B	SIS	Motor	Gate/2	C	ECCS injection flowpath
2SIS*MOV865A/B/C	+8808A/B/C	SIS	Motor	Gate/2	O	Accumulator discharge isolation
2SIS*MOV8809A/B	8809A/B	SIS	Motor	Gate/2	O	LHSI pump suction isolation
2SIS*MOV8811A/B	8811A/B	SIS	Motor	Gate/2	C	Recirculation pump discharge
2SIS*MOV863A/B	8812A/B	SIS	Motor	Gate/2	C	LHSI header to HHSI pump suction
2SIS*MOV869A	8814	SIS	Motor	Gate/2	C	Cont. iso. H.L. injection
2SIS*MOV869B	8816	SIS	Motor	Gate/2	C	Cont. iso. H.L. injection
2SIS*MOV842	8871	SIS	Motor	Globe/2	C	Containment isolation
2GNS*AOV101-1/-2	8880A/B	SIS	Air	Globe/2	C	Containment isolation
2SIS*MOV836	8885	SIS	Motor	Gate/2	C	ECCS flowpath to cold legs
2SIS*MOV8887A/B	8887A/B	SIS	Motor	Gate/2	O	LHSI C.L. to H.L. crossconnect
2SIS*MOV8888A/B	8888A/B	SIS	Motor	Gate/2	O	LHSI C.L. injection path
2SIS*MOV8889	8889	SIS	Motor	Gate/2	C	H.L. injection path

+ Procured as active; during normal operation, valves are open with power removed.

TABLE 3.9N-10 (Cont)

Mark No.	Vendor Mark No.	System	Actuated By	Type/ Safety Class	Normal Position	Function
2SIS*MOV8890A/B	8890A/B	SIS	Motor	Gate/2	C	LHSI miniflow
2SIS*MOV841	8892	SIS	Motor	Gate/2	O	ECCS injection flowpath
2SIS*27	8926	SIS	Self	Check/2	-	RWST to CCP pumps suction
2SIS*141,145,151	8948A/B/C	SIS	Self	Check/1	-	RCPB, accumulator injection
2SIS*142,147,148	8956A/B/C	SIS	Self	Check/1	-	RCPB, accumulator injection
2SIS*AOV889	8961	SIS	Air	Globe/2	C	Containment isolation
2SIS*107,108,109	8973A/B/C	SIS	Self	Check/1	-	RCPB
2SIS*6,7	8976A/B	SIS	Self	Check/2	-	LHSI pump discharge
2SIS*128,129	8988A/B	SIS	Self	Check/1	-	RCPB
2SIS*122,123,124	8990A/B/C	SIS	Self	Check/1	-	RCPB
2SIS*125,126,127	8992A/B/C	SIS	Self	Check/1	-	RCPB
2SIS*545,546,547	8993A/B/C	SIS	Self	Check/1	-	RCPB
2SIS*134,135,136	8995A/B/C	SIS	Self	Check/1	-	RCPB
2SIS*137,138,139	8997A/B/C	SIS	Self	Check/1	-	RCPB
2SIS*548,550,552	8998A/B/C	SIS	Self	Check/1	-	RCPB
2SIS*HCV868A/B	HCV-937A/B	SIS	Solenoid	Globe/2	C	Emergency boration, safe shutdown

NOTE: This valve listing is for design applicability of "active" components.

For test requirements of those valves contained in the BVPS-2 IST Program, see the BVPS-2 Inservice Testing Program for Pumps and Valves (ASME OM Code).

TABLE 3.9N-11

MAXIMUM DEFLECTIONS ALLOWED FOR REACTOR INTERNAL SUPPORT STRUCTURES

Component	Allowable Deflections (in)	No-Loss-of Functions Deflections (in)
Upper barrel		
Radial inward	4.1	8.2
Radial outward	1.0	1.5
Upper package	0.10	0.15
Rod cluster guide tubes	1.00	1.75

TABLE 3.9N-12

STRESS LIMITS FOR REACTOR VESSEL INTERNAL STRUCTURES

Operating Condition	Stress Categories and Limit of Stress Intensities
Normal and Upset	Figure NG 3221.1
Faulted	Appendix F, Section 3 Rules for Evaluating Faulted Conditions and NG 3,200 as applicable.

TABLE 3.9N-13

LOADING COMBINATIONS FOR CORE SUPPORT STRUCTURES

Condition Classification	Loading Combination
Normal	Normal condition transients, temperature, pressure deadweight
Upset	Upset condition transients, temperature, pressure deadweight, operating basis earthquake
Faulted	Faulted condition transients, temperature, pressure deadweight safe shutdown earthquake or (safe shutdown earthquake and pipe rupture loads)

TABLE 3.9N-14

ALLOWABLE STRESSES FOR CORE SUPPORT STRUCTURES

Operating Condition Classification	Stress Limits ASME Section III
Normal and Upset	$P_m \leq S_m$; $P_m + P_a \leq 1.5\ S_m$; $P_m + P_b + Q \leq 3\ S_m$
Faulted	$P_m \leq 2.4\ S_m$; $P_m + P_b \leq 3.6\ S_m$

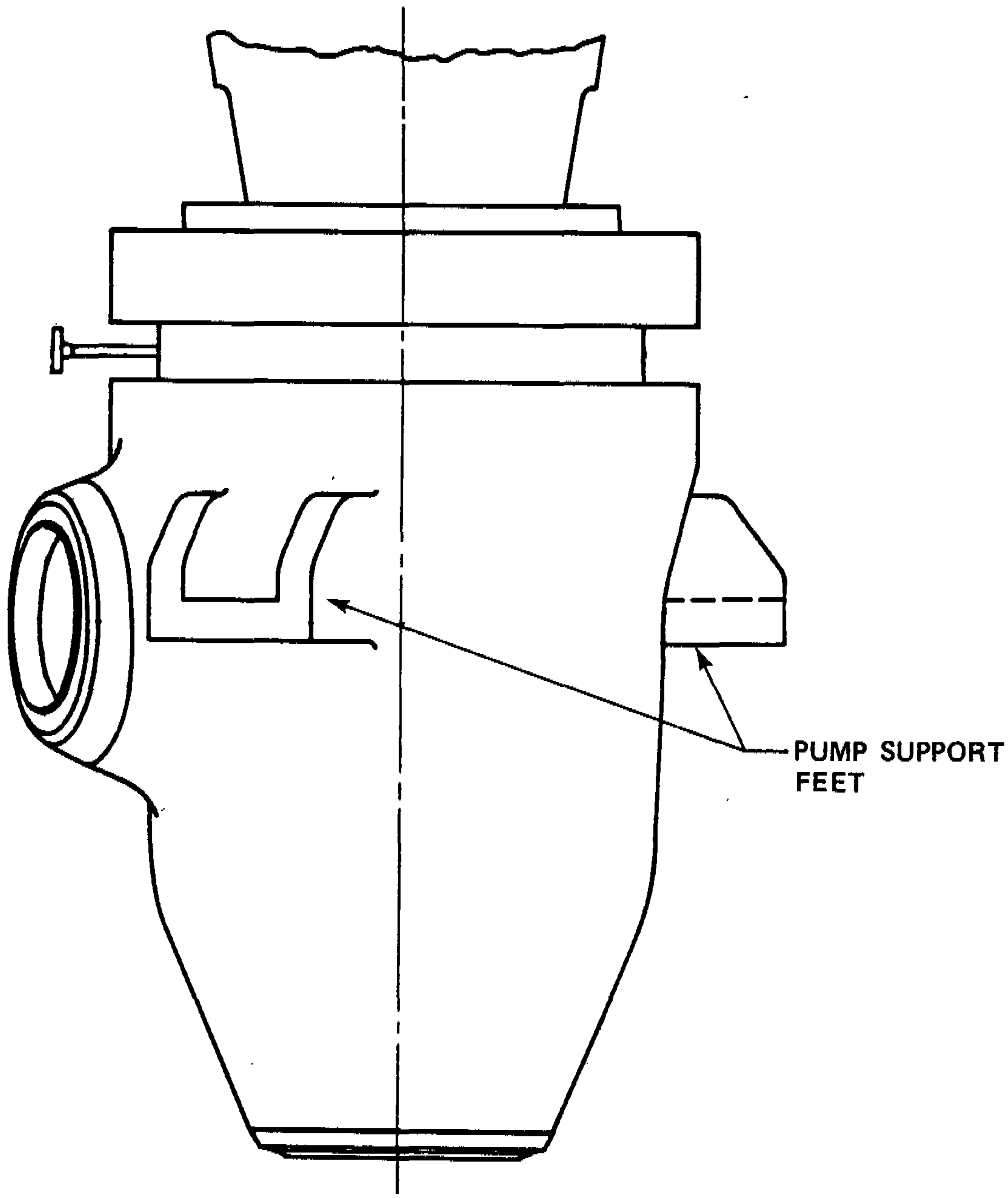


FIGURE 3.9N-1

REATOR COOLANT PUMP
CASING WITH SUPPORT FEET

BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

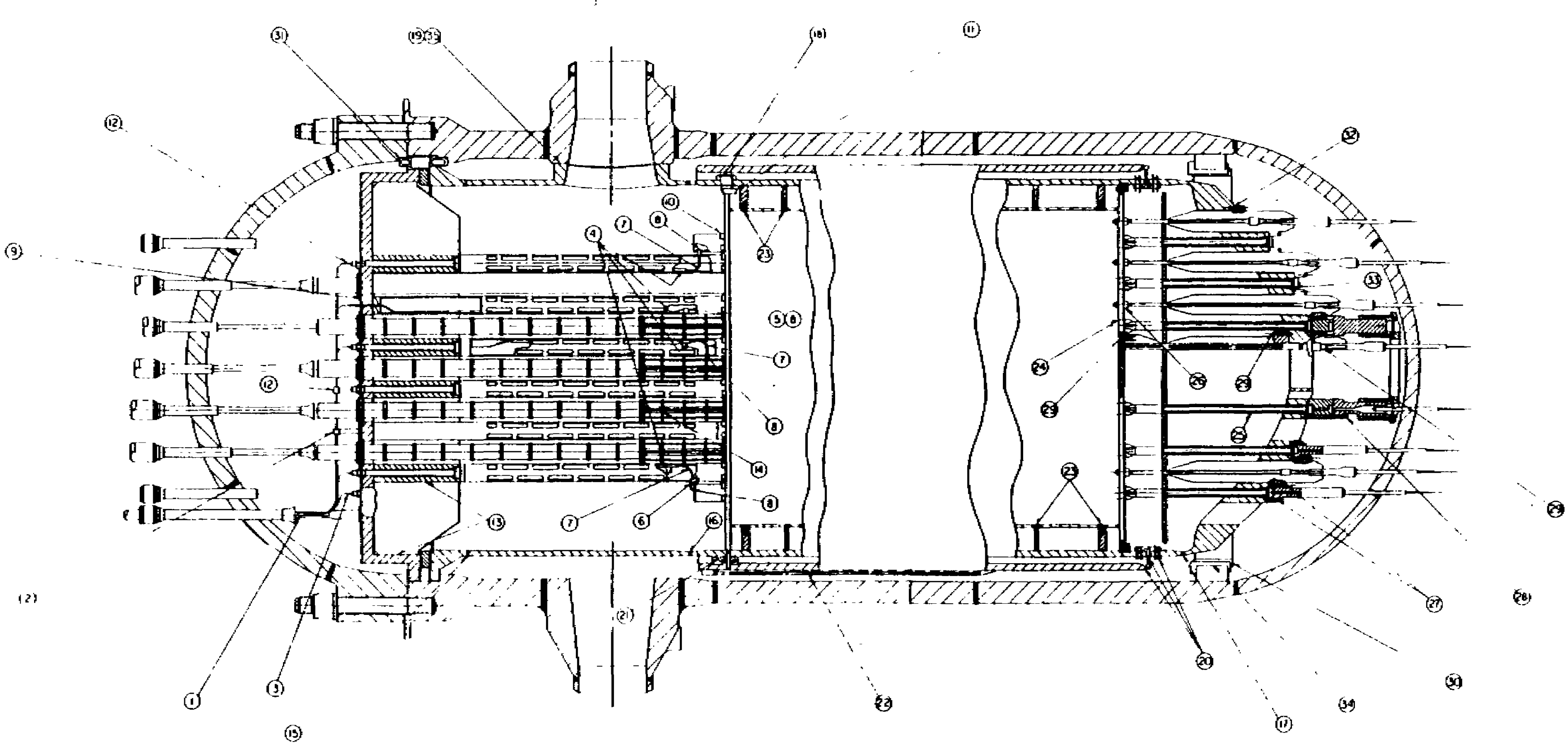


FIGURE 3.9N-2 (SHEET 1)
VIBRATION CHECK-OUT FUNCTIONAL
TEST INSPECTION POINTS
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

Group	No.	FEATURES TO BE EXAMINED
UPPER INTERNALS	1	THERMOCOUPLE CONDUIT CLAMPS INSIDE THE THERMOCOUPLE COLUMN
	2	CLAMP ARRANGEMENTS AT THE MOUNTING BRACKET LOCATIONS
	3	PLUG TO CONDUIT WELD AT THE FIVE SUPPORT COLUMNS ADJACENT TO THE THERMOCOUPLE COLUMNS
	4	ACCESSIBLE ANGLE CONDUIT CLAMPS INSIDE THE UPPER SUPPORT COLUMNS
	5	ACCESSIBLE WELD JOINTS AT THE THERMOCOUPLE STOP FOR THE SELF INSTRUMENTED COLUMNS
	6	WELD JOINTS ON ACCESSIBLE SUPPORT COLUMN AND MIXING DEVICE GUSSETS (THERMOCOUPLE SUPPORT HARDWARE)
	7	RIGIDITY OF EXPOSED PORTION OF THERMOCOUPLE CONDUIT RUNS. AT ACCESSIBLE LOCATIONS. (INSIDE SUPPORT COLUMNS – LOWER END)
	8	RIGIDNESS OF THE ACCESSIBLE PROTRUDING THERMOCOUPLE TIPS
	9	THERMOCOUPLE COLUMN AND GUIDE TUBE SCREW LOCKING DEVICES
	10	ACCESSIBLE SUPPORT COLUMN MIXING DEVICE. ORIFICE PLATE. AND CORE PLATE INSERT SCREW LOCKING DEVICES
	11	UPPER CORE PLATE INSERTS
	12	CONDUIT CONNECTOR FITTINGS AND CROSS RUN CLAMP ARRANGEMENTS
	13	DEEP BEAM WELDS AT THE SKIRT AND AT THE OUTER HOLLOW ROUNDS
	14	ACCESSIBLE GUIDE TUBE WELDS
LOWER INTERNALS	15	UPPER BARREL TO FLANGE GIRTH WELD
	16	UPPER BARREL TO LOWER BARREL GIRTH WELD
	17	LOWER BARREL TO CORE SUPPORT GIRTH WELD
	18	UPPER CORE PLATE ALIGNING PIN WELDS AND BEARING SURFACES
	19	OUTLET NOZZLE INTERFACE SURFACE CONDITION
	20	THERMAL SHIELD FLEXURE ARM. ATTACHMENTS TO BARREL, AND WELD TO THE THERMAL SHIELD. DYE PENETRANT INSPECT ALL SIX
	21	THERMAL SHIELD INTERFACE AT THE HANG OFF PADS
	22	IRRADIATION SPECIMEN BASKET WELDS
	23	BAFFLE ASSEMBLY SCREW LOCKING ARRANGEMENTS AT THE TWO TOP AND THE TWO BOTTOM FORMER ELEVATIONS
	24	CORE SUPPORT COLUMN TO LOWER CORE PLATE SCREW LOCKING DEVICES. (24 RANDOMLY CHOSEN)
	25	CORE SUPPORT COLUMN ADJUSTING SLEEVES
	26	ACCESSIBLE (2) INSTRUMENTATION GUIDE COLUMN LOCKING COLLARS NEAREST THE MANWAY
	27	LOCKING DEVICES OF THE BOTTOM INSTRUMENTATION GUIDE COLUMNS
	28	LOCKING DEVICES OF THE SECONDARY CORE SUPPORT
	29	ACCESSIBLE LOCKING DEVICES OF THE OFF-SET INSTRUMENTATION COLUMN (UPPER AND LOWER ENDS)
	30	RADIAL SUPPORT KEY LOCKING ARRANGEMENTS AND BEARING SURFACES
	31	HEAD AND VESSEL ALIGNING PIN SCREW LOCKING DEVICES AND BEARING SURFACES
	32	CONTACT AT INTERFACE OF THE ACCESSIBLE INSTRUMENTATION GUIDE COLUMNS
	33	CONTACT AT INTERFACE OF THE ACCESSIBLE CORE SUPPORT COLUMN NUTS
VESSEL	34	VESSEL CLEVIS LOCKING ARRANGEMENTS AND BEARING SURFACES
	35	VESSEL NOZZLE INTERFACE SURFACE CONDITION

FIGURE 3.9N-2 (SHEET 2)
VIBRATION CHECK-OUT FUNCTIONAL TEST INSPECTION POINTS
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

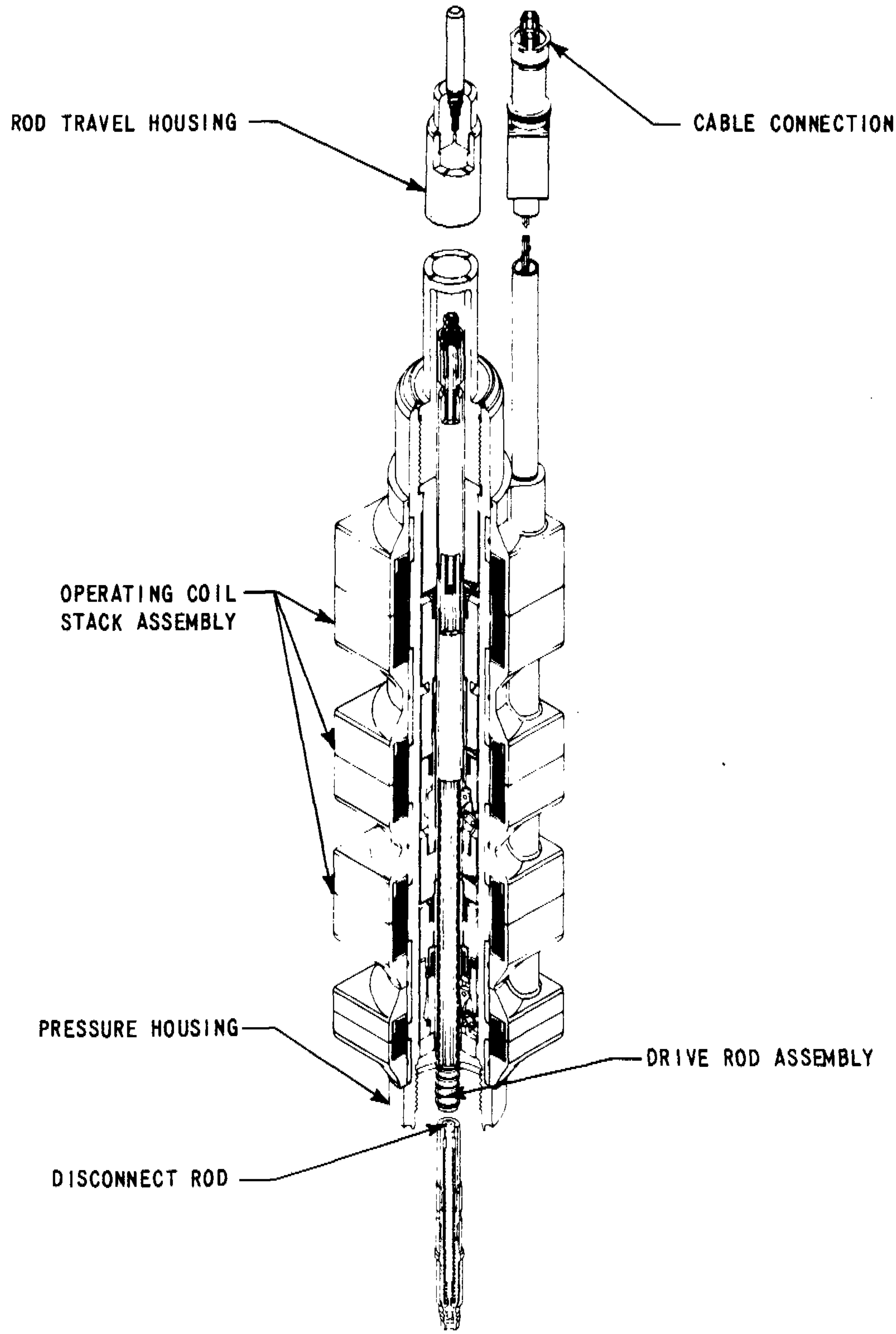


FIGURE 3.9 N - 3
CONTROL ROD DRIVE MECHANISM
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

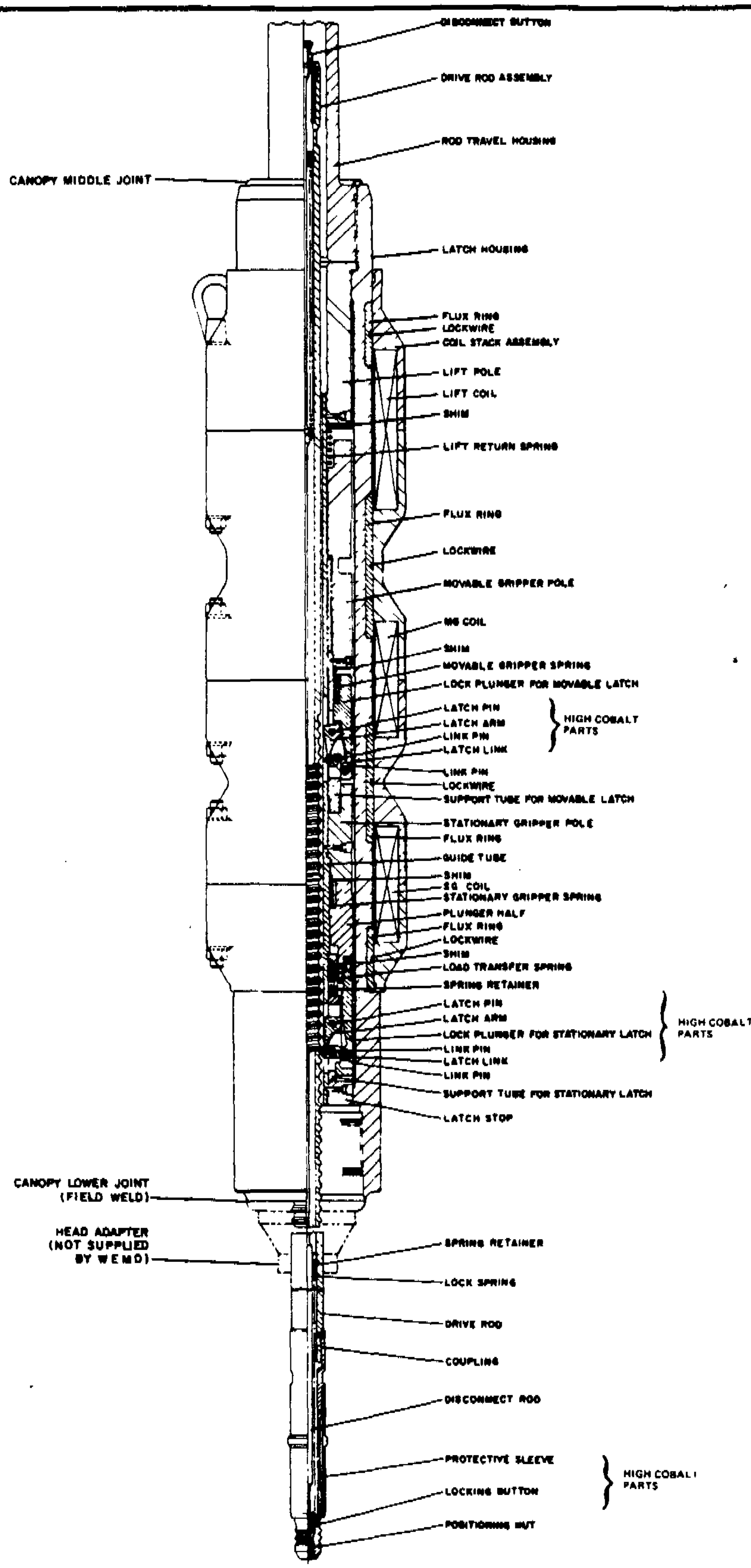
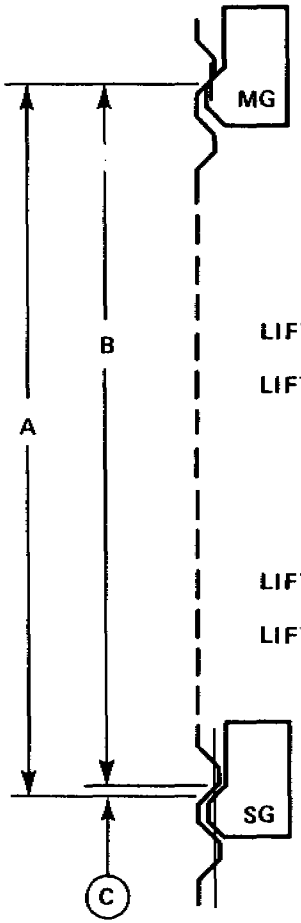


FIGURE 3.9 N -4
CONTROL ROD DRIVE
MECHANISM SCHEMATIC
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

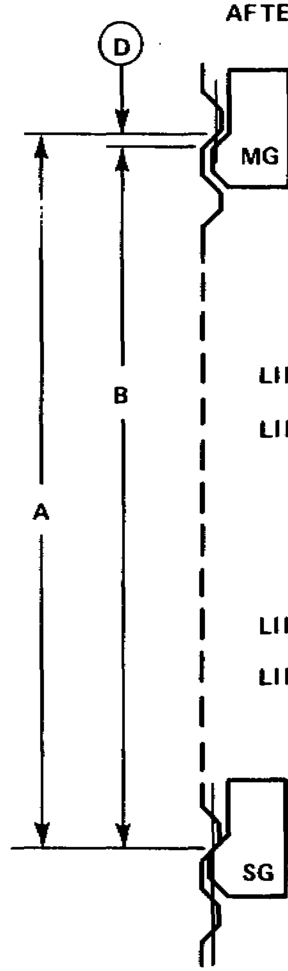
BEFORE LOAD TRANSFER



	AT 70°		
	A	B	C
LIFT COIL OFF	13.133	13.125	0.008
LIFT COIL ON	13.758	13.750	0.008

	AT 650°		
	A	B	C
LIFT COIL OFF	13.203	13.174	0.029
LIFT COIL ON	13.848	13.802	0.046

AFTER LOAD TRANSFER



	AT 70°		
	A	B	D
LIFT COIL OFF	13.125	13.070	0.055
LIFT COIL ON	13.750	13.695	0.055

	AT 650°		
	A	B	D
LIFT COIL OFF	13.174	13.123	0.051
LIFT COIL ON	13.802	13.768	0.034

FIGURE 3.9N-5
NOMINAL LATCH CLEARANCE MINIMUM AND MAXIMUM TEMPERATURE
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

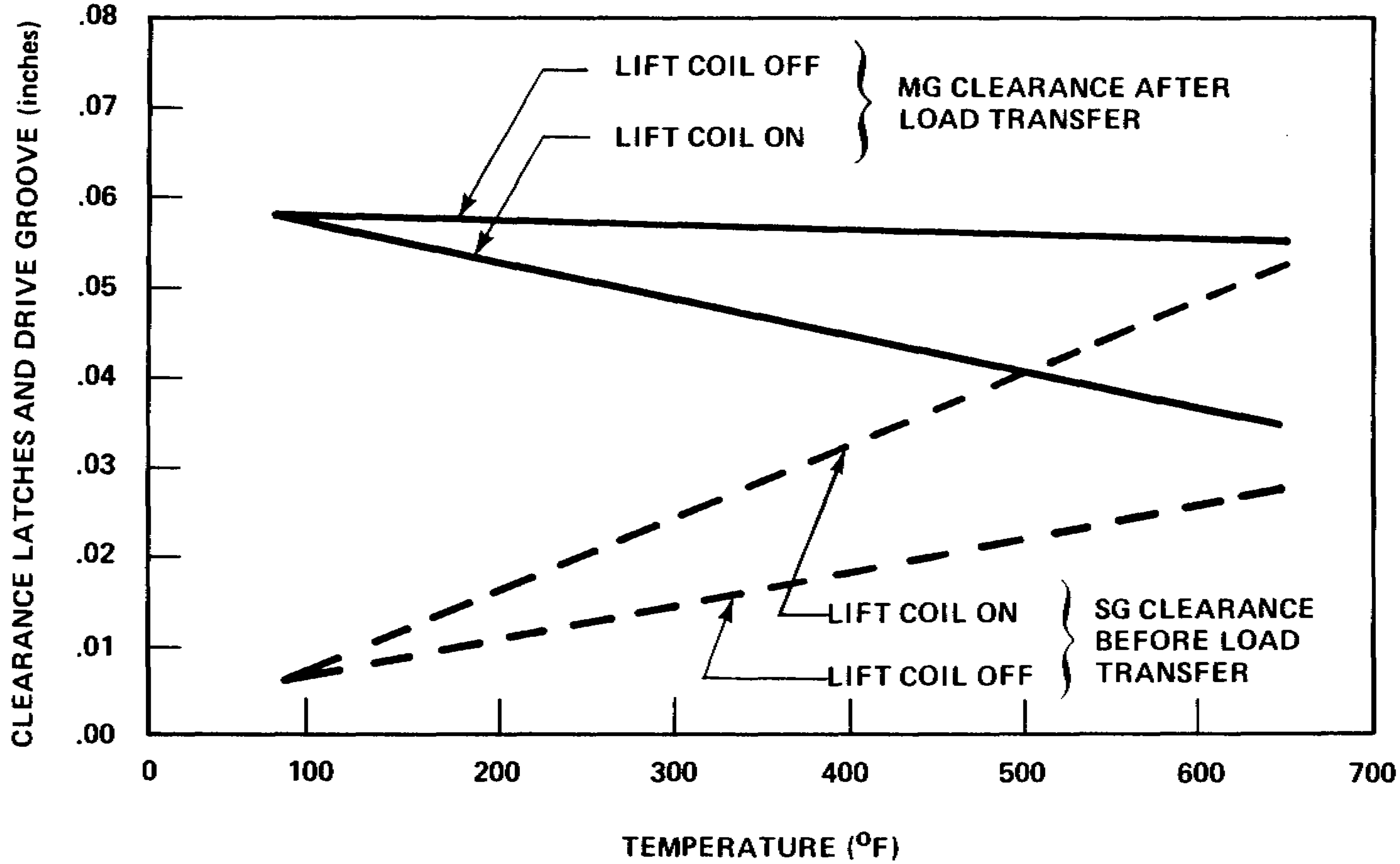


FIGURE 3.9N-6
CONTROL ROD DRIVE MECHANISM
LATCH CLEARANCE THERMAL EFFECT
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

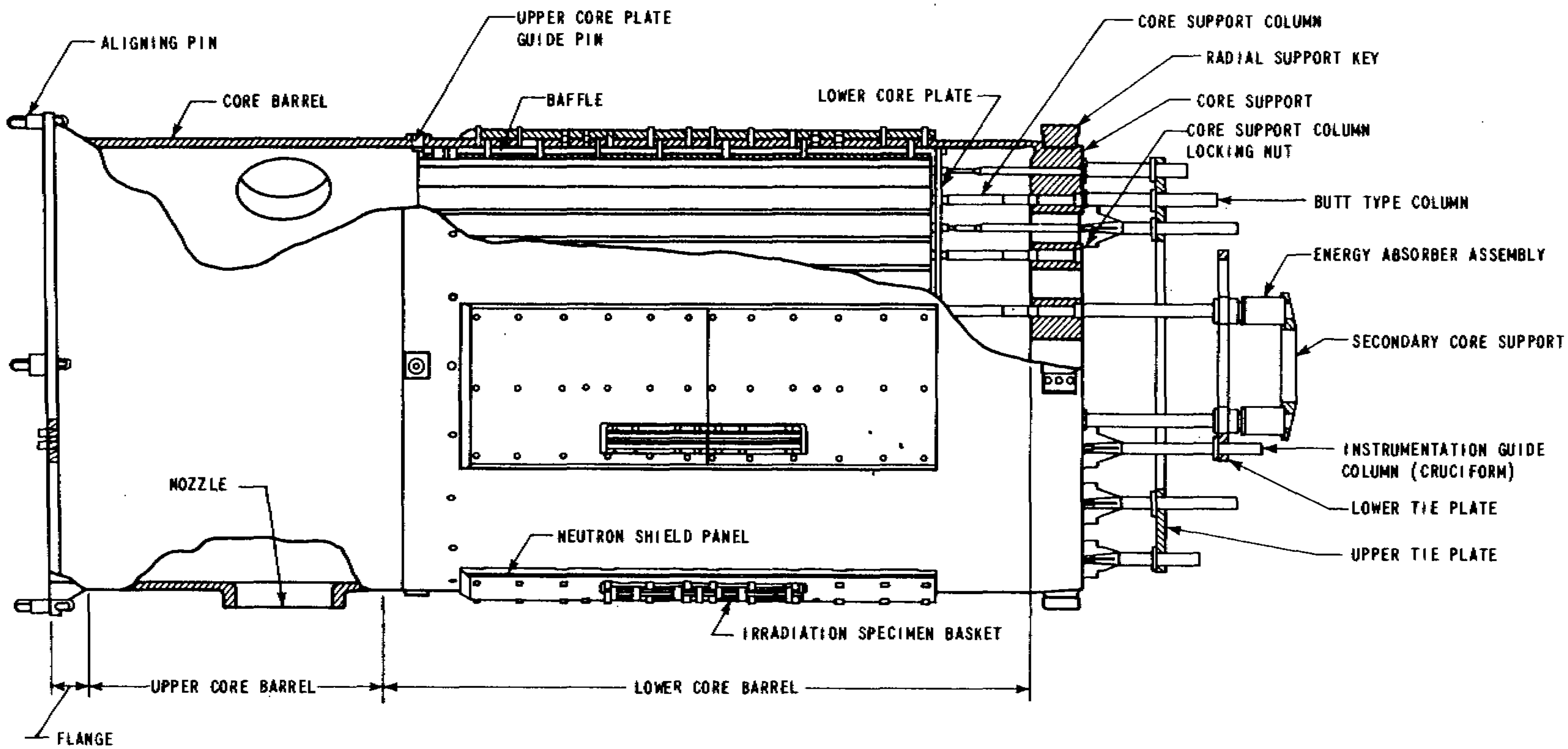


FIGURE 3.9N-7
LOWER CORE SUPPORT ASSEMBLY
(CORE BARREL ASSEMBLY)
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

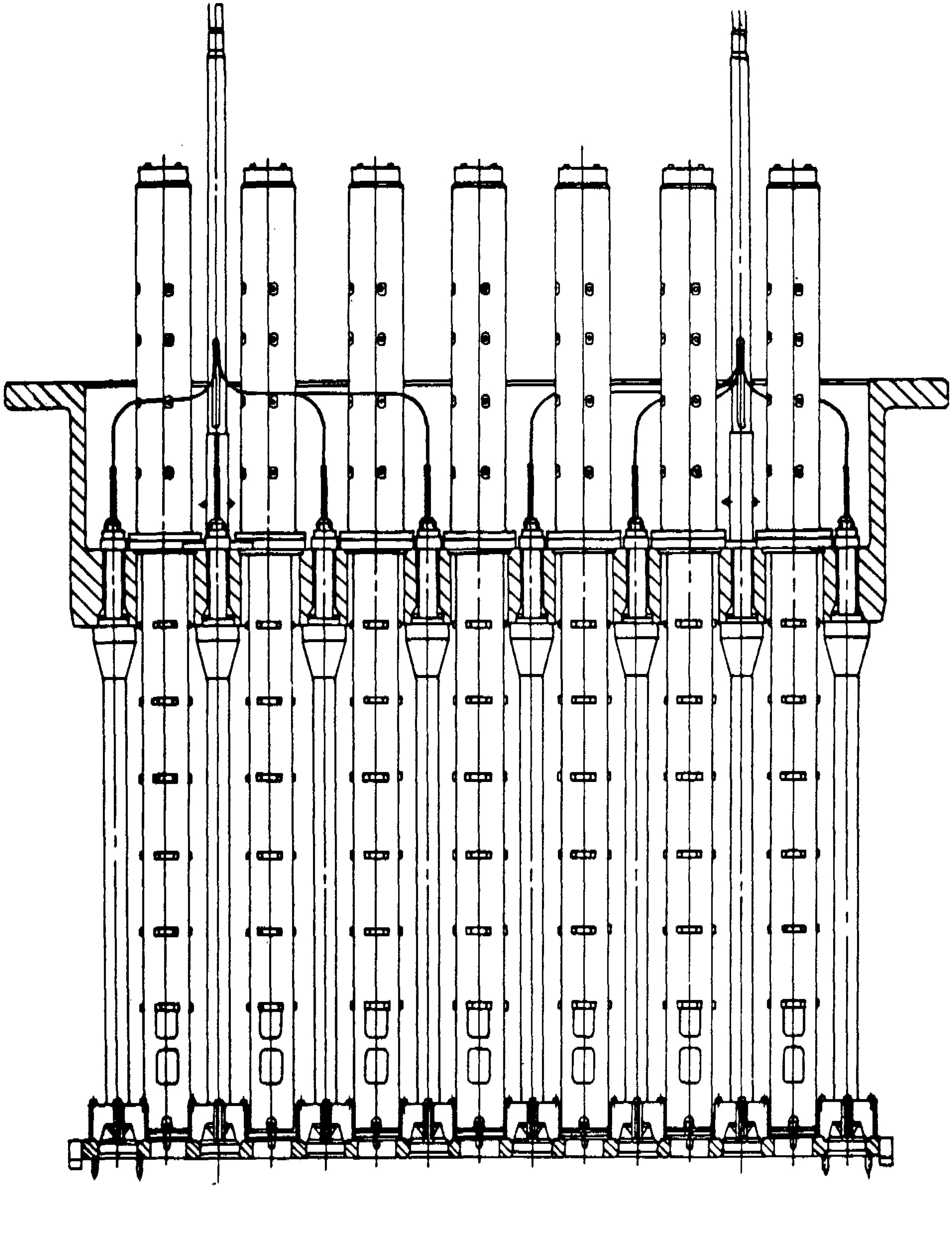


FIGURE 3.9N-8
UPPER CORE SUPPORT STRUCTURE
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

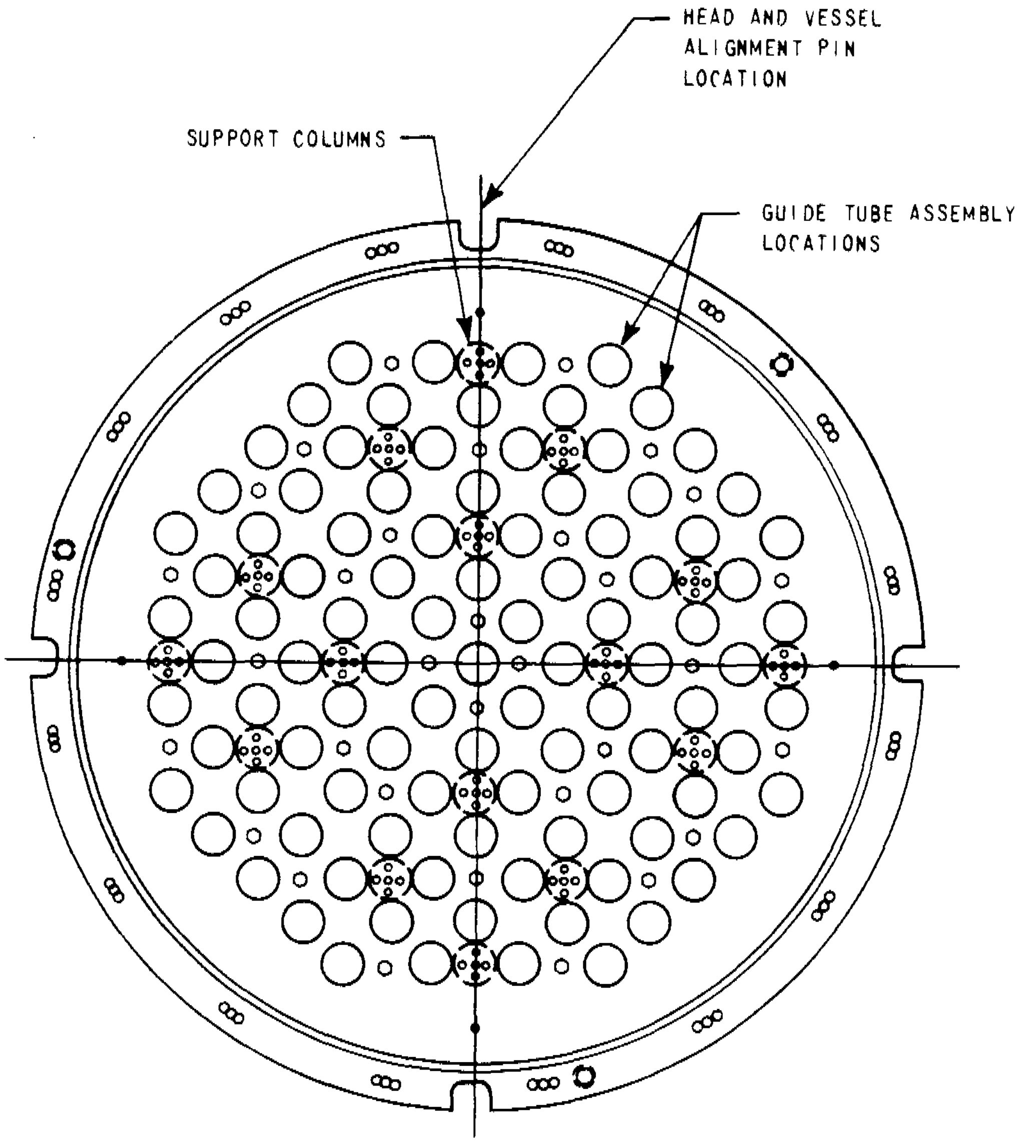


FIGURE 3.9N-9
PLAN VIEW OF UPPER CORE SUPPORT STRUCTURE
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

3.10 SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

Sections whose identification numbers include the letter B contain material within balance-of-plant (BOP) scope, while sections whose identification numbers include the letter N contain material within the nuclear steam supply system (NSSS) scope.

3.10B.1 Seismic Qualification Criteria

Seismic Category I instrumentation and electrical equipment are designed to maintain the capability to:

1. Initiate a protective action during the safe shutdown earthquake (SSE),

2. Withstand seismic disturbances during post-accident operation without loss of safety function.

There is no requirement that a SSE and Design Basis Accident (DBA) be considered simultaneously for accident analytical purposes. The above criteria apply to safety related SSC design considerations only.

Instrumentation and electrical equipment are seismically qualified in accordance with general instructions for earthquake requirements (Section 3.7B.3.1). These requirements conform with, or exceed, those outlined in IEEE Standard 344-1971, and are in agreement with the acceptance criteria in SRP 3.10, Rev. 1, 11-75 (NUREG-75-087). Although not required (due to Beaver Valley's docket date being before October 27, 1972), IEEE 344-1975 was employed for seismic qualification of Seismic Category I electrical equipment when feasible. Instrumentation and electrical equipment may be tested as individual components, as part of a simulated structural section, or as part of a completely assembled module or unit.

3.10B.2 Methods and Procedures for Qualifying Electrical Equipment and Instrumentation

The response of racks, panels, cabinets, and consoles is considered in assessing the seismic capability of instrumentation and electrical equipment. As a minimum, mounted equipment is qualified to acceleration levels consistent with those transmitted by supporting structures. A design objective is to minimize amplification of floor acceleration by supporting members to mounted equipment.

Determination of amplification and seismic adequacy of instrumentation and electrical equipment are implemented by the analysis and testing methods outlined in Section 3.7B.3.1.

3.10B.3 Methods and Procedures of Analysis or Testing of Supports of Electrical Equipment and Instrumentation

Supports for Seismic Category I electrical equipment, instrumentation, and control systems are seismically qualified by the analysis and testing procedures outlined in Section 3.7B.3.1. Supports are designed to withstand the combined effects of normal operating loads

acting simultaneously with horizontal and vertical components of earthquake loading and must retain their functional capability and structural integrity as applicable. When qualified by analysis, stress levels permitted under applicable codes. If there are no applicable codes, the stress level under the combined loading for an operating basis earthquake (OBE) does not exceed 75 percent of the minimum yield strength of the material in accordance with the ASTM specification.

A design objective is to provide supports for electrical equipment, instrumentation, and control systems that are seismically rigid (that is, with fundamental natural frequencies above the cutoff frequency of the relevant amplified response spectra curves). This ensures that amplification of floor accelerations through supporting members to mounted equipment is minimized.

The dynamic analysis method is typically used to establish support spacing for cable trays and conduit. Additionally, restraints are used as necessary to limit the horizontal lateral loads to allowable design values established on the basis of raceway loading and unsupported span lengths. Design provisions for significant differential motions between buildings are made by breaks in raceways if these relative displacements would result in unacceptable equipment or support loadings.

3.10B.4 Operating License Review

The results of testing and analyses that ensure proper implementation of the criteria in Section 3.10B.1 and demonstrate adequate seismic qualification of Seismic Category I instrumentation and electrical equipment were made available at the Seismic Qualification Review Team (SQRT) and Pump and Valve Operability Review Team (PVORT) site audit.

3.10N SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

This section presents information to demonstrate that instrumentation and electrical equipment classified as Seismic Category I is capable of performing designated safety-related functions in the event of an earthquake. The information presented includes identification of the Category I instrumentation and electrical equipment that are within the scope of the Westinghouse NSSS, the qualification criteria employed, and for each item of equipment; the designated safety-related functional requirements, definition of the applicable seismic environment and documentation of the qualification process employed to demonstrate the required seismic capability.

There is no requirement that a Safe Shutdown Earthquake (SSE) and Design Basis Accident (DBA) be considered simultaneously for accident analytical purposes. The above provisions apply to safety related SSC design considerations only.

3.10N.1 Seismic Qualification Criteria

3.10N.1.1 Qualification Standards

The U.S. Nuclear Regulatory Commission's (USNRC) recommendations concerning the methods to be employed for seismic qualification of electrical equipment endorse IEEE Standard 344-1971. Westinghouse Electric Corporation (Westinghouse) meets or exceeds the requirements

appropriate combination of these methods. Westinghouse meets this commitment either under the Westinghouse Supplemental Qualification Program (Westinghouse 1975) or as defined in the final version of WCAP 8587 (Butterworth and Miller 1979). Regulatory Guide 1.100 Seismic Qualification of Electrical Equipment for Nuclear Power Plants does not directly apply to Beaver Valley Power Station-Unit-2 (BVPS-2) due to its docket date being before July, 1974.

Morrone (1971) presents the Westinghouse testing procedure used to qualify equipment by type testing. Seismic qualification testing of equipment to IEEE Standard 344-1971 is documented according to Potochnik (1971), Reid (1972), and Vogeding (1971a, 1971b, 1971c, 1974). Fisher and Jarecki (1974) present the theory, practice, and justification for the use of single axis sine beats test inputs used in seismic qualification. In addition, it is noted that Westinghouse has conducted a seismic qualification Demonstration Test Program to confirm equipment operability during a seismic event (Westinghouse 1975).

For the seismic qualification of Westinghouse electrical equipment outside of the containment, the previously noted demonstration test program, in conjunction with the justification for the use of single-axis sine-beat tests (Figenbaum and Vogeding 1974) and the original tests documented by Vogeding (1971a, 1971b, 1971c, 1974), Potochnik (1971), Reid (1972); and Figenbaum and Vogeding (1974) meets or exceeds the requirements of IEEE Standard 344-1971.

Thus, since the Demonstration Test Program was successfully completed, the equipment's operability has been demonstrated to meet or exceed the requirements of IEEE Standard 344-1971.

The acceptability criteria for the SSE notes that there may be permanent deformation of the equipment, provided that the capability to perform its function is maintained.

The BOP instrumentation and electrical equipment which are designed to withstand the SSE horizontal and vertical accelerations at each floor level are discussed in Section 3.10B.

3.10N.1.2 Performance Requirements for Seismic Qualification

For NSSS instrumentation and electrical equipment classified as Seismic Category I and covered under the supplemental qualification program, qualification requirements can be found in Westinghouse (1975). WCAP 8587 (Westinghouse 1978) contains an Equipment Qualification Data Package (EQDP) for that NSSS instrumentation and electrical Seismic Category I equipment which has been upgraded to the criteria defined in WCAP-8587 (Butterworth and Miller 1979). Each EQDP in WCAP 8587 (Westinghouse 1978) contains a section entitled Performance Specifications. This specification establishes the safety-related functional requirements of the equipment to be

demonstrated during and after a seismic event. The required response spectrum (RRS) employed by Westinghouse for generic seismic qualification is also identified in the specification, as applicable. Complete results of qualification of safety-related electrical and mechanical equipment were made available at the NRC's Seismic Qualification Review Team (SQRT) and Pump and Valve Operability Review Team (PVORT) site audits.

3.10N.1.3 Acceptance Criteria

Seismic qualification must demonstrate that Category I instrumentation and electrical equipment is capable of performing designated safety-related functions during and after an earthquake of magnitude up to and including the OBE and SSE without the initiation of undesired spurious actuation which might result in consequences adverse to safety. The qualification will also demonstrate the structural integrity of mechanical supports and structures at the OBE level. Some permanent mechanical deformation of supports and structures is acceptable at the SSE level providing that the ability to perform the designated safety-related functions is not impaired.

3.10N.2 Methods and Procedures for Qualifying Electrical Equipment and Instrumentation

In order to meet or exceed the requirements of IEEE Standard 344-1971, seismic qualification of safety-related electrical equipment is demonstrated by either type testing, analysis, or a combination of these methods. The choice of qualification method employed by Westinghouse for a particular item of equipment is based upon many factors; including practicability, complexity of equipment, economics, availability of previous seismic qualification earlier standards, etc. The qualification method employed for a particular item of equipment is identified in the applicable qualification document or EQDP.

3.10N.2.1 Seismic Qualification by Type Test

From 1969 to mid-1974, Westinghouse seismic test procedures employed single axis sine beat inputs in accordance with IEEE Standard 344-1971 to seismically qualify equipment. The input form selected by Westinghouse was chosen following an investigation of building responses to seismic events as reported by Morrone (1971). In addition, Westinghouse has conducted seismic retesting of certain items of equipment as part of the Supplemental Qualification Program (Westinghouse 1975). This retesting was performed at the request of the USNRC staff on agreed selected items of equipment employing multi-frequency, multi-axis test inputs (Jarecki 1975) to demonstrate the conservatism of the original sine-beat test method with respect to the modified methods of testing for complex equipment recommended by IEEE Standard 344-1975, which is not required for BVPS-2, but where possible, was employed.

Where possible, testing utilizes multi-frequency, multi-axis, inputs, developed by the general procedures outlined by Jarecki (1975). The test results contained in the individual EQDPs Westinghouse (1978) demonstrate that the measured test response spectrum envelopes the applicable RRS defined for generic testing as specified in Section 1 of the EQDP (Westinghouse 1978). Qualification for plant specific use is established by verification that the generic RRS specified by Westinghouse envelops the applicable plant specific response spectrum. Alternative test methods, such as single frequency, single axis inputs, are used in selected cases.

3.10N.2.2 Seismic Qualification by Analysis

Employing motors as an example, the structural integrity of safety-related motors is demonstrated by a static seismic analysis meeting or exceeding the requirements of IEEE Standard 344-1971.

The analytical models employed and the results of the analysis are described in the applicable qualification document.

3.10N.3 Method and Procedures for Qualifying Supports of Electrical Equipment and Instrumentation

Where supports for the electrical equipment and instrumentation are within the Westinghouse NSSS scope of supply, the seismic qualification tests and/or analysis are conducted including the supplied supports. Where applicable, the appropriate qualification documents identify the equipment mounting employed for qualification purposes and establish interface requirements for the equipment to ensure that the subsequent in-plant installation does not prejudice the qualification established by Westinghouse.

3.10N.4 Operating License Review

The results of tests and analyses that ensure that the criteria established in Section 3.10N.1 have been satisfied, employing the qualification methods described in Sections 3.10N.2 and 3.10N.3, are included in the individual qualification documents or EQDPs.

3.10N.5 References for Section 3.10N

Butterworth, G. and Miller, R. B. 1979. Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety-Related Electrical Equipment. WCAP-8587.

Figenbaum, E. K. and Vogeding, E. L. 1974. Seismic Testing of Electrical and Control Equipment (Type DB Reactor Trip Switchgear) WCAP-7817; Supplement 6, August 1974.

Fischer, E. G. and Jarecki, S. J. 1974. Qualification of Westinghouse Seismic Testing Procedure for Electrical Equipment Tested Prior to May 1974. WCAP-8373.

Jarecki, S. J. 1975. General Method of Developing Multi-Frequency Biaxial Test Inputs for Bistables. WCAP-8634 Proprietary.

Morrone, A. 1971. Seismic Vibration Testing with Sine Beats. WCAP-7558.

Potochnik, L. M. 1971. Seismic Testing of Electric and Control Equipment (Low Seismic Plants). WCAP-7817, Supplement 2.

Reid, J. B. 1972. Seismic Testing of Electrical and Control Equipment (WCID NUCANA 7300 Series) (Low Seismic Plants) WCAP-7817, Supplement 4.

Vogeding, E. L. 1971. Seismic Testing of Electrical and Control Equipment 1970. WCAP-7397-L (Proprietary) January 1970 and WCAP-7817 (Non-Proprietary).

Vogeding, E. L. 1971. Seismic Testing of Electrical and Control Equipment (WCID Process Control Equipment). WCAP-7397-L, Supplement 1 (Propriety) January 1971 and WCAP-7871, Supplement 1 (Non-Proprietary). December 1971.

Vogeding, E. L. 1971. Seismic Testing of Electric and Control Equipment (Westinghouse Solid State Protection System) (Low Seismic Plants). WCAP-7817, Supplement 3.

Vogeding, E. L. 1974. Seismic Testing of Electrical and Control Equipment (Instrument Bus Distribution Panel) (Low Seismic Plants). WCAP-7817, Supplement 5.

Westinghouse 1975. Personal Communication Between C. Eicheldinger (Westinghouse) to D. B. Vassallo (USNRC), letter NS-CE-692, dated July 10, 1975.

Westinghouse 1978. EQDP - Equipment Qualification Data Packages. Supplement 1 to WCAP 8587.

3.11 ENVIRONMENTAL QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT

Information on the environmental qualification of Class lE equipment and safety-related mechanical equipment has been provided separately (refer to Table 1.7-3).

The Environmental Qualification Report, as referenced in Table 1.7-3, discusses the initial implementation of environmental qualification of electrical equipment. This document is now considered a historical report only, and is no longer being updated to reflect current EQ program requirements.

The current, ongoing program of environmental qualification for electrical equipment is in accordance with the provisions of 10CFR50.49 and is implemented at BVPS-2 through several administrative procedures.

Tables for Section 3.11

TABLE 3.11-1

ENVIRONMENTAL QUALIFICATION PARAMETERS FOR SAFETY-RELATED EQUIPMENT

Table 3.11-1 has been deleted and information on the environmental qualification of Class 1E equipment and safety-related mechanical equipment has been provided separately (refer to Table 1.7-3).

TABLE 3.11-2

PLANT ENVIRONMENTAL CONDITIONS

Table 3.11-2 has been deleted and information on the environmental qualification of Class 1E equipment and safety-related mechanical equipment has been provided separately (refer to Table 1.7-3).

APPENDIX 3A

COMPUTER PROGRAM FOR DYNAMIC AND STATIC ANALYSIS OF SEISMIC CATEGORY I STRUCTURES, EQUIPMENT, AND COMPONENTS

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APPENDIX 3A

COMPUTER PROGRAMS FOR DYNAMIC AND STATIC ANALYSIS
OF SEISMIC CATEGORY I STRUCTURES, EQUIPMENT
AND COMPONENTS, AND PIPING SYSTEMS

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LIST OF FIGURES (Cont)

LIST OF FIGURES (Cont)

APPENDIX 3A

COMPUTER PROGRAMS FOR DYNAMIC AND STATIC ANALYSIS OF SEISMIC CATEGORY I STRUCTURES, EQUIPMENT, AND COMPONENTS AND PIPING SYSTEMS

3A.1 STRUCTURES

The following computer programs are used in dynamic and static analysis of Seismic Category structures:

1. STRUDL II - Structural Analysis Program,
2. SHELL 1 - Shell Analysis,
3. STRUDL-SW - Structural Analysis Program,
4. ASAAS - Asymmetric Stress Analysis of Axisymmetric Solids,
5. TAC2D - Heat Transfer Program,
6. Time History Program - Dynamic Analysis,
7. PLAXLY - Finite Element Soil-Structure Analysis for Plain Strain Problems,
8. MAT - Axisymmetric Mat Analysis,
9. ANSYS - Engineering Analysis System,
10. MEMBRANE - Membrane Stresses Analysis,
11. SBMMI - Single Barrier Mass Missile Impact,
12. Baseplate Analysis Processor (BAP),
13. GTSTRUDL.

3A.1.1 STRUDL II

3A.1.1.1 General Description

The finite element method (Cheung and Ziankiewicz 1967) provides for the solution of a wide range of solid mechanics problems. Its implementation within the context of the STRUDL analysis facilities expands these for the treatment of plane stress, plane strain, plate bending, shallow shell, and three-dimensional stress analysis problems.

STRUDL II (Massachusetts Institute of Technology (MIT) 1968; 1971) has been designed as a modified subsystem of the Integrated Civil Engineering System (ICES), which was designed and formulated at the MIT Department of Civil Engineering.

STRUDL II also provides a dynamic analysis capability for linear elastic structures undergoing small displacements. Either free or forced vibrational response may be obtained and, in the latter case, the forcing functions may be in the form of time histories or response spectra.

Seismic Category I structures are analyzed for seismic effects using the dynamic analysis capability of STRUDL II. The analysis yields frequencies of vibrations, mode shapes, displacements, velocities, accelerations, and forces.

STRUDL II has been documented by benchmarking procedures against the GTSTRUDL computer code STRUDL-SW. GTSTRUDL is a recognized public domain program.

3A.1.2 SHELL 1

3A.1.2.1 General Description

This program is based upon the general numerical procedure proposed by B. Budiansky and P.P. Radkowski (1963) and Greenbaum (1963) to analyze a shell of revolution subjected to arbitrary loadings.

This is a finite difference stress analysis computer code. It can be used to determine the forces, moments, shears, displacements, rotations, and stress in a thin shell of revolution subject to arbitrary loads expanded in a Fourier series of up to 150 terms. Single layer shells with up to 30 simply connected branches may be analyzed. Poisson's ratio may change at discontinuity points, and Young’s modulus and the thermal coefficient of expansion may be different at each point. The allowed types of loading include elastic restraints, pressures in three orthogonal directions, temperature changes which may have a gradient through the shell thickness, and simplified input for weight of the shell or earthquake forces.

The equilibrium equations for a thin shell are based on Sanders linear theory (Sanders 1959). Sanders' equations are expanded and modified slightly to handle a broader range of problems. All pertinent load, stress, and deformation variables are expanded into a Fourier series. The individual Fourier components of stress and deflection are found separately by solution of the finite difference forms of the appropriate differential equations. The algorithm used to solve these equations is a minor modification of the Gaussian elimination method.

3A.1.2.2 Sample Problem - Thin-Wall Cylinder

A long thin-walled circular cylinder is subjected to a constant internal pressure distribution. A solution of this problem may be obtained (Roark 1965).

The pertinent parameters of the cylinder are presented in Table 3A.1.2-1.

The following solution can be verified (Roark 1965).

$$\delta_R = \frac{PR^2}{Et} \tag{3A.1.2-1}$$

$$\sigma_\theta = \frac{PR}{t} \tag{3A.1.2-2}$$

where:

σ_θ = hoop stress

δ_R = radial displacement

P = pressure

R = radius

t = thickness

E = Young's modulus

The cylinder is idealized by 100 elements, as shown on Figure 3A.1.2-1. Computer results are presented in Table 3A.1.2-2 along with the results obtained from Equations 3A.1.2-1 and 3A.1.2-2. As can be seen, the computer results compare favorably; therefore, this problem demonstrates the accuracy of SHELL 1.

3A.1.3 STRUDL-SW

3A.1.3.1 General Description

The STRUDL-SW computer code uses the stiffness analysis method to analyze a wide range of structural problems. It handles two- and three-dimensional trusses and frames, having linear elastic members and statically applied loading.

STRUDL-SW has been documented by bench marking procedures against the GTSTRUDL computer code. GTSTRUDL is a recognized program in the public domain.

3A.1.4 ASAAS

3A.1.4.1 General Description

ASAAS is a finite element computer code (Crose 1971). It can be used to determine stresses and displacements in arbitrary axisymmetric solids, including problems involving asymmetric mechanical and thermal loads and asymmetric temperature-dependent mechanical properties. All dependent variables, including the mechanical properties, are input by Fourier series expansions of the

circumferential coordinate. The mechanical loads can be surface pressures, surface shears, and nodal point forces.

The explicit stiffness relations for the axisymmetric solid ring elements of the triangular cross section are based on the classical theorem of potential energy and the assumption that, within any element, the displacement variation in the R-Z plane is linear. All dependent variables, including the material properties, are expanded into the Fourier series. The harmonics are coupled and all the equilibrium equations are solved simultaneously. The algorithm used to solve the equations is a block modified square root Cholesky method with iterative refinement (Crose 1971).

3A.1.4.2 Sample Problem - Harmonic Axisymmetric Plane Strain

An infinitely long, solid, circular cylinder is subjected to $P_o\cos\theta$ and $P_o\cos\alpha\theta$ pressure distributions plus a $P_o\sin\theta$ shear distribution. A closed-form solution of this problem may be obtained (Love 1944).

The pertinent parameters of the cylinder are presented in Table 3A.1.4-1.

The following solution can be verified (Love 1944):

$$\sigma_r = P_o\left(\frac{r}{a}\cos\theta + \cos 2\theta\right) \tag{3A.1.4-1}$$

$$\sigma_\theta = P_o\left[\frac{3r}{a}\cos\theta + \frac{2r^2 - a^2}{a^2}\cos 2\theta\right] \tag{3A.1.4-2}$$

$$\sigma_{r\theta} = P_o\left[\frac{r}{a}\sin\theta - \frac{r^2 - a^2}{a^2}\sin 2\theta\right] \tag{3A.1.4-3}$$

$$U_r = U_{r_1} + U_{r_2} \qquad (3A.1.4\text{-}4)$$

$$U_{r_1} = \frac{P_o(1-4\nu)(1+\nu)r^2}{2Ea}\cos\theta$$

$$U_{r_2} = P_o\left(\frac{1+\nu}{E}\right)\left(r - \frac{2\nu r^3}{3a^2}\right)\cos 2\theta$$

$$U_\theta = U_{\theta_1} + U_{\theta_2} \qquad (3A.1.4\text{-}5)$$

$$U_{\theta_1} = \frac{P_o(5-4\nu)(1+\nu)r^2\sin\theta}{2Ea}$$

$$U_{\theta_2} = P_o\left(\frac{1+\nu}{E}\right)\left[\left(1-\frac{2\nu}{3}\right)\frac{r^3}{a^2} - r\right]\sin 2\theta$$

where:

σ_r = radial stress

σ_θ = circumferential stress

$\sigma_{r\theta}$ = shear stress

U_r = radial deformation

U_θ = circumferential deformation

ν = Poisson's ratio

a = outer radius of solid cylinder

E = Young's Modulus

r = radius

The cylinder is idealized by 16 elements, as shown on Figure 3A.1.4-1. Computer results are depicted on Figure 3A.1.4-2, along with the exact results obtained from Equations 3A.1.4-4 and 3A.1.4-5. As can be seen, the computer results are very close to the exact results. Therefore, this problem verifies the accuracy of ASAAS for mechanical loading problems where material properties are not variable.

3A.1.5 TAC2D

3A.1.5.1 General Description

This is a finite difference computer code (Peterson 1969) which can be used to determine steady-state and transient temperatures in two- dimensional problems. The configuration of the body to be analyzed is described in the rectangular, cylindrical, or circular (polar) coordinate system by orthogonal lines of constant coordinate called grid lines. These grid lines specify an array of nodal elements. Nodal points are defined as lying midway between the bounding grid lines of these elements. A finite difference equation is formulated for each nodal point in terms of its capacitance, heat generation, and heat flow paths to neighboring nodal points. The equations for all the nodal points are assembled and solved using an implicit alternating gradient algorithm.

3A.1.5.2 Sample Problem

A sample problem is presented to compare the results from TAC2D with an analytical solution. The objective is to show that the TAC2D program yields the correct solution.

The problem is to determine the transient temperature distribution in a right circular cylinder which is initially at temperature T_1. At time, t = 0, the temperature at the surface is instantaneously changed to T_2 and maintained at that value.

Mathematically, the problem is defined by the following equations:

$$\frac{1}{r}\left(\frac{d}{dr}\left(\frac{rdT}{dr}\right)\right)+\frac{d^2T}{dZ^2}=\frac{1}{\alpha}\left(\frac{dT}{dt}\right) \qquad \text{(3A.1.5-0)}$$

$$0 \le \mathrm{r} \le \mathrm{R} \qquad \text{(3A.1.5-1)}$$

$$\mathrm{T}(\mathrm{r,z}) = \mathrm{T}_1 \text{ at } \mathrm{t} \le 0 \qquad \text{(3A.1.5-2)}$$

$$\mathrm{T}(\mathrm{R,z}) = \mathrm{T}_2 \text{ at } \mathrm{t} \ge 0 \qquad \text{(3A.1.5-3)}$$

$$T\left(r, \pm \frac{L}{2}\right) = T_2 \;\; at \;\; t \ge 0 \qquad \text{(3A.1.5-4)}$$

where:

t = the time,

r = the radius,

z = the axial coordinate,

R = the outside radius of the cylinder,

L = the length of the cylinder, and

α = the diffusivity.

Further,

$$\alpha = \frac{k}{\rho c} \qquad (3A.1.5\text{-}5)$$

where :

k = the thermal conductivity,

ρ = the density,

c = the specific heat capacity,

For the specific problem analyzed, the following numerical values were used:

R = 12.0 inches
L = 48.0 inches
k = 20.0 Btu/hr-ft-°F
ρ = 40.0 Btu/cu ft-°F
T_1 = 0.0 °F
T_2 = 1,000.0°F

3A.1.5.2.1 Analytical Solution

It may be shown in Carlslaw and Jaeger (1959) that the solution is:

$$\frac{T - T_1}{T_2 - T_1} = 1 - f(z,t)\, g(r,t) \qquad (3A.1.5\text{-}6)$$

$$f(z,t) = \frac{4}{\pi} \sum_{n=0}^{\infty} \frac{(-1)^n}{(2n+1)} e^{-\alpha t \left[\frac{\pi}{2L}(2n+1)\right]^2} \cos\left[\frac{\pi z}{2L}(2n+1)\right] \qquad (3A.1.5\text{-}7)$$

$$g(r,t) = \frac{2}{R} \sum_{m=1}^{\infty} \frac{J_o(r\psi m)}{\psi_m J_1(R\psi_m)} e^{-\alpha\psi_m^2 t} \qquad (3A.1.5\text{-}8)$$

where the Ψm are the roots of

$$J_O\ (R\Psi) = 0 \qquad (3A.1.5\text{-}9)$$

The roots Ψm of Equation 3A.1.5-9 and the Bessel functions (J_O) and (J_1) are tabulated by Jahnke and Emde (1945) and need not be computed.

From the definition of the problem there is symmetry about the geometric center of the cylinder and the origin of the coordinate system taken at that point, as is reflected in the boundary conditions, Equations 3A.1.5-3 and 3A.1.5-4.

3A.1.5.2.2 Numerical Solution With TAC2D

A cross section of the problem model for TAC2D is shown on Figure 3A.1.5-1. The model extends only to the axial midplane of the cylinder where an adiabatic boundary may be specified by virtue of the symmetry condition described previously. The solid material is represented by one material block. The boundary conditions on the four external boundaries are described by Coolants 1 through 4 (specifically, Coolant Blocks 1 through 4). The material and coolant thermal parameters, as specified by the input functions, are given in Table 3A.1.5-1. All coolants have the standard specific heat of 1.0 Btu per pound-°F (Btu/lb-°F). Coolants 1 and 2, which represent the adiabatic external boundaries, have the standard heat transfer coefficient of 10^{-6} Btu/hr-sq ft-°F and the standard flow rate of 10^{6} pounds per hour.

3A.1.5.2.3 Comparison of TAC2D Solution with the Analytical Solution

A comparison of the output from the code with the series solution is shown on Figure 3A.1.5-2. The temperature-versus-time function is plotted at three representative points within the cylinder. It can be seen that the results from TAC2D are almost identical to the

series solution results. The maximum difference between the two sets of results is about 2°F out of a mean magnitude of 100°F.

3A.1.6 Time History Program

3A.1.6.1 General Description

The Time History Program computes time history response and amplified response spectra (ARS) at any mass location of a lumped mass system due to a synthetic earthquake input. The responses are computed by integration of the modal equations of the system by exact methods (Nigam and Jennings 1968). The program's main application is the generation of ARS used in the design of Seismic Category I equipment and piping.

3A.1.6.2 Sample Problem

The Time History Program's solution to a test problem is essentially identical to the solution obtained using STARDYNE. STARDYNE is a recognized program in the public domain. The sample problem used consists of a structure subjected to an earthquake time history record. The structure is idealized by five lumped masses interconnected by five elastic beam elements, as shown on Figure 3A.1.6-1.

Peak acceleration and displacement, as well as the horizontal ARS at the top mass point, are compared in Tables 3A.1.6-1 and 3A.1.6-2, respectively.

3A.1.7 PLAXLY

3A.1.7.1 General Description

The PLAXLY program provides a numerical solution for the dynamic analysis of plane systems under general dynamic loadings. This program works with a two-dimensional plane-strain finite element idealization of the soil structure interaction problem.

The original version of PLAXLY was developed at the University of California in Berkeley (Waas 1972). It was later modified and extended at Stone & Webster Engineering Corporation (SWEC) to incorporate transient seismic excitations, nonlinear soil behavior, and lumped mass representations of the structures.

3A.1.7.2 Sample Problem

The PLAXLY program's solution to a test problem is essentially identical to the solution obtained by using the FLUSH program. FLUSH is a recognized program in the public domain.

The sample problem used consists of a structure represented by five lumped masses interconnected by four elastic beam elements. This

structural model is connected to a finite element representation of the soil, as shown on Figure 3A.1.7-1. The horizontal amplified response spectra (ARS) at the top mass point are compared on Figure 3A.1.7-2.

3A.1.8 MAT6

3A.1.8.1 General Description

This program is based upon the general numerical procedures proposed by Zhemoshkin (1962) to analyze a circular plate on an elastic foundation. It is used to determine moments, shears, vertical deflections, radial displacements, tangential and radial in-plane forces, plus rotations of the circular plate subjected to axisymmetric loadings.

The soil subgrade may be modeled as either a Winkler (1867) or a Boussinesqu (1885) type elastic foundation.

3A.1.8.1 Sample Problem

As a sample problem, the case of a circular plate with two axisymmetric loadings will be considered. Figure 3A.1.8-1 presents the pertinent parameters of the problem and a cross-sectional sketch of the plate.

The results from a hand calculation using the methods of Ulickii (1972) are compared to those from MAT6 in Tables 3A.1.8-1 and 3A.1.8-2.

3A.1.9 ANSYS

3A.1.9.1 General Description

ANSYS is a general purpose finite element analysis program with structural and heat transfer capabilities. ANSYS is used as-needed for engineering analysis. ANSYS is a recognized program in the public domain.

3A.1.10 MEMBRANE

3A.1.10.1 General Description

This program computes membrane stresses and strains in containment structures due to dead loads, internal pressure, and temperature

gradients across the wall. It analyzes cylinders, cones, and spherical domes which consist of a fully cracked reinforced concrete section with a steel liner.

Stresses are computed by shell membrane theory (Billington 1965). The program automatically considers the effect of the uplift force acting on the roof of a cone or cylinder.

3A.1.10.2 Sample Problem

As an example, a cylindrical shell was analyzed by the program and by a hand calculation. Table 3A.1.10-1 presents the pertinent parameters of the cylinder. The comparison of results is given in Table 3A.1.10-2.

As can be seen, the computer program's results compare very favorably. This problem verifies the accuracy of MEMBRANE.

3A.1.11 SBMMI

3A.1.11.1 General Description

This program computes the elasto-plastic structural response of a barrier due to the following types of loads: (a) static loads; (b) suddenly applied constant dynamic loads which remain permanently on the structure; (c) suddenly applied constant dynamic loads representing missile impact with a finite force and specific momentum; and (d) suddenly applied dynamic load of zero time duration and specific momentum representing missile impact. The barrier is modelled as a single barrier mass and a non-linear spring, with the above loads applied. The equation of motion is integrated in time assuming constant acceleration in each time step (Biggs 1964).

3A.1.11.2 Sample Problem

As an example, the program SBMMI was run for each load case separately. The computer's results were compared to those obtained from a hand calculation. The hand calculation was based on the elasto-plastic response charts found in Biggs (1964). Table 3A.1.11-1 presents the pertinent model and load parameters. The hand and computer program results are compared in Table 3A.1.11-2. It can be seen that the results compare very favorably. This problem verifies the accuracy of the SBMMI computer program.

3A.1.12 Baseplate Analysis Processor (BAP)

3A.1.12.1 General Description

The computer program BAP is a preprocessor/postprocessor which works in conjunction with the program ANSYS. The purpose of BAP is to generate the ANSYS input necessary for the static, non-linear

analysis of baseplates subjected to out-of-plane loads, to distribute in-plane loads to the anchor bolts assuming an infinitely rigid baseplate, and to postprocess the ANSYS results into a report style format.

BAP has been documented by bench marking procedures against the Baseplate Investigation Processors computer code, which is a recognized program in the public domain.

3A.1.13 GTSTRUDL

3A.1.13.1 General Description

The computer program GTSTRUDL provides for the solution of a wide range of solid mechanics problems. It is capable of analyzing for various static and dynamic loadings

GTSTRUDL is a recognized program in the public domain.

3A.1.14 References for Appendix 3A.1

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Tables for Section 3A.1

TABLE 3A.1.2-1

THIN-WALL CYLINDER,
PERTINENT PARAMETERS

Dimensions and Properties	Loading and Boundary Conditions
R = 25 inches	At z = 0 inch; Fr = M = δ_z = 0
ℓ = 200 inches	
t = 0.5 inch	At z = ℓ = 200 inches; Fr = M = Fz = 0
E = 28 x 10^6 psi	
Poisson's ratio = 0.3	P = 75 psi
M = Moment on free edge	
Fr = Radial force	
F_z = Force in Z-direction	
δ_z = Displacement in Z-direction	

TABLE 3A.1.2-2

EXACT AND COMPUTER STRESSES FOR THIN-WALL CYLINDER

Variable	Exact	Shell 1
δ_R (inch)	3.348×10^{-3}	3.348×10^{-3}
σ_θ (psi)	3,750	3,750

TABLE 3A.1.4-1

INFINITELY LONG SOLID CYLINDER, PERTINENT PARAMETERS

Dimensions and Properties	Loading and Boundary Conditions
$a = b$	$P_r = P_o (\cos\theta + \cos 2\theta)$
$\ell = b$	$\tau_{r\theta} = P_0 \sin\theta$
$E = 10 \times 10^6$ psi	$Uz = 0$
$\nu = 0.25$	At $r = 0$, $U_r = 0$
$b = 1$ inch	$P_o = 10{,}000$ psi

TABLE 3A.1.5-1

INPUT THERMAL PARAMETER FUNCTIONS FOR TAC2D SAMPLE PROBLEM

Material Thermal Parameters	Coolant Thermal Parameters
SPEC1 (X) = 40.0 Btu/lb-°R	H3A (X) = 1.0×10^8 Btu/hr-ft^2-°R
RCON1 (X) = 20.0 Btu/hr-ft-°R	FL03A (X) = 1.0×10^8 lb/hr
ACON1 (X) = 20.0 Btu/hr-ft-°R	TIN3A (X) = 1,460°R
	H4A (X) = 1.0×10^8 Btu/hr-ft^2-°R
	FLO4A (x) = 1.0×10^8 lb/hr
	TIN4A (x) = 1,460°R

Explanation of terms:

SPEC1(X)	=	specific heat capacity of material 1
RCON1(X)	=	thermal conductivity in the X (or radial) direction of material 1
ACON1(X)	=	thermal conductivity in the Y (or theta, or axial direction) of material 1
H3A(x)	=	heat transfer coefficient for coolant 3
FL03A(x)	=	mass flow rate of coolant 3
TIN3A(x)	=	inlet temperature of coolant 3
HYA(X)	=	heat transfer coefficient for coolant 4
FLO4A(X)	=	mass flow rate of coolant 4
TIN4A(X)	=	inlet temperature of coolant 4

TABLE 3A.1.6-1

PEAK ACCELERATION AND DISPLACEMENT

	Time History Program	Stardyne Program
Peak Acceleration	0.922 g	0.922 g
Peak Displacement	0.352 in.	0.352 in.

TABLE 3A.1.6-2

HORIZONTAL AMPLIFIED RESPONSE SPECTRA
(TWO-PERCENT OSCILLATOR DAMPING)

Period (Seconds)	Time History Program (g)	Stardyne Program (g)
0.02	0.929	0.928
0.04	0.999	0.999
0.06	1.040	1.039
0.08	1.301	1.301
0.10	1.253	1.252
0.12	1.694	1.697
0.14	2.419	2.419
0.16	3.158	3.153
0.18	5.962	5.961
0.20	11.104	11.101
0.22	6.777	6.802
0.24	4.964	5.025
0.26	3.441	3.451
0.28	2.576	2.576
0.30	2.417	2.428
0.34	1.613	1.603
0.38	1.720	1.731
0.42	1.493	1.491
0.46	1.487	1.507
0.50	1.201	1.201
0.70	0.653	0.660

TABLE 3A.1.8-1

FORCE QUANTITIES

Station (ai)	Radius (r, ft)	Radial Moment Mr (ft-K/ft)		Tangential Moment Mt (ft-K/ft)		Shear Force Qr (U/ft)	
		Hand	MAT6	Hand	MAT6	Hand	MAT6
0.0	0.0	-864.47	-864.81	-864.47	-864.81	0.0	0.0
1.0	5.0	-864.47	-864.81	-864.47	-864.81	0.0	0.0
2.0	10.0	-864.47	-864.81	-864.47	-864.81	0.0	0.0
3.0	15.0	-864.47	-864.81	-864.47	-864.81	0.0	0.0
4.0	20.0	-864.47	-864.81	-864.47	-864.81	0.0	0.0
5.0	25.0	-864.47	-864.81	-864.47	-864.81	0.0	0.0
6.0	30.0	-864.47	-864.81	-864.47	-864.81	0.0	0.0
7.0	35.0	-864.47	-864.81	-864.47	-864.81	0.0	0.0
8.0	40.0	-864.47	-864.81	-864.47	-864.81	0.0	0.0
9.0	45.0	-864.47	-864.81	-864.47	-864.81	0.0	0.0
10.0	50.0	-864.47	-864.81	-864.47	-864.81	0.0	0.0
10.5	52.5					-30.32	-30.32
11.0	55.0	-718.41	-718.79	-833.50	-833.82	-28.94	
11.5	57.5					-27.68	-27.68
12.0	60.0	-593.84	-594.23	-796.47	-796.77	-26.63	
12.5	62.5					-25.46	-25.46
13.0	65.0	-485.44	-485.85	-756.20	-756.40	-24.49	
13.5	67.5					-23.58	-23.58
14.0	70.0	-389.60	-390.01	-714.40	-714.68	-22.74	
14.5	72.5					-21.96	-21.95
15.0	75.0	-303.72	-304.14	-672.13	-672.40	-21.22	
15.5	77.5					-20.54	-20.54
16.0	80.0	-225.94	-226.37	-630.04	-630.31	-19.89	
16.5	82.5					-19.29	-19.29
17.0	85.0	-154.85	-155.29	-588.53	-588.79	-18.72	
17.5	87.5					-18.19	-18.19
18.0	90.0	-89.37	-89.82	-547.85	-548.10	-17.63	
18.5	92.5					-17.21	-17.21
19.0	95.0	-28.68	-29.12	-508.13	-508.37	-16.75	
19.5	97.5	0.0	-0.35	-488.65	-488.84	-16.32	-16.32

TABLE 3A.1.8-2

DISPLACE QUANTITIES,
A COMPARISON OF RESULTS

Station (ai)	Radius (r, ft)	Relative Vertical Displ. U (ft)		Mat rotation 0 (rad)	
		Hand	MAT6	Hand	MAT6
0.0	0.0	0.0	0.0	0.0	0.0
1.0	5.0	-0.00	-0.00031	-0.0001	-0.0001
2.0	10.0	-0.00100	-0.00106	-0.0002	-0.0002
3.0	15.0	-0.00225	-0.00231	-0.0003	-0.0003
4.0	20.0	-0.00400	-0.00406	-0.0004	-0.0004
5.0	25.0	-0.00625	-0.00632	-0.0005	-0.0005
6.0	30.0	-0.00900	-0.00907	-0.0006	-0.0006
7.0	35.0	-0.01225	-0.01232	-0.0007	-0.0007
8.0	40.0	-0.01600	-0.01607	-0.0008	-0.0008
9.0	45.0	-0.02025	-0.02032	-0.0009	-0.0009
10.0	50.0	-0.02500	-0.02507	-0.0010	-0.0010
11.0	55.0	-0.03024	-0.03030	-0.0010911	-0.001045
12.0	60.0	-0.03588	-0.03593	-0.0011631	-0.0011628
13.0	65.0	-0.04185	-0.04188	-0.00122	-0.00122
14.0	70.0	-0.04806	-0.04809	-0.00126	-0.00126
15.0	75.0	-0.05446	-0.05448	-0.001300	-0.001300
16.0	80.0	-0.06100	-0.06101	-0.001317	-0.001317
17.0	85.0	-0.06719	-0.06720	-0.001329	-0.001329
18.0	90.0	-0.07428	-0.07427	-0.001333	-0.001333
19.0	95.0	-0.08094	-0.08092	-0.001329	-0.001329
19.5	97.5	-0.08426	-0.08424	-0.001324	-0.001324

TABLE 3A.1.10-1

PERTINENT PARAMETERS OF CYLINDRICAL SHELL
SAMPLE PROGRAM

Input Data	Parameter
Radius of outer layer of rebars	r_o = 66.5 ft
Radius of inner layer of rebars	r_i = 63.67 ft
Radius of liner	r_1 = 63 ft
Radius of reference surface	r = 65.25 ft
Height of cylinder	h = 122 ft
Meridional steel area/unit length, outer layer	A_o = 4 in^2
Meridional steel area/unit length, inner layer	A_i = 4 in^2
Liner area/unit length	A_1 = 4.5 in^2
Circumferential steel area, unit length, outer layer	A'_o = 8 in^2
Circumferential steel area, unit length, inner layer	A'_i = 8 in^2
Internal pressure	p = 9.72 ksi
Wall weight per unit surface	q = 0.68 ksi
Total load at top	w = 9,726.5 k
Temperature increment, outer rebars	ΔT_o = -12°F
Temperature increment, inner rebars	ΔT_i = 27°F
Temperature increment, liner	ΔT_1 = 230°F

TABLE 3A.1.10-2

SUMMARY OF RESULTS

Input Data	Hand Calculation	Computer Run	% Difference
Liner strain:			
Meridional	0.001308	0.001308	0.0
Circumferential	0.001412	0.001412	0.0
Membrane stresses: (ksi)			
Meridional rebars:			
Outer layer	41.58	41.59	0.02
Inner layer	33.975	33.99	0.02
Liner	-6.9857	-6.97	0.2
Hoop rebars:			
Outer layer	42.47	42.47	0.0
Inner layer	36.65	35.65	0.0
Liner	-4.586	-4.586	0.0
Membrane Forces: (k/ft)			
Meridional	271.76	271.90	0.05
Circumferential	612.32	612.36	0.0

TABLE 3A.1.11-1

TEST PROBLEM DATA

	Load Type	Load
1.	Static load	-15.k k*
2.	Suddenly applied constant load (remains on structure permanently)	62.9 k*
3.	Missile impact - finite force	264 k*
	specific momentum	1.4 k*/sec
4.	Suddenly applied dynamic load w/zero time duration (applied impulse)	1.2 k*/sec

NOTES:

*Equivalent barrier weight = 16.66 k.
Barrier resistance function - From zero displacement to a displacement of 0.003 foot, the barrier resistance increases linearly from 0 to 87.2 k. For displacements greater than 0.0003 foot, the resistance remains at a constant value of 87.2 k.

TABLE 3A.1.11-2

A COMPARISON OF HAND CALCULATION AND COMPUTER PROGRAM RESULTS

Load Number		Results from Hand Calculation	Results from Computer Run
1	Barrier deflection	0.000533 ft	0.0005 ft
2	Barrier deflection	0.0054 ft	0.0054 ft
	Time of maximum deflection	0.01802 sec	0.018378 sec
3	Barrier deflection	0.0183 ft	0.0187 ft
	Time of maximum deflection	0.0180 sec	0.018391 sec
4	Barrier deflection	0.0171 ft	0.0172 ft
	Time of maximum deflection	0.01481 sec	0.014419 sec

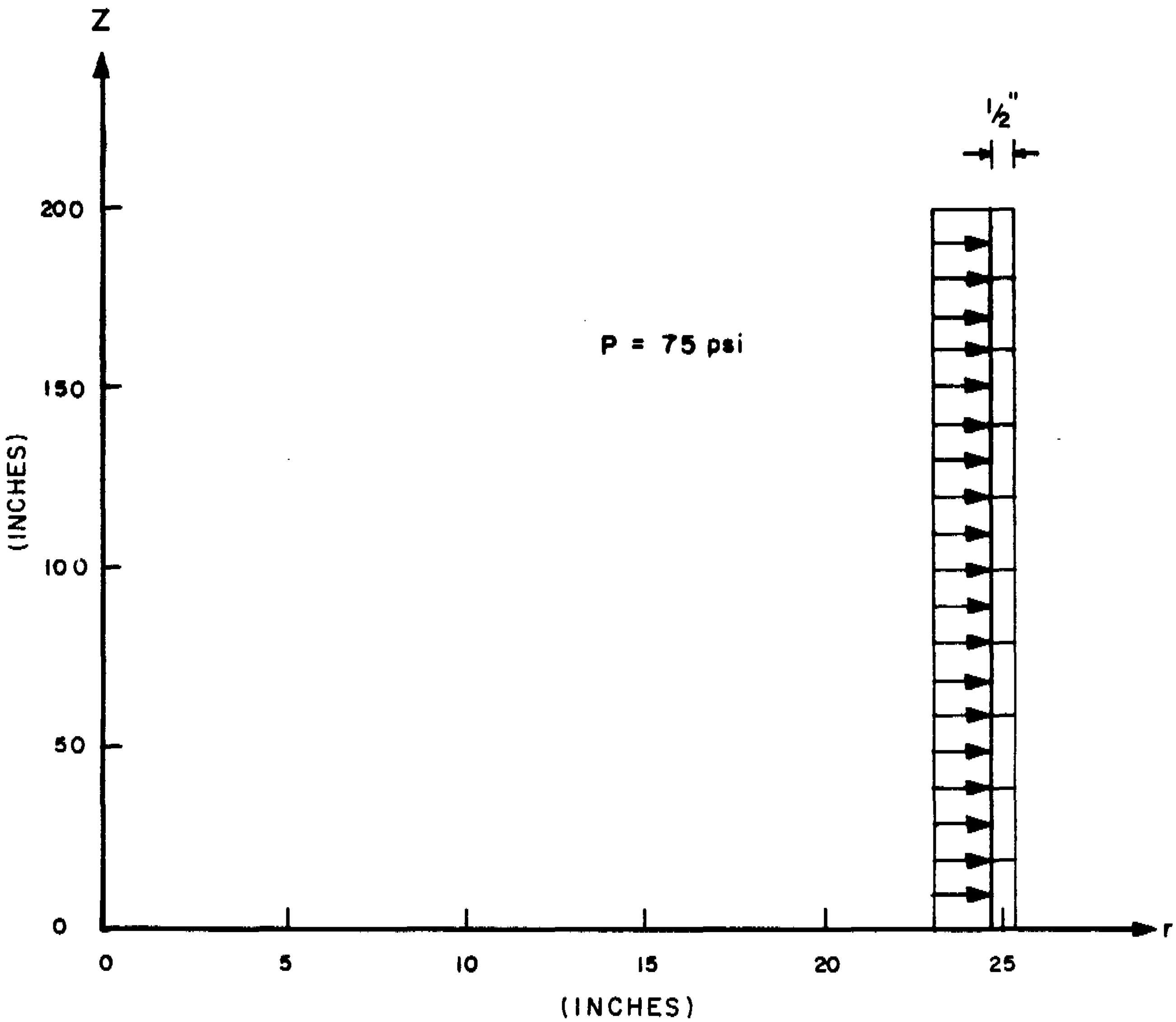


FIGURE 3A.1.2-1

ONE HUNDRED-ELEMENT IDEALIZATION OF THIN-WALL CYLINDER

BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

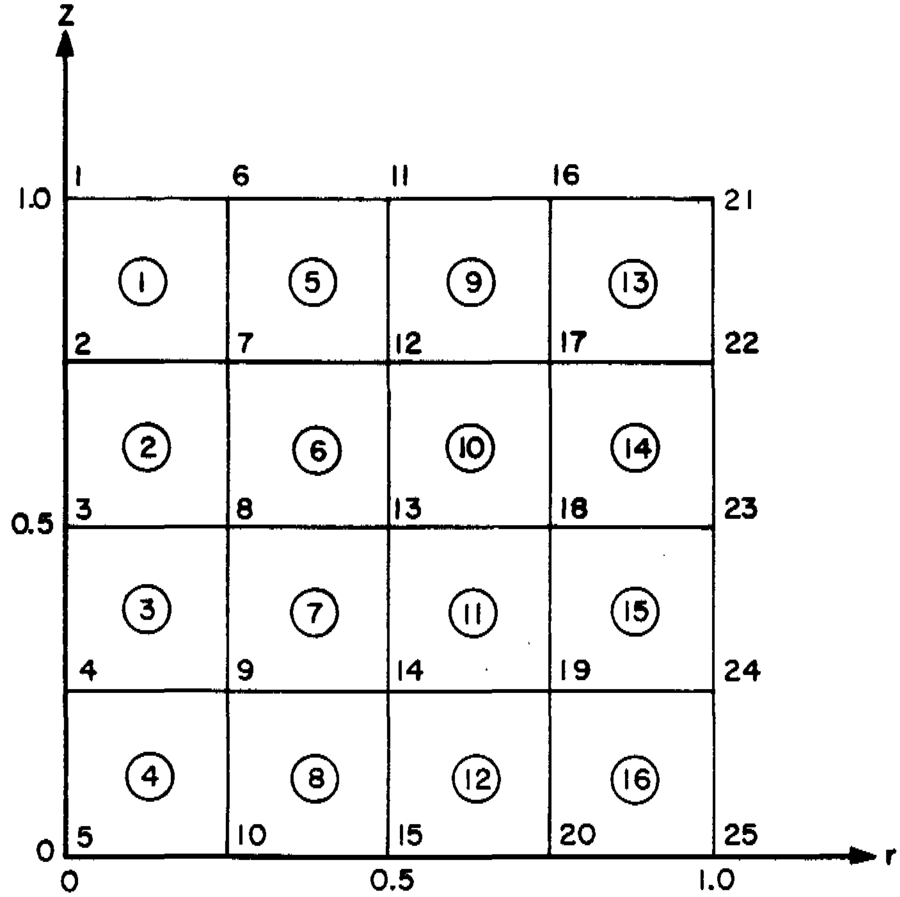


FIGURE 3A.1.4-1
ELEMENT PLOT
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

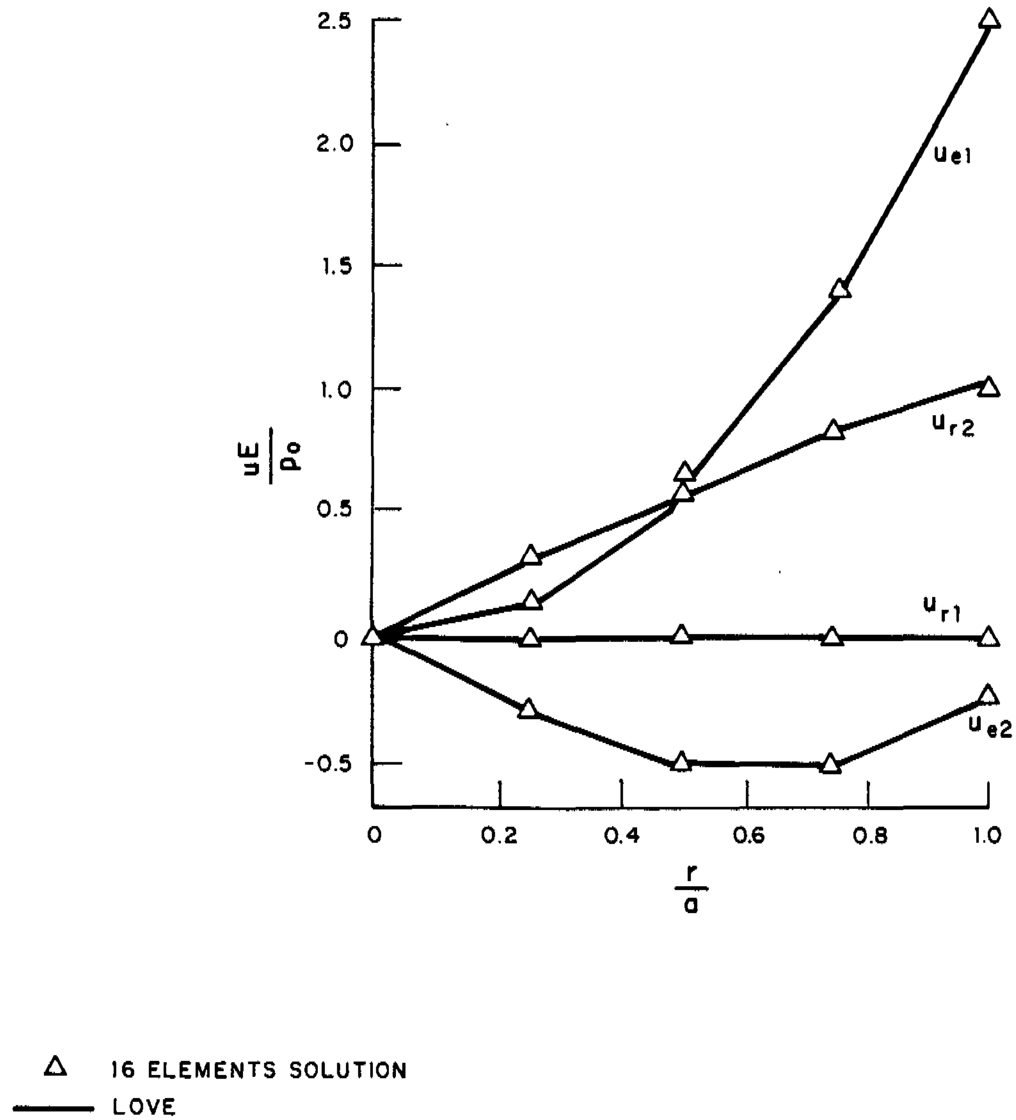


FIGURE 3A.1.4-2
HARMONIC AXISYMMETRIC
PLAN STRAIN
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

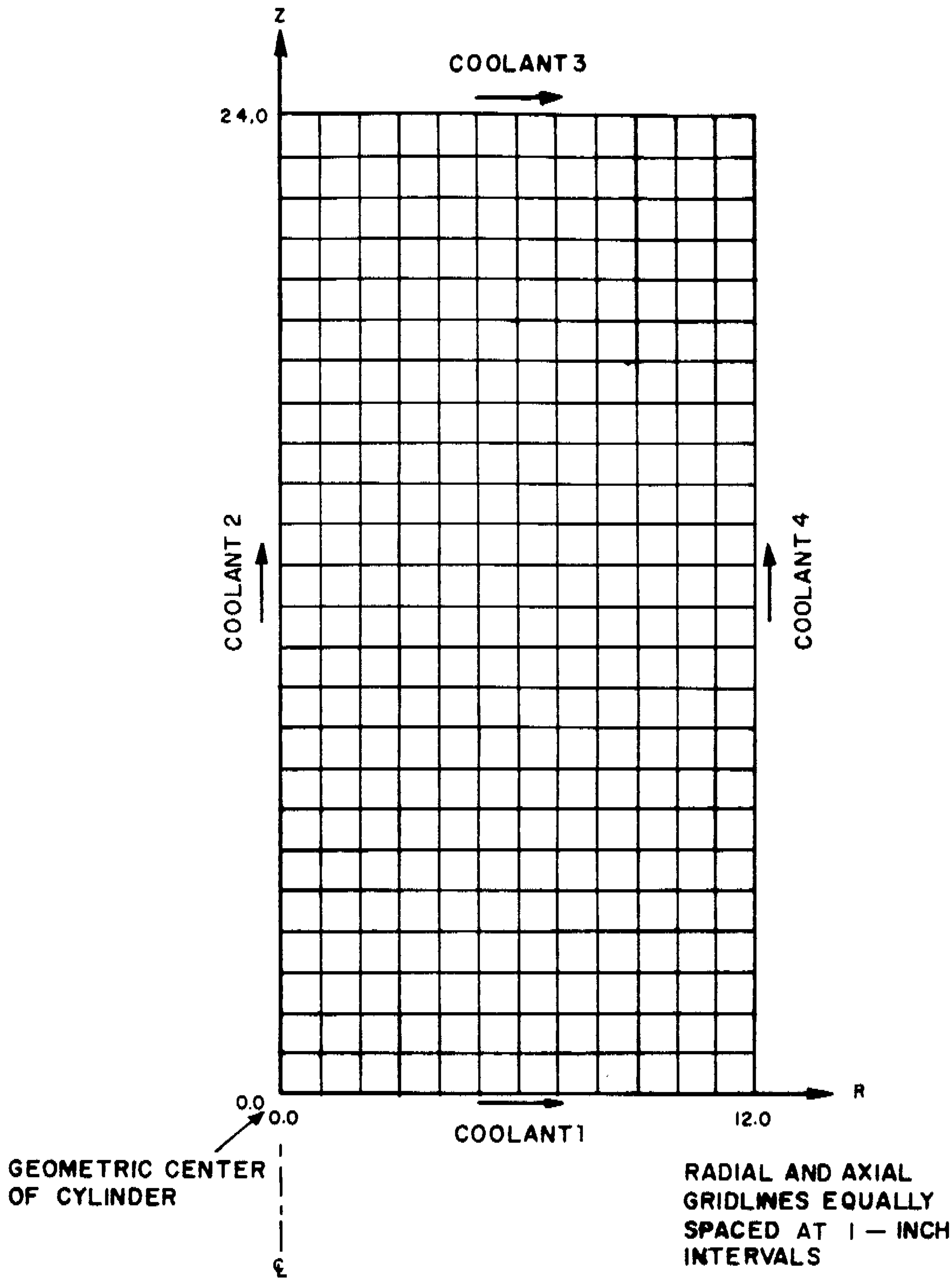


FIGURE 3A.1.5-1
TAC2D SAMPLE PROBLEM
THERMAL MODEL
BEAVER VALLEY POWER STATION- UNIT 2
FINAL SAFETY ANALYSIS REPORT

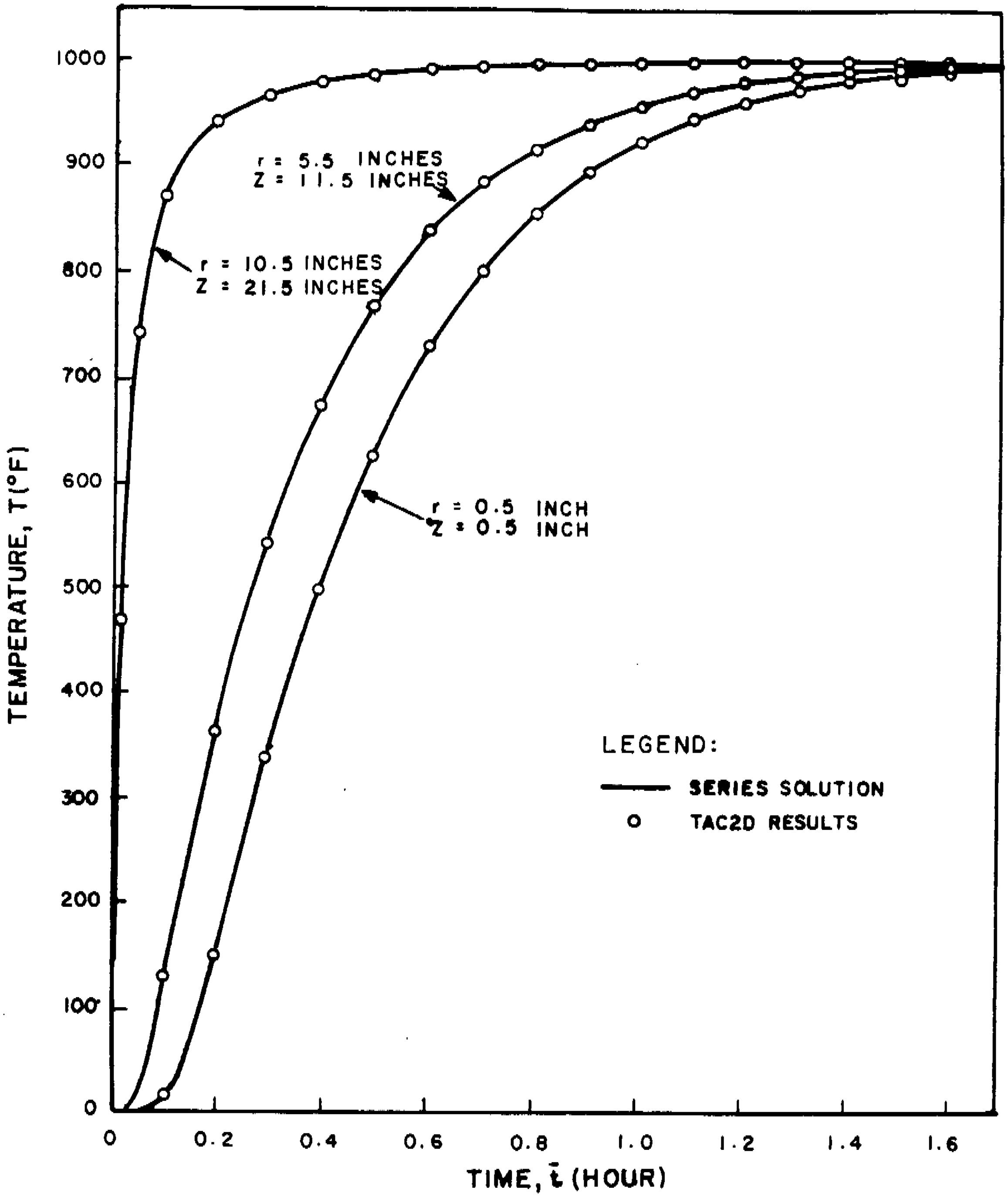


FIGURE 3A.1.5-2
TRANSIENT TEMPERATURES IN A RIGHT CIRCULAR CYLINDER-COMPARISON OF TAC 2D RESULTS WITH SERIES SOLUTION

BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

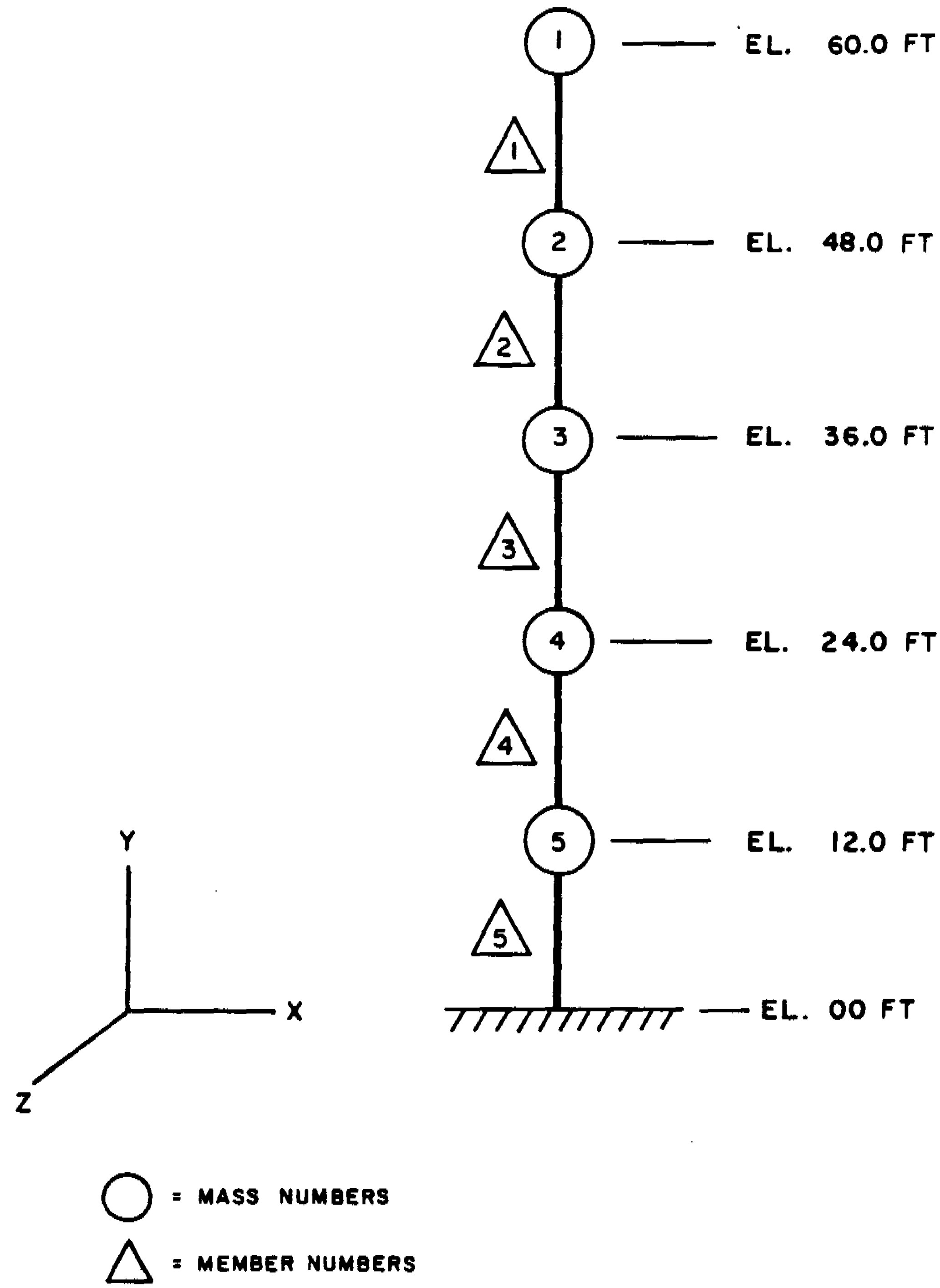
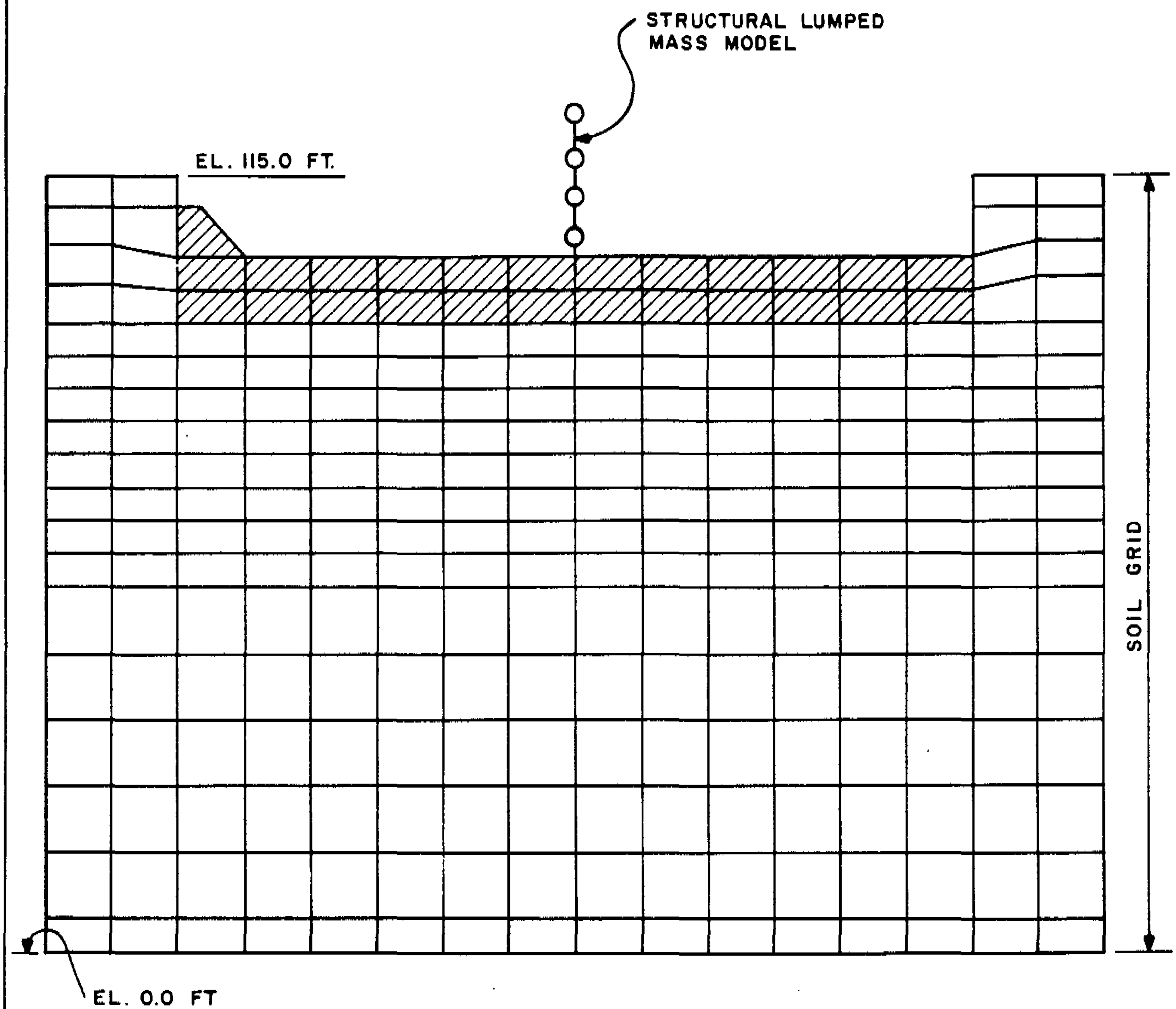


FIGURE 3A.1.6-1
STRUCTURAL MODEL
TIME HISTORY PROGRAM
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT



NOT TO SCALE

LEGEND

MAT

FIGURE 3A.1.7-1
FINITE ELEMENT GRID FOR
PLAXLY - FLUSH COMPARISON
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

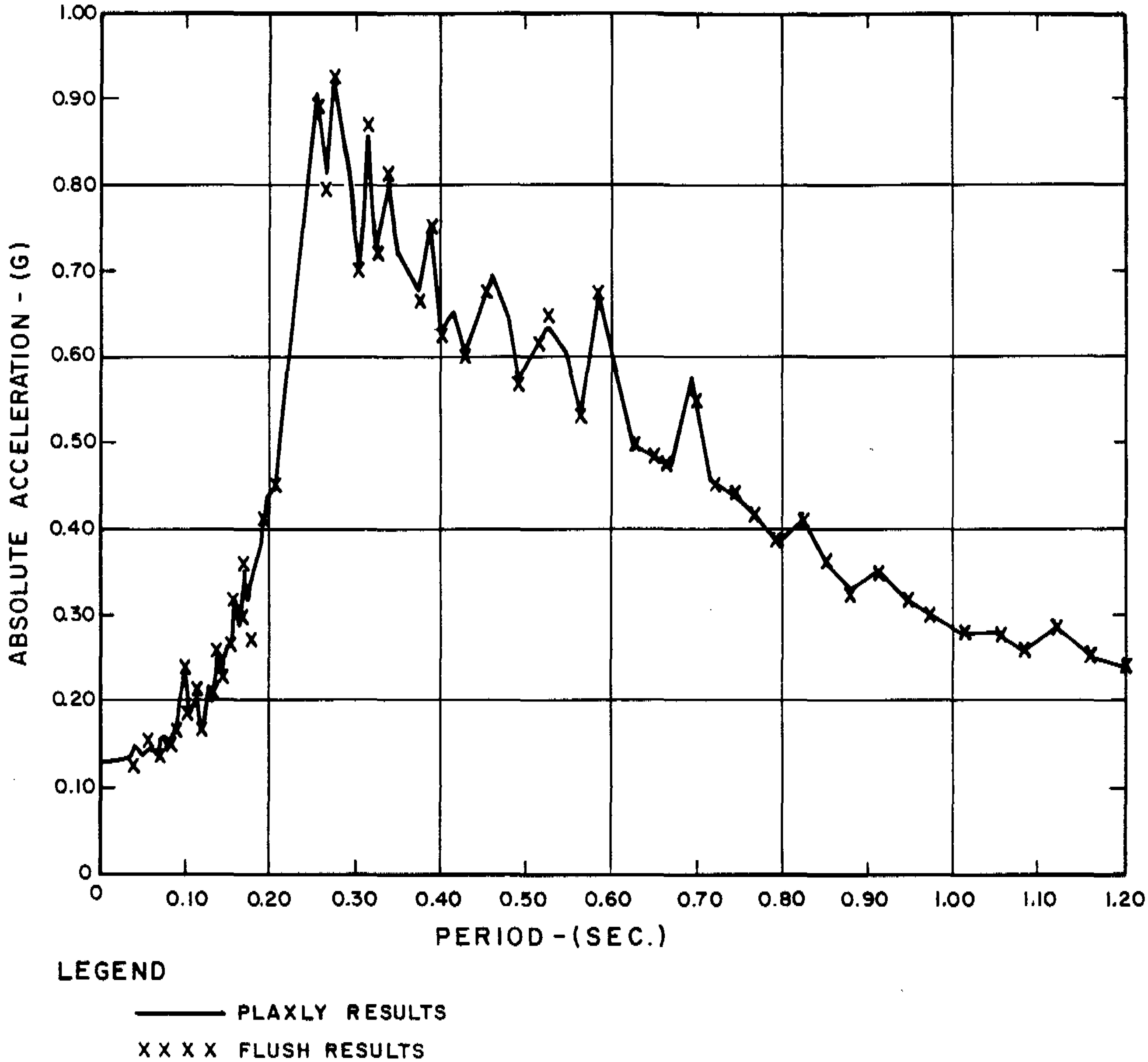
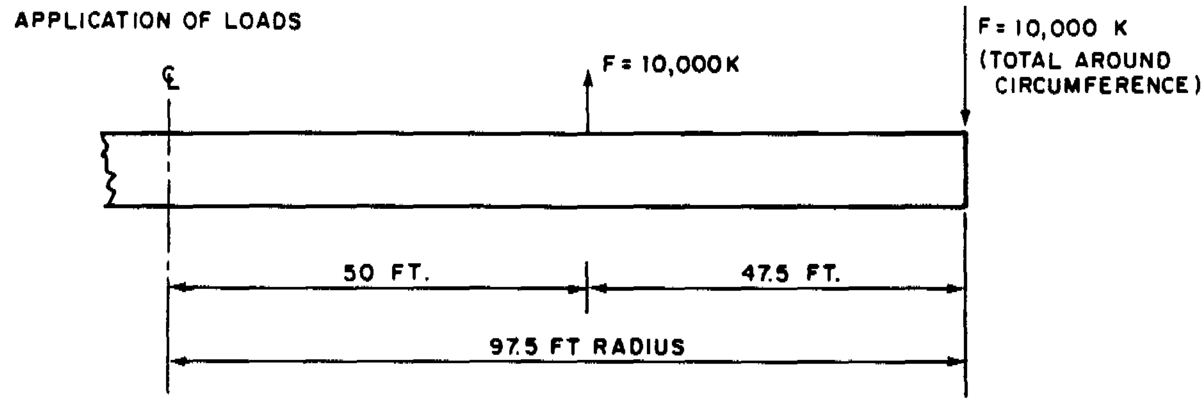


FIGURE 3A.1.7-2
HORIZONTAL AMPLIFIED RESPONSE SPECTRA AT TOP MASS IN BEAM STRUCTURE
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT



NOTES

1. MAT THICKNESS = 10.0 FT
2. MAT RADIUS = R = 97.5 FT
3. YOUNG'S MODULUS OF MAT = E_m = 432,000.0 KSF
4. POISSON'S RATIO OF MAT = υ = 0.167
5. SOIL SHEAR MODULUS = G_0 = 2,678.6 KSF
6. POISSON'S RATIO OF SOIL = μ_0 = 0.40
7. YOUNG'S MODULUS OF SOIL = E_0 = 7,500.0 KSF
8. SOIL LAYER THICKNESS = 10^6 FT

FIGURE 3A.1.8-1
MAT 6 SAMPLE PROBLEM
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

3A.2 EQUIPMENT AND COMPONENTS

The following computer programs are used for the analysis of Seismic Category I equipment and components as well as for pipe rupture design and analysis:

1. DINASAW - Dynamic Inelastic Nonlinear Analysis,
2. LIMITA2 - 2-D Nonlinear Transient Dynamic Analysis,
3. LIMITA3 - 3-D Nonlinear Transient Dynamic Analysis,
4. STARDYNE - Linear and Nonlinear Elastic Structure Analysis,
5. NOZZLE - Vessel Penetration Analysis,
6. LION - 3-D Heat Transfer Analysis,
7. SLOSH - Simplified Tank Sloshing Analysis,
8. MISSILE - Turbine Missile Probability Program,
9. PSPECTRA - Combining Amplified Response Spectra,
10. LIMITA2S - Nonlinear Static Analysis of Plane Frames,
11. STRUDL-SW - Structural Analysis Package,
12. STRUDL II - Structural Analysis Program,
13. TAC2D - Heat Transfer Program,
14. TAP-A - 3-D Heat Transfer Program,
15. ANSYS - Nonlinear Transient Thermal/Structural Analysis Program,
16. GTSTRUDL - Structural Analysis Program,
17 LIDOP - Pipe Crush Characteristics,
18. ASYMPR - Asymmetric Pressure Force Time History, and
19. DLF - Dynamic Load Factor

PSPECTRA, STRUDL-SW, STRUDL II, ANSYS, GTSTRUDL and TAC2D program descriptions and verifications are presented in Sections 3A.3.12, 3A.1.3, 3A.1.1, 3A.1.9, 3A.1.13 and 3A.1.5; respectively, and are not duplicated here.

3A.2.1 DINASAW

3A.2.1.1 General Description

DINASAW is a modification and extension of a lumped-mass elastic-plastic dynamic analysis code (Wu and Witmer 1972) used to predict the large-deflection behavior of beams and rings. DINASAW extends this analysis to cover pipes (tubular cross-sections) which may impact walls or restraints.

The analysis, as derived (Wu and Witmer 1972; Collins and Witmer 1973), employs the spatial finite-element method in which the tangential and normal displacement fields are represented by cubic interpolations. By applying the principle of virtual work in conjunction with D'Alembert's principle, the equations of motion may be derived in the form:

$$[M]\ \{\ddot{q}\} = \{F\} - \{P\} - [H]\ \{q\} \tag{3A.2.1-1}$$

where:

$\{q\}$ and $\{\ddot{q}\}$	=	the generalized displacements and generalized accelerations, respectively, for the complete assembled discretized structure defined with respect to a global coordinate system,
$[M]$	=	the lumped mass matrix for the complete assembled discretized structure,
$\{F\}$	=	the assembled vector of externally-applied loading,
$\{P\}$	=	an assembled internal force matrix (replaces conventional stiffness matrix),
$[H]\ \{q\}$	=	generalized loads arising from both large deflection and plastic behavior.

3A.2.1.2 Program Verification

Three examples are discussed here. The first (Wu and Witmer 1972) involves a ring subjected to a radial blast wave over a portion of its circumference (Figure 3A.2.1-1). The resulting deformation severely distorts the ring, flattening it considerably. Still, the computer code follows very closely not only the displacement field, but also the strain time history (Figure 3A.2.1-2 and 3A.2.1-3).

The second case (Collins and Witmer 1973) involves the impact of a rotor segment onto a ring or shroud. Again the program, in conjunction with the Collision Imparted Velocity Method (CIVM), follows experimental results very closely (Figures 3A.2.1-4 and 3A.2.1-5).

The third case was analyzed by DINASAW and LIMITA2 (Section 3A.2.2). It consisted of a cantilever pipe (Figure 3A.2.2-1) subjected to an impulsive load at its free end. The impulse is imparted by the detonation of a sheet of high explosive, separated from the pipe by a buffer material. A nearly uniform initial velocity is produced in the loaded region and is determined by high-speed photography. The results of the DINASAW analyses were compared with experimental data and the output from LIMITA2.

The stress-strain curves used in the DINASAW and LIMITA2 calculations are shown on Figure 3A.2.2-2 with the experimentally derived curve. Figure 3A.2.2-3 shows the lumped mass model used for both computer solutions. The impulsive load, idealized as initial nodal velocities, is also shown on Figure 3A.2.2-3. Time history plots of the X and Y displacements of the free end of the pipe for the LIMITA2 and DINASAW runs are shown on Figures 3A.2.2-4 and 3A.2.2-5, respectively. The moment reaction at the clamped end of the pipe is shown on Figure 3A.2.2-6. A comparison of the permanent pipe deformations predicted by experiment, DINASAW and LIMITA2, is illustrated on Figure 3A.2.2-7. Agreement is good in all cases as shown on Figures 3A.2.2-4, 3A.2.2-5, 3A.2.2-6 and 3A.2.2-7.

3A.2.2 LIMITA2

3A.2.2.1 General Description

LIMITA2 (ST-223) is a two-dimensional, nonlinear, transient dynamic computer code developed and fully documented by Stone & Webster Engineering Corporation (SWEC) for in-house use. A plane frame is simulated as a lumped parameter system, represented mathematically by an assembly of discrete lumped masses connected by beam members. Under any loading, the equilibrium at the rth mass point is ensured by the equation of motion:

$$m_r \ddot{q}_r + \sum^{i} K_{ri} q_i = f_{\bar{r}} \qquad (3A.2.2\text{-}1)$$

where:

$\sum$ = a series with one term for each of the i displacements,

K_{ri} = the member stiffness, which is defined as the force necessary to hold the structural member from moving in the rth degree of freedom when the ith degree of freedom is given a unit displacement when all other degrees of freedom are restrained from moving, (Martin 1966; Przemieniecki 1968),

$f_{\bar{r}}$ = the external load factor,

m_r = the rth discrete mass point of the structure,

$q_i, \ddot{q}_r$ = the generalized displacement and accelerations, respectively, for the complete assembled discretized structure defined with respect to a global system.

To take account of nonlinear effects, such as plasticity and large deflections, Equation 3A.2.2-1 is solved by an incremental method (Clough and Wilson 1962). At any particular time, t, the displacement increment is obtained from:

$$m_r \ddot{q}_r^t + \sum^{i} K_{ri}^t \,\Delta q_i^t = f_{ri}^{\,t} - \sum_{s=o}^{t-\Delta t} \left(\sum^{i} K_{ri}^s \,\Delta q_i^s \right) \qquad (3A.2.2\text{-}2)$$

where:

K_{ri}^t = the member stiffness

f_{ri}^t = the forcing function

which are calculated based on the current deformed structure (Martin 1966) and assumed constant through the time step, Δt. The displacement and member forces are thus given by:

$$q_r^t = \sum_{s=o}^{t} \Delta q_r^s$$

$$Q_r^t = \sum_{s=o}^{t} \left(\sum^{i} K_i^s \,\Delta q_i^s \right) \qquad (3A.2.2\text{-}3)$$

where:

q_r = member r displacement vector

Q_r = member r force vector

The second order differential system equations (Equation 3A.2.2-2) are solved by a linear acceleration method (Hildebrand 1956).

Since no external loading is applied to a member between nodes, the maximum value of the internal force acting on a member occurs at its end sections. The transition from the elastic to the fully plastic state is disregarded and the end sections are assumed to remain linearly elastic up to the full plastic yield surface.

The yield surface is defined by a scalar function of the internal member forces, Q, of the form (Hodge 1959; Neal 1961; and Stokey, Peterson, and Wruder, 1966):

$$\Phi(Q^t) = 1 \qquad (3A.2.2\text{-}4)$$

Here the function Φ is obtained by integrating the stress across the section with the stress fully developed over the section and satisfying the von Mises (or Tresca) yield criterion:

$$\sigma^2 + \gamma^2 \tau^2 = \sigma_y^2 \qquad (3A.2.2\text{-}5)$$

where:

σ = normal stress,

τ = shear stress,

σ_y = yield stress in simple tension,

γ^2 = 3 (von Mises) or 4 (Tresca).

Thus, the function Φ depends on the shape of the cross section and the force components being considered.

For a frame structure, the yielding normally occurs due to either a predominant bending moment or to a predominant tension or compression. Thus, two plastic models are provided:

1. Bending predominant members

   Since a section is either elastic or fully plastic, there are four possible states:

   a. Both ends A and B are elastic,

   b. End A is yielding and B is elastic,

   c. End A is elastic and B is yielding, or

   d. Both ends A and B are yielding.

   A plastic hinge is introduced at any end section which is yielding. The force-displacement relation of the plastic hinge follows an ideal bilinear strain-hardening curve (Clough, Benuska, and Wilson 1965; Giberson 1967). In

situations where the force unloads, the elastic stiffness of the hinged member is restored (isotropic strain-hardening model).

2. Tension or compression predominant members

   There are only two possible states:

   a. The entire member is elastic, or

   b. The entire member is plastic.

   When the member yields, Young's Modulus is replaced by a plastic tangent modulus and the force-displacement curve follows a bilinear curve. If the member unloads, the elastic modulus is restored.

3A.2.2.2 Program Verification

SWEC sponsored an experimental investigation performed by the Massachusetts Institute of Technology (MIT) (Pirotin and East 1977). The problem consisted of the cantilevered pipe (Figure 3A.2.2-1) subjected to an impulsive load at its free end. The impulse is imparted by the detonation of a sheet of high explosive, separated from the pipe by a buffer material. A nearly uniform initial velocity is produced in the loaded region and is determined by high speed photography. This problem was analyzed by LIMITA2. The results were compared with experimental data and output from another computer program, DINASAW.

The stress-strain curves used in the LIMITA2 and DINASAW calculations are shown on Figure 3A.2.2-2 with the experimentally derived curve. Figure 3A.2.2-3 shows the lumped-mass models used for both computer solutions. The impulsive load, idealized as initial nodal velocities, is also shown on Figure 3A.2.2-3. Time history plots of the x and y displacements of the free end of the pipe for the LIMITA2 and DINASAW runs are shown on Figures 3A.2.2-4 and 3A.2.2-5, respectively. The moment reaction at the clamped end of the pipe is shown on Figure 3A.2.2-6. A comparison of the permanent pipe deformations predicted by the experiment, DINASAW, and LIMITA2 is illustrated on Figure 3A.2.2-7. Agreement is good in all cases, as seen on Figures 3A.2.2-4, 3A.2.2-5, 3A.2.2-6 and 3A.2.2-7. Additional problems were also evaluated to ensure that all program options were exercised, and thus demonstrate the function and adequacy of this program.

3A.2.3 LIMITA3

3A.2.3.1 General Description

LIMITA3 (ST-225) is a computer code developed and fully documented by SWEC for in-house use. Its formulation is identical to that of LIMITA2 (Section 3A.2.2), with the exception that the equations are

applicable to a general three-dimensional problem. For a space frame, yielding normally occurs due to either a predominant bending moment or a predominant torsion (combined with axial load). Therefore, two plastic models are provided.

1. Bending Yield Model

   Since a beam section is either elastic or fully plastic, there are four possible states:

   a. Both ends A and B are elastic,

   b. End A is plastic, end B is elastic,

   c. End A is elastic, end B is plastic, or

   d. Both ends A and B are plastic.

   A plastic hinge is introduced at any end section which is yielding. The force-displacement relation of the plastic hinge follows an ideal bilinear strain-hardening curve (Clough, Benuska, and Wilson 1965; Giberson 1967). In situations where the force unloads, the elastic stiffness of the hinged member is restored (isotropic strain-hardening model).

2. Torsional-Axial Yield Model

   There are only two possible states:

   a. The entire member is elastic, or

   b. The entire member is plastic.

   When the member yields, the Young's Modulus is replaced by a plastic tangent modulus and the force-displacement relation follows a bilinear curve. If the member unloads, the elastic modulus is restored.

3A.2.3.2 Program Verification

3A.2.3.2.1 Elastic Example

Consider the dynamic response of a space frame (Figure 3A.2.3-1) subjected to a step load of 30 kips at joint 6. This problem was analyzed by LIMITA3 and STRUDL II elastically. The results of displacements and moment Z at joint 6 were plotted against each other on Figures 3A.2.3-2 and 3A.2.3-3, respectively. As shown, there is excellent agreement.

3A.2.3.2.2 Plastic Example

This example is provided to illustrate the ability of the program to determine the inelastic transient response of a three-dimensional structure. The structure considered consists of cantilevered steel tubes (Figure 3A.2.3-4) subjected to force transients, causing bending and torsion in the structure. The results obtained from an analysis using the LIMITA3 code are compared with data obtained experimentally (Larson 1973).

The experiment was a drop test in which the cantilevered tubes were loaded by weights at each tube end. The results tabulated were the peak deflections and their corresponding times and the permanent deflections. These results are compared to those obtained using the LIMITA3 code in Table 3A.2.3-1.

Additional problems, elastic and inelastic, were analyzed to ensure that all program options were exercised, and thus demonstrate the function and adequacy of this program.

3A.2.4 STARDYNE

3A.2.4.1 General Description

The STARDYNE (ST-330) Structural Analysis System, written by System Development Corporation (SDC) of Santa Monica, California, is a fully warranted and documented computer program available at Control Data Corporation. The latest version of this program became available in 1984.

The SDC STARDYNE Analysis System consists of a series of compatible digital computer programs, which includes the DYNRE1, DYNRE4, and DYNRE6 codes, designed to analyze linear and nonlinear elastic structural models. The system encompasses the full range of static and dynamic analyses.

The basic equation used by STARDYNE is:

$$[m]\{\ddot{\delta}\} + [c]\{\dot{\delta}\} + [k]\{\delta\} = \{F(t)\} \quad (3A.2.4\text{-}1)$$

where:

$[m]$ = mass matrix

$[c]$ = damping matrix

$[k]$ = stiffness matrix

$\{\delta\}$ = displacement vector

$\{\dot{\delta}\}$ = velocity vector

$\{\ddot{\delta}\}$ = acceleration vector

$\{F(t)\}$ = time dependent forcing function

The static capability includes the computation of structural deformations and member loads and stresses caused by an arbitrary set of thermal, nodal-applied loads, and prescribed displacements (Cybernet Services 1974, Section I-C).

Utilizing the normal mode technique (Cybernet Services 1974, Sections III-A and IV-A) linear dynamic response analyses can be performed for a wide range of loading conditions, including transient (Cybernet Services 1974, Section IV-B), steady-state harmonic (Cybernet Services 1974, Section IV-C), random (Cybernet Services 1974, Section IV-D), and shock spectra excitation types (Cybernet Services 1974, Sections III-E and III-F). Dynamic response results can be presented as structural deformations and internal member loads.

The nonlinear dynamic analysis program is integrated in the rest of the STARDYNE system. The equations of motion for the linear portion of the structural model are generated and modified to account for the nonlinear springs. The resulting nonlinear equations of motion are directly integrated using either the Newmark or Wilson implicit integration operators (Newmark 1959; Bathe and Wilson 1973). The user may enter sets of structural loadings which vary with time, and specify time points at which the program is to output the structural response.

3A.2.5 NOZZLE

3A.2.5.1 General Description

The vessel penetration analysis computer code, NOZZLE (ST-147), is a code written and fully documented by SWEC for in-house use. This code performs various analyses on tanks and pressure vessels. All of the analyses are concerned with local stresses at penetrations. Typical problems which can be handled include the following:

1. Applied load stresses at vessel-nozzle junction for:
    a. Rigid attachment to cylinder,
    b. Rigid attachment to sphere, or
    c. Hollow attachment to sphere,
2. Pressure discontinuity analysis for thin shell interaction,
3. Allowable load functions on nozzles for each case.

Local stresses due to nozzle loads are found by the method prescribed by P. P. Bijlaard (Wichman, Hopper, and Mershon 1965). The method prescribed by Johns and Orange (1961) is used for pressure discontinuity stresses.

3A.2.5.2 Program Verification

A sample problem of a thin-walled cylindrical vessel is subjected to applied loads from a rigid cylindrical attachment. This problem may be solved using Wichman, Hopper, and Mershon's (1965) method. The pertinent parameters of the problem are presented in Figure 3A.2.5-1.

A summary of the manual calculations is shown on Figure 3A.2.5-1. The computer calculations are summarized on Figure 3A.2.5-2. As can be seen, the computer results are very close to the exact results.

The second sample problem is of a thin-walled spherical vessel subjected to applied loads at a hollow nozzle. The pertinent parameters and a summary of the manual calculations are shown on Figure 3A.2.5-3. The computer calculations are summarized on Figure 3A.2.5-4. The comparison shows excellent agreement for this type of analysis.

The third sample problem is of a thin-walled cylindrical vessel. The purpose of this problem was to analyze the discontinuity stresses at the nozzle to shell junction. The pertinent parameters are provided on Figure 3A.2.5-5. The comparison of manual calculations to computer calculations are summarized on Figure 3A.2.5-6.

3A.2.6 LION

3A.2.6.1 General Description

LION (ME-112) is a digital computer program which is used to solve three-dimensional transient and steady-state temperature distribution problems. The program may also consider subcooled nucleate boiling and coolant heat transfer effects. The surface conditions may be forced convection, free convection, or radiation, and heat may be externally or internally generated. Input to the program consists of structural geometry, physical properties, boundary conditions, internal heat generation rates, coolant flow properties, and flow rates. The program solves the transient heat conduction equations for a three-dimensional field using a first forward difference method.

Since the original program by Bray (1954) was developed, subsequent versions have evolved to solve larger and more complex problems (Bray 1954; Bray and McCraken 1959; Briggs 1963; Lechliter, Liedel, and Schmid 1964; Schmid, Lechliter, and Fisher 1969).

LION is a recognized program developed by the General Electric Company, Knolls Atomic Power Laboratory, Schenectady, N.Y., and

obtained by SWEC from the Argonne Code Center. Verification of the program is contained in the user's manual. This includes problems which were provided by the Argonne Code Center, hand-calculated results produced by Knolls Atomic Power Laboratory, and the computer output of the test runs.

3A.2.7 SLOSH

3A.2.7.1 General Description

SLOSH (ME-111) is a computer code written and fully documented by SWEC for in-house use. The purpose of this program is to compute the seismically-induced liquid pressures and the maximum vertical displacement of the liquid surface in a container under horizontal acceleration. The mathematical procedures and formulas used in developing the program were taken from AEC Report TID-7024 (U.S. Atomic Energy Commission (USAEC) 1963). The program uses data for intensity of ground motion taken from average-acceleration-spectrum curves, as used in the analysis in TID-7024.

The program is used for circular or rectangular, shallow or slender, ground-supported tanks and circular or rectangular, shallow (not slender) tower-supported tanks.

3A.2.7.2 Program Verification

There are three examples provided in TID-7024 (USAEC 1963). These problems cover 1) tanks supported on the ground, 2) slender tanks supported on the ground, and 3) elevated tanks. The pertinent parameters of the problems are presented in Table 3A.2.7-1. The comparison of computer calculations to those given in TID-7024 are provided in Table 3A.2.7-2. This comparison shows excellent agreement between the SLOSH results and those provided in TID-7024.

3A.2.8 MISSILE

3A.2.8.1 General Description

MISSILE (MA-057) is a computer code written and fully documented by SWEC for in-house use. The MISSILE program calculates the impact probability (P_2) of postulated turbine missiles on specified targets. The solid angle method is used to calculate P_2. The following is the methodology:

The turbine spins about the Z-axis of the reference system as shown on Figure 3A.2.8-1. Based on data provided by Westinghouse Electric Corporation, a postulated missile is thrown from the turbine with initial velocity V , as shown. The variable angles required to describe the resulting motion are displayed as shown on Figure 3A.2.8-1. Deflection angles δ^1 and δ^2 provided by Westinghouse limit 0 to the range:

$$\frac{\pi}{2} - \delta_1 \leq \theta \leq \frac{\pi}{2} + \delta_2 \qquad (3A.2.8\text{-}1)$$

The probability that a single disc fragment strikes a critical area
A is defined as:*f*

$$P_2(A_0) = \int_{\Omega_o} f(\Omega)\, d\Omega \qquad (3A.2.8\text{-}2)$$

where:

Ω_0 = the solid angle which must be subtended by the initial velocity vector for a missile to strike A,

$d\Omega$ = the differential solid angle, and

$f(\Omega)$ = the probability density function.

From Figure 3A.2.8-1:

$$d\,\Omega = \cos\Phi\; d\Phi\; d\psi \qquad (3A.2.8\text{-}3)$$

Given V_o, the elevation angle Φ necessary to hit any point on A_o (described by r, y, and Ψ on Figure 3A.2.8-1 is determined from classical trajectory theory as:

$$\Phi = \tan^{-1}\left[\frac{1 \pm \left[1-\left(rg/v_0^2\right)^2 - 2\left(yg/v_0^2\right)\right]^{1/2}}{\left(rg/v_0^2\right)}\right] \qquad (3A.2.8\text{-}4)$$

In Equation 3A.2.8-4, air resistance is neglected and the ± refers to high and low trajectory missiles respectively.

The probability density function f (Ω) is determined by assuming:

$f(\Omega)$ = constant = f_o for $0 \leq \beta \leq 2\pi$ and (3A.2.8-5)

$$\pi/2 - \delta_1 \leq \Phi \leq \pi/2 + \delta_2$$

$f(\Omega)$ = O, for all other θ

From probability theory it is required that:

$$\int_{\Omega} f(\Omega) \quad d\Omega = 1 \tag{3A.2.8-6}$$

Therefore,

$$f_o = \frac{1}{2\prod \ (\sin \delta_1 + \sin \delta_2)} \tag{3A.2.8-7}$$

The probability that n disc fragments strike a critical area, A is then:

$$P_2(A_o) = \frac{n}{2\prod \ (\sin \delta_1 \ + \ \sin \delta_2)} \int_{\Omega_o} d\Omega \tag{3A.2.8-8}$$

MISSILE has been developed to calculate the strike probability using Equation 3A.2.8-7. Following Bush (1973), the analysis considers high trajectory hits on the tops of all critical targets and low trajectory hits on the sides of all critical targets. Figures 3A.2.8-2 and 3A.2.8-3 represent the top and side views of an idealized target. The strike probability of the target is found by numerically integrating Equation 3A.2.8-7, which gives:

$$P_2 = \frac{n}{2\pi \ (\sin \delta_1 + \sin \delta_2)} \sum_{i=1}^{n_\Psi} (\cos \phi_i)(\Delta\phi_i)\Delta\psi \tag{3A.2.8-9}$$

for

$$(\prod/2 - \delta_1) \leq \theta_1 \leq (\prod/2 \ + \ \delta_2)$$

and

$$P_2 = 0$$

for

$$O \leq \theta_i < (\prod/2 - \delta_1), \ (\prod/2 + \delta_2) < \theta_1 \leq \prod$$

where:

$$\theta_i = \cos^{-1} (\cos \phi_i \quad \cos \psi_i)$$

and

n_ψ = number of ground angle increments taken through the target.

From Figures 3A.2.8-2 and 3A.2.8-3,

$$\Delta\psi = \frac{\psi \max - \psi \min}{n_\psi}$$

$$\psi_i = \psi_{min} + (i - 1/2)\ \Delta\psi \qquad (3A.2.8\text{-}10)$$

$$\Delta\phi_1 = \left|\phi_2^i - \phi_1^i\right|$$

$$\phi_i = 1/2\left(\phi_1^i + \phi_2^i\right)$$

Equation 3A.2.8-4 is used to determine ϕ_1^i, ϕ_2^i. The low and high trajectory probabilities are calculated separately and added to obtain the final probability.

3A.2.8.2 High Trajectory Verification

Westinghouse has derived a formula to predict the probability of impact for high trajectory missiles (Westinghouse 1974). Some adjustments to the formula are necessary to enable direct comparison of hand calculated results with the program results. The formula has been derived on the basis that the initial velocity is random and uniformly distributed between V_1 and V_2. The program uses a deterministic initial velocity. The formula may be specialized to this condition by setting V_1 equal to V_2 after applying L'Hopital's Rule. Also, the formula has been derived assuming a missile fragment occurs in the quadrant of the target, whereas the program assumes a missile fragment can occur in any of the four quadrants. These differing assumptions can be reconciled by using four fragments for program input.

After making the above adjustments, the high trajectory formula becomes:

$$P = G^2/(2\pi\Delta V^4) \qquad (3A.2.8\text{-}11)$$

where:

P = Impact probability per square foot of target,

G = Acceleration of gravity (ft/sec^2),

Δ = Deflection angle range (radians),

V = Initial velocity (fps).

Comparison of the probability calculated by the formula and the results of the computer program are given in Table 3A.2.8-1.

3A.2.8.3 Low Trajectory Verification

The probability of impact for low trajectory missiles (LTM) calculated by this program was verified by comparison with Bush (1973). The LTM was identified as four fragments of the outer disk resulting from a turbine failure. The two different initial ejection velocities were 300 and 600 fps. The geometry is shown on Figure 3A.2.8-4.

Since only the half of Bush's (1973) 4,800 square foot target lies in the reported interval ($O \leq \delta < 25°$), only a 2,400 square foot portion was modeled in MISSILE. A comparison of the probability listed in Bush (1973) and the results of the computer program is provided in Table 3A.2.8-1.

Additional problems were analyzed to ensure that all program options were exercised and demonstrate the function and adequacy of this program.

3A.2.9 PSPECTRA

3A.2.9.1 General Description

A description of the program and sample problem used for verification are provided in Section 3A.3.12.

3A.2.10 LIMITA2S

3A.2.10.1 General Description

LIM1TA2S (ST-224) is a computer code written and fully documented by SWEC which predicts the nonlinear, static behavior of two-dimensional structures. A plane frame is represented mathematically as a discrete system of beam members. Under loading, the equilibrium at each joint is ensured by the system equilibrium equations (Martin 1960):

$$[K]\ \{q\} = \{F\} \qquad (3A.2.10\text{-}1)$$

where:

$[K]$ = System stiffness matrix

{q} = Global displacement vector

{F} = External force vector

An element of the stiffness matrix, K_{ij}, situated in row i and column j, is the force in the i th degree of freedom required to produce a unit displacement in the j th degree of freedom when all other degrees of freedom are restrained from moving (Martin 1966; Przemieniecki 1968).

To account for nonlinear effects, such as plasticity and large deflections, Equation 3A.2.10-1 is solved by an incremental method. At any particular load step i, Equation 3A.2.10-1 may be written:

$$[K]_i \ \{\Delta q\} = \{\Delta F\}_i \tag{3A.2.10-2}$$

where:

$$[\Delta q]_i = [q]_i = [q]_{i-1} \ [\Delta F]_i = [F]_i - [F]_{i-1}$$

$$[\Delta q]_O = [q]_O, \qquad [\Delta F]_O = [F]_O$$

The stiffness matrix [K] , calculated based on the deformed structure at load step i in Martin's work (1966), is assumed constant through the load step. Displacements and member forces are given at load step i by:

$$\{q\} = \sum_{\alpha=o}^{i} \{\Delta q\}_\alpha$$

(3A.2.10-3)

$$\{Q\} = \sum_{\alpha=o}^{i} [k]_\alpha \ \{\Delta \bar{q}\}_\alpha$$

where :

{Q} = Member force vector

[k] = Member stiffness matrix

{q} = Member displacement vector

The equilibrium equations, Equation 3A.2.10-2, are solved by a standard elimination technique.

Since no external loading is applied to a member between joints, the maximum value of the internal force acting on a member occurs at its ends. The transition from the elastic to the fully plastic state is disregarded, and the end sections are assumed to remain linearly elastic until a fully plastic state is reached. The yield surface is defined by a scalar function, , of the internal member force, $\{Q\}$, having the form (Hodge 1959; Neal 1961; Stokey, Peterson, and Wruder, 1966):

$$\phi\ (\{Q\}_i) = 1 \qquad (3A.2.10\text{-}4)$$

ϕ is obtained by integrating stress across the member section with the stress fully developed over the section and satisfying the von Mises (or Tresca) yield criterion:

$$\sigma^2 + \alpha^2 \tau^2 = \sigma_y^2 \qquad (3A.2.10\text{-}5)$$

where:

σ = Normal stress

τ = Shear stress

σ_y = Yield stress in simple tension

α^2 = 3 (von Mises) or 4 (Tresca)

Thus the function depends on the shape of the cross section and the force components being considered. For a plane frame, the yielding normally occurs due to either a predominant bending moment or a predominant axial force. Therefore, two plastic models are used.

Bending Yield Model:

Since a section is either elastic or fully plastic, there are four possible states:

1. Both ends A and B are elastic,
2. End A is plastic, end B is elastic,
3. End A is elastic, end B is plastic, or
4. Both ends A and B are plastic.

A plastic hinge is introduced at any end section which is yielding. The force-displacement relation of the plastic hinge follows an ideal

bilinear curve (Clough, Benuska, and Wilson 1965; Giberson 1967). In situations where force reversal occurs, the elastic stiffness of the hinged member is restored, providing elastic unloading (isotropic strain-hardening model).

Axial Yield Model:

There are only two possible states:

1. The entire member is elastic.

2. The entire member is plastic.

When the member yields, the member elastic Young's modulus is replaced by a plastic tangent modulus and the force-displacement relation follows a bilinear curve. If the member unloads, the elastic modulus is restored.

3A.2.10.2 Program Verification

A center-loaded beam, built in at one end, and supported at the other end, is analyzed for comparison to data obtained analytically. The analytical solution was obtained using limit analysis as explained in Hodge's work (1959). The displacement at the point of loading calculated using LIMITA2S and calculated analytically is shown on Figure 3A.2.10-1.

Additional problems were analyzed to ensure that all program options were exercised and thus demonstrate the function and adequacy of the program.

3A.2.11 STRUDL-SW

3A.2.11.1 General Description

A description of the program is provided in Section 3A.1.3.

3A.2.12 STRUDL II

3A.2.12.1 General Description

A description of the program is provided in Section 3A.1.1.

3A.2.13 TAC2D

3A.2.13.1 General Description

A description of the program and sample problem used for verification are provided in Section 3A.1.5.

3A.2.14 TAP-A

3A.2.14.1 General Description

TAP-A is a general thermal analysis program developed to solve problems involving transient and steady-state heat transfer in multi- dimensional systems. The systems may have arbitrary geometric configuration, boundary conditions, initial conditions, and physical properties. The program has the capability to consider the following modes of heat transfer and boundary conditions:

1. Internal conduction and radiation,

2. Free and forced convection,

3. Radiation at external surfaces,

4. Specified time dependent surface temperatures, heat fluxes and boundary conditions, and

5. Space and time dependent thermal conductivity and heat capacity.

The TAP-A program solves the general heat conduction equation:

$$\nabla \bullet K\nabla t + q = \rho\ C_p\ \frac{\partial t}{\partial \theta} \qquad (3A.2.14\text{-}1)$$

by either of two unconditionally stable finite difference methods. The first method is the implicit method:

$$Q_i + \sum_j Y_{ij}\,(t_i^* - t_j^*) = \frac{C_i\,(t_i^* - t_i)}{\Delta\theta} \qquad (3A.2.14\text{-}2)$$

and the second technique is the exponential explicit method:

$$t_i^* = Z_i t_i + \frac{(1-Z_i)}{\sum_j Y_{ij}}\,\{\sum_j Y_{ij}\,t_j^* + \sum_j Y_{ij}\ t_j + Q_i\} \qquad (3A.2.14\text{-}3)$$

where:

t_i = temperature of node i at time θ

t_i^* = temperature of node i at time $\theta + \Delta\theta$

θ = time

K = thermal conductivity

C_p = heat capacity

q = heat generation rate per unit volume

Y_{ij} = admittance between node i and node j

Q_i = $q_i V_i P\theta$

V_i = volume of node i = $L_i W_i D_i$

Pe = dimensionless time function used when the heat generation rate is a function of time.

C_i = $C_{p_i} \rho_i V_i$

ρ_i = density of material of node i

L_i, W_i, D_i = dimension of node i

$$Z_i = \exp\left\{\frac{-\Delta\theta}{C_i}\Sigma Y_{ij}\right\}$$

When the implicit method is used to determine the temperature of a number of nodes n, a system of n linear equations (3A.2.14-1) must be solved. The Gauss-Seidel procedure is used to solve this system of n linear equations.

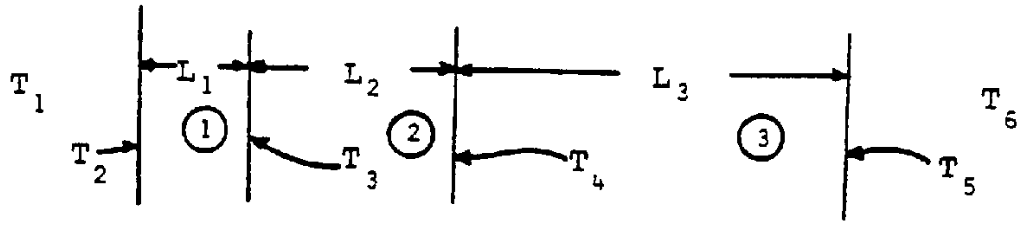
3A.2.14.2 Program Verification

TAP-A is a recognized program disseminated under the sponsorship of the National Aeronautics and Space Administration (NASA) and produced by Westinghouse Astronuclear Laboratory. The program or document was obtained by SWEC through Computer Software Management and Information Center (COSMIC), Athens, Georgia.

TAP-A was verified by a multi-dimensional sample problem provided in the user's manual (Pierce and Stumpf 1969) which utilizes all of the program options. As a check or bench mark, several additional sample problems were run and compared to hand-calculated results.

3A.2.14.2.1 Sample Problem

The following sample problem is a typical problem selected to verify steady-state temperature distribution in a composite wall:



where :

T_1 = 300°F (temp of left environment)

h_1 = 25 Btu/hr-ft^2-°F (film coefficient)

T_6 = 0°F (temp of right environment)

h_6 = 2 Btu/hr-ft^2-°F

K_1 = 1 Btu/hr-ft-°F (thermal conductivity of material No. 1)

K_2 = 5

K_3 = 0.1

L_1 = 0.5 ft (length of material No. 1)

L_2 = 1 ft

L_3 = 2 ft

Hand Solution:

$$Q = \frac{T_1 - T_6}{\sum R_{th}} \quad (R_{th} = \text{thermal resistance})$$

$$R_{th} = \frac{1}{h_1} + \frac{L_1}{K_1} + \frac{L_2}{K_2} + \frac{L_3}{K_3} + \frac{1}{h_6}$$

= 21.24

therefore, Q = 14.124 Btu/hr

The general solution for T_n is:

$T_n + 1 = T - Q/Rth$

The results are as follows:

	Hand Calculation	TAP-A
T_1	299.4°F	299.4
T_2	292.3	293.0
T_3	289.5	289.7
T_4	7.0	7.1

Verification is concluded from the above results.

3A.2.15 ANSYS

3A.2.15.1 General Description

A description of the program is provided in Section 3A.1.9.1.

3A.2.16 GTSTRUDL

3A.2.16.1 General Description

A description of the program is provided in Section 3A.1.13.1.

3A.2.17 LIDOP (ME-184)

3A.2.17.1 General Description

LIDOP (ME-184) will generate crush rigidities and deformation energies for pressurized or unpressurized piping in the following geometries:

1. Ring crush against flat rigid surface
2. Indent of straight pipe against rigid cylinder
3. 1.5D pipe elbow (extrados) against flat rigid surface
4. Pipe bend (extrados) against flat rigid surface
5. Indent of straight pipe against rectangular block

Both dynamic and material properties are considered in generation of the crush characteristics.

Unpressurized force-displacement and energy-displacement character-istics of pipe and elbows are generated from empirical equations which are based on experimental data. Pressurization effects, based on fluid displacement during deformation, are superimposed on the unpressurized characteristics. The overall dimensions of the contact area, where applicable, are generated by empirically corrected geometric relationships. Dynamic effects of elbows are empirically determined from an experimental comparison of static and dynamic impact of spheres. Dynamic effects of all other geometries and elbows in certain cases are based on the results of finite element computer simulations of rings impacting flat, rigid surfaces. The effects of material properties are determined from empirical relationships based on computer predictions.

3A.2.17.2 Program Verification

Several sample problems were performed by the use of the code ME-184 to ensure that all options were exercised and thus demonstrate the function and adequacy of the program. Analytical results from ME-184 were verified by comparison to hand calculations.

3A.2.18 ASYMPR (ME-171)

3A.2.18.1 General Description

This computer program is written to calculate the time history of the resulting forces and moments at assigned nodes in the dynamic model of the RPV support system. These forces are produced due to the external asymmetric pressure in the reactor cavity, resulting from a LOCA near a hot leg or cold leg nozzle. The pressure forces may be acting at the reactor pressure vessel, the primary shield wall, or the neutron shield tank.

The program performs the following calculation:

$$P(t) = \sum Pj(t) = \sum A_i \, x \, P_i(t)$$

$$M(t) = \sum Pj(t) \; x \; R_j$$

where :

$P_i(t)$ = A pressure time history for pressure area No. i

A_i = An area vector corresponding to the pressure time history p (t)

$Pj(t)$ = Force vector due to pressure acting on No. j area

$P(t)$ = Resultant force vector due to pressures acting on all areas

Rj = Displacement vector from a force application point on an area to the point of rotation

M(t) = Resultant moment vector due to all pressures

To calculate the force and moment time history at a node in a given structural model, the surface area on which the pressure acts is divided into several regions, such that only a constant pressure acts on one region at any time.

The projection of a surface area, multiplied by the pressure acting perpendicular to it, gives the pressure force. This force is broken into three global components by defining the direction cosines of the pressure force vector.

The centroid coordinates of the projected area are then calculated, and when these are subtracted from the coordinates of the node, the displacement components are obtained, which are used in calculating the three moments at the node.

The forces and moments for all projected areas, due to corresponding pressure forces, are thus calculated and summed for each time point to get the force/moment time history for the node.

3A.2.18.2 Program Verification (ME-171)

Sample problems were performed using ME-171 to ensure that all options were exercised and thus demonstrate the functional adequacy of the program. Analytical results from ME-171 were verified by comparison to hand calculations.

3A.2.19 Dynamic Load Factors (DLF ME-185)

3A.2.19.1 General Description

The ME-185 program determines the dynamic load factor (DLF) for a single degree-of-freedom harmonic oscillator subject to an arbitrary force history. At time zero, the oscillator is assumed to be in equilibrium and at rest. Its response to the force history, defined by a series of force-time pairs, is then computed. If the force is not specified at time zero, it is automatically set at zero and ramps up linearly to the first specified force-time coordinate. If the force is specified at time zero, it is assumed to be suddenly applied.

Using the initial conditions at time zero, ME-185 solves the equation of motion and searches for maxima during the interval up to the next specified force-time pair. It then determines the boundary conditions (position and velocity) at the end of the interval and uses these as initial conditions for the solution during the next time increment. The process is repeated until the last force-time pair is reached. Assuming this last force is applied as a continuing

load, the steady state response is computed and maxima determined. The greatest maxima is then divided by the greatest applied load to determine the maximum load factor. Since the solution method searches for the greatest absolute amplitude of the system response and applied force, the applied force may be positive or negative and may arbitrarily change signs during the specified force history.

The dynamic load factor depends on the natural frequency of the single degree of freedom oscillator. In order to provide the DLF at the frequency of the structural system being analyzed, as well as to show how the DLF changes to an error in the calculated frequency or in the duration of the applied force, DLFs are computed for a range of frequencies. The above calculation method is repeated for each of several discrete frequencies in the range. The frequency range extends approximately one order of magnitude to either side of the frequency corresponding to the period (duration) of the applied force history.

For a force which varies linearly between two specified force-time pairs, the equation of motion is:

$$M\ddot{x} + kx = F_o + a_o t$$

where:

M = mass of the oscillator

k = stiffness of the oscillator

F_o = applied force at the start of the interval

a_o = rate of change in the applied force (F/t)

The solution of this equation is:

$$X = C_1 \sin(wt) + C_2 \cos(wt) + C_3 + C_4 t$$

where:

w = $\sqrt{K/M,}$ the natural frequency

C_1 = $(V_o - a_o/k)/w$

C_2 = $X_o - F_o/k$

C_3 = F_o/k

C_4 = a_o/k

x_o = initial position

V_o = initial velocity

The above relations are simplified and the magnitude of the spring force is made identical to displacement of K=l. This relation is used in ME-185.

When transferring from one time interval to the next, the position and velocity must be determined before the new coefficients, $C_1 \ldots C_4$ can be calculated. The position at the end of the previous time interval can be computed from the above equation for X and the velocity may be determined from:

$$V = C_1W \cos(wt) - C_2W \sin(wt) + C_4$$

In any interval, the maxima or minima may be computed by substituting the times of zero velocity in the equation for X. These times may be computed from the relation:

$$O = C_1w \cos(wt) - C_2w \sin(wt) + C_4$$

If C_2 = O then,

$$wt = \cos^{-1} (-C_4/(C_1w))$$

If C_4 = O then,

$$wt = \tan^{-1} (C_1/C_2)$$

If C_1 = O then,

$$Wt = \sin^{-1} (C_4/C_2w))$$

otherwise:

$$wt = \cos^{-1} \left[\frac{-C_1C_4\,w \pm C_2w\sqrt{(C_1^2+C_2^2)w^2-C_4^2}}{(C_1^2+C_2^2)w^2}\right]$$

3A.2.19.2 Program Verification

Several problems were performed by the use of DLF (ME-185) to ensure that all options were used and thus demonstrate the function and adequacy of the program. Analytical results from ME-185 were verified by comparison to hand calculations.

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Tables for Section 3A.2

TABLE 3A.2.3-1

COMPARISON OF EXPERIMENTAL DATA AND ANALYTICAL DATA USING LIMITA3

	Experimental Values	LIMITA3 Computer Results	Difference (%)
Peak deflection-in node 6	.297	.310	4.2
Peak deflection-in node 9	(Not determined)	.870	-
Time at peak deflection-sec node 6	.004	.0042	4.8
Permanent deflection-in node 6	.144	.140	2.8
Permanent deflection-in node 9	.302	.310	2.6

TABLE 3A.2.7-1

SAMPLE PROBLEMS-INPUT PARAMETERS

	Example 1 - Tank on Ground	Example 2 - Slender Tank	Example 3 - Tank on Tower
Tank			
Type	Cylinderical	Cylinderical	Cylinder
Diameter	26 ft	26 ft	26 ft
Height of water	15 ft	30 ft	15 ft
Type of support	Ground	Ground	Steel tower
Critical damping	15%	15%	5%
Output desired	Impulsive and convective forces and movements and displacement surface.	Seismic forces and movements on the tank and displacement of water surface.	Maximum deflection, seismic shear on the tower, and displacement of water surface.

TABLE 3A.2.7-2

COMPARISON OF RESULTS OF SLOSH VS AEC ANALYSIS*

Example 1	Page 188*	SLOSH
W_o, Eq. impulsive force (kips)	298.5	299.7
P_o, Impulse force (kips)	105.4	105.8
M (EBP), Impulsive moment (kip-ft)	594	595
M (IBP), Impulsive moment (kip-ft)	1,120	1,118
W_1, Convective force (kips)	133	133
M_1 (EBP), Convective moment (kip-ft)	212	214
M_1 (IBP), Convective moment (kip-ft)	252	253
M_{max}, Maximum moment (kip-ft)	1,372	1,371
P_{max}, Maximum shear (kips)	127.9	128.5

Example 2	Page 192*	SLOSH
W_o, Eq. impulsive force (kips)	458	458
P_o, Impulsive force (kips)	277	278
M (EBP), Impulsive moment (kip-ft)	3,460	3,470
M (IBP), Impulsive moment (kip-ft)	4,070	4,074
W_1, Convective force (kips)	139	137
M_1(EBP), Convective moment (kip-ft)	552	547
M_1(IBP), Convective moment (kip-ft)	560	552
M_{max}, Maximum moment (kip-ft)	4,630	4,626
P_{max}, Maximum shear (kips)	301	301

Example 3	Page 197*	SLOSH
W_o, Eq. impulsive force (kips)	298.5	299.7
W_1, Convective force (kips)	133	133
Mode 1		
Frequency (cps)	0.333	0.331
FB1, seismic force (kips)	23.46	23.80
FA1, seismic force (kips)	1.26	1.28
Mode 2		
Frequency (cps)	1.486	1.482
FB2, seismic force (kips)	-3.90	-3.91
FA2, seismic force (kips)	174.39	174.54
P_{max}, Maximum shear (kips)	195.21	195.72

NOTE:

*U.S. Atomic Energy Commission 1963.

TABLE 3A.2.8-1

MISSILE PROGRAM VERIFICATION

Comparison of High Trajectory Probabilities

Velocity (fps)	Deflection Angle (°)	Formula*	Program
300	5	0.254×10^{-3}	0.259×10^{-3}
300	25	0.508×10^{-4}	0.534×10^{-4}
600	5	0.162×10^{-4}	0.160×10^{-4}
600	25	0.318×10^{-5}	0.330×10^{-5}

Comparison of Low Trajectory Probabilities

Velocity (fps)	Deflection Angle (°)	Bush**	Program
300	0° <25°	0.113	0.111
600	0° <25°	0.11	0.112

NOTES:

*Westinghouse Electric Corporation 1974.
**Bush, S.H. 1973.

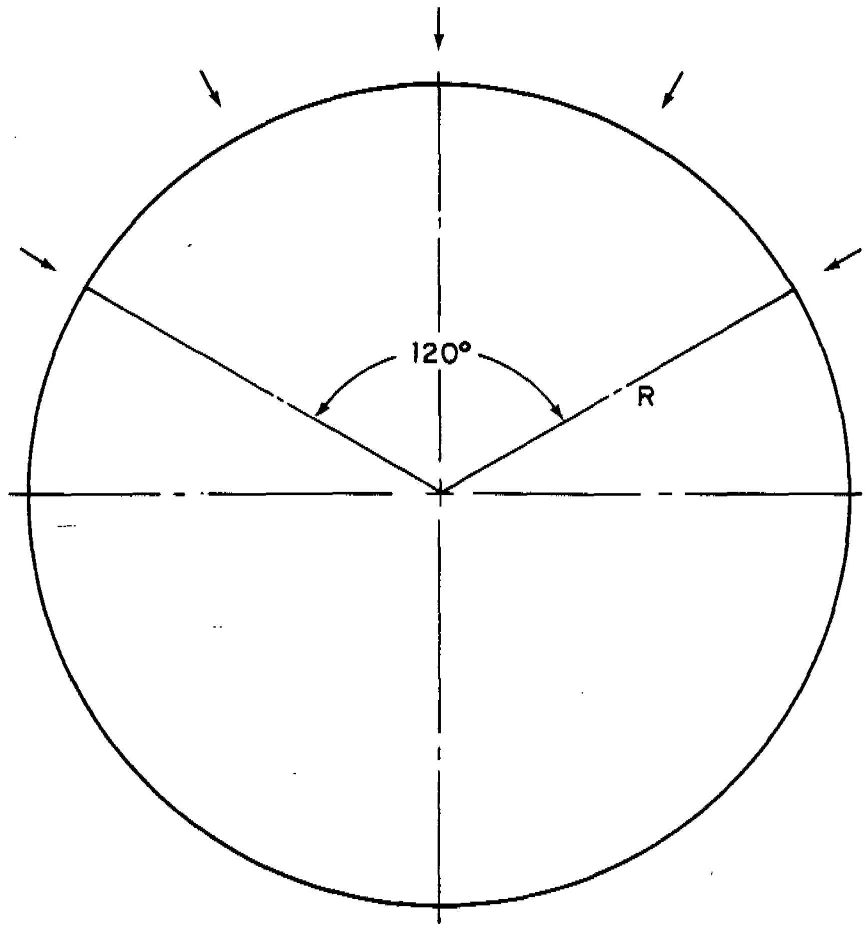


FIGURE 3A.2.1-1
CIRCULAR RING SUBJECTED TO
RADIAL BLAST WAVE
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

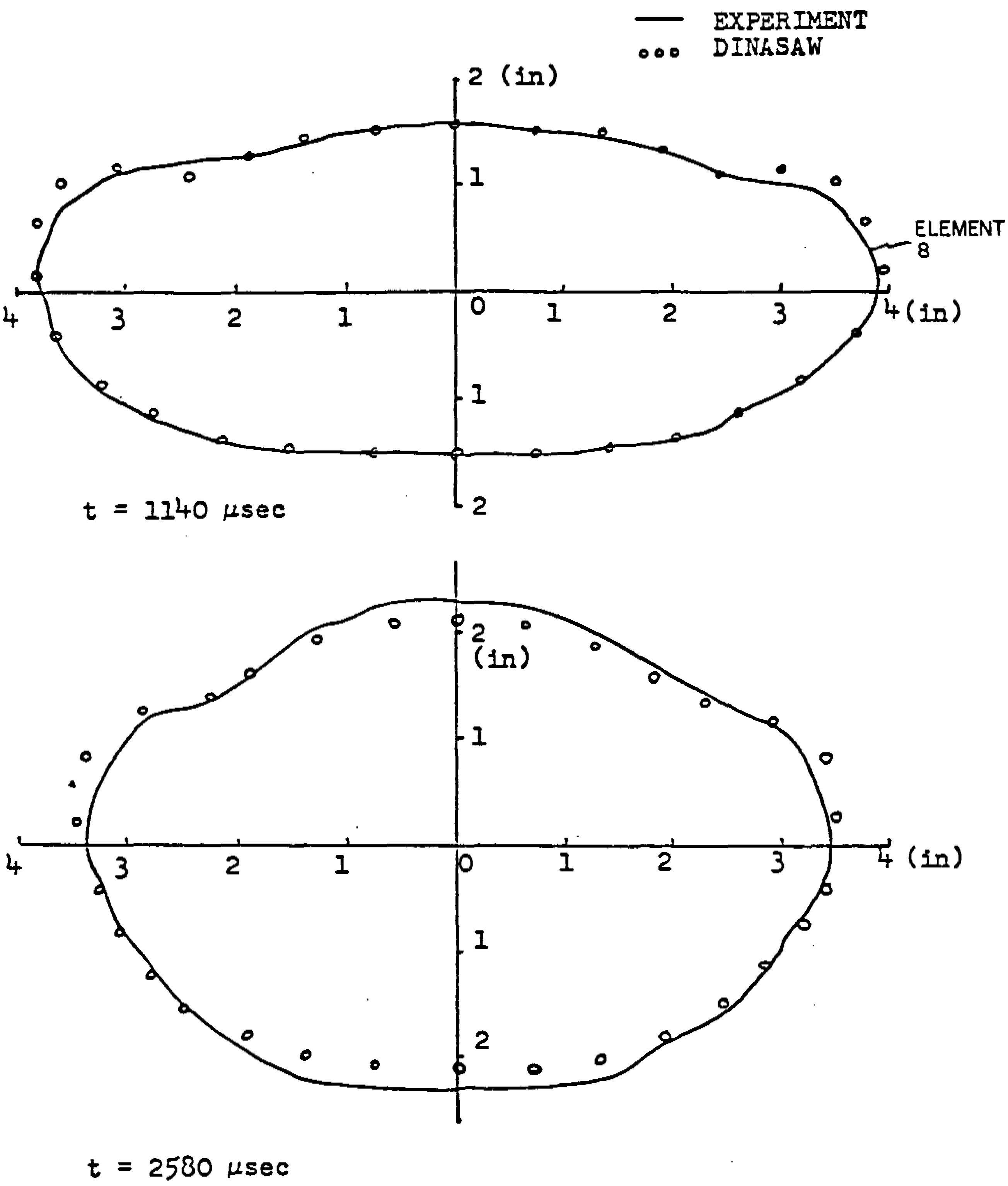


FIGURE 3A.2.1-2
DEFORMATION PROFILES
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

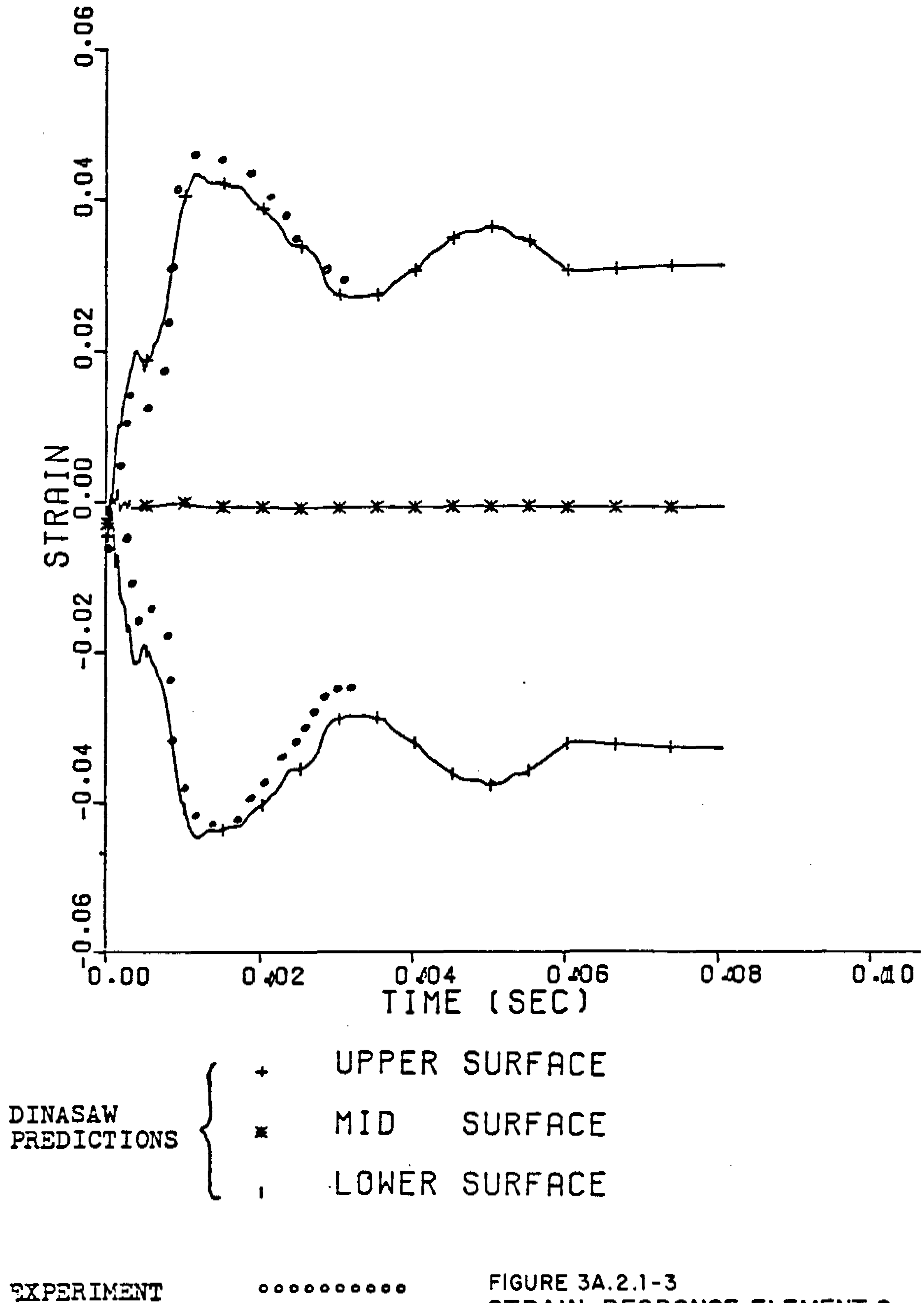


FIGURE 3A.2.1-3
STRAIN RESPONSE ELEMENT 8
(ANGLE = 90°)
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

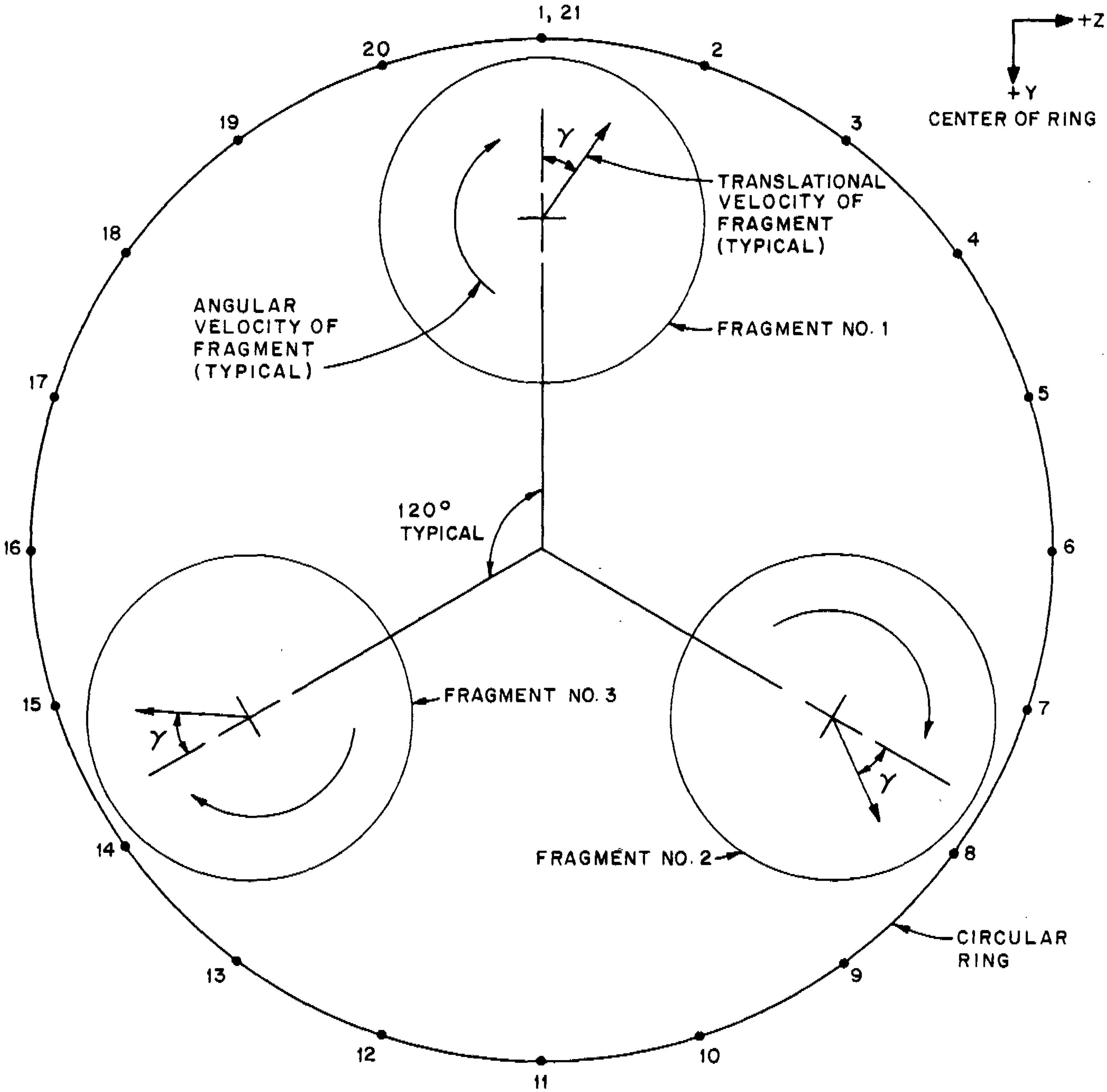


FIGURE 3A.2.1-4
FRAGMENT-RING IMPACT MODEL
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

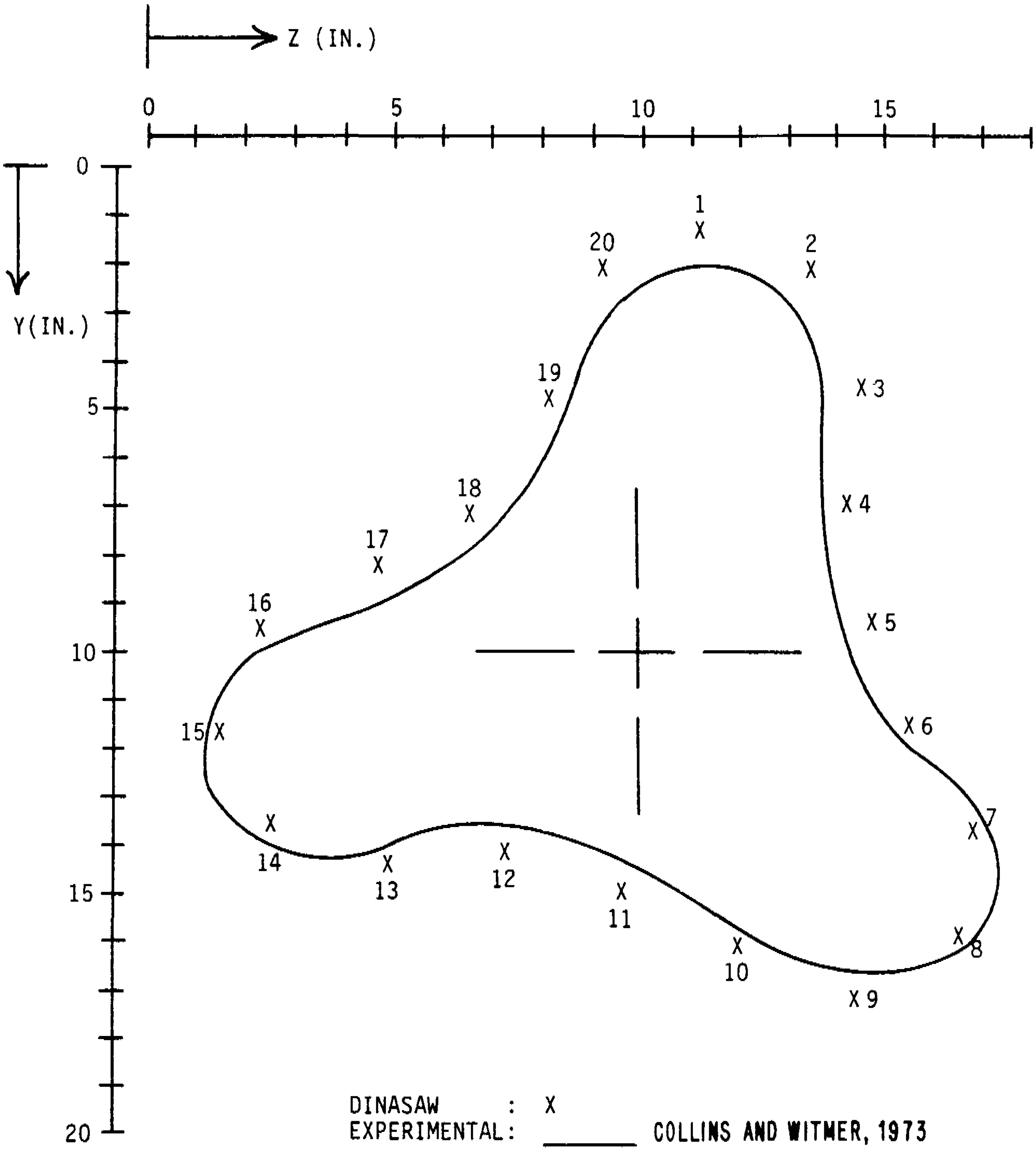
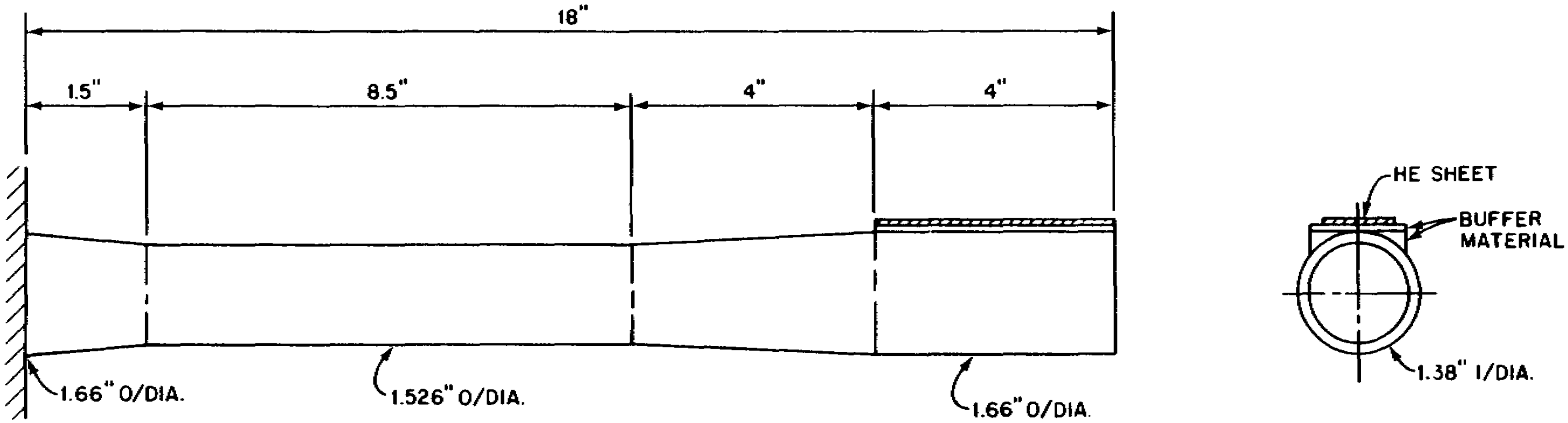


FIGURE 3A.2.1-5
SHAPE OF RING AT 700 μSEC
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT



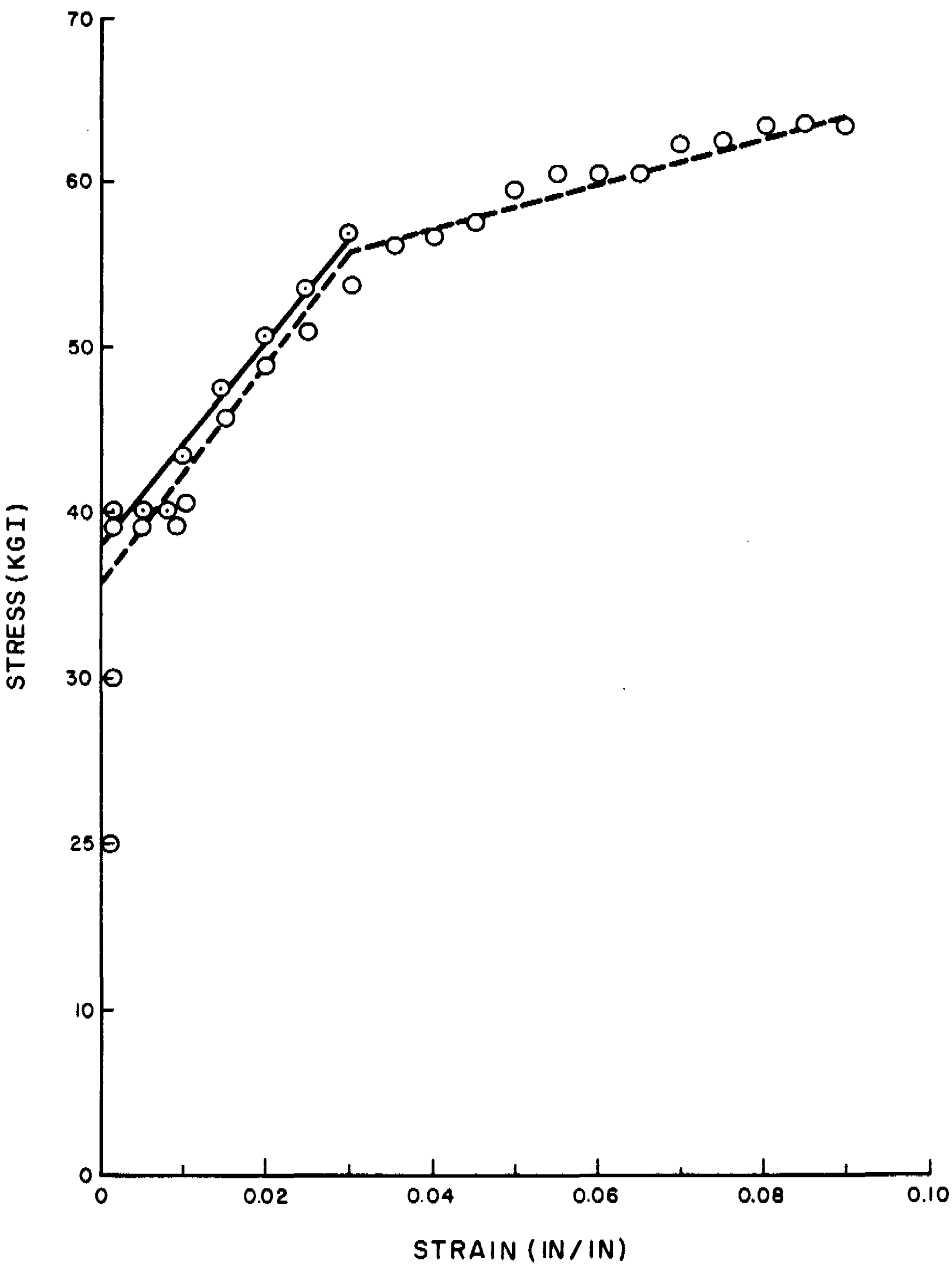
PIPE MATERIAL: A106 STEEL.

NOTE:
NOT DRAWN TO SCALE.

FIGURE 3A.2.2-1

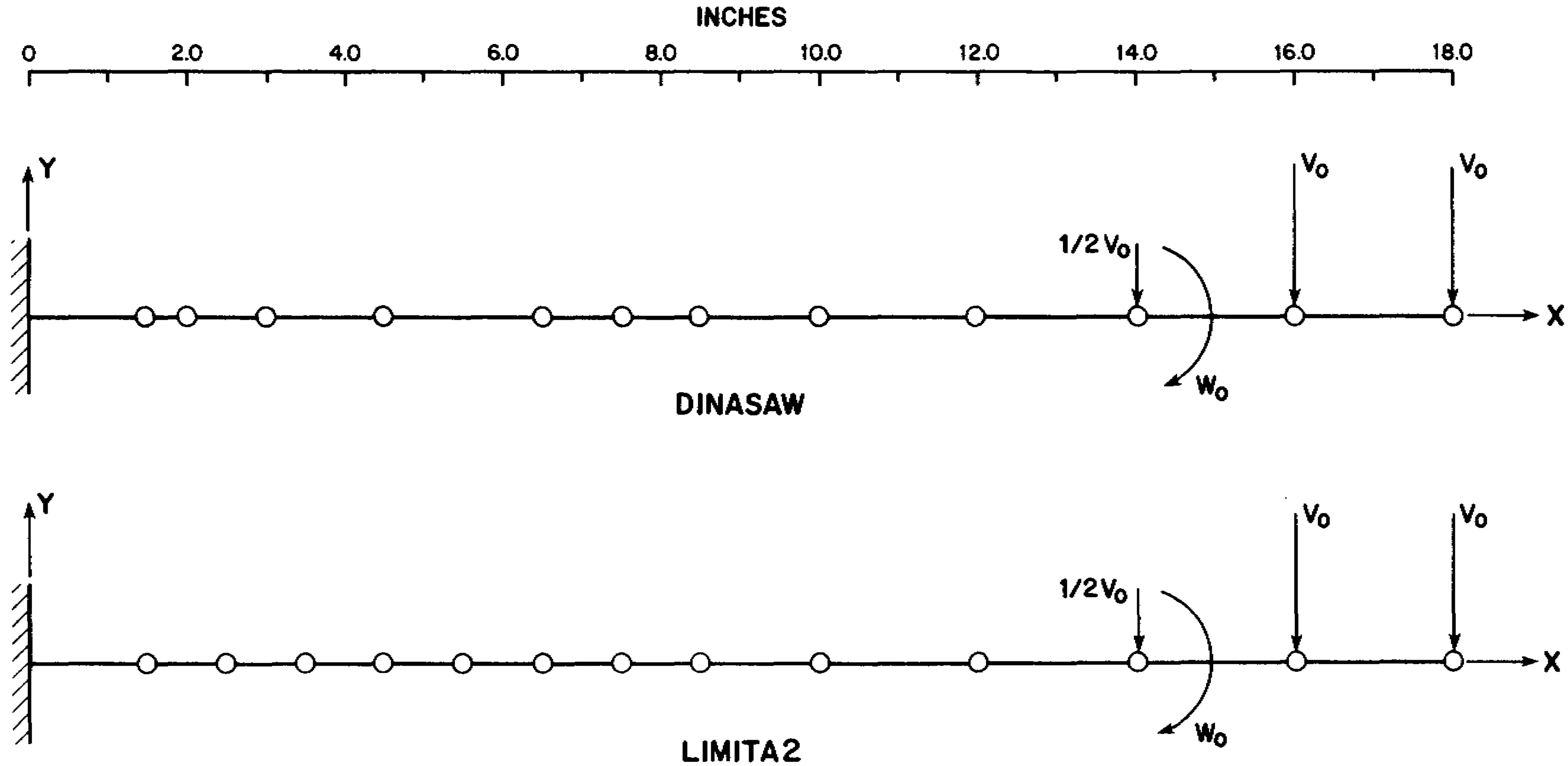
CANTILEVER PIPE
USED IN MIT EXPERIMENT

BEAVER VALLEY POWER STATION – UNIT 2
FINAL SAFETY ANALYSIS REPORT



LEGEND
⊙ + O EXPERIMENT
------ DINASAW
—— LIMITA2

FIGURE 3A.2.2-2
MIT EXPERIMENT
STRESS-STRAIN CURVES
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANLYSIS REPORT



NOTES:
V_o = 1,400 IN/SEC
W_o = 600 RAD/SEC

FIGURE 3A.2.2-3
DINASAW AND LIMITA2 LUMPED MASS MODELS
BEAVER VALLEY POWER STATION – UNIT 2
FINAL SAFETY ANALYSIS REPORT

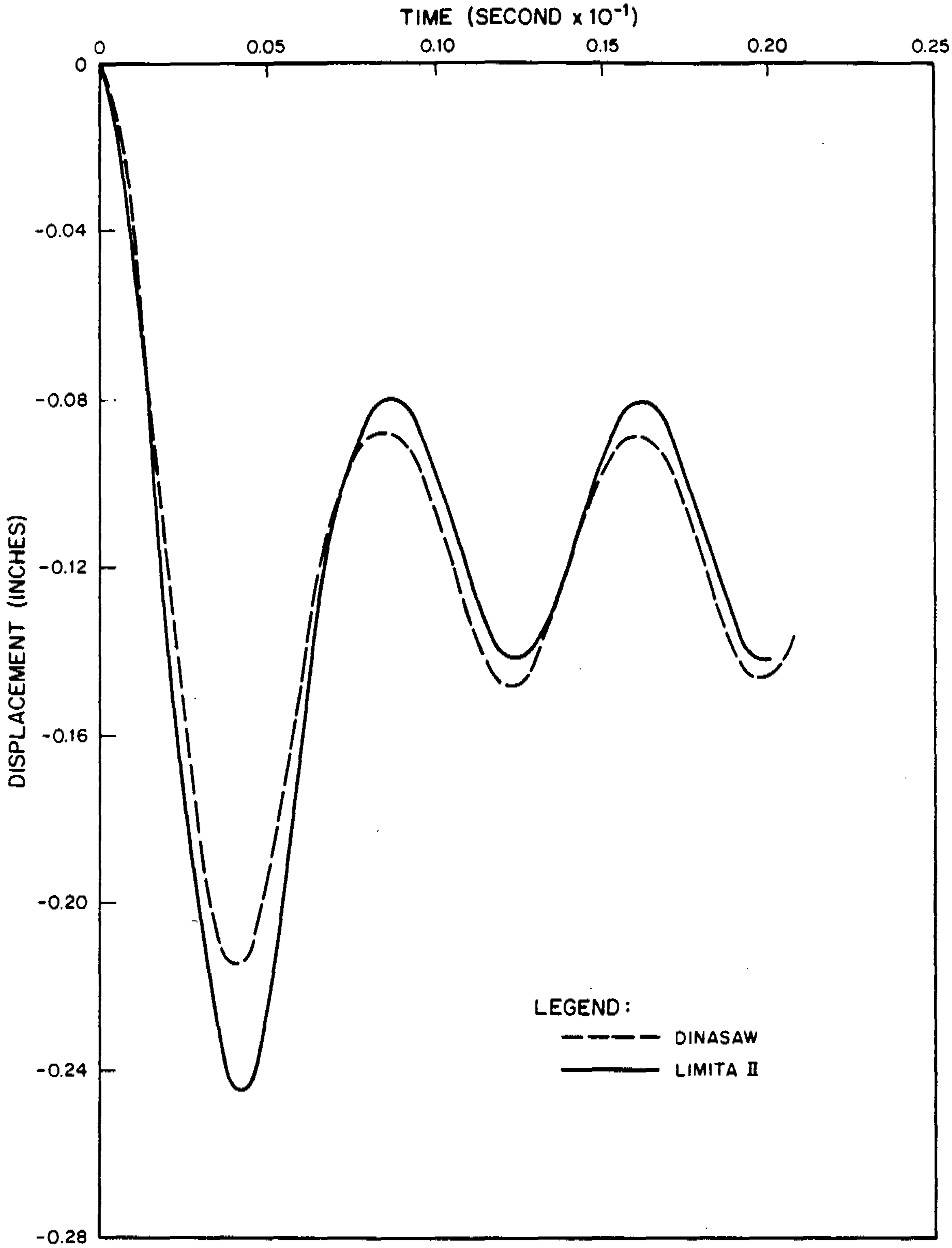


FIGURE 3A.2.2-4

TIME HISTORY PLOTS OF THE X-DISPLACEMENT AT THE FREE END

BEAVER VALLEY POWER STATION – UNIT 2
FINAL SAFETY ANALYSIS REPORT

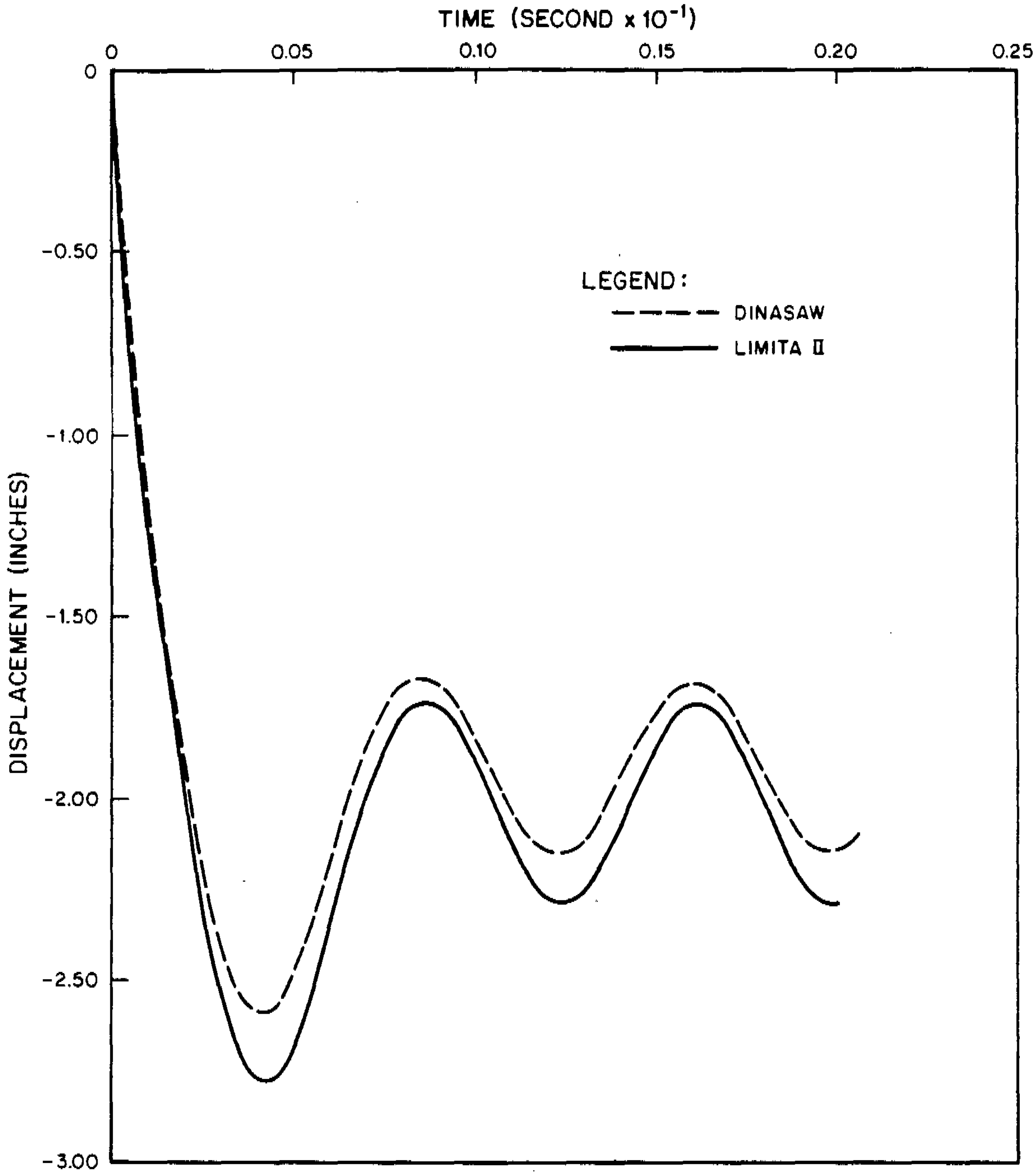
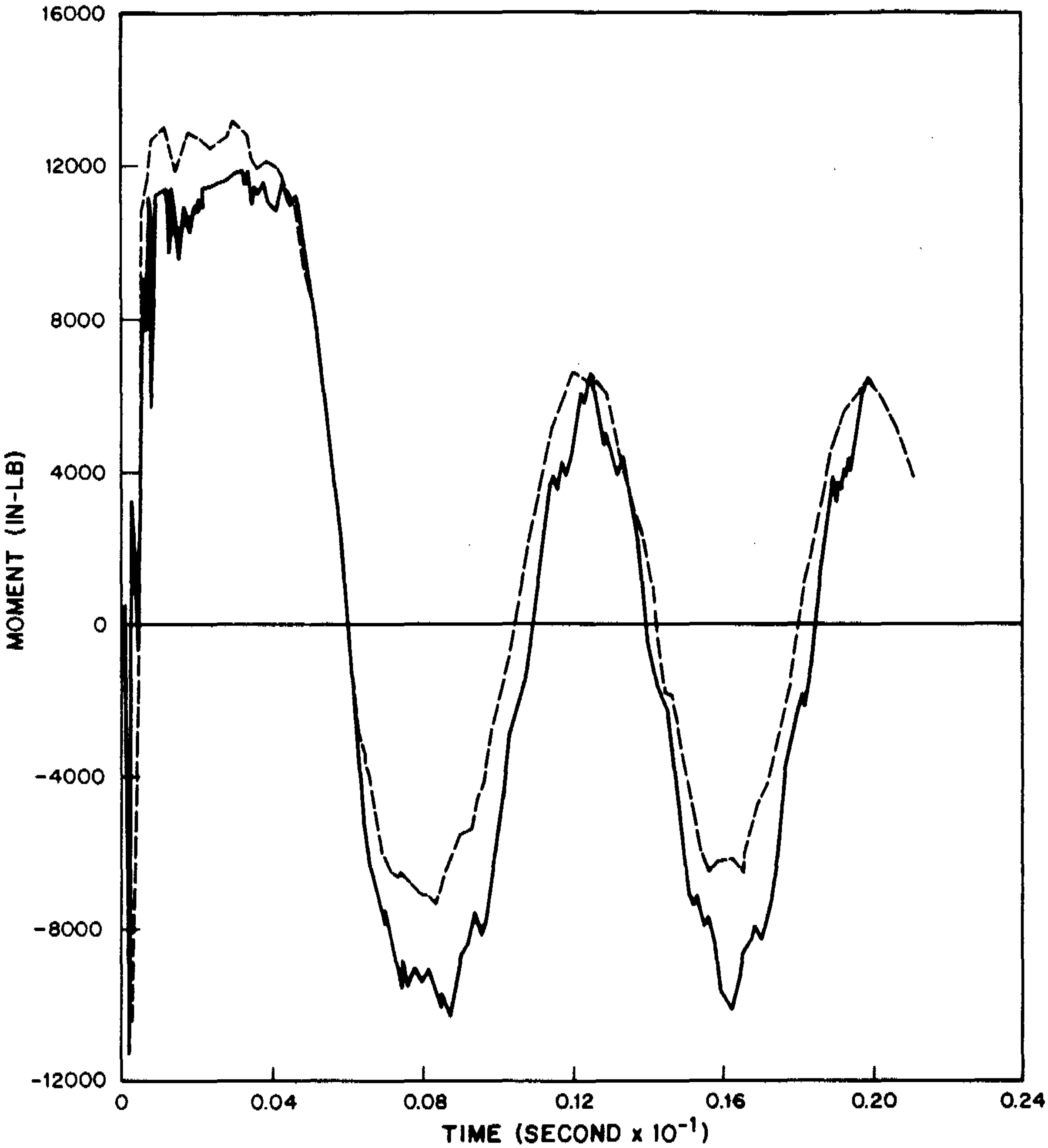


FIGURE 3A. 2.2-5

TIME HISTORY PLOTS OF THE Y-DISPLACEMENT AT THE FREE END

BEAVER VALLEY POWER STATION — UNIT 2
FINAL SAFETY ANALYSIS REPORT



LEGEND:

- - - - DINASAW

——— LIMITA II

FIGURE 3A. 2.2-6

TIME HISTORY PLOTS OF THE MOMENT REACTION AT THE CLAMPED END

BEAVER VALLEY POWER STATION—UNIT 2

FINAL SAFETY ANALYSIS REPORT

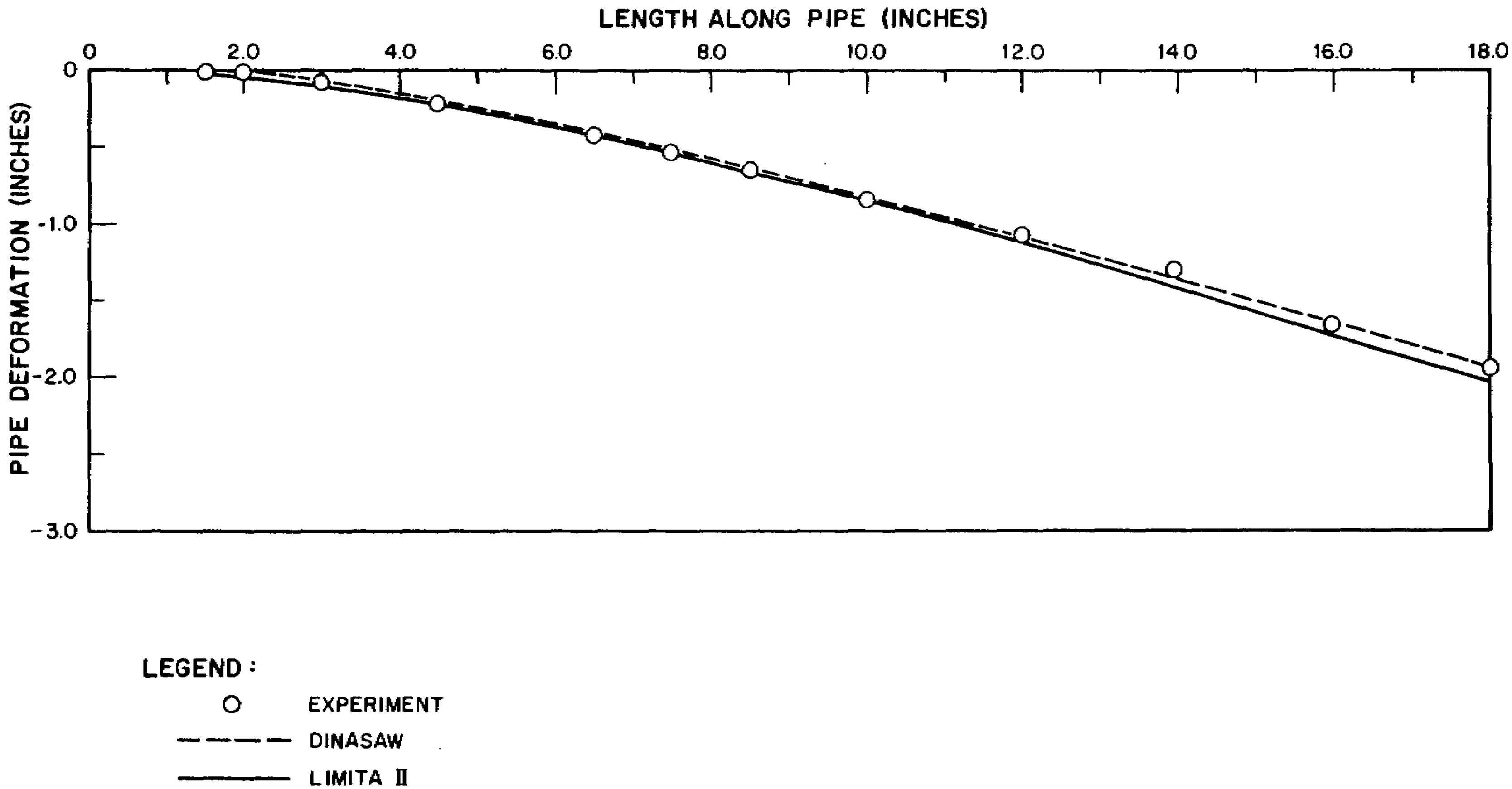
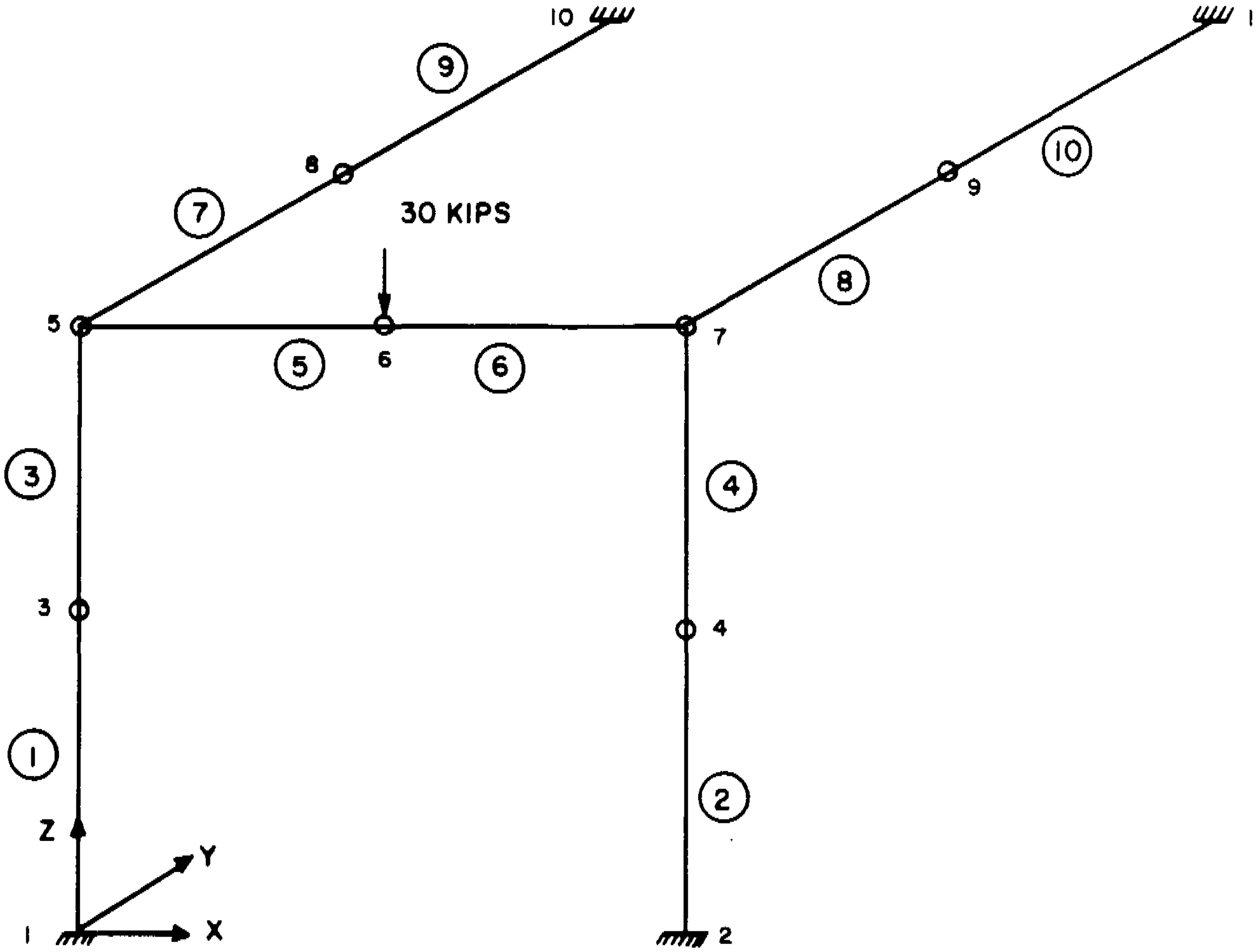


FIGURE 3A.2.2-7

PERMANENT PIPE DEFORMATIONS FOR THE MIT EXPERIMENT

BEAVER VALLEY POWER STATION – UNIT 2
FINAL SAFETY ANALYSIS REPORT



NOTE:

ALL MEMBERS W14 X 500

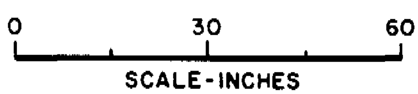


FIGURE 3A.2.3-1

LIMITA3 ELASTIC SPACE FRAME

BEAVER VALLEY POWER STATION-UNIT 2

FINAL SAFETY ANALYSIS REPORT

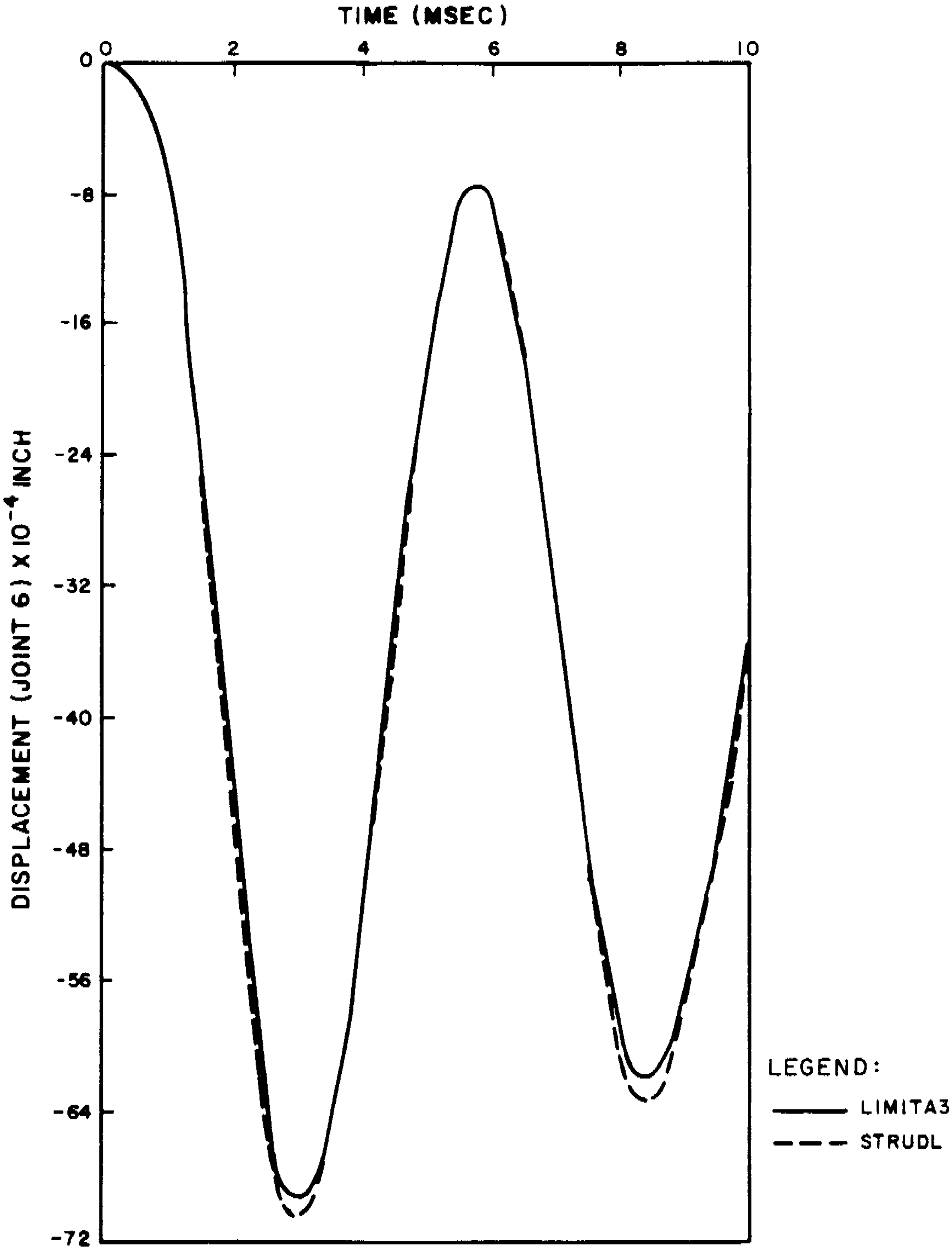


FIGURE 3A.2.3-2

LIMITA3 DISPLACEMENT TIME HISTORY AT JOINT 6

BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

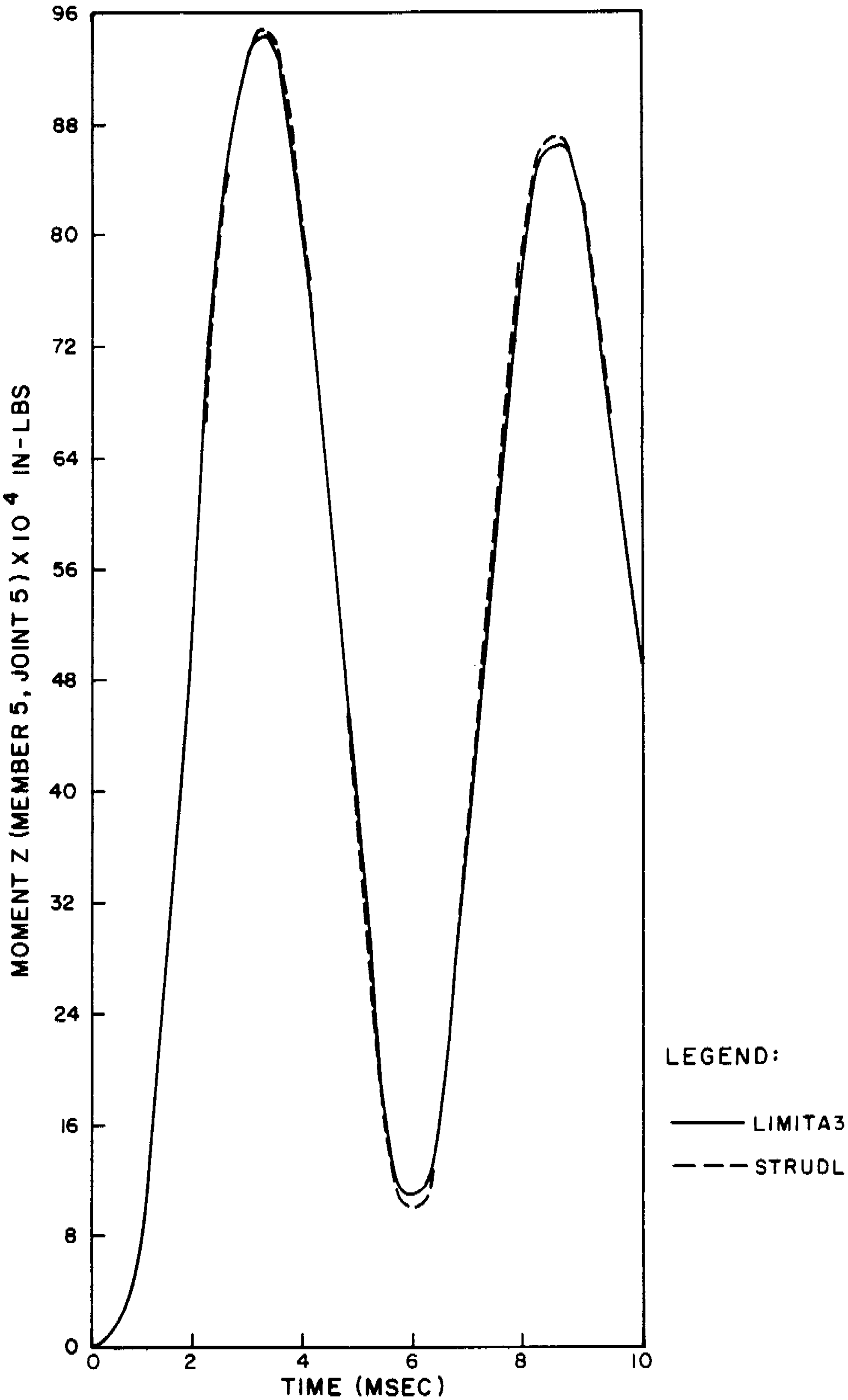
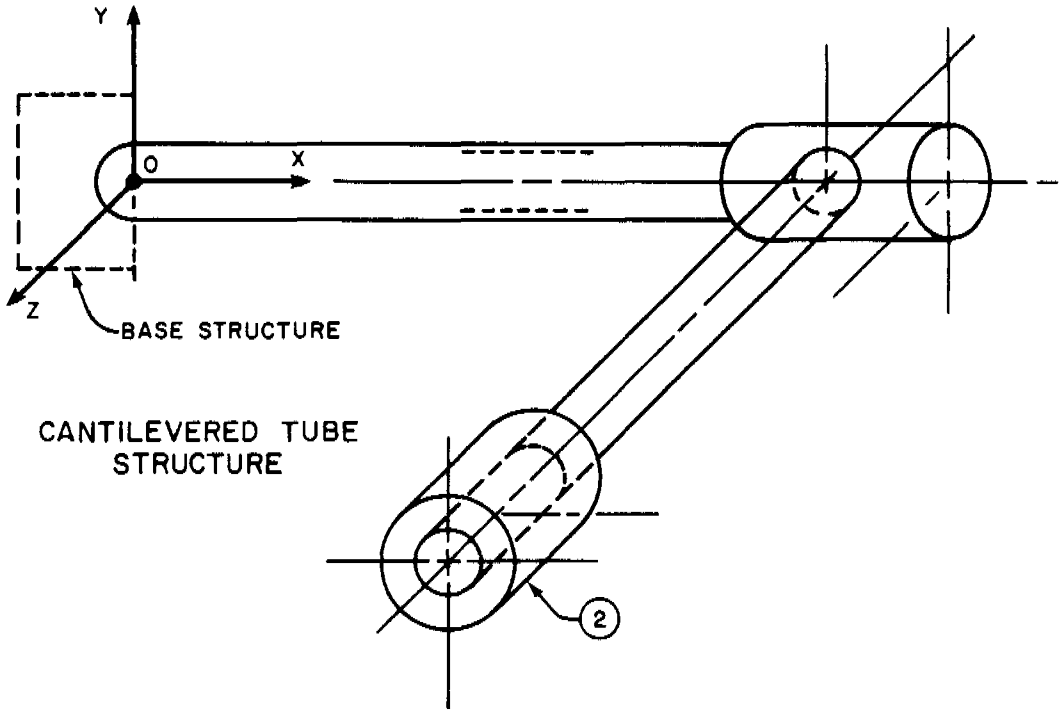
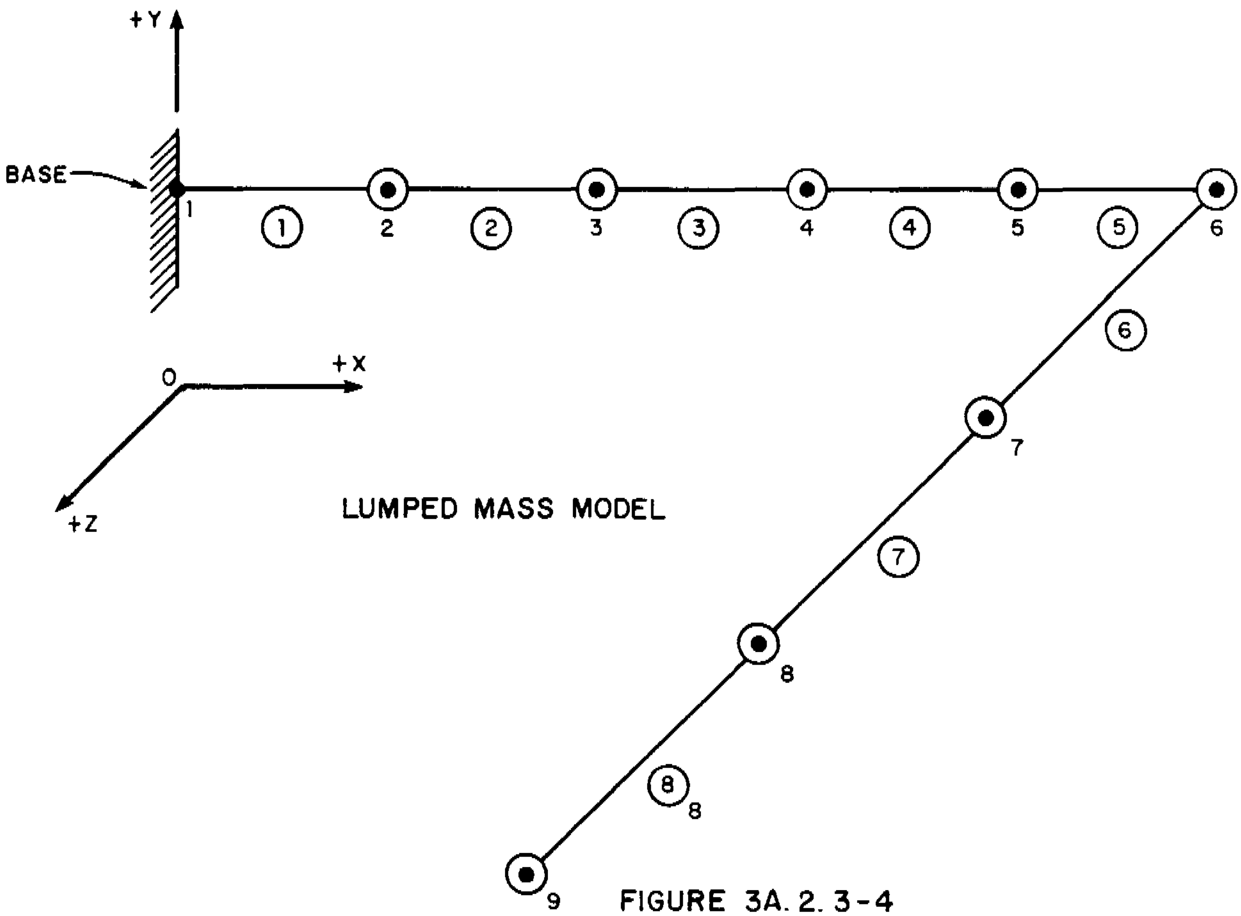


FIGURE 3A.2.3-3

LIMITA3 TIME HISTORY OF MOMENT Z IN NUMBER 5, JOINT 6

BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT





NOT TO SCALE

FIGURE 3A. 2. 3-4
STRUCTURE AND MODEL USED FOR LIMITA3 ANALYSES
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

1. APPLIED LOADS

Radial Load	P	- 1000 lb.
Circ. Moment	M_C	- 1000 in. lb.
Long. Moments	M_L	- 1000 in. lb.
Torsion Moment	M_T	- 1000 in. lb.
Shear Load	V_C	- 1000 lb.
Shear Load	V_L	- 1000 lb.

2. GEOMETRY

Vessel Thickness	T	- 0.375 in
Attachment Radius	r_o	- 8.0 in.
Vessel Radius	R_m	- 19.1875 in.

3. GEOMETRIC PARAMETERS

$\gamma = \dfrac{R_m}{T}$ 51.167

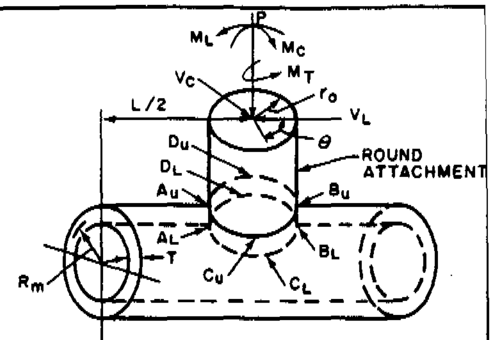
$\beta = (0.875)\dfrac{r_o}{R_m}$ 0.3648

STRESS CONCENTRATION DUE TO:

a) Membrane Load Kn - 1.0

b) Bending Load Kb - 1.0

NOTE: Enter all force values in accordance with sign convention.



FROM FIG.(*)	READ CURVES FOR	COMPUTE ABSOLUTE VALUES OF STRESS AND ENTER RESULTS	STRESSES - if load is opposite that shown, reverse signs shown							
			A_u	A_L	B_u	B_L	C_u	C_L	D_u	D_L
3C	$\dfrac{N\phi}{P/R_m} = 1.78$	$Kn\left(\dfrac{N\phi}{P/R_m}\right)\cdot\dfrac{P}{R_mT} = 247$	-247	-247	-247	-247	-247	-247	-247	-247
1C	$\dfrac{M\phi}{P} = 0.023$	$Kb\left(\dfrac{M\phi}{P}\right)\cdot\dfrac{6P}{T^2} = 981$	-981	+981	-981	+981	-981	+981	-981	+981
3A	$\dfrac{N\phi}{M_C/R_m^2\beta} = 1.41$	$Kn\left(\dfrac{N\phi}{M_C/R_m^2\beta}\right)\cdot\dfrac{M_C}{R_m^2\beta T} = 28$					-28	-28	+28	+28
1A	$\dfrac{M\phi}{M_C/R_m\beta} = 0.059$	$Kb\left(\dfrac{M\phi}{M_C/R_m\beta}\right)\cdot\dfrac{6M_C}{R_m\beta T^2} = 360$					-360	+360	+360	-360
3B	$\dfrac{N\phi}{M_L/R_m^2\beta} = 2.63$	$Kn\left(\dfrac{N\phi}{M_L/R_m^2\beta}\right)\cdot\dfrac{M_L}{R_m^2\beta T} = 52$	-52	-52	+52	+52				
1B or 1B-1	$\dfrac{M\phi}{M_L/R_m\beta} = 0.0067$	$Kb\left(\dfrac{M\phi}{M_L/R_m\beta}\right)\cdot\dfrac{6M_L}{R_m\beta T^2} = 41$	-41	+41	+41	-41				
Add algebraically for summation of ϕ stresses $\sigma\phi$			-1321	+723	-1135	+745	-1616	+1066	-840	+402
4C	$\dfrac{N_x}{P/R_m} = 4.4$	$Kn\left(\dfrac{N_x}{P/R_m}\right)\cdot\dfrac{P}{R_mT} = 612$	-612	-612	-612	-612	-612	-612	-612	-612
2C	$\dfrac{M_x}{P} = 0.0088$	$Kb\left(\dfrac{M_x}{P}\right)\cdot\dfrac{6P}{T^2} = 375$	-375	+375	-375	+375	-375	+375	-375	+375
4A	$\dfrac{N_x}{M_C/R_m^2\beta} = 5.0$	$Kn\left(\dfrac{N_x}{M_C/R_m^2\beta}\right)\cdot\dfrac{M_C}{R_m^2\beta T} = 99$					-99	-99	+99	+99
2A	$\dfrac{M_x}{M_C/R_m\beta} = 0.0235$	$Kb\left(\dfrac{M_x}{M_C/R_m\beta}\right)\cdot\dfrac{6M_C}{R_m\beta T^2} = 143$					-143	+143	+143	-143
4B	$\dfrac{N_x}{M_L/R_m^2\beta} = 1.52$	$Kn\left(\dfrac{N_x}{M_L/R_m^2\beta}\right)\cdot\dfrac{M_L}{R_m^2\beta T} = 30$	-30	-30	+30	+30				
2B or 2B-1	$\dfrac{M_x}{M_L/R_m\beta} = 0.01$	$Kb\left(\dfrac{M_x}{M_L/R_m\beta}\right)\cdot\dfrac{6M_L}{R_m\beta T^2} = 61$	-61	+61	+61	-61				
Add algebraically for summation of X stresses σ_x			-1078	-206	-896	-268	-1229	-193	-745	-281
Shear stress due to torsion M_T		$\tau\phi_x = \tau_x\phi = \dfrac{M_T}{2\pi r_o^2 T} = 7$	+7	+7	+7	+7	+7	+7	+7	+7
Shear stress due to load V_C		$\tau_x\phi = \dfrac{V_C}{\pi r_o T} = 106$	+106	+106	-106	-106				
Shear stress due to load V_L		$\tau_x\phi = \dfrac{V_L}{\pi r_o T} = 106$					-106	-106	+106	+106
Add algebraically for summation of shear stresses τ			+113	+113	-99	-99	-99	-99	+113	+113

NOTE: WICHMAN et al; 1965

FIGURE 3A.2.5-1

SUMMARY OF MANUAL CALCULATIONS FOR LOCAL STRESSES IN CYLINDRICAL SHELLS

BEAVER VALLEY POWER STATION-UNIT 2

FINAL SAFETY ANALYSIS REPORT

1. APPLIED LOADS

Radial Load	P	- 1000 lb.
Circ. Moment	M_C	- 1000 in. lb.
Long. Moments	M_L	- 1000 in. lb.
Torsion Moment	M_T	- 1000 in. lb.
Shear Load	V_C	- 1000 lb.
Shear Load	V_L	- 1000 lb.

GEOMETRY

Vessel Thickness	T	- 0.375 in
Attachment Radius	r_o	- 8.0 in.
Vessel Radius	R_m	- 19.1875 in.

3. GEOMETRIC PARAMETERS

$\gamma = \frac{R_m}{T}$ 51.167

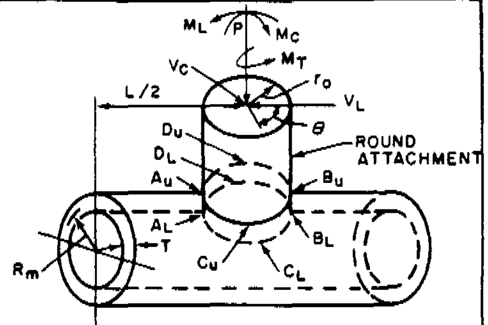
$\beta = (0.875)\frac{r_o}{R_m}$ 0.3648

STRESS CONCENTRATION DUE TO:

a) Membrane Load Kn - 1.0

b) Bending Load Kb - 1.0

NOTE: Enter all force values in accordance with sign convention.



CYLINDRICAL SHELL

FROM FIG. (*)	READ CURVES FOR	COMPUTE ABSOLUTE VALUES OF STRESS AND ENTER RESULTS	STRESSES - if load is opposite that shown, reverse signs shown							
			A_u	A_L	B_u	B_L	C_u	C_L	D_u	D_L
3C	$\frac{N\phi}{P/R_m} = 1.77$	$Kn\left(\frac{N\phi}{P/R_m}\right)\cdot\frac{P}{R_m T} = 246$	-246	-246	-246	-246	-246	-246	-246	-246
1C	$\frac{M\phi}{P} = 0.022$	$Kb\left(\frac{M\phi}{P}\right)\cdot\frac{6P}{T^2} = 937$	-937	+937	-937	+937	-937	+937	-937	+937
3A	$\frac{N\phi}{M_C/R_m^2\beta} = 1.41$	$Kn\left(\frac{N\phi}{M_C/R_m^2\beta}\right)\cdot\frac{M_C}{R_m^2\beta T} = 28$					-28	-28	+28	+28
1A	$\frac{M\phi}{M_C/R_m\beta} = 0.059$	$Kb\left(\frac{M\phi}{M_C/R_m\beta}\right)\cdot\frac{6M_C}{R_m\beta T^2} = 357$					-357	+357	+357	-357
3B	$\frac{N\phi}{M_L/R_m^2\beta} = 2.62$	$Kn\left(\frac{N\phi}{M_L/R_m^2\beta}\right)\cdot\frac{M_L}{R_m^2\beta T} = 52$	-52	-52	+52	+52				
1B or 1B-1	$\frac{M\phi}{M_L/R_m\beta} = 0.0066$	$Kb\left(\frac{M\phi}{M_L/R_m\beta}\right)\cdot\frac{6M_L}{R_m\beta T^2} = 40$	-40	+40	+40	-40				
Add algebraically for summation of ϕ stresses $\sigma\phi$			-1275	+1679	-1091	+703	-1568	+1020	-798	+362
4C	$\frac{N_x}{P/R_m} = 4.4$	$Kn\left(\frac{N_x}{P/R_m}\right)\cdot\frac{P}{R_m T} = 614$	-614	-614	-614	-614	-614	-614	-614	-614
2C	$\frac{M_x}{P} = 0.0087$	$Kb\left(\frac{M_x}{P}\right)\cdot\frac{6P}{T^2} = 373$	-373	+373	-373	+373	-373	+373	-373	+373
4A	$\frac{N_x}{M_C/R_m^2\beta} = 5.0$	$Kn\left(\frac{N_x}{M_C/R_m^2\beta}\right)\cdot\frac{M_C}{R_m^2\beta T} = 99$					-99	-99	+99	+99
2A	$\frac{M_x}{M_C/R_m\beta} = 0.0233$	$Kb\left(\frac{M_x}{M_C/R_m\beta}\right)\cdot\frac{6M_C}{R_m\beta T^2} = 142$					-142	+142	+142	-142
4B	$\frac{N_x}{M_L/R_m^2\beta} = 1.53$	$Kn\left(\frac{N_x}{M_L/R_m^2\beta}\right)\cdot\frac{M_L}{R_m^2\beta T} = 30$	-30	-30	+30	+30				
2B or 2B-1	$\frac{M_x}{M_L/R_m\beta} = 0.01$	$Kb\left(\frac{M_x}{M_L/R_m\beta}\right)\cdot\frac{6M_L}{R_m\beta T^2} = 62$	-62	+62	+62	-62				
Add algebraically for summation of X stresses σ_x			-1079	-209	-895	-280	-1228	-198	-746	-284
Shear stress due to torsion M_T		$\tau\phi_x = \tau_x\phi = \frac{M_T}{2\pi r_o^2 T} = 7$	+7	+7	+7	+7	+7	+7	+7	+7
Shear stress due to load V_C		$\tau_x\phi = \frac{V_C}{\pi r_o T} = 106$	+106	+106	-106	-106				
Shear stress due to load V_L		$\tau_x\phi = \frac{V_L}{\pi r_o T} = 106$					-106	-106	+106	+106
Add algebraically for summation of shear stresses τ			+113	+113	-99	-99	-99	-99	+113	+113

NOTE: WICHMAN et al; 1965

FIGURE 3A.2.5-2

SUMMARY OF COMPUTER CALCULATIONS FOR LOCAL STRESSES IN CYLINDRICAL SHELLS

BEAVER VALLEY POWER STATION-UNIT 2

FINAL SAFETY ANALYSIS REPORT

1. Applied Loads*

Radial Load,	P = 1.0
Shear Load,	V_1 = 1.0
Shear Load,	V_2 = 1.0
Overturning Moment,	M_1 = 1.0
Overturning Moment,	M_2 = 1.0
Torsional Moment,	M_T = 1.0

2. Geometry

Vessel Thickness,	T = .25
Vessel Mean Radius,	R_m = 30.25
Nozzle Thickness,	t = .25
Nozzle Mean Radius,	r_m = 1.25
Nozzle Outside Radius,	r_o = 1.375

3. Geometric Parameters

$\gamma = \dfrac{r_m}{t} = 5.0$

$\rho = \dfrac{T}{t} = 1.0$

$U = \dfrac{r_o}{\sqrt{R_m T}} = 0.5$

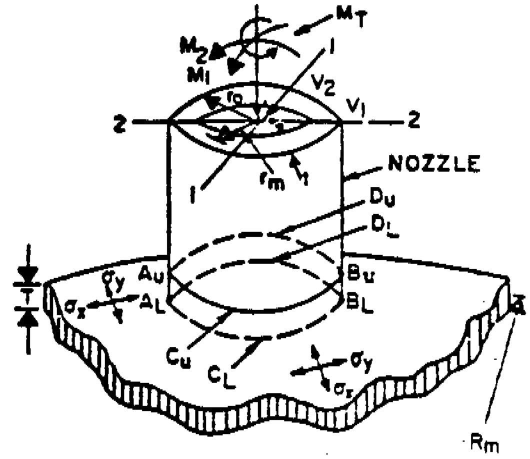
4. Stress Concentration Factors

due to:

membrane load, K_n = 1.0

bending load, K_b = 1.0

NOTE: Enter all force values in accordance with sign convention



HOLLOW ATTACHMENT

From Fig.	Read curves for	Compute absolute values of stress and enter result –	STRESSES – if load is opposite that shown, reverse signs shown							
			Au	AL	Bu	BL	Cu	CL	Du	DL
SP-1 to 10	$\frac{N_x T}{P}$ = .087	$K_n \left(\frac{N_x T}{P}\right) \cdot \frac{P}{T^2}$ =	-1.392	-1.392	-1.392	-1.392	-1.392	-1.392	-1.392	-1.392
↓	$\frac{M_x}{P}$ = .087	$K_b \left(\frac{M_x}{P}\right) \cdot \frac{6P}{T^2}$ =	-8.352	+8.352	-8.352	- 8.352	-8.352	+8.352	-8.352	+8.352
SM-1 to 10	$\frac{N_x T\sqrt{R_m T}}{M_1}$ = .135	$K_n \left(\frac{N_x T\sqrt{R_m T}}{M_1}\right) \cdot \frac{M_1}{T^2\sqrt{R_m T}}$ =					- .785	- .785	+ .785	+ .785
	$\frac{M_x\sqrt{R_m T}}{M_1}$ = .220	$K_b \left(\frac{M_x\sqrt{R_m T}}{M_1}\right) \cdot \frac{6M_1}{T^2\sqrt{R_m T}}$ =					-7.680	- 7.680	+ 7.680	-7.680
	$\frac{N_x T\sqrt{R_m T}}{M_2}$ = .135	$K_n \left(\frac{N_x T\sqrt{R_m T}}{M_2}\right) \cdot \frac{M_2}{T^2\sqrt{R_m T}}$ =	-.785	- .785	+.785	+ .785				
↓	$\frac{M_x\sqrt{R_m T}}{M_2}$ = .220	$K_b \left(\frac{M_x\sqrt{R_m T}}{M_2}\right) \cdot \frac{6M_2}{T^2\sqrt{R_m T}}$ =	-7.680	-7.680	+7.680	-7.680				
Add algebraically for summation of σ_x =										
SP-1 to 10	$\frac{N_y T}{P}$ = .139	$K_n \left(\frac{N_y T}{P}\right) \cdot \frac{P}{T^2}$ =	-2.224	-2.224	-2.224	-2.224	-2.224	-2.224	-2.224	-2.224
↓	$\frac{M_y}{P}$ = .037	$K_b \left(\frac{M_y}{P}\right) \cdot \frac{6P}{T^2}$ =	-3.552	+3.552	-3.552	+3.552	-3.552	+3.552	-3.552	+3.552
SM-1 to 10	$\frac{N_y T\sqrt{R_m T}}{M_1}$ = .146	$K_n \left(\frac{N_y T\sqrt{R_m T}}{M_1}\right) \cdot \frac{M_1}{T^2\sqrt{R_m T}}$ =					- .849	- .849	- .849	.849
	$\frac{M_y\sqrt{R_m T}}{M_1}$ = .092	$K_b \left(\frac{M_y\sqrt{R_m T}}{M_1}\right) \cdot \frac{6M_1}{T^2\sqrt{R_m T}}$ =					-3.212	-3.212	-3.212	-3.212
	$\frac{N_y T\sqrt{R_m T}}{M_2}$ = .146	$K_n \left(\frac{N_y T\sqrt{R_m T}}{M_2}\right) \cdot \frac{M_2}{T^2\sqrt{R_m T}}$ =	-.849	- .849	.849	.849				
↓	$\frac{M_y\sqrt{R_m T}}{M_2}$ = .092	$K_b \left(\frac{M_y\sqrt{R_m T}}{M_2}\right) \cdot \frac{6M_2}{T^2\sqrt{R_m T}}$ =	-3.212	3.212	+3.212	3.212				
Add algebraically for summation of σ_y =										
Shear stress due to load, V_1		$\tau_1 = \frac{V}{\pi r_o T} = \frac{1.0}{\pi(1.375)(.25)}$					.926	.926	+ .926	+ [illegible]
Shear stress due to load, V_2		$\tau_2 = \frac{V}{\pi r_o T} = \frac{1.0}{\pi(1.3\ldots)}$	-.926	- .926	.426	- 476				
Shear stress due to torsion, M_T		$\tau_1 = \tau_2 = \frac{M}{2\pi r_o^2 T}$ =	+ .337	+ .337	+ .337	+ .337	+ .337	+ [illegible]	+ [illegible]	+ .337

FIGURE 3A.2.5-3

SUMMARY OF MANUAL CALCULATIONS FOR LOCAL STRESSES IN SPHERICAL SHELLS

BEAVER VALLEY POWER STATION - UNIT 2

FINAL SAFETY ANALYSIS REPORT

PROBLEM NO. 1 - STRESSES IN SHELL AT NOZZLE ASSUMING SHELL THICKNESS IS EQUAL TO INSERT THICKNESS

DIMENSIONLESS CURVE FACTORS BASED UPON 3A ABOVE

NY/P	MY/P	NY/M1	MY/M1	NY/M2	MY/M2	NX/P	MX/P	NX/M1	MX/M1	NX/M2	MX/M2
0.140894	0.033549	0.145850	0.096415	0.145850	0.096415	0.086875	0.086470	0.127766	0.214108	0.127766	0.214108

STRESS / LOCATION		AU	AL	BU	BL	CU	CL	DU	DL
N-X	P	-1.390	-1.390	-1.390	-1.390	-1.390	-1.390	-1.390	-1.390
M-X	P	-8.301	8.301	-8.301	8.301	-8.301	8.301	-8.301	8.301
N-X	MC	XXXXXXXXXX	XXXXXXXXXX	XXXXXXXXXX	XXXXXXXXXX	-0.743	-0.743	0.743	0.743
M-X	MC	XXXXXXXXXX	XXXXXXXXXX	XXXXXXXXXX	XXXXXXXXXX	-7.474	7.474	7.474	-7.474
N-X	ML	-0.743	-0.743	0.743	0.743	XXXXXXXXXX	XXXXXXXXXX	XXXXXXXXXX	XXXXXXXXXX
M-X	ML	-7.474	7.474	7.474	-7.474	XXXXXXXXXX	XXXXXXXXXX	XXXXXXXXXX	XXXXXXXXXX
N-PHI	P	-2.254	-2.254	-2.254	-2.254	-2.254	-2.254	-2.254	-2.254
M-PHI	P	-3.221	3.221	-3.221	3.221	-3.221	3.221	-3.221	3.221
N-PHI	MC	XXXXXXXXXX	XXXXXXXXXX	XXXXXXXXXX	XXXXXXXXXX	-0.849	-0.849	0.849	0.849
M-PHI	MC	XXXXXXXXXX	XXXXXXXXXX	XXXXXXXXXX	XXXXXXXXXX	-3.366	3.366	3.366	-3.366
N-PHI	ML	-0.849	-0.849	0.849	0.849	XXXXXXXXXX	XXXXXXXXXX	XXXXXXXXXX	XXXXXXXXXX
M-PHI	ML	-3.366	3.366	3.366	-3.366	XXXXXXXXXX	XXXXXXXXXX	XXXXXXXXXX	XXXXXXXXXX
SHEAR STRESS	VC	0.926	0.926	-0.926	-0.926	XXXXXXXXXX	XXXXXXXXXX	XXXXXXXXXX	XXXXXXXXXX
SHEAR STRESS	VL	XXXXXXXXXX	XXXXXXXXXX	XXXXXXXXXX	XXXXXXXXXX	-0.926	-0.926	0.926	0.926
SHEAR STRESS	MT	0.337	0.337	0.337	0.337	0.337	0.337	0.337	0.337

FIGURE 3A.2.5-4
SUMMARY OF COMPUTER CALCULATIONS FOR LOCAL STRESSES IN SPHERICAL SHELLS
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

VESSEL THICKNESS (h_s) = 0.375 IN.

NOZZLE THICKNESS (h_c) = 0.365 IN.

VESSEL INNER RADIUS (R_i) = 23.625 IN.

VESSEL MEAN RADIUS (R_m) = 23.8125 IN.

NOZZLE INNER RADIUS (r_i) = 5.010 IN.

NOZZLE MEAN RADIUS (a) = 5.1925 IN.

INTERNAL PRESSURE (P) = 50 PSI

FIGURE 3A.2.5-5

SAMPLE INPUT DATA-NOZZLE

BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

MERIDIONAL STRESSES (S_1)

LOCATION	Σ OF STRESSES	HAND CALC.	TEST PROBLEM 2*
AU	$\sigma_{\phi o} + \sigma_{AV}$	4842.688	4844.043
AL	$\sigma_{\phi I} + \sigma_{AV}$	-3694.314	-3695.641
BU	$\sigma_{\phi o} + \sigma_{AV}$	4842.688	4844.043
BL	$\sigma_{\phi I} + \sigma_{AV}$	-3694.314	-3695.641
CU	$\sigma_{\phi o} + \sigma_{HV}$	6417.688	6419.043
CL	$\sigma_{\phi I} + \sigma_{HV}$	-2119.314	-2120.641
DU	$\sigma_{\phi o} + \sigma_{HV}$	6417.688	6419.043
DL	$\sigma_{\phi I} + \sigma_{HV}$	-2119.314	-2120.641

HOOP STRESSES (S_2)

LOCATION	Σ OF STRESSES	HAND CALC.	TEST PROBLEM 2*
AU	$\sigma_{\theta o} + \sigma_{HV}$	6660.645	6660.332
AL	$\sigma_{\theta I} + \sigma_{HV}$	4099.545	4098.430
BU	$\sigma_{\theta o} + \sigma_{HV}$	6660.645	6660.332
BL	$\sigma_{\theta I} + \sigma_{HV}$	4099.545	4098.430
CU	$\sigma_{\theta o} + \sigma_{AV}$	5085.645	5085.332
CL	$\sigma_{\theta I} + \sigma_{AV}$	2524.545	2523.430
DU	$\sigma_{\theta o} + \sigma_{AV}$	5085.645	5085.332
DL	$\sigma_{\theta I} + \sigma_{AV}$	2524.545	2523.430

* TEST PROBLEM 2 COMPUTER RESULTS (RUN 001)

FIGURE 3A.2.5-6
COMPARISONS OF MERIDIONAL
AND HOOP STRESSES
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

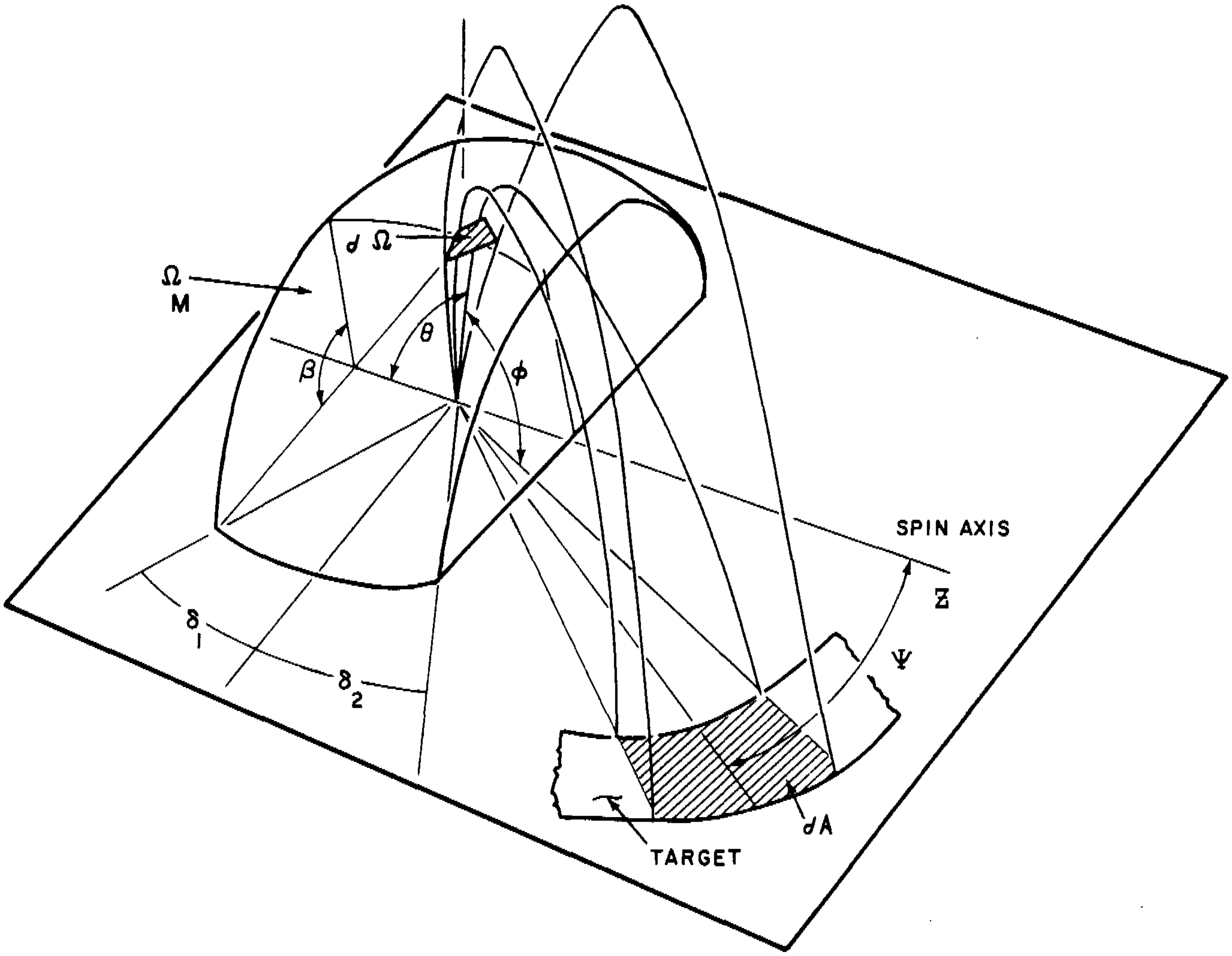


FIGURE 3A. 2.8-1

NOMENCLATURE FOR MISSILE PROGRAM

BEAVER VALLEY POWER STATION-UNIT 2

FINAL SAFETY ANALYSIS REPORT

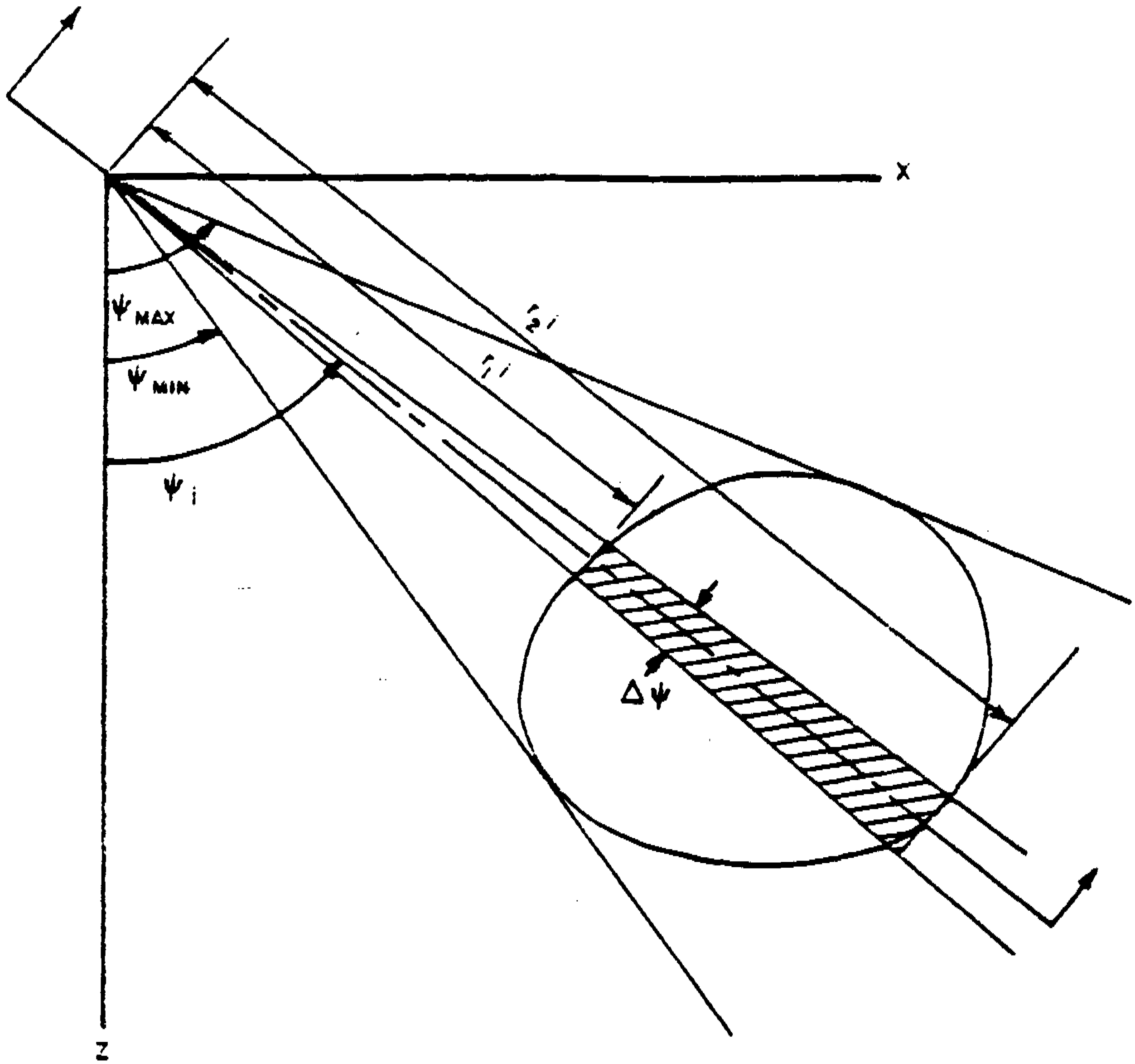


FIGURE 3A.2.8-2
TOP VIEW OF IDEALIZED TARGET
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

E

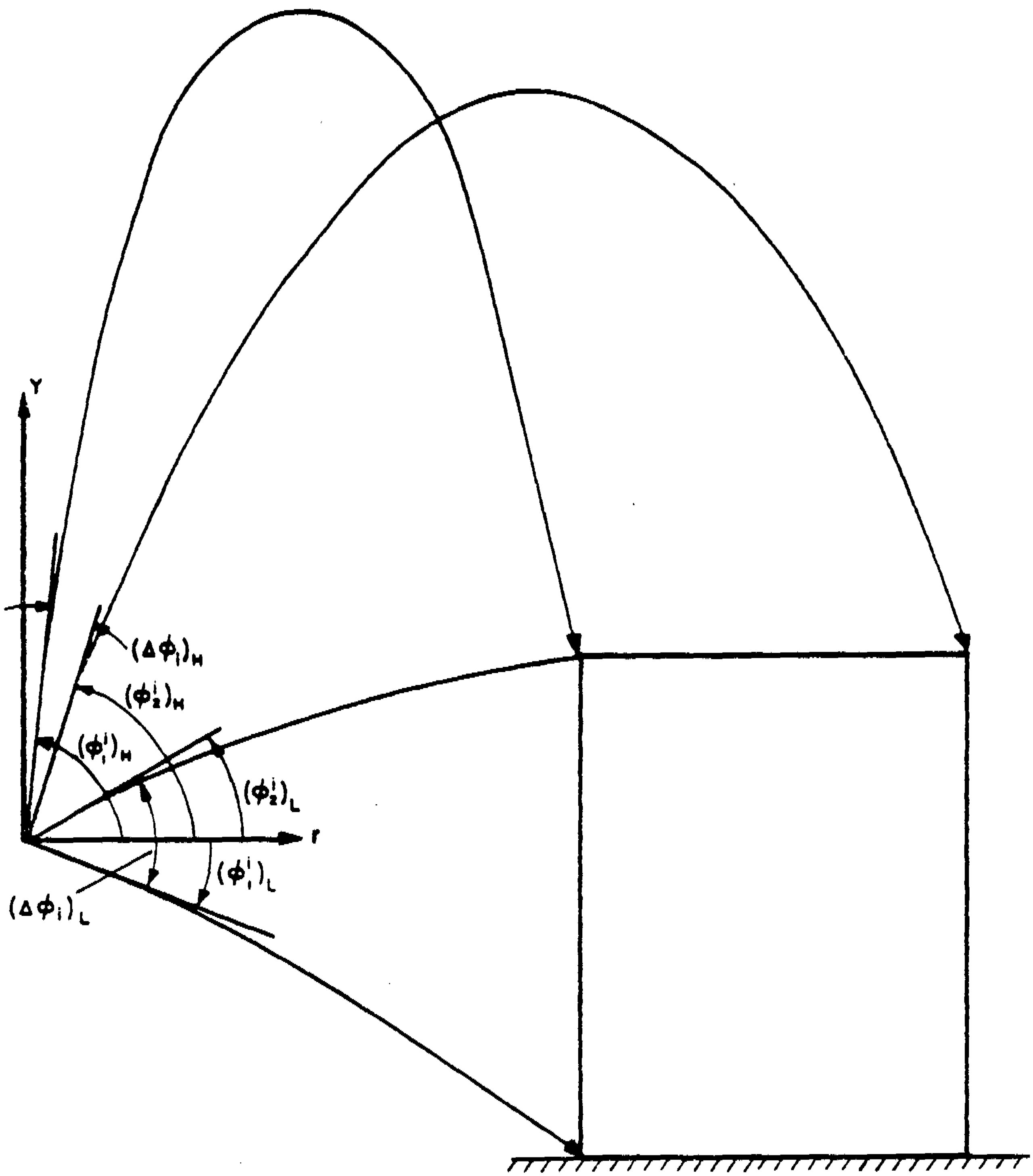
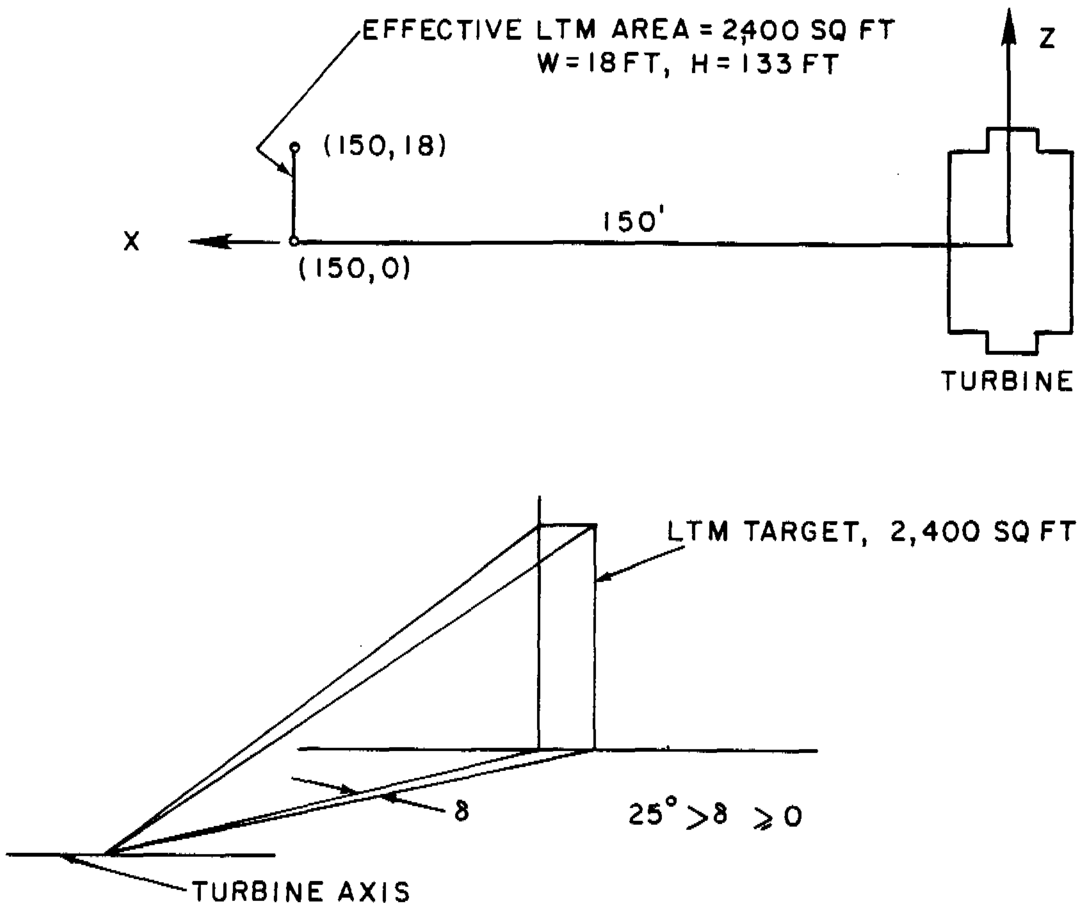


FIGURE 3A.2.8-3
SIDE VIEW OF IDEALIZED TARGET
BEAVER VALLEY POWER STATION–UNIT 2
FINAL SAFETY ANALYSIS REPORT



NOTE: BUSH 1973

FIGURE 3A.2.8-4

TARGET FOR LOW TRAJECTORY MISSILE

BEAVER VALLEY POWER STATION—UNIT 2
FINAL SAFETY ANALYSIS REPORT

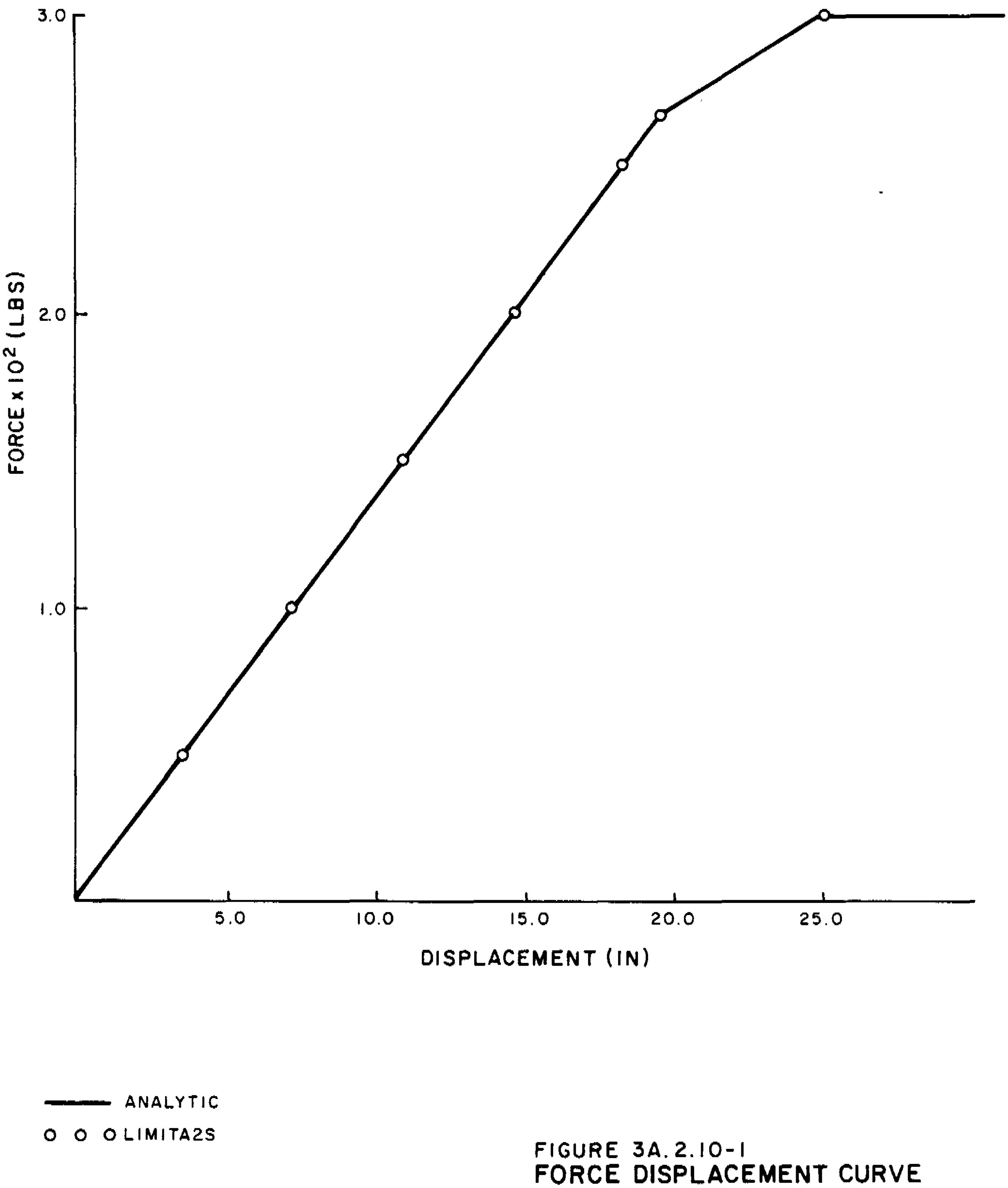


FIGURE 3A.2.10-1
FORCE DISPLACEMENT CURVE
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

3A.3 PIPING SYSTEMS

The following computer programs are used for the analysis of Seismic Category I piping systems:

1. NUPIPE-SW - Linear Elastic Analysis of 3-D Piping System Subjected to Thermal, Static, and Dynamic Loads,
2. HTLOAD - Thermal Transient Analysis,
3. PITRUST - Local Stress Analysis at Junction of Two Cylindrical Vessels,
4. PILUG - Local Stress Analysis at Junction of Lug with Pipe for a Cylindrical Vessel,
5. PITRIFE - Finite Element Analysis of Integral Weld Attachment,
6. STEHAM - Steamhammer Transient Analysis,
7. WATHAM - Waterhammer Transient Analysis,
8. WATSLUG - Water Slug Transient Analysis,
9. ELBOW - Detailed Stresses in Elbows,
10. BENDCORD - Bend Coordinate Program,
11. CDC - BASEPLATE II,
12. PSPECTRA - Data Generator for NUPIPE,
13. ANSYS - Engineering Analysis System,
14. STRUDL II - Structural Analysis Program,
15. STRUDL-SW - Structural Analysis Program,
16. NUPIPE II - Linear Elastic Analysis of 3-D Piping System Subjected to Thermal, Static, and Dynamic Loads,
17. BIP - Baseplate Information Processor,
18. APE - Anchor Plate Evaluation,
19. CHPLOT - Data Plotting Program,
20. BAP - Baseplate Analysis Processor,
21. TRUNPIPE - Analysis of Local Pipe Stresses at Trunnion Attachments,

22. LUGAPIPE - Analysis of Local Pipe Stresses at Lug Attachments,

23. CCN318 - ASME III Code Case N-318 Analysis of Rectangular Attachments to Pipe, and

24. CCN392 - ASME III Code Case N-392 Analysis of Circular Attachments to Pipe.

3A.3.1 NUPIPE-SW

3A.3.1.1 General Description

The NUPIPE-SW (SWEC 1982) piping program performs a linear elastic analysis of three-dimensional piping systems subjected to thermal, static, and dynamic loads. It utilizes the finite element method of analysis.

NUPIPE-SW handles all loading conditions required for complete nuclear piping analyses. A given piping configuration may be analyzed successively for a number of static and dynamic load conditions in a single computer run. Separate load cases, such as thermal expansion and anchor displacements, may be combined to form additional analysis cases. The piping deadload analysis considers both distributed weight properties of the piping and any added concentrated weights.

A lumped mass model of the system is used for all dynamic analysis, and both translational and rotational degrees of freedom may be considered. Location of lumped masses and degrees of freedom at each mass point are preselected by the analyst. The program automatically computes values of translational lumped masses.

Program input consists basically of program control, piping configuration description, and load specification information. Output includes certain computed system information and a listing of calculated forces, moments, deflections, and stresses for each individual load case. Output from seismic analyses includes system normal mode information. NUPIPE-SW output data also contains pipe stress and pipe support summaries and piping isometric plots. Output data of NUPIPE-SW can be saved on a separate tape for further analysis, if required.

The NUPIPE-SW program is designed to perform analysis in accordance with ASME Section III, Nuclear Power Plant Components (Code). Features ensuring code conformance include use of accepted analysis methods, incorporation of specified stress indices and flexibility factors, proper combination of moment resultants, and provision to generate (automatically) results of combined loading cases. A program option is available to specify among:

1. Class 1 analysis per Article NB-3600 of the Code,

2. Class 2 analysis per Article NC-3600 of the Code,

3. Analysis per ANSI B31.1.0 power piping code, and

4. Combined Class 1 and Class 2 analysis per Articles NB-3600 and NC-3600 of the Code.

3A.3.1.2 Program Verification

NUPIPE-SW program has been verified with ADLPIPE (ADL 1972) for thermal, weight, and response spectrum seismic analysis. The results from both programs are presented in Tables 3A.3.1-1, 3A.3.1-2, 3A.3.1-3, 3A.3.1-4, 3A.3.1-5, 3A.3.1-6 and 3A.3.1-7. The model used for this comparison is shown on Figure 3A.3.1-1.

The comparison is also made with the ASME (1972) Benchmark solution for force time-history dynamic response. The model used for this comparison is shown on Figure 3A.3.1-2. The results for comparisons are plotted an Figure 3A.3.1-2. The natural frequencies are given in Table 3A.3.1-8.

The Class 1 pipe stresses computed by NUPIPE-SW agree with those calculated by hand. The model used is shown on Figure 3A.3.1-3. The results are listed in Tables 3A.3.1-9 and 3A.3.1-10.

3A.3.2 HTLOAD

3A.3.2.1 General Description

HTLOAD is a computer program which performs a finite difference method analysis of piping system response to thermal transients of its contained fluid. The output gives overall thermal growth, linear and nonlinear temperature distribution through the pipe wall, gross discontinuity information (T_A-T_B), and Equation 10 and Equation 11 stress analysis results for Article NB-3600 of ASME Section III.

HTLOAD can analyze piping, with or without a thermal sleeve, that is subject to changes in fluid temperature, velocity, and/or state. The properties of subcooled or saturated water and superheated or saturated steam are taken from the ASME steam tables (Meyer, McClintock, Silvestri, and Spencer 1967). The pressure range is from 0.45 to 6,210 psia.

This computer program also performs thermal analysis for pipes ranging from non-insulated to perfectly insulated. It has stored properties for insulation such as unibestos, asbestos, reflective aluminum, reflective stainless steel, and calcium silicate. Provision is made for hand input properties of other insulation types.

Also stored in the program are the piping material properties of carbon steel, austenitic stainless steel, low-chrome steel, high-chrome steel, and nickel-chrome iron for the temperature range of 32° to 1,600°F.

Program input includes piping material insulation information, time lapse for initial to final fluid temperature, calculation time limit, fluid velocities, initial and final temperature and pressure, pipe and thermal sleeve dimensions.

HTLOAD requires that each thermal transient be input as a step change, a ramp change, or as a 50-point (maximum) arbitrary function.

Output results are used in the calculation of piping stress in accordance with Article NB-3600 of ASME Section III. HTLOAD also calculates stress for the local thermal transient terms for equations 10 and 11 of NB-3650. This information must be combined with other loading terms applicable to equations 10 and 11 and must be ranged with other transients in order to perform a complete code check.

3A.3.2.2 Program Verification

The sample problem selected for solution by HTLOAD consists of a 2-inch, Schedule 160, stainless steel pipe, with one end connected to a 1/2-inch thick socket-welded fitting. Saturated water flowing within the piping system changes temperature from 400° to 500°F in a period of 10 seconds. Velocity of fluid is 7,560 feet per hour. Input properties are listed in Tables 3A.3.2-1 and 3A.3.2-2.

Reynolds number and heat transfer coefficients are compared with hand calculations by Krieth (1964) and are given in Table 3A.3.2-3.

Comparison between HTLOAD and Brock and McNeill's charts (1971) for ΔT_1 and ΔT_2 are given in Table 3A.3.2-4. Table 3A.3.2-5 represents the comparison between TRHEAT in Nuclear Services Corporation (1972) and HTLOAD for ΔT_1, ΔT_2, and T_A-T_B.

3A.3.3 PITRUST

3A.3.3.1 General Description

PITRUST is a computer program which calculates the local stress intensity at the junction of two cylindrical vessels; typically, where a trunnion is welded to a run pipe or where a branch pipe exists from a run pipe or vessel. Code-specified loading conditions are applied, and, if overstressing occurs, there is a program option that will redesign the pad thickness to attain acceptability. Input consists of run pipe cross-section dimensions, pipe internal operating pressure, trunnion outside diameter, trunnion-run pipe orientation, code classification, and load type. PITRUST computes stresses in accordance with the method outlined by Wichman (et al 1965).

If the design criteria for the stresses are exceeded, the program will incrementally increase the pad thickness and recalculate the stresses until the pipe passes or until the pad reaches a maximum of 1.5 times the pipe wall thickness.

PITRUST is capable of complying with the following code requirements: ASME Section III, Class 2 and 3, and ANSI B31.1 Power Piping Code.

Program output tabulates the applied loadings and local stress at the junction of the trunnion and the run pipe.

3A.3.3.2 Program Verification

Program PITRUST has been verified by comparing its solution of a test problem to the solution of the same problem by an independently written piping local stress program, CYLNOZ, in the public domain. The CYLNOZ piping local stress program was written by Franklin Institute, Philadelphia, Pennsylvania, and is presently used by engineering companies. The test problem is a 72.375-inch (outside diameter) by 0.375-inch thick run pipe, reacting under an external loading condition of 1,000 pounds of force (normal and shear) and 1,000 inch-pounds bending and torsional moments transmitted by a 16-inch (outside diameter) nozzle. A comparison of results is listed in Table 3A.3.3-1. Program PITRUST has also been verified by comparing its solution of a test problem to the experimental results obtained by Corum and Greenstreet (1971). A comparison of these results is shown on Figure 3A.3.3-1.

3A.3.4 PILUG

3A.3.4.1 General Description

PILUG is a computer program which calculates local stress intensity at the junction of a lug with a pipe or other cylindrical vessel; specifically, where a rectangular lug is welded to a run pipe. Input consists of run pipe cross-section dimensions, pipe internal operating pressure, lug dimensions, lug-run pipe orientation, code classification, and load type. PILUG computes stresses in accordance with the method outlined by Wichman, Hopper, and Mershon (1965).

If the design criteria for the stresses are exceeded, the program incrementally increases the pad thickness and recalculates the stresses until the lug passes or until the pad reaches 1.5 times the pipe wall thickness.

PILUG is capable of complying with the following code requirements: ASME Section III, Class 2 and Class 3; and ANSI B31.1 Power Piping Code.

Program output tabulates the applied loadings and applied stresses at the junction of the lug and the run pipe.

3A.3.4.2 Program Verification

PILUG has been verified by comparing its solution to a test problem to results obtained by hand calculations using the formulations specified by Wichman, Hopper, and Mershon (1965). A comparison of results is presented in Table 3A.3.4-1.

3A.3.5 PITRIFE

3A.3.5.1 General Description

PITRIFE (SWEC 1982) is a computer program for calculating the local discontinuity stresses in a pipe at the intersection with a circular trunnion due to loads applied to the trunnion. It is a post-processor program that utilizes the results of a finite element model of two intersecting cylinders. Based upon the stresses calculated with the finite element model, non-dimensional stress coefficients are computed for a size-on-size pipe-trunnion configuration for three different values of average pipe radius to wall thickness (R/t = 5,10,20). Additionally, non-dimensional stress coefficients are computed for a trunnion radius equal to 0.707 times the pipe radius (0.707 size-on-size) for the three values of R/t. To facilitate the determination of non-dimensional stress coefficients for other values of R/t, a rotated parabola curve that fits the three R/t data points is generated for both the size-on-size and the 0.707 size-on-size data. The PITRIFE program reconstructs these curves and uses them to interpolate and extrapolate for stress coefficients for different values of R/t. The finite element models are analyzed using the STRUDL-II computer program (ICES) SWEC 1977.

3A.3.5.2 Program Verification

The PITRIFE computer program has been verified by demonstrating that the maximum stress intensities as given by PITRIFE equal the values given by the finite element analysis for specific size-on-size and 0.707 size-on-size models. A comparison of these results is tabulated in Table 3A.3.5-1. The program was verified for other ratios of trunnion to pipe radius by demonstrating that the stress coefficients and maximum stress intensities derived by hand calculation equal the coefficients used in the program to calculate maximum stress intensity. A comparison of these results is given in Table 3A.3.5-2.

3A.3.6 STEHAM

3A.3.6.1 General Description

STEHAM is a computer program which is used to determine the steamhammer transients of piping systems. This program uses the method of characteristics with finite difference approximations both in space and in time (Jonsson, Matthews, and Spaulding 1973; Luk 1975; and Moody 1973). It calculates the one-dimensional transient flow responses and the flow-induced forcing functions in a piping system caused by rapid operational changes of piping components, such as the actuation of a stop valve or a safety/relief valve. Flow characteristics of piping components are mathematically formulated as boundary conditions in the program. These components include the flow control valve, the stop valve, the safety/relief valve, the steam manifold, and the steam reservoir. Frictional effects are taken into consideration in this program.

This program accepts the following as input:

1. The flow network representation of the piping system,

2. The initial flow conditions along the piping system, and

3. Time-dependent flow characteristics of piping components.

Output consists of time-histories of flow pressures, flow densities, flow velocities, inertia, and momentum functions.

3A.3.6.2 Program Verification

STEHAM is verified by comparing its solutions of a test program (Figures 3A.3.6-1 and 3A.3.6-2) to the results of the same problem obtained by an independent analytical approach, as well as an experimental measurement (Progelhof and Owczanek 1963a; 1963b). A comparison of results for time-history pressure responses is plotted on Figures 3A.3.6-3, 3A.3.6-4, and 3A.3.6-5. The forcing functions

developed for nodal points of the piping system, calculated from the relation

$$F = \left(P + \frac{YV^2}{g}\right)A$$

have also been checked by hand calculations (Table 3A.3.6-1).

3A.3.7 WATHAM

3A.3.7.1 General Description

WATHAM is a computer program which is used to determine the flow-induced forcing functions acting on piping systems due to waterhammer. These forcing functions may then be used as input to a structural dynamic analysis, such as a NUPIPE program run.

WATHAM is applicable to a waterhammer problem or, more generally, any unsteady, incompressible fluid flow. These events may be caused by normal or abnormal operational changes of piping components, such as the start-up and trip of pumps, or the rapid opening and closing of valves.

The analysis is based upon the method of characteristics with finite-difference approximations both in time and space for the solution of one-dimensional liquid flows. Influences of piping components, including flow valves, pipe connections, reservoirs, and pumps, have been considered in the analysis.

WATHAM input requires the geometry of the piping system, pipe properties, water properties, operational characteristics of pump and valve, flow frictional coefficients, and the initial water flow conditions. The output provides the time history functions of piezometric heads, velocities, and nodal forces for all nodes and the inertial unbalanced force for each segment. It also gives the maximum values of all the previously mentioned functions and their time of occurrence in the process of flow-transient.

3A.3.7.2 Program Verification

For the verification of WATHAM, a problem from Streeter and Wylie (1967) is employed. Figure 3A.3.7-1 depicts the flow network with nine pipes, its geometrical properties, and steady state flow conditions. The flow-transient mode analyzed is the sudden closure of a valve at the downstream end. Figure 3A.3.7-2 shows the hydraulic network for WATHAM. Table 3A.3.7-1 illustrates the input data needed for a WATHAM run. Figures 3A.3.7-3 and 3A.3.7-4 show the comparison of head-time curves obtained from Streeter and Wylie (1967), Fabic (1967), and WATHAM. Table 3A.3.7-2 presents the comparison of nodal forces derived by hand calculation and WATHAM computation.

In general, WATHAM 3 results are in good agreement with Streeter and Wylie's (1967) results. The small discrepancy is attributed to the modeling of the reservoir boundary condition. In WATHAM, the energy

equation between the reservoir is utilized rather than assuming the head of the pipe entrance is the same as that of the reservoir.

3A.3.8 WATSLUG

3A.3.8.1 General Description

The purpose of WATSLUG (Hsieh and van Duyne 1982) is to determine forcing functions on piping systems during water slug discharge events for subsequent input to piping dynamic analysis.

The analysis is based upon rigid body motion of the generally subcooled water slug and ideal gas representations of the steam or air using rigid column theory to facilitate tracking the several water-steam or water-air interfaces. The driving force is the steam pressure between the valve and the slug, less friction and other losses, and back pressure. Density changes due to possible local flashing of the water slug are considered. Having recourse to the control volume theory, the subsequent segment-forced calculation is carried out.

The input consists of complete piping system geometry, pipe dimensions (Table 3A.3.8-1), valve flow characteristics, valve opening time, detail upstream steam conditions, and initial downstream steam or air conditions, (Table 3A.3.8-2), while the output contains forcing functions for each piping segment based upon flow velocities, pressures, and densities during the water slug discharge event. Forces are written on tape for direct input to NUPIPE-SW.

3A.3.8.2 Program Verification

The WATSLUG model of the test problem is diagrammed in Figure 3A.3.8-1, while the NUPIPE-SW model is diagrammed in Figure 3A.3.8-2. WATSLUG is verified by comparing the solution of this test problem to the results for the same problem obtained by an independent analytical approach (RELAP5/MOD 1) (House 1982), as shown in Figures 3A.3.8-3 and 3A.3.8-4, and by comparison of predicted versus measured support reactions. NUPIPE-SW (ME-110)-generated support reactions due to WATSLUG forcing functions were compared with experimental measurements from a test run of this problem, EPRI Test 908 (RELAP5/MOD 1) shown in Figures 3A.3.8-5 and 3A.3.8-6.

The WATSLUG-generated forcing functions and the resultant NUPIPE-SW support reactions compare favorably with the RELAP5/MOD 1 predicted forcing functions and the EPRI-measured support reactions, respectively.

3A.3.9 ELBOW

3A.3.9.1 General Description

ELBOW calculates the circumferential and longitudinal stresses on the inside and outside surfaces of an elbow subjected to internal

pressure, in-plane bending, out-of-plane bending, torsion, and linear temperature gradient through the wall. Stress indices and flexibility factors for the elbows are also calculated. Results can be used directly for design and analysis of elbows in accordance with Article NB-3600 of ASME Section III.

The solution method utilizes Table NB-3685.1-1 relative to internal pressure and Table NB-3685.l-2, with modifications as indicated by Dodge and Moore (1972) relative to moment loadings and flexibility factors.

The complete analysis of Rodabaugh and George (1957), based on minimum potential energy method, was written in terms of infinite series. A modified version of the analytical method by Rodabaugh and George (1957), which considered both in-plane and out-of-plane bending as well as the influence of internal pressure, was selected by ORNL as the most appropriate basis in determining the stresses and flexibility for elbows. The analysis is a generation of the work done by Von Karman. The modifications to the analysis method of Rodabaugh and George (1957) include a generalization of the "correction for transverse compression" recommended by Gross. The ORNL computer program ORNL-ELBOW was written by Dodge and Moore (1973) to implement this analysis procedure.

The SWEC computer program ELBOW uses the same theoretical considerations to obtain the flexibility factor and detailed stresses in the elbows.

When this program is used, it is considered as a detailed analysis. For elbows free from local discontinuities, ELBOW solutions are the detailed solutions to Equation 10, Table NB-3653, including the consideration of C_1 and C_2, but without $\left|\alpha_a T_a - \alpha_b T_b\right|$ term. The solutions are also the detailed solutions to Equation 11 if $|\Delta T2|$ term is negligible.

The program does not take into account the effects of discontinuities on the elbows. The influence length of a concentrated force or moment in a shell structure is about $2.5\sqrt{rt}$, where r is the radius of curvature of the shell surface, or for a pipe r is the mean radius and t is the thickness of the pipe. For the portion of elbow at a distance of $2.5\sqrt{rt}$ away from local discontinuities, detailed stresses can be obtained by using this program.

In general, for an elbow welded to tangent pipe of the same thickness, the effects of straight tangent pipe on the elbow can be neglected (Table NB-3683.2). However, if two elbows are welded together or joined by a piece of straight pipe that is less than one pipe diameter in length, some intensification effects may have to be considered.

This program is an efficient and easy to use program for determining stresses, stress indices, and flexibility factors for elbows. Comparison with experimental results indicates that the results accurately represent the maximum stresses which occur at the center

of the bend. Since end effects are not included in the analytical solution on which this program is based, the calculated stresses and flexibility of the elbow may be larger than the actual values.

Required input data includes elbow dimensions, material properties, applied moments and forces, internal pressure, and linear temperature gradients.

Output includes stress indices, flexibility factor for the elbow, and circumferential and longitudinal stresses on the inside and outside surfaces at specified locations.

3A.3.9.2 Program Verification

Sample problems were selected for solution by ELBOW, and these results were compared with those obtained from hand calculations. The following cases were selected for purposes of verification:

Case 1 - Elbow is subjected to internal pressures. Results are given in Table 3A.3.9-2.

Case 2 - Elbow is subjected to a linear temperature gradient through the pipe wall. Results are given in Table 3A.3.9-3.

Case 3 - Elbow is subjected to combined loadings at one end. Results are given in Table 3A.3.9-4.

Elbow properties utilized for the analyses are given in Table 3A.3.9-1.

3A.3.10 BENDCORD

3A.3.10.1 General Description

BENDCORD is a FORTRAN IV program which supplies, in printed and card form, data for coding segments of a circle for use in the NUPIPE-SW piping program (Section 3A.3.1).

BENDCORD has the capacity of operating on an arc which lies in any one of three planes defined by the cartesian coordinate system.

The piping system may be divided into equal or unequal segments so as to aid the user in placing supports. For seismic piping systems, the user is provided the option of locating mass points at alternating nodes or, if desired, at every node.

Various elbow types may be provided as input to reflect the requirements of Class 1 analysis.

The method utilized by BENDCORD divides an arc into tangent lines, the lengths of which are then calculated by subtracting the coordinates of the tangent point from the tangent intersection point of the tangent lines.

For output, BENDCORD provides a table of the tangent intersection coordinates of the arc.

3A.3.10.2 Program Verification

The BENDCORD program is verified by calculating distances (or offsets) between nodal points by hand and comparing these values to those calculated by BENDCORD.

The problem consists of a 180 degree piping arc in the X-Z plane beginning with node 10 at phi = 45 degrees, and ending at node 36. There are 26 included angles in the arc, all equal. Mass points are at every other node. Refer to Figure 3A.3.10-1 for a graphic representation of the problem.

Partial results are shown in Table 3A.3.10-1. It can be seen that there are no significant differences between the results from BENDCORD and the hand calculations.

3A.3.11 CDC - BASEPLATE II

3A.3.11.1 General Description

BASEPLATE II is a combination of a pre- and post-processor to the STARDYNE program for the purpose of analyzing flexible baseplates on a geometrically nonlinear foundation. The program employs an automatic mesh generation technique with the user in control of mesh size and element configuration.

Input includes plate geometry, nonstandard and standard (library) attachments, anchor locations, anchor and concrete stiffnesses, material properties, anchor allowables, and up to fifty loading conditions.

Output consists of a printer plot of the baseplate showing attachment and anchor locations, plate deformations and principal stresses, anchor bolt tension and resultant shear load for each anchor together with calculated tension/shear interaction and factor of safety.

3A.3.11.2 Program Verification

BASEPLATE II is verified and qualified through control data corporation quality assurance programs, which are periodically evaluated by Stone & Webster Engineering Corporation and which have been found to be satisfactory.

3A.3.12 PSPECTRA

3A.3.12.1 General Description

The PSPECTRA program peak spreads and envelops amplified response spectra (ARS) curves of earthquakes or other dynamic events. The program reads ARS curves from magnetic tape, card, or disk files. The created curves are stored on a disk file and are optionally

printed, plotted, or punched on a card file. The disk and card file format is compatible with the NUPIPE-SW program (Section 3A.3.1). One important application of PSPECTRA is the creation of disk or card data sets of operating basis earthquake (OBE) and safe shutdown earthquake (SSE) curves for input to NUPIPE-SW.

There are two methods of peak spreading - either the sides of the spread peaks can be vertical or they can be parallel to the sides of the original peaks. There are three methods of enveloping - maximum value, absolute sum, and square root of the sum of the squares (SRSS). Another program option allows enveloping E-W and N-S direction curves to form one horizontal curve for each curve set on a disk file. ARS curves can also be input from as many as four disk files in one run for the purpose of enveloping or plotting curves with up to four different damping values, and superimposing curves with different damping values on the same plot.

Input data includes ARS data tapes, optional disk file input, and card input.

Output data includes tables of edited ARS data as retrieved from tape; tables of enveloped curves; and plots of original, peak spread, and enveloped curves.

3A.3.12.2 Program Verification

PSPECTRA is verified by manually enveloping and peak spreading input ARS curves and comparing these results with those calculated by PSPECTRA. The following presents the results of four sample verification problems.

1. Problem I - This problem verifies the maximum value enveloping option. The data points were selected from each of three sets of ARS curves, and the maximum value of the acceleration was manually determined for each set. These values were then compared with those from PSPECTRA. Results are shown in Table 3A.3.12-1.

2. Problem II - This problem verifies the absolute sum value enveloping option. The same procedure as Problem I was utilized. Results are shown in Table 3A.3.12-2.

3. Problem III - This problem verifies the SRSS enveloping option. Same procedure as Problem I was utilized. Results are shown in Table 3A.3.12-3.

4. Problem IV - This problem verifies the peak spreading option. Two different peaks were selected from the input ARS and were spread manually. The values of these spreads were then compared with the values from PSPECTRA. Results are shown in Table 3A.3.12-4.

3A.3.13 ANSYS

3A.3.13.1 General Description

A description of the program is provided in Section 3A.1.9.

3A.3.14 STRUDL II

3A.3.14.1 General Description

A description of the program is provided in Section 3A.1.1.

3A.3.15 STRUDL-SW

3A.3.15.1 General Description

A description of the program is provided in Section 3A.1.3.

3A.3.16 NUPIPE II

3A.3.16.1 General Description

The NUPIPE II piping program performs a linear elastic analysis of three-dimensional piping systems subjected to thermal, static, and dynamic loads. It utilizes the finite element method of analysis.

NUPIPE II is a recognized program in the public domain.

3A.3.17 BIP

3A.3.17.1 General Description

BIP (baseplate information processor) is a set of two programs, BIP1 and BIP2, which are pre- and post-processors to ANSYS. These programs reduce the time required for baseplate analysis. Baseplates are analyzed for in-plane and out-of-plane loads that are transferred through the attachments. The baseplate may be treated as infinitely rigid for in-plane loads, resulting in a statically determinate solution for anchor bolt shear loads. Out-of-plane loads are analyzed by ANSYS to account for plate flexibility as well as gaps between the baseplate and concrete and interference fit with anchors.

Input to BIP includes plate and attachment geometry, anchor locations, anchor and concrete stiffness values, anchor and concrete gaps, material properties, anchor allowables with reduction factors, and up to ten loading conditions.

Output consists of an input echo, a printer plot of the baseplate showing attachment and anchor locations, resultant shear at each anchor together with reduced tension allowables based on tension-shear interaction, plate deformations and stresses, and reactions (including bolt pullout loads).

3A.3.17.2 Program Verification

BIP program is a publicly available program and is verified and qualified through Boeing Computer Services Quality Assurance Programs. These are periodically evaluated by Stone & Webster Engineering Corporation and have been found to be satisfactory.

3A.3.18 APE

3A.3.18.1 General Description

Computer program APE (anchor plate evaluation) calculates the shear and tension loads for each anchor of a group of drilled-in anchors of a baseplate subjected to in-plane and out-of-plane loads. A reduced tension allowable, based on the calculated shear load and tension-shear interaction, is also calculated.

Program input includes plate and attachment geometry, anchor bolt locations and allowables, and applied loads.

Output consists of anchor bolt pattern center of gravity and polar moment of inertia, resultant shear and allowable tension load for each anchor, load factors for out-of-plane loading, and anchor tension loads, including load factors, for each anchor.

3.A.3.18.2 Program Verification

Verification of program APE (version 01, level 00) was performed by comparing results of APE analyses with similar results obtained from Boeing Computer Service Program BIP (refer to Section 3A.3.17). Comparisons of results from APE (version 01, level 00) and Boeing Computer Service Program BIP are shown in Table 3A.3.18-1. APE results are shown to be conservative or comparable with respect to BIP. All results are based on loadings at the critical anchor.

3A.3.19 CHPLOT

3A.3.19.1 General Description

CHPLOT is a program which will plot any number of data values (variables) versus time. Although the plot input data file can be in the form of card data, the more appropriate application of this program is to be used in conjunction with a program that creates a plot data file (on disk or tape) having the format required for input to this program.

Plots are available in two sizes; one with axes of 5 inches ordinate by 8 inches (abscissa) that fits the standard 8-1/2 inches by 11 inches page, and the other is 8 inches by 12 inches for fitting an 11 inches by 15 inches page. Plots are normally one data value versus time per graph, although up to 14 data values (plots) can be plotted on one graph.

Each graph's abscissa will be labeled as TIME (SEC) and the ordinate labels for each graph can be input. Ordinate axes can be selectively suppressed. Scaling is performed automatically to fit the size selected. Graphs can be grouped into a maximum of nine groups, within which each group of graphs will be scaled to the same scale factor.

An optional label is available for labeling all graphs at the bottom of each graph.

3A.3.19.2 Program Verification

CHPLOT is qualified by analyzing sample problems which utilize all three methods for invoking CHPLOT and verifying results by inspection. These methods include invoking CHPLOT in the STEHAM or WATHAM computer programs, plotting functions from a NUPIPE computer program tape, or plotting manually input functions.

3A.3.20 BAP

3A.3.20.1 General Description

A description of the program is provided in Section 3A.1.12.

3A.3.21 TRUNPIPE

3A.3.21.1 General Description

TRUNPIPE (SWEC 1986) is a computer program run on the HP-41CX desk computer, which analyzes integral circular attachments on ASME III Class 2 and 3 piping in accordance with ASME III Code Case N-392.

3A.3.21.2 Program Verification

Verification of the program was performed by comparison to results of manual calculations.

3A.3.22 LUGAPIPE

3A.3.22.1 General Description

LUGAPIPE (SWEC 1986) is a computer program run on the HP-41CX desk computer, which analyzes integral rectangular attachments on ASME III Class 2 and 3 piping in accordance with ASME III Code Case N-318.

3A.3.22.2 Program Verification

Verification of the program was performed by comparison to results of manual calculations.

3A.3.23 CCN318

3A.3.23.1 General Description

CCN318 is a computer program that performs analysis of local stresses at rectangular cross section welded attachments on ASME III Class 2 and 3 and ANSI B31.1 piping using Code Case N-318. Various types of welds and loading conditions may be evaluated.

Program output consists of an echo print of the input data followed by local stress indices, calculated pressure stresses, and a tabulation of calculated stresses based on the code case equations and the corresponding allowable stresses.

3A.3.23.2 Program Verification

Verification of the program was performed by comparison to results of manual calculations.

3A.3.24 CCN392

3A.3.24.1 General Description

CCN392 is a computer program that performs analysis of local stresses at hollow circular cross section welded attachments on ASME III Class 2 and 3 and ANSI B31.1 piping using Code Case N-392. Various types of welds and loading conditions may be evaluated.

Program output consists of an echo print of the input data followed by local stress indices, calculated pressure stresses, and a tabulation of calculated stresses based on the code case equations and the corresponding allowable stresses.

3A.3.24.2 Program Verification

Verification of the program was performed by comparison to results of manual calculations.

3A.3.25 References for Appendix 3A.3

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Tables for Section 3A.3

TABLE 3A.3.1-1

COMPARISON OF SUPPORT REACTION DUE TO THERMAL, ANCHOR MOVEMENT, AND EXTERNAL FORCE LOADING

		Forces (lb)			Moments (in-lb)		
Node	Program	FX	FY	FZ	MX	MY	MZ
170	NUPIPE-SW	-9,154	7,541	4,492	-5,952	-823,420	1,241,512
	ADLPIPE	-9,178	7,540	4,492	-5,529	-823,420	1,241,512
218	NUPIPE-SW		16,650				
	ADLPIPE		16,622				
330	NUPIPE-SW	34,532	-33,620	-31,750	-486,338	-1,516,811	573,673
	ADLPIPE	34,511	-33,608	-31,736	-486,386	-1,519,359	573,438
390	NUPIPE-SW		8,631				
	ADLPIPE		8,678				
430	NUPIPE-SW	1,702	798	12,553	-28,147	164,346	248,852
	ADLPIPE	1,746	768	12,541	-26,917	166,180	250,956

TABLE 3A.3.1-2

COMPARISON OF DEFLECTIONS AND ROTATIONS DUE TO THERMAL, ANCHOR MOVEMENT, AND EXTERNAL FORCE LOADING

		Deflection (inch)			Rotation (rad)		
Node	Program	DX	DY	DZ	RX	RY	RZ
197	NUPIPE-SW	0.348	-0.141	0.230	-0.0026	0.0025	-0.0084
	ADLPIPE	0.348	-0.141	0.229	-0.0026	0.0025	-0.0084
212	NUPIPE-SW	1.120	0.052	-0.023	-0.0092	-0.0051	-0.0115
	ADLPIPE	1.120	0.052	-0.023	-0.0092	-0.0051	-0.0115
230	NUPIPE-SW	1.276	-0.028	-0.548	-0.0066	-0.0044	0.0024
	ADLPIPE	1.276	-0.027	-0.548	-0.0066	-0.0044	0.0024
260	NUPIPE-SW	0.512	-0.001	-0.520	-0.0034	-0.0005	0.0035
	ADLPIPE	0.512	-0.000	-0.520	-0.0035	-0.0005	0.0035
390	NUPIPE-SW	0.066	-0.000	0.249	-0.0010	0.0026	-0.0020
	ADLPIPE	0.067	-0.000	0.248	-0.0010	0.0026	-0.0020
420	NUPIPE-SW	-0.029	-0.079	0.011	-0.0002	-0.0002	-0.0007
	ADLPIPE	-0.029	-0.079	0.011	-0.0002	-0.0002	-0.0007

TABLE 3A.3.1-3

COMPARISON OF STRESS DUE TO THERMAL, ANCHOR MOVEMENT AND EXTERNAL FORCE LOADING

Node	NUPIPE-SW (psi)	ADLPIPE (psi)
180	18,989	19,013
199	17,703	17,731
214	23,958	23,955
236	14,427	14,416
265	6,254	6,251
305	12,539	12,532
344	11,845	11,838
370	6,295	6,296
395	3,476	3,473
430	3,282	3,308

TABLE 3A.3.1-4

COMPARISON OF INTERNAL FORCES DUE TO DEADWEIGHT ANALYSIS

		Forces (lb)			Moments (in-lb)		
Node	Program	FX	FY	FZ	MX	MY	MZ
197	NUPIPE-SW	295	2,337	14	-35,864	5,218	51,979
	ADLPIPE	290	2,341	15	-35,108	5,231	52,081
212	NUPIPE-SW	295	3,306	14	59,390	-5,394	14,010
	ADLPIPE	299	3,310	15	59,735	-5,500	14,542
360	NUPIPE-SW	330	2,781	-29	30,930	-22,748	-84,971
	ADLPIPE	326	2,783	-32	31,920	-23,105	-82,784
390	NUPIPE-SW	330	4,933	-29	-255,351	701	126,476
	ADLPIPE	336	4,707	-32	-256,444	916	126,716
420	NUPIPE-SW	330	-492	-29	8,972	27,075	82,202
	ADLPIPE	336	-497	-32	-9,181	27,724	80,676

TABLE 3A.3.1-5

COMPARISON OF DEFLECTIONS AND ROTATION
DUE TO DEADWEIGHT ANALYSIS

		Deflections (inch)			Rotations (rad)		
Node	Program	DX	DY	DZ	RX	RY	RZ
197	NUPIPE-SW	0.007	-0.014	-0.004	0.0001	0.0001	0.0002
	ADLPIPE	0.007	-0.014	-0.004	0.0001	0.0001	0.0002
212	NUPIPE-SW	-0.005	-0.013	0.013	0.0006	0.0001	0.0004
	ADLPIPE	-0.005	-0.013	0.013	0.0006	0.0001	0.0004
360	NUPIPE-SW	-0.008	-0.068	0.024	0.0004	-0.0000	-0.0004
	ADLPIPE	-0.009	-0.069	0.024	0.0004	0.0000	-0.0004
390	NUPIPE-SW	-0.015	-0.000	-0.003	0.0002	-0.0002	-0.0005
	ADLPIPE	-0.015	-0.000	-0.003	0.0002	-0.0002	-0.0005
420	NUPIPE-SW	-0.001	0.002	-0.001	-0.0000	-0.0001	-0.0002
	ADLPIPE	-0.001	0.002	-0.001	-0.0000	-0.0001	-0.0002

TABLE 3A.3.1-6

COMPARISON OF STRESS DUE TO DEADWEIGHT ANALYSIS

Node	NUPIPE-SW (psi)	ADLPIPE (psi)
180	685	694
199	448	458
214	667	679
236	2,472	2,449
265	530	524
305	515	522
344	635	631
370	679	677
395	575	580
430	1,101	1,091

TABLE 3A.3.1-7

COMPARISON OF NATURAL FREQUENCIES (Hz)
NUPIPE-SW VERSUS ADLPIPE

Mode	1st	2nd	3rd	4th	5th
NUPIPE-SW	7.109	9.328	12.297	14.681	18.043
ADLPIPE	7.118	9.329	12.492	14.427	17.714

TABLE 3A.3.1-8

COMPARISON OF NATURAL FREQUENCIES (Hz)
NUPIPE-SW VERSUS BENCHMARK

Mode	1st	2nd
NUPIPE-SW	2.407	13.537
BENCHMARK	2.3288	13.0808

TABLE 3A.3.1-9

COMPARISON OF CLASS 1 PIPE STRESS ANALYSIS

Point No. 20	Hand Calculation	NUPIPE
Minimum wall thickness (in)	0.032	0.032
Primary stress (Eq. 9) (psi)	3,713	3,712
Primary and secondary stress (Eq. 10)(psi)	16,041	16,038
Alternating stress (Eq. 11 and 14) (psi)	13,468	13,465
Usage factor	0.0654	0.0631
Point No. 30		
Minimum wall thickness (in)	0.047	0.047
Primary stress (Eq. 9) (psi)	8,748	8,741
Primary and secondary stress (Eq. 10)(psi)	117,655	117,546
Expansion stress (Eq. 12) (psi)	99,884	99,781
Eq. 13 (psi)	18,252	18,246
Alternate stress (Eq. 14) (psi)	218,258	217,811
Usage factor	Not in Range	

TABLE 3A.3.1-10

INDIVIDUAL PAIR USAGE FACTOR FOR POINT NO. 30

Pair	Hand Calculation	NUPIPE
1, 5	0.183	0.1803
1, 8	1.660	1.7361
1, 9	0.0001	0.0001
1, 10	Not in Range	
5, 8	Not in Range	
5, 9	0.221	0.2646
5, 10	0.747	0.8051
8, 9	0.857	0.8832
8, 10	5.5518	5.8608
9, 10	0.0001	0.0001

TABLE 3A.3.2-1

PIPE MATERIAL PROPERTIES

Property	Temperature (°F)	Value
Thermal conductivity	450	10.01 Btu/°F-hr-ft
Thermal diffusivity	450	0.164 sq ft/hr
Young's Modulus	70	28.3×10^{6} psi
Coefficient of thermal expansion	70	9.11×10^{-6} in/in/°F

TABLE 3A.3.2-2

FLUID MATERIAL/THERMAL PROPERTIES

Property	Temperature (°F)	Value
Density	450	51.300 lb/cu ft
Viscosity	450	0.2920 lb/hr/ft
Specific heat	450	1.135 Btu/lb-°F
Conductivity	450	0.3650 Btu/°F-hr-ft
Volume expansion coefficient	450	0.0009/°F

TABLE 3A.3.2-3

COMPARISON OF HTLOAD WITH HAND CALCULATION

Property	HTLOAD	Hand Calculation
Reynolds number	186,700	186,700
Heat transfer coefficient Btu/°F-hr-sq ft	946.8	946.8

TABLE 3A.3.2-4

COMPARISON OF HTLOAD WITH CHARTS OF BROCK AND MCNEILL

Parameter	Charts	HTLOAD
Maximum ΔT_1 (°F)	43.31	45.14
Maximum ΔT_2 (°F)	8.50	8.36

TABLE 3A.3.2-5

COMPARISON OF HTLOAD WITH TRHEAT

Parameter	TRHEAT	HTLOAD
Maximum ΔT_1 (°F)	44.70	45.14
Maximum ΔT_2 (°F)	8.69	8.36
Maximum T_A - T_B (°F)	19.03	19.08

TABLE 3A.3.3-1

COMPARISON OF PITRUST WITH
FRANKLIN INSTITUTE PROGRAM CYLNOZ AND HAND CALCULATION

Source of Stress	Franklin Institute Corrected Values	Output from PITRUST	Hand Calculation
Circumferential			
p (Normal)(lb)	395.00	399.00	399.99
p (Bending)(lb)	1,875.00	1,883.00	1,877.30
M_C (Normal)(in-lb)	35.85	35.57	36.06
M_C (Bending)(in-lb)	364.70	366.60	354.30
M_L (Normal)(in-lb)	79.05	79.66	79.54
M_L (Bending)(in-lb)	90.52	80.57	79.42
Axial			
p (Normal)(lb)	813.00	812.00	814.80
p (Bending)(lb)	812.30	827.00	810.60
M_C (Normal)(in-lb)	91.79	105.00	95.45
M_C (Bending)(in-lb)	158.80	160.00	158.80
M_L (Normal)(in-lb)	37.06	37.00	37.12
M_L (Bending)(in-lb)	117.90	105.00	103.85
Shear stress by M_C (psi)	6.63	6.63	6.63
Shear stress by V_C (psi)	106.10	106.10	106.10
Shear stress by V_C (psi)	106.10	106.10	106.10

TABLE 3A.3.4-1

COMPARISON OF PILUG COMPUTER PROGRAM OUTPUT WITH HAND CALCULATIONS[1]

Figure[2]	P	Stress From Hand Calculation	Computer Output	Remarks
		Stress in Circumferential Direction		
3C	0.5485	387	330	Membrane stress due to P[3]
1C	0.326	2,165	2,160	Bending stress due to P[3]
3A	0.294	671	629	Membrane stress due to M_C[4]
1A	0.388	18,976	19,904	Bending stress due to M_C[4]
3B	0.467	3,014	2,961	Membrane stress due to M_L[5]
1B	0.416	6,143	5,969	Bending stress due to M_L[5]
		Stress in Axial Direction		
4C	0.4447	683	690	Membrane stress due to P
2C	0.4632	773	792	Bending stress due to P
4A	0.294	1,897	1,864	Membrane stress due to M_C[4]
2A	0.550	6,357	5,942	Bending stress due to M_C[4]
4B	0.467	2,365	2,328	Membrane stress due to M_L[5]
2B	0.582	4,989.7	4,842	Bending stress due to M_L[5]
		Shear Stress		
		1,304.8	1,304.8	Shear stress due to M_T[6]
		-366.99	-366.99	Shear stress due to V_C[7]
		127.15	127.16	Shear stress due to V_L[8]

TABLE 3A.3.4-1 (Cont)

NOTES:

1. Test problem dimensions:

   a. Run pipe outside diameter = 17 inches

   b. Run pipe thickness = 0.812 inch

   c. Axial length of lug = 12 inches

   d. Width of lug along circumference = 3 inches

2. Wichman et al 1965, WRC Bulletin No. 107

3. Test problem loading P = 3,399 lb

4. Test problem loading M_C = 81,834 lb

5. Test problem loading M_L = 103,320 in/lb

6. Test problem loading M_T = 76,284 in/lb

7. Test problem loading V_C = 1,788 lb

8. Test problem loading V_L = 2,478 lb

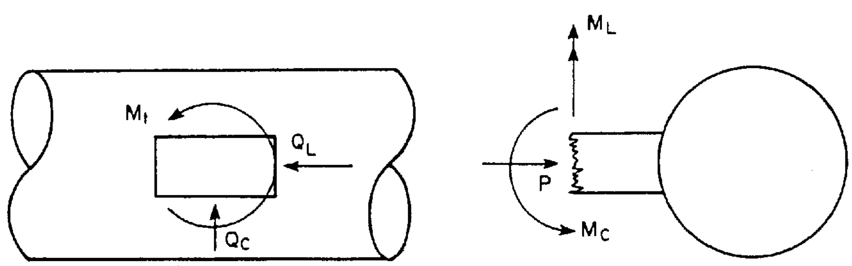
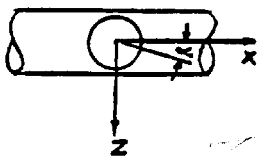


TABLE 3A.3.5-1

COMPARISON OF PITRIFE COMPUTER PROGRAM OUTPUT WITH STRUDL-II OUTPUT

Test Problem:



	Size-On-Size	0.707 Size-On-Size
Avg Pipe Radius (in)	3.00	3.00
Avg Trunnion Radius (in)	3.00	2.12
Pipe Wall Thick-ness (in)	0.30	0.30
Trunnion Wall Thickness (in)	0.30	0.21

SIZE-ON-SIZE
MAXIMUM STRESS INTENSITY-PSI ($\alpha = 30°$)

Load	PITRIFE Output	STRUDL-II Output
FX = 10,000 lbs	5,763	5,768
FY = 10,000 lbs	7,844	7,846
FZ = 10,000 lbs	6,507	6,506
MX = 10,000 in-lb	1,329	1,329
MY = 10,000 in-lb	1,688	1,687
MZ = 10,000 in-lb	4,066	4,068

0.707 SIZE-ON-SIZE
MAXIMUM STRESS INTENSITY - PSI ($\alpha = 30°$)

Load	PITRIFE Output	STRUDL-II Output
FX = 10,000 lbs	13,471	13,458
FY = 10,000 lbs	9,616	9,611
FZ = 10,000 lbs	20,105	20,030
MX = 10,000 in-lb	4,371	4,368
MY = 10,000 in-lb	2,467	2,467
MZ = 10,000 in-lb	6,178	6,176

TABLE 3A.3.5-2

COMPARISON OF PITRIFE COMPUTER PROGRAM OUTPUT WITH HAND CALCULATIONS

Test Problem:

Avg Pipe Radius	=	1.5 in
Avg Trunnion Radius	=	1.35 in
Pipe Wall Thickness	=	0.30 in
Trunnion Wall Thickness	=	0.27 in

LOADS FOR EACH LOAD TYPE COMBINED (DL, OBEI, THER, OCCU, ETC)

FX = FY = FZ = 10,000 lbs
MX = MY = MZ = 10,000 in-lbs
MNS Stress = 200 psi
Internal Pressure = 100 psi

STRESS COEFFICIENTS - 0.9 SIZE-ON-SIZE - FX LOADING (α = 30°)

Stress Type	Coefficient By Hand Calculation	Coefficient From PITRIFE
Longitudinal - Inside Fiber	-1.2652	-1.2652
Circumferential - Inside Fiber	-0.2764	-0.2764
Shear - Inside Fiber	0.2041	0.2041
Longitudinal - Outside Fiber	0.7454	0.7454
Circumferential - Outside Fiber	1.3509	1.3509
Shear - Outside Fiber	0.2041	0.2041

MAXIMUM STRESS INTENSITY - 0.9 SIZE-ON-SIZE (α = 30°C)

	Maximum Stress Intensity - psi	
Load Condition	Hand Calculation	PITRIFE
P + DL + MNS_1	28,181	28,182
P + DL + SRSS (OBEI, OCCU) + MNS_2	73,220	73,220
P + DL + OBEA + THER + MNS_3	88,216	88,216
P + DL + OCCE + MNS_4	59,853	59,853
P + DL + SRSS (SSEI, OCCF) + MNS_5	73,220	73,220

TABLE 3A.3.6-1

NODAL FORCE COMPARISON*

Node No.	Pressure (psia)	Velocity (fps)	Density (lb/cu ft)	Force (STEHAM) (lb)	Force (Hand Calculation) (lb)
1	42.523	0.0	0.23954	196.57	196.67
5	42.785	5.7843	0.24076	198.43	198.53
10	44.231	31.219	0.24647	209.00	209.11
15	47.003	78.172	0.25737	230.62	230.73
20	50.214	129.89	0.26979	257.84	257.97
25	52.095	159.43	0.27697	274.93	275.06
30	52.209	161.97	0.27742	276.09	276.23
35	52.168	162.21	0.27731	275.83	275.97

NOTE:

*This comparison is based on the following:

Diameter D = 0.25 feet

Area A = $\pi D^2/4$
= 0.0490874 square foot

Nodal Force F = $(p + \gamma V^2 /g) A - p_{atm} A$

where p = pressure (lb/sq ft)

γ = density (lb/cu ft)

V = velocity (fps)

g = gravitational constant (32.2 ft/sec^2)

P_{atm} = 14.7 x 144 lb/sq ft

at time t = 0.00650 second

TABLE 3A.3.7-1

INPUT DATA FOR WATHAM*

Pipe No.	Total Length (ft)	Inside Diameter (ft)	Friction Factor	No. of Nodes	Nodal Span (ft)	Thickness (in)	Velocity (fps)
1	2,000	3.0	0.030	7	333.33	0.30824	4.24413
2	3,000	2.5	0.028	9	375.00	0.44	2.92132
3	2,000	2.0	0.024	6	400.00	0.50026	4.98473
4	1,800	1.5	0.020	7	300.00	0.11108	3.59336
5	1,500	1.5	0.022**	5	375.00	0.264	4.52142
6	1,600	1.5	0.025	6	320.00	0.13796	2.29183
7	2,200	2.5	0.040	8	314.29	0.21534	3.65878
8	1,500	2.0	0.030	6	300.00	0.14811	3.83245
9	2,000	3.0	0.024	7	333.33	0.30824	4.24413

NOTES:

*The initial heads of all nodes are calculated by using the Darcy-Weisbach equation.

**Friction factor in Pipe 5 should be 0.022 instead of 0.02.

TABLE 3A.3.7-2

COMPARISON OF NODAL FORCE COMPARISON*

Pipe No.	Node No.	Force (WATHAM) (kips)	Force (Hand Calculation) (kips)
1	1	276.34	276.48
1	2	300.46	300.62
1	3	317.78	317.94
1	4	329.59	329.76
1	5	341.39	341.56
1	6	355.31	355.49
1	7	369.52	369.71

NOTE:

*Nodal Force Calculation is based on the following equation:

$$F = A\,(H + \gamma \frac{V^2}{g})$$

where:

F = nodal force (lb)

γ = density (lb/cu ft)

H = nodal head (ft)

g = 32.2 ft/sec^2

V = nodal velocity (fps)

A = pipe area (sq ft)

At time = 2.34 seconds

TABLE 3A.3.8-1

INPUT DATA FOR NUPIPE-SW*

Cutoff Mode	Cutoff Frequency	Time Step	Integration Time	Damping Ratio
53	433 Hz	0.0009 sec.	0.5 sec	10%

Pipe Section	Total Length (ft)	Outside Diameter (in)	Thickness (in)	Weight (lb/ft)
1	4.73	8.625	0.906	74.71
2	12.31	6.625	0.864	53.16
3	12.43	6.625	0.28	18.97
4	69.0	12.75	0.688	88.60
5	1.1	12.75	1.5	-
6	1.0	8.625	0.322	28.55
7	0.83	6.625	0.432	28.57

$F_{hot} = E_{cold}$ = Young's Modulus of pipe = 28.3×10^{6} psi

NOTE:

*See Figure 3A.3.8-2 for sketch of NUPIPE-SW model

TABLE 3A.3.8-2

INPUT DATA FOR WATSLUG*

Pipe No.	Total Length (ft)	Inside Diameter (ft)	Friction Factor
1	16.125	0.408	0.015
2	12.563	0.5054	0.015
3	63.562	0.948	0.013

Valve Characteristics

Orifice Area (ft^2)	Opening Time (sec)	Discharge Coefficient	Flow Rate (lbm/sec)
0.0253	0.015	0.805	120.83

Upstream Steam Conditions

Pressure (psia)	Temperature	Density $\frac{lbm}{ft^3}$	Pressure Rise Rate $\frac{psi}{sec}$
2690	679°F (1139°R)	8.862	-40.**

Downstream Gas Conditions

Pressure (psia)	Temperature	Density $\frac{lbm}{ft^3}$
15	80°F (540°R)	0.09975

Waterslug Weight = 69.8 lbs

NOTES:

*See Figure 3A.3.8-1 for sketch of WATSLUG model.
**Pressure is decreasing after valve opens.

TABLE 3A.3.9-1

ELBOW PROGRAM - ELBOW PROPERTIES USED FOR VERIFICATION PROBLEMS

Outside diameter	30.0 inches
Minimum wall thickness	0.5239 inches
Bend radius	44.214 inches
Pipe radius	14.738 inches
Young's modulus	28.3×10^{6} psi
Poisson's ratio	0.3
Coefficient of thermal expansion	9.11×10^{-6} in/in °F

TABLE 3A.3.9-2

ELBOW PROGRAM - CASE 1 RESULTS

Internal Pressure equals 413.58 psi

	Circumferential Stresses - psi		Longitudinal Stresses - psi		Stress Intensities - psi	
	Inside	Outside	Inside	Outside	Inside	Outside
ELBOW Program	11,676	11,676	5,714	5,714	12,090	11,676
Hand Calculation*	11,676	11,676	5,714	5,714	12,090	11,676

NOTE:

*Hand calculation based on Article NB-3685.1 of ASME Section III, 1974.

TABLE 3A.3.9-3

ELBOW PROGRAM - CASE 2 RESULTS

Linear Temperature Gradient Through Wall Equals 100°F

	Circumferential Stresses - psi		Longitudinal Stresses - psi		Stress Intensities - psi	
	Inside	Outside	Inside	Outside	Inside	Outside
ELBOW Program	18,415	-18,415	18,415	-18,415	18,415	18,415
Hand Calculation	*18,415	-18,415	18,415	-18,415	18,415	18,415

NOTE:

*Hand calculation of Timoshenko and Goodier 1970.

TABLE 3A.3.9-4

ELBOW PROGRAM - CASE 3 RESULTS

Combined Loadings as Follows:

1. Internal pressure equals 413.58 psi
2. Linear temperature gradient equals 100°F
3. Axial force equal to 60,000 lbs
4. Torsional moment equal to 3,500,000 in-lbs

	Circumferential Stresses - psi		Longitudinal Stresses - psi		Stress Intensities - psi	
	Inside	Outside	Inside	Outside	Inside	Outside
ELBOW Program	30,091	-6,739	24,129	-12,701	32,165	14,362
Hand Calculation	30,091	-6,739	24,129	-12,701	32,071	14,362

TABLE 3A.3.10-1

BENDCORD PROGRAM - VERIFICATION PROBLEM

	Offsets Between Nodal Points (ft)	
	Manual	BENDCORD
X_1	4.27721	4.27715
Z_1	4.27721	4.27713
X_2	3.73046	3.73042
Z_2	4.76158	4.76151
X_3	3.73043	3.73041
Z_3	4.76152	4.76151
X_4	3.12932	3.12929
Z_4	5.17652	5.17642
X_5	3.12932	3.12929
Z_5	5.17652	5.17642
X_6	2.48254	2.48254
Z_6	5.51598	5.51588
X_7	2.48254	2.48254
Z_7	5.51598	5.51587
X_8	1.79956	1.77956
Z_8	5.77500	5.77489
X_9	1.79956	1.79959
Z_9	5.77500	5.77490

TABLE 3A.3.12-1

PSPECTRA PROGRAM VERIFICATION PROBLEM

MAXIMUM VALUE ENVELOPING

			ARS Curves			PSPECTRA	
Pt.	Dir.	Quake	El (ft)	Period (sec)	Accel. (ft/sec^2)	Period (sec)	Accel. (ft/sec^2)
1	N-S	OBE	718.5	0.168	0.550	0.168	0.896
			737.5	0.168	0.455		
			758.0	0.168	0.896		
2	E-W	SSE	718.5	0.282	0.670	0.282	1.360
			737.5	0.282	1.010		
			758.0	0.282	1.360		
3	N-S	SSE	718.5	0.075	0.150	0.075	0.230
			737.5	0.075	0.170		
			758.0	0.075	0.230		

TABLE 3A.3.12-2

PSPECTRA PROGRAM VERIFICATION PROBLEM

ABSOLUTE SUM ENVELOPING

		ARS Curves			PSPECTRA	
Dir.	Quake	El (ft)	Period (sec)	Accel. (ft/sec^2)	Period (sec)	Accel. (ft/sec^2)
N-S	OBE	718.5	0.282	0.670	0.282	
		737.5	0.282	1.010		
		758.0	0.282	1.360		
		Abs. Sum		3.040		3.040

TABLE 3A.3.12-3

PSPECTRA PROGRAM VERIFICATION PROBLEM

SRSS ENVELOPING

		ARS Curves			PSPECTRA	
Dir.	Quake	El	Period	Accel.	Period	Accel.
N-S	OBE	718.5	0.282	0.670	0.282	
		737.5	0.282	1.010		
		758.0	0.282	1.360		
			SSRS	1.822		1.822

TABLE 3A.3.12-4

PSPECTRA PROGRAM VERIFICATION PROBLEM

PEAK SPREADING*,**

	Hand Calculation		PSPECTRA	
	TLD (sec)	THI (sec)	TLD (sec)	THI (sec)
Peak No. 1	0.0768	0.120	0.076	0.120
Peak No. 2	0.278	0.435	0.278	0.435

NOTES:

*This table is based on the following values taken from the input ARS for two peaks as follows:

1. For Peak No. 1 - T(period) = 0.096 sec,
2. For Peak No. 2 - T(period) = 0.348 sec,
3. Percent spread on high side = 25, and
4. Percent spread on low side = 20.

**TLD represents lower bound period for spread peak.
THI represents upper bound period for spread peak.

TABLE 3A.3.18-1

APE PROGRAM VERIFICATION PROBLEM
COMPARISON OF DRILLED-IN ANCHOR LOADS

Attachment	Attachment Location	Loading	BIP (ST 361) Anchor Shear	BIP (ST 361) Anchor Tension	APE (ST 378, V01L00) Anchor Shear	APE (ST 378, V01L00) Anchor Tension
TS 4X4	Corner of Surface Mounted Plate	F_z = 3,000 lb	0	2,269.36	0	2,296.87
TS 4X4	Corner of Surface Mounted Plate	M_x = 12,000 in. lb	0	1,256.61	0	2,625.00
TS 4X4	Corner of Surface Mounted Plate	M_x = 8,000 in. lb M_y = 8,000 in. lb	0	1,204.13	0	3,500.0
L 4X4	Corner of Surface Mounted Plate	F_z = 3,000 lb	0	2,116.39	0	2,268.52
L 4X4	Corner of Surface Mounted Plate	M_x = 12,000 in. lb	0	1,418.15	0	2,333.33
L 4X4	Corner of Surface Mounted Plate	M_x = 8,000 in. lb M_y = 8,000 in. lb	0	1,506.74	0	3,111.11
TS 4x4	Center of Embedded Plate	F_z = 20,000 lb	0	12,742.6	0	12,752.1
TS 4x4	Center of Embedded Plate	M_x = 50,000 in. lb	0	4,285.7	0	5,934.86
TS 4x4	Center of Embedded Plate	M_x = 25,000 in. lb M_y = 25,000 in. lb	0	3,756.76	0	5,934.86
TS 4x4	Corner of Embedded Plate	F_z = 20,000 lb	0	17,591.00	0	17,607.22
TS 4x4	Corner of Embedded Plate	M_x = 50,000 in. lb	0	9,735.84	0	10,937.50
TS 4x4	Corner of Embedded Plate	M_x = 25,000 in. lb M_y = 25,000 in. lb	0	5,575.9	0	10,937.5
L 4x4	Center of Embedded Plate	F_z = 20,000 lb	0	14,585.3	0	15,940.12
L 4x4	Center of Embedded Plate	M_x = 50,000 in. lb	0	4,959.99	0	5,934.86
L 4x4	Center of Embedded Plate	M_x = 25,000 in. lb M_y = 25,000 in. lb	0	4,344.17	0	5,934.86

TABLE 3A.3.18-1 (Cont)

Attachment	Attachment Location	Loading	BIP (ST 361) Anchor Shear	BIP (ST 361) Anchor Tension	APE (ST 378, V01L00) Anchor Shear	APE (ST 378, V01L00) Anchor Tension
L 4x4	Corner of Embedded Plate	F_z = 20,000 lb	0	18,596.5	0	22,009.02
L 4x4	Corner of Embedded Plate	M_x = 50,000 in. lb	0	10,559.60	0	10,937.50
L 4x4	Corner of Embedded Plate	M_x = 25,000 in. lb M_y = 25,000 in. lb	0	8,648.15	0	10,937.50
C 4x5.4	Corner of Embedded Plate	F_x = F_y = 2,800 lb F_z = 1970 lb M_x = 30,430 in. lb M_y = 27,390 in. lb M_z = 0	1,763.35	11,067	1,763.35	32,351.37
TS 4x4	Along Edge of Surface Mounted Plate	F_z = 3,000 lb	0	1,237.02	0	1,458.33
TS 4x4	Along Edge of Surface Mounted Plate	M_x = 12,000 in. lb	0	786.89	0	1,250.0
TS 4x4	Along Edge of Surface Mounted Plate	M_x = 8,000 in. lb M_y = 8.000 in. lb	0	856.54	0	1,833.33

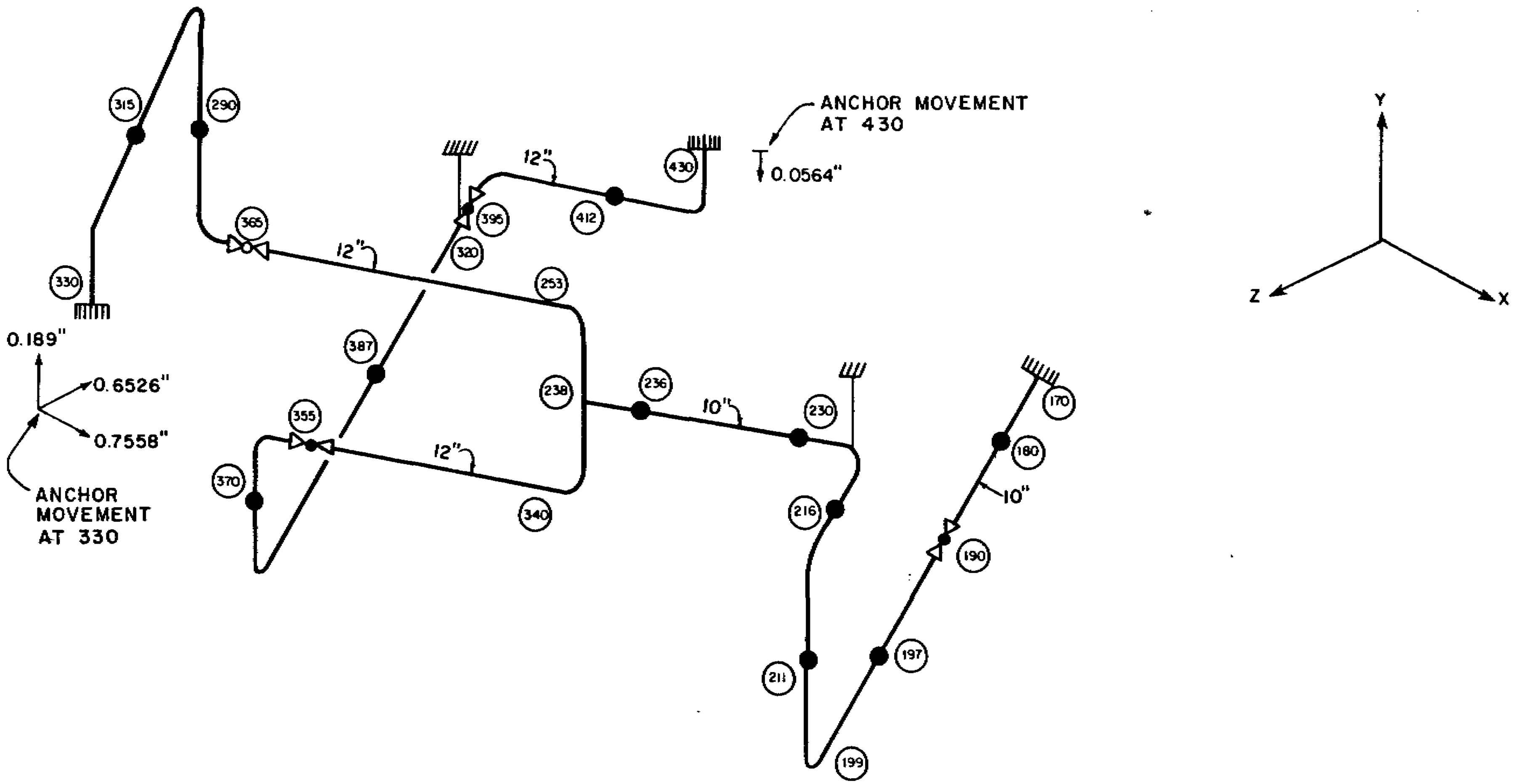


FIGURE 3A.3.1-1
MATHEMATICAL MODEL FOR FLEXIBILITY ANALYSIS VERIFICATION
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

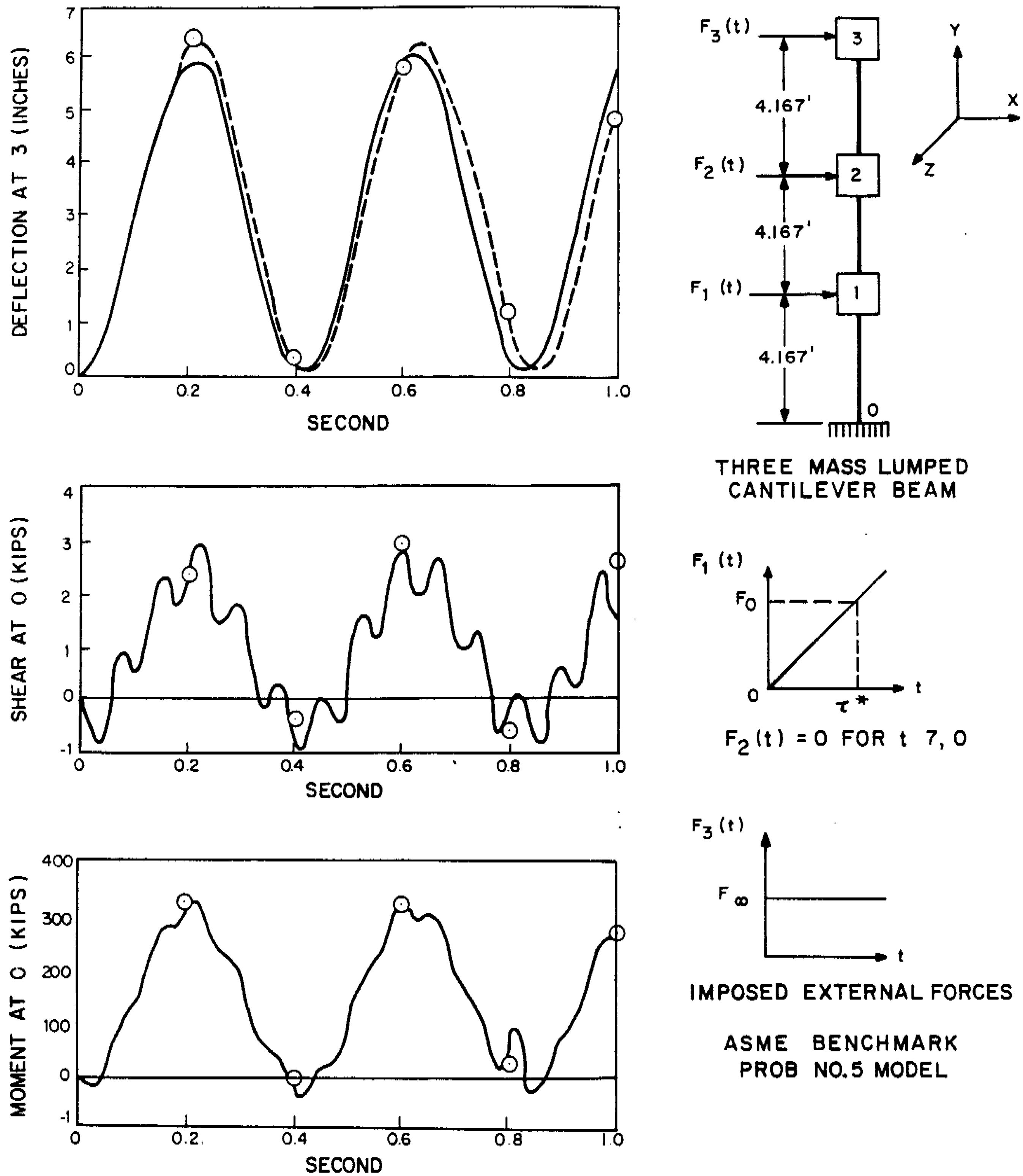
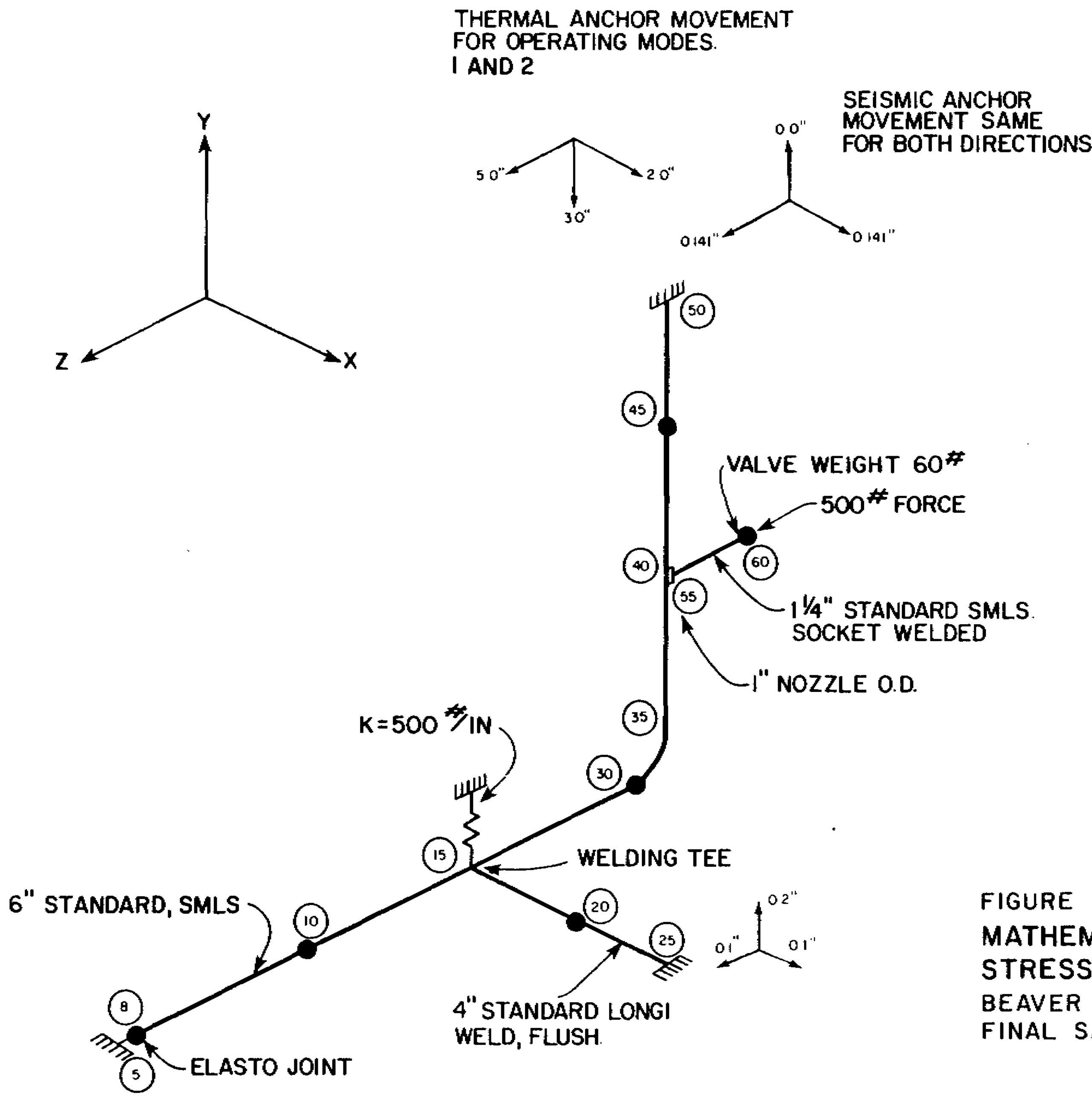


FIGURE 3A.3.1-2
NUPIPE PROGRAM FORCE TIME-HISTORY VERIFICATION
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

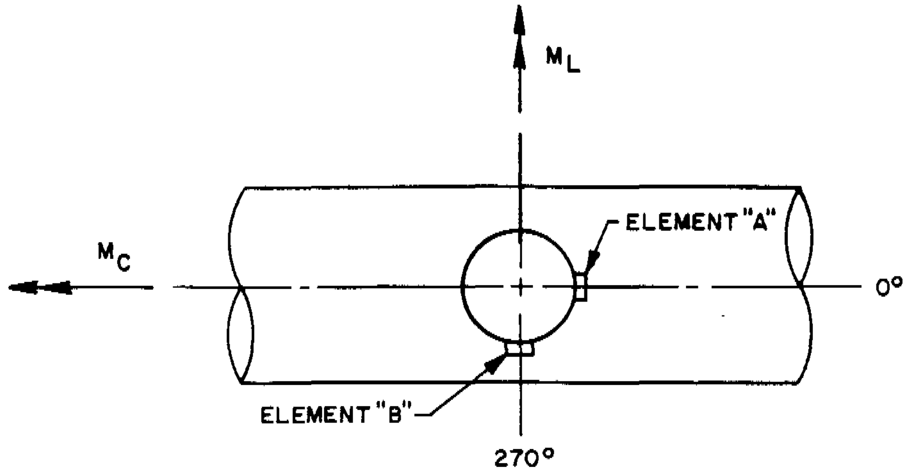


OPERATING CONDITIONS			
OPER MODE	PIPE	PRESSURE (PSI)	TEMPERATURE °F
1	6"	200	400
	4"	"	"
	1¼"	"	"
$E/\alpha_1 Ta - \alpha b Tb$ / AT 15 = 440 PSI			
2	6"	200	700
	4"	0	70
	1¼"	200	700
$\alpha \Delta T_1 = 0.0002$ $\alpha \Delta T_2 = 0.0004$ IN/FT			
3	6"	700	70
	4"	700	70
	1¼"	700	70

FIGURE 3A.3.1-3

MATHEMATICAL MODEL FOR CLASS I STRESS VERIFICATION

BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

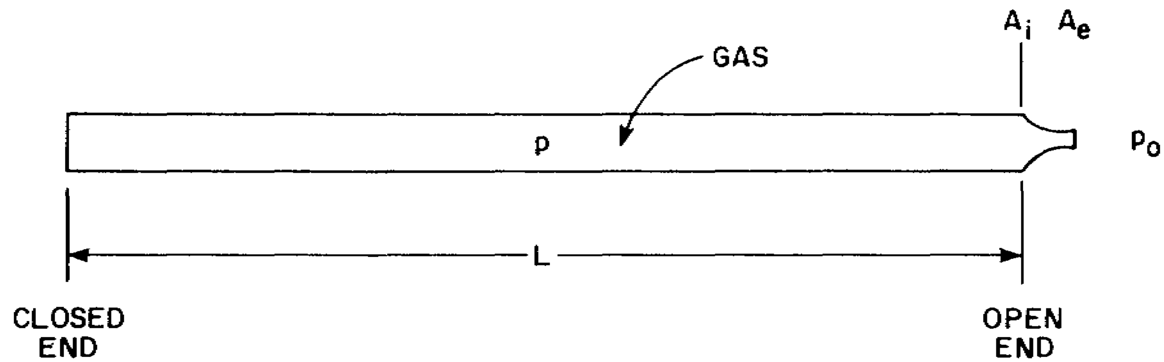


LOCATION AND CAUSE	PITRUST RESULTS	EXP. RESULTS (1)	
ELEMENT "A"			
LONGT. MOMENT			
CIRCUMF. STRESS	20,438.9 psi	20,000 psi	(HODGE-FIG.16)
AXIAL STRESS	26,292.6 psi	25,000 psi	
ELEMENT "B"			
CIRCUMF. MOMENT			
CIRCUMF. STRESS	22,016.2 psi	24,000 psi	(HODGE-FIG.15)
AXIAL STRESS	13,105.8 psi	13,000 psi	

SOURCE:

1. HODGE, P.G. PLASTIC ANALYSIS OF STRUCTURES. McGRAW-HILL BOOK COMPANY, INC., NEW YORK, N.Y., 1959.

FIGURE 3A.3.3-1
COMPARISON OF
PITRUST WITH HODGE'S RESULTS
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT



CASE (A) FOR COMPARISON WITH ANALYTICAL RESULTS.

INITIAL CONDITIONS:

$p_1/p_o = 4.72$, $a_o/a_1 = 0.80$

DIMENSIONS:

$A_e/A_i = 0.6$, L = PIPE LENGTH

SPECIFIC HEAT RATIO:

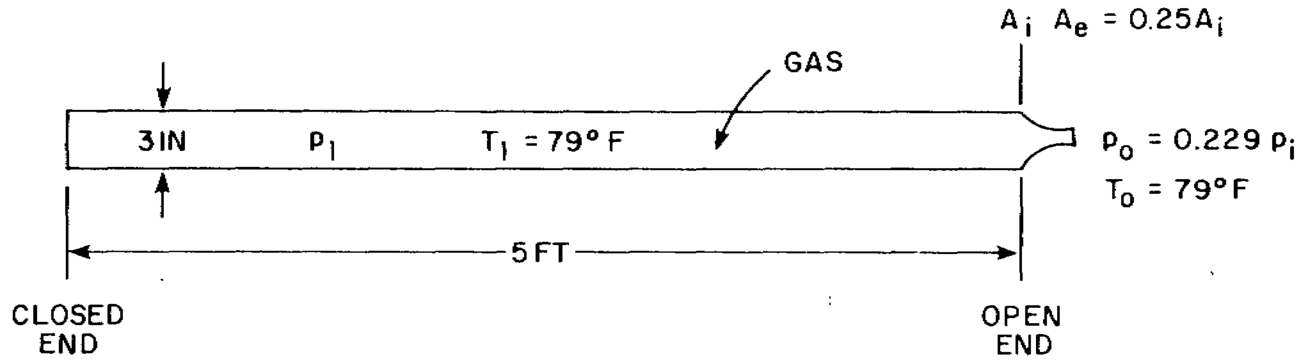
$\gamma = C_p/C_v = 1.4$

p = PRESSURE, a = SOUND VELOCITY, A = FLOW AREA

FIGURE 3A.3.6-1

SUDDEN DISCHARGE OF A GAS FROM A PIPELINE THROUGH A NOZZLE

BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT



CASE (B) FOR COMPARISON WITH EXPERIMENTAL DATA AND HAND CALCULATION.

PRESSURE = $p_0 = 14.7$ psia, $p_i = 14.7/0.229 = 64.2$ psia

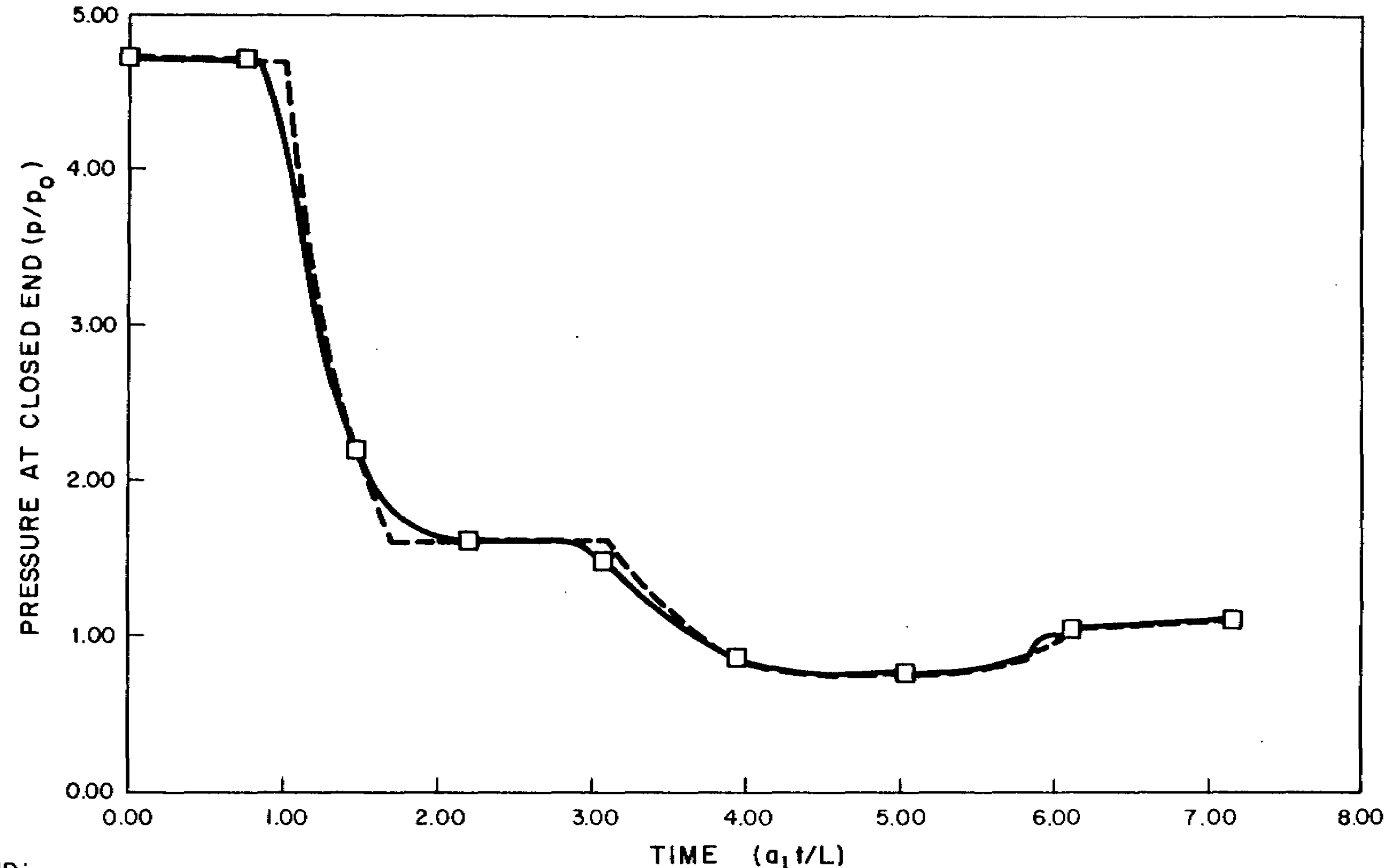
AREA = $A_i = \frac{\pi}{4}(0.25)^2 = 0.0491\ ft^2$, $A_e = 0.25A_i = 0.0123\ ft^2$

GAS CONSTANT = $R = 53.35\ ft\text{-}lb_f/lb°R$

TEMPERATURE = $T_0 = 79°F = 539°R$, $T_i = T_0$

DENSITY = $\rho_0 = \frac{p_0}{RT_0} = 0.0736\ lb/ft^3$, $\rho_i = \frac{p_i}{RT_i} = 0.3215\ lb/ft^3$

FIGURE 3A.3.6-2
SUDDEN DISCHARGE OF A GAS FROM A PIPELINE THROUGH A NOZZLE
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT



LEGEND:

(CASE A)

——— STEHAM

– – – – GIBERSON (1) AND WILSON (2) - ANALYTICAL SOLUTION

SOURCES:

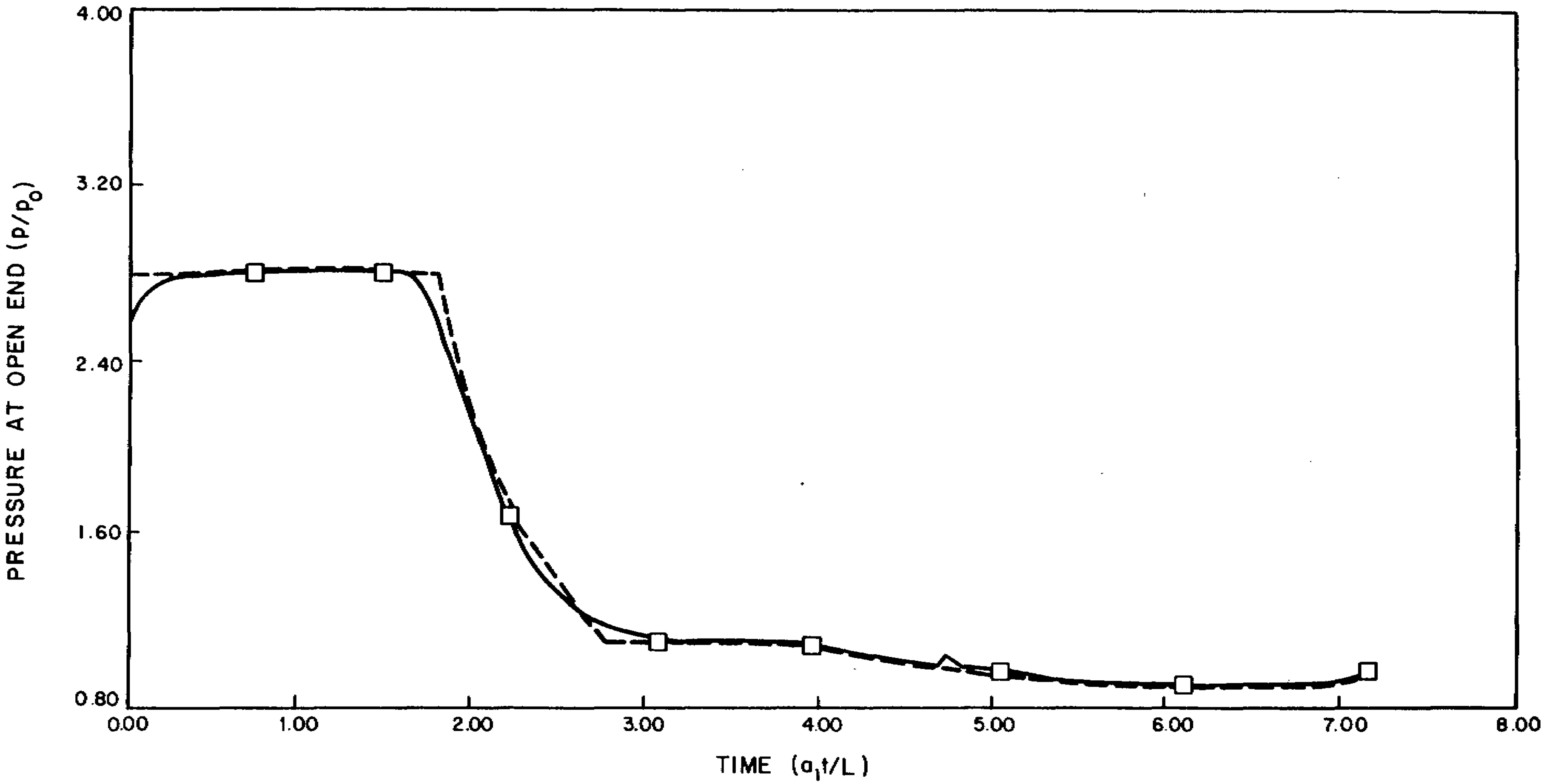
1. GIBERSON, M.F. THE RESPONSE OF NON-LINEAR MULTI-STORY STRUCTURES SUBJECTED TO EARTHQUAKE EXCITATION. EARTHQUAKE ENGINEERING RESEARCH LAB, CALIFORNIA INSTITUTE OF TECHNOLOGY, PASADENA, JUNE 1967.
2. WILSON, E.L. A COMPUTER PROGRAM FOR THE DYNAMIC STRESS ANALYSIS OF UNDERGROUND STRUCTURES. REPORT TO WATERWAYS EXPERIMENTAL STATION, U.S. ARMY CORPS OF ENGINEERS, REPORT NO. 68-1, STRUCTURAL ENGINEERING LABORATORY, UNIVERSITY OF CALIFORNIA, BERKELEY, JANUARY 1968.

FIGURE 3A.3.6-3

COMPARISON OF PRESSURE RESPONSE AT THE CLOSED END

BEAVER VALLEY POWER STATION-UNIT 2

FINAL SAFETY ANALYSIS REPORT



LEGEND:

(CASE A)

——— STEHAM

– – – – GIBERSON (1) AND WILSON (2) – ANALYTICAL SOLUTION

SOURCES:

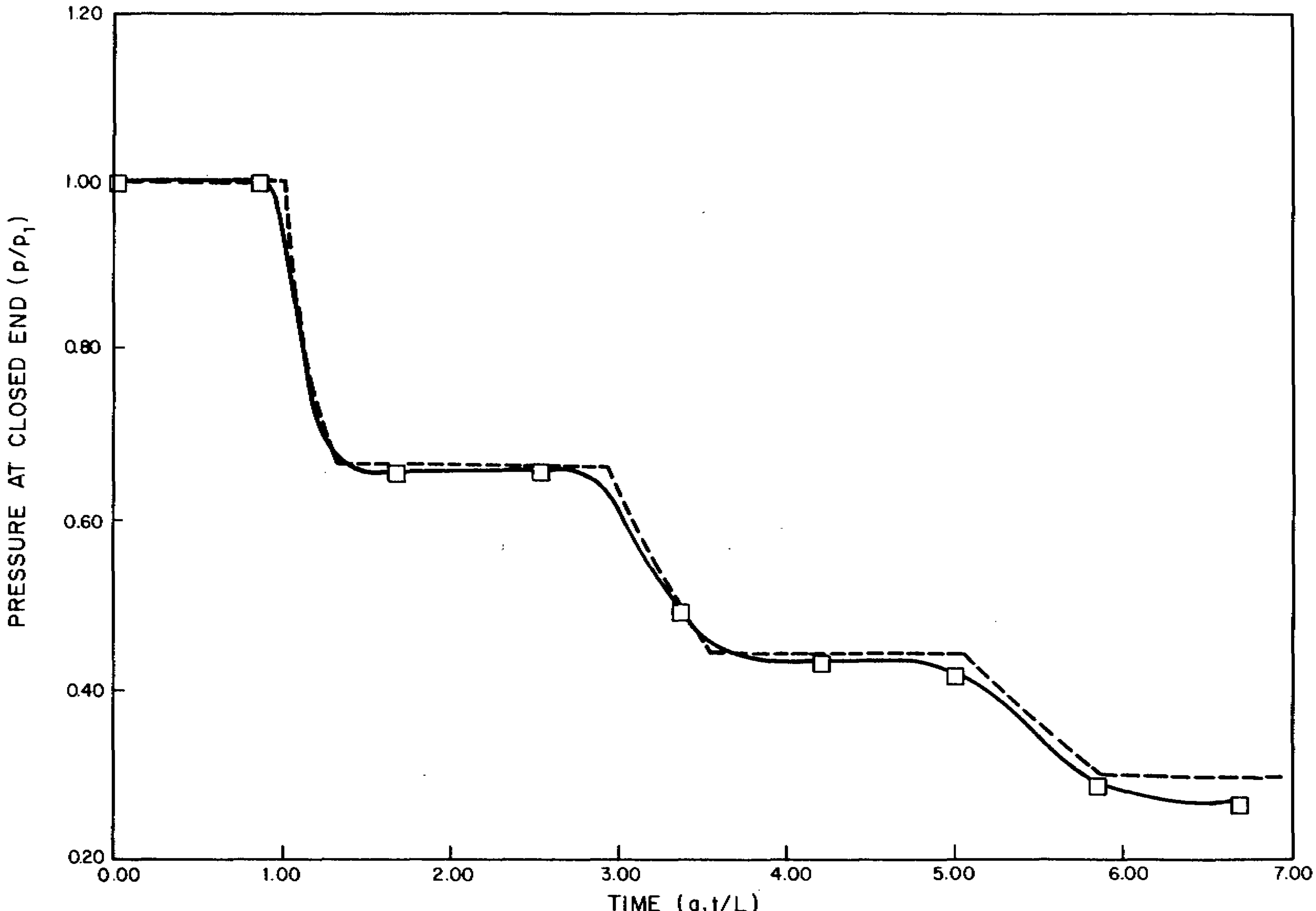
1. GIBERSON, M.F. THE RESPONSE OF NON-LINEAR MULTI-STORY STRUCTURES SUBJECTED TO EARTHQUAKE EXCITATION. EARTHQUAKE ENGINEERING RESEARCH LAB, CALIFORNIA INSTITUTE OF TECHNOLOGY, PASADENA, JUNE 1967.
2. WILSON, E.L. A COMPUTER PROGRAM FOR THE DYNAMIC STRESS ANALYSIS OF UNDERGROUND STRUCTURES. REPORT TO WATERWAYS EXPERIMENTAL STATION, U.S. ARMY CORPS OF ENGINEERS, REPORT NO. 68-1, STRUCTURAL ENGINEERING LABORATORY, UNIVERSITY OF CALIFORNIA, BERKELEY, JANUARY 1968.

FIGURE 3A.3.6-4

COMPARISON OF PRESSURE RESPONSE AT THE OPEN END

BEAVER VALLEY POWER STATION-UNIT 2

FINAL SAFETY ANALYSIS REPORT

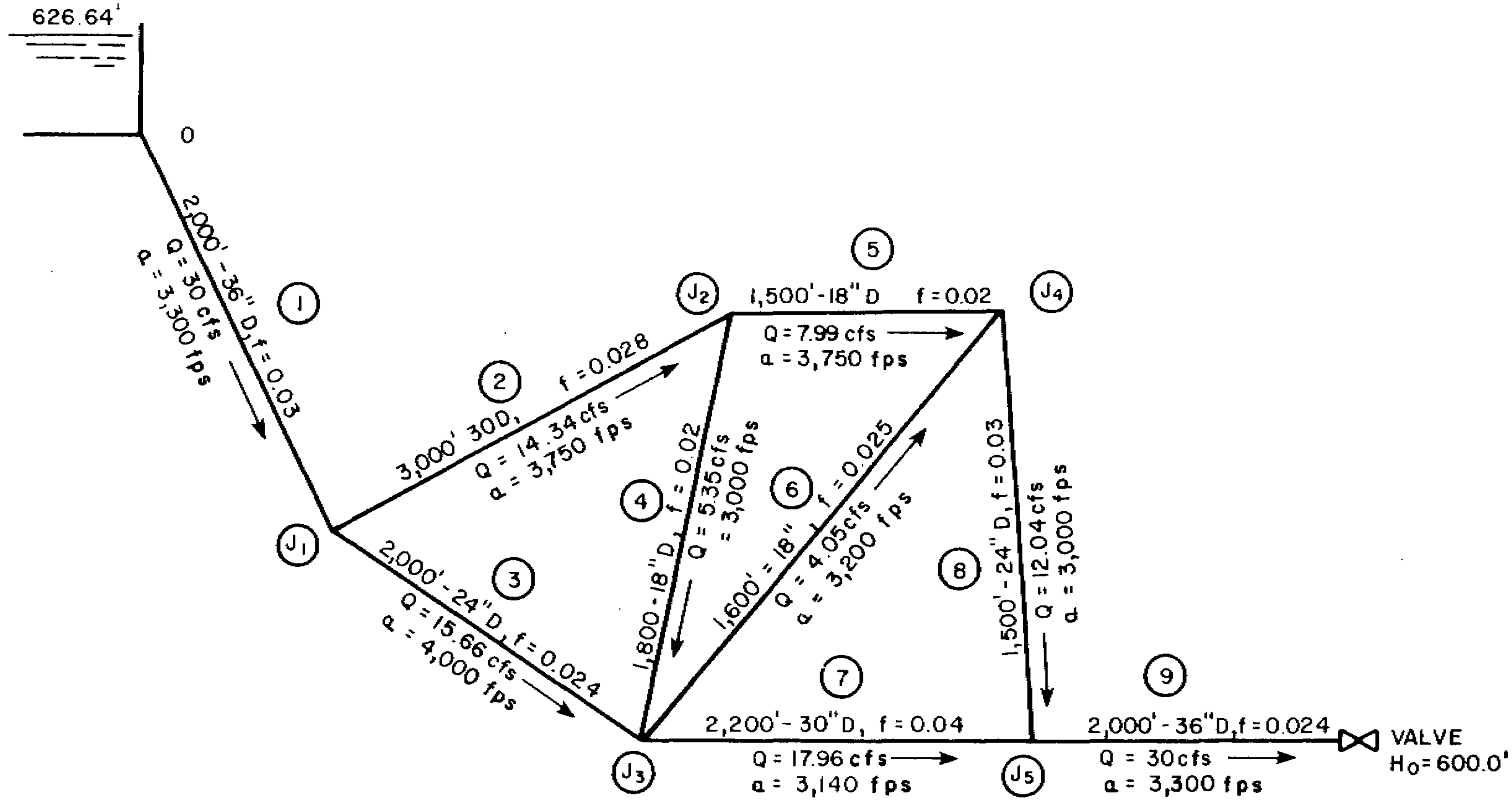


LEGEND:
(CASE B)
——— STEHAM
– – – GIBERSON (1) AND WILSON (2) - ANALYTICAL SOLUTION

SOURCES:

1. GIBERSON, M.F. THE RESPONSE OF NON-LINEAR MULTI-STORY STRUCTURES SUBJECTED TO EARTHQUAKE EXCITATION. EARTHQUAKE ENGINEERING RESEARCH LAB, CALIFORNIA INSTITUTE OF TECHNOLOGY, PASADENA, JUNE 1967.
2. WILSON, E.L. A COMPUTER PROGRAM FOR THE DYNAMIC STRESS ANALYSIS OF UNDERGROUND STRUCTURES. REPORT TO WATERWAYS EXPERIMENTAL STATION, U.S. ARMY CORPS OF ENGINEERS, REPORT NO. 68-1, STRUCTURAL ENGINEERING LABORATORY, UNIVERSITY OF CALIFORNIA, BERKELEY, JANUARY 1968.

FIGURE 3A.3.6-5
COMPARISON OF PRESSURE RESPONSES BY STEHAM AND EXPERIMENT
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT



LEGEND:

(5) PIPE NUMBER

(J_1) PIPE JUNCTION

D = INSIDE DIAMETER

f = FRICTION FACTOR

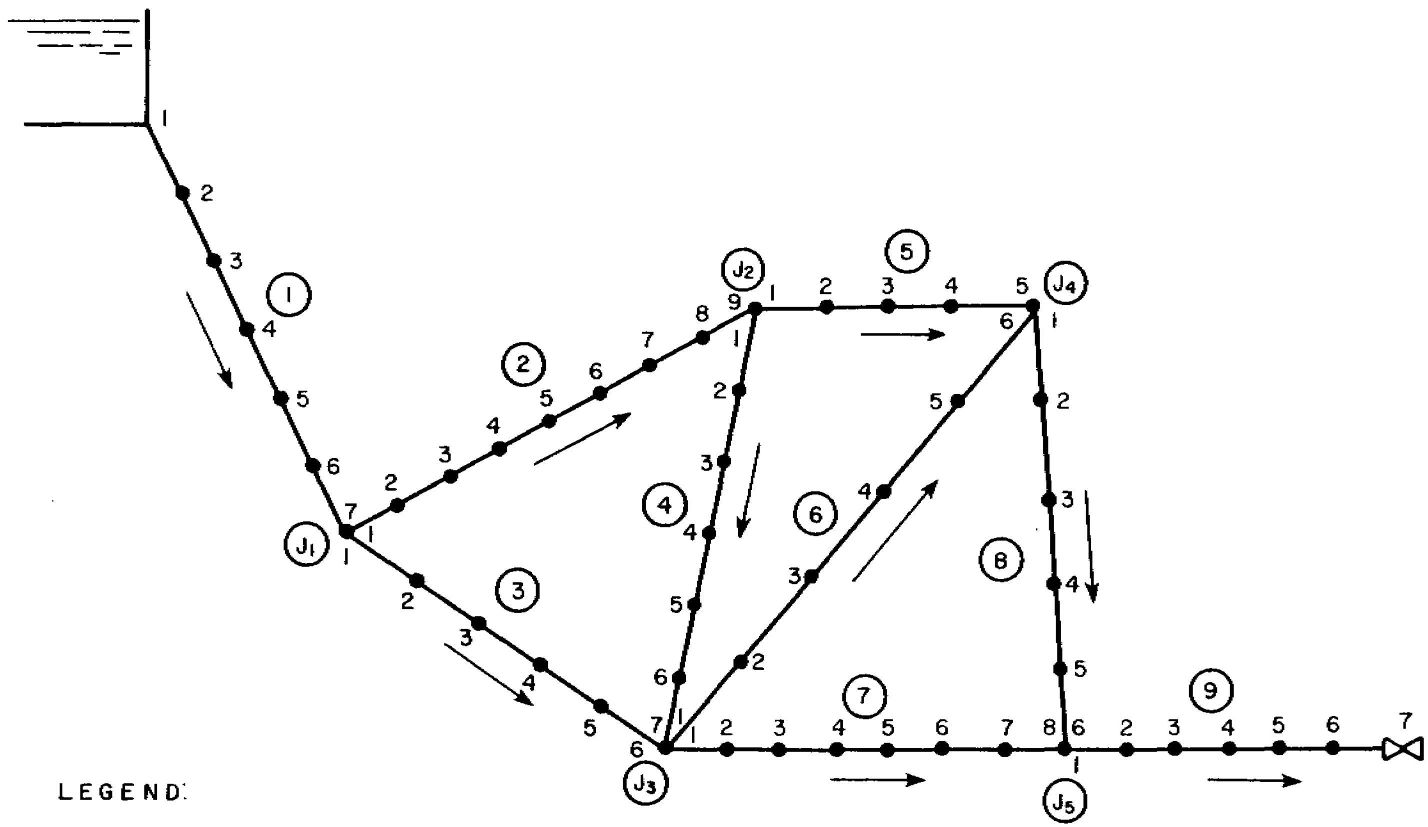
Q = FLOW RATE

a = SONIC SPEED (WATER)

FIGURE 3A.3.7-1

HYDRAULIC NETWORK FOR VERIFICATION PROBLEM

BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT



LEGEND:

(5) PIPE NUMBER

(J1) PIPE JUNCTION

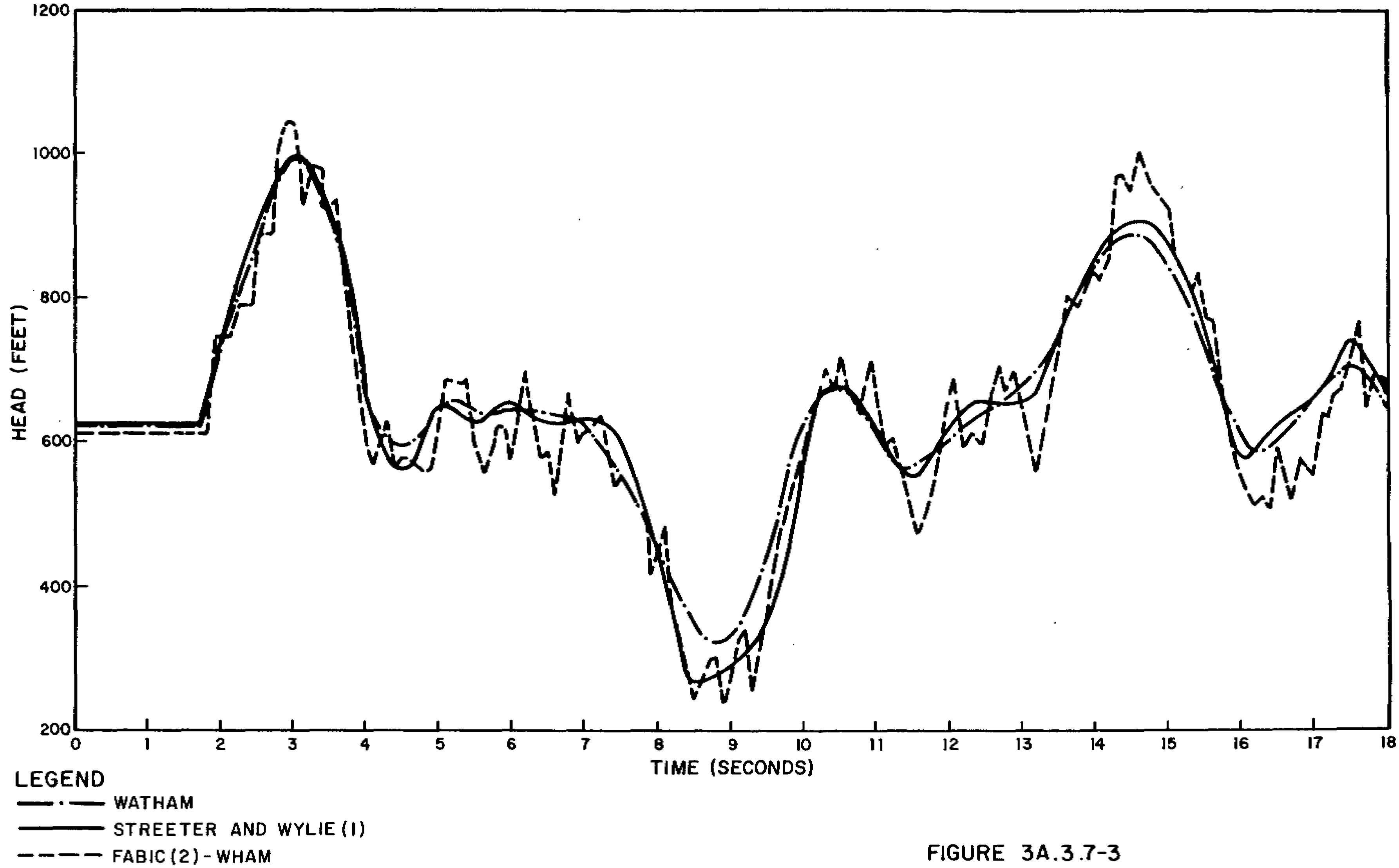
NODE NUMBER

VALVE

FIGURE 3A.3.7-2

HYDRAULIC NETWORK FOR WATHAM VERIFICATION

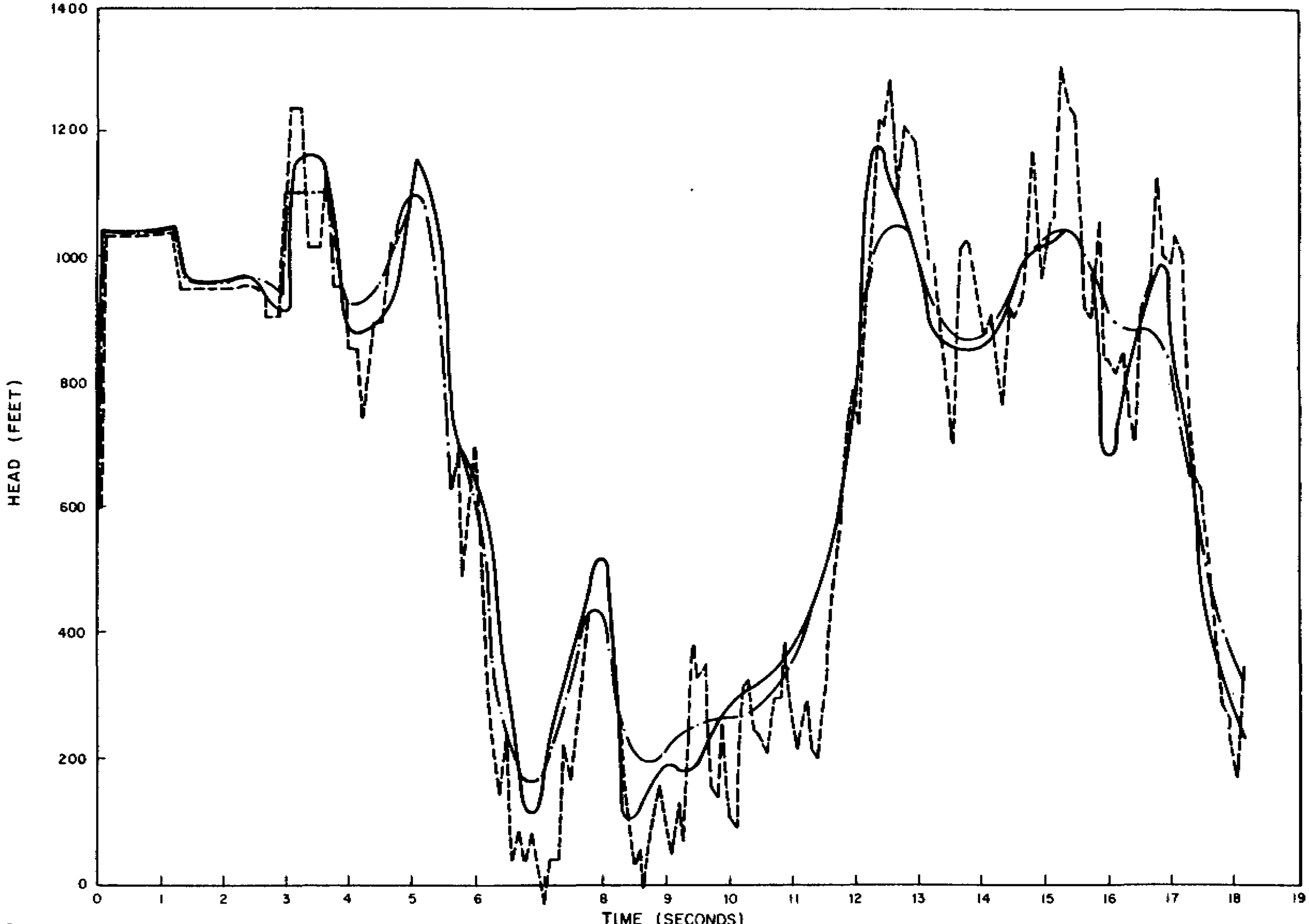
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT



SOURCES:

1. STREETER, V.L. AND WYLIE, E.G. HYDRAULIC TRANSIENTS. McGRAW-HILL BOOK COMPANY, INC., NEW YORK, N.Y., 1967.
2. FABIC, S. COMPUTER PROGRAM WHAM FOR CALCULATION OF PRESSURE, VELOCITY, AND FORCE TRANSIENTS IN LIQUID-FILLED PIPING NETWORKS. REPORT NO. 67-49-R, KAISER ENGINEERS, NOVEMBER 1967.

FIGURE 3A.3.7-3
HEAD VERSUS TIME-PLOT FOR JUNCTION J
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT



LEGEND

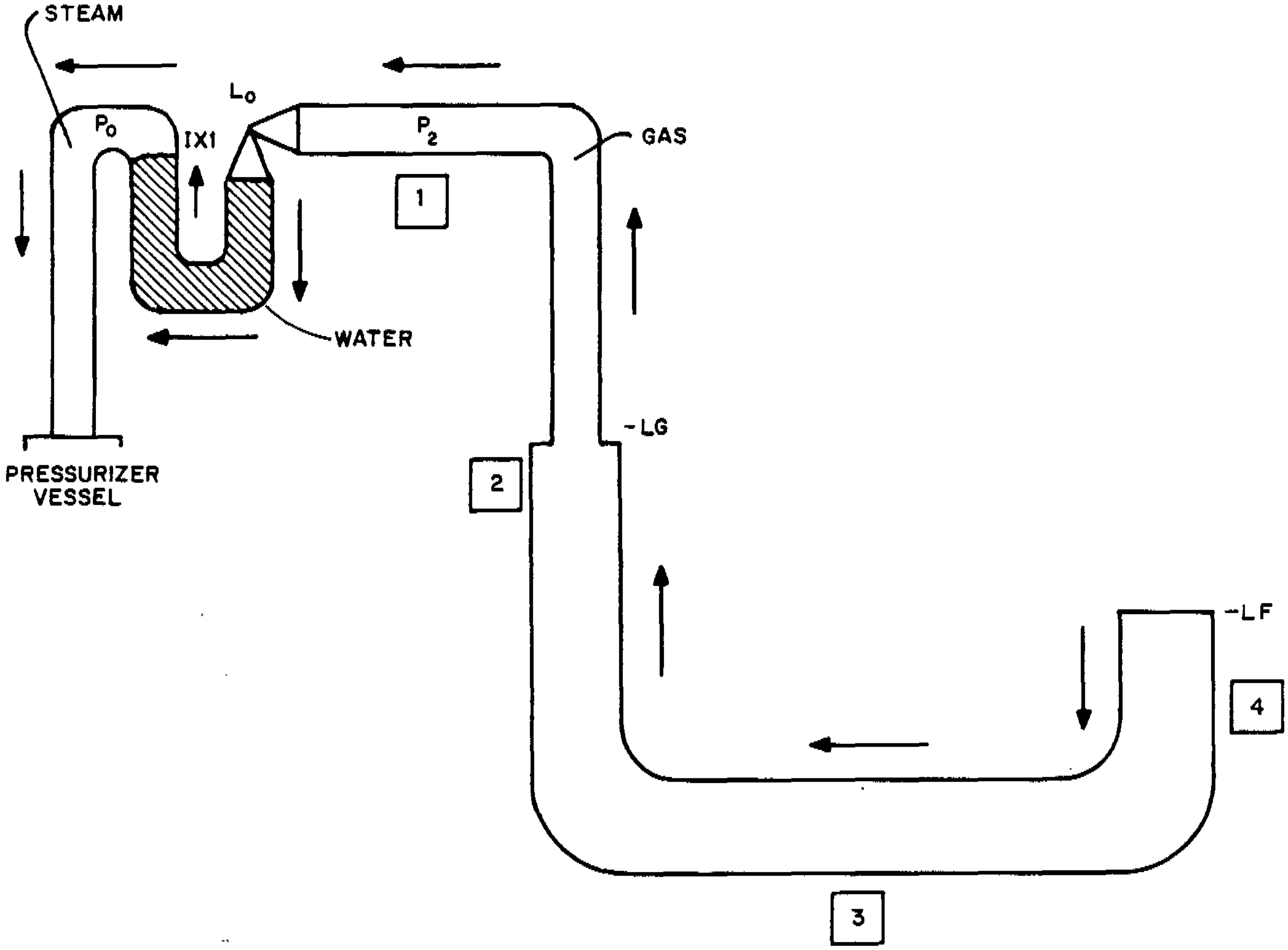
—·— WATHAM
—— STREETER AND WYLIE (1)
– – – FABIC (2) - WHAM

SOURCES:

1. STREETER, V.L. AND WYLIE, E.G. HYDRAULIC TRANSIENTS. McGRAW-HILL BOOK COMPANY, INC., NEW YORK, N.Y., 1967.
2. FABIC, S. COMPUTER PROGRAM WHAM FOR CALCULATION OF PRESSURE, VELOCITY, AND FORCE TRANSIENTS IN LIQUID-FILLED PIPING NETWORKS. REPORT NO. 67-49-R, KAISER ENGINEERS, NOVEMBER 1967.

FIGURE 3A.3.7-4
HEAD VERSUS TIME-PLOT AT VALVE

BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT



DISTANCES FROM PRESSURIZER VESSEL:

L_o = SAFETY/RELIEF VALVE

LG = DOWNSTREAM AREA CHANGE

LF = PIPE EXIT

IX1 = TRAILING EDGE OF SLUG 1

← FORCING FUNCTION DIRECTION, +

□ EPRI TEST 908
SEGMENT NO.

WATSLUG MODEL PIPE NUMBER

1 - PRESSURIZER TO L_o

2 - L_o TO LG

3 - LG TO LF

FIGURE 3A.3.8-1
WATSLUG MODEL OF EPRI SAMPLE PROBLEM

BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

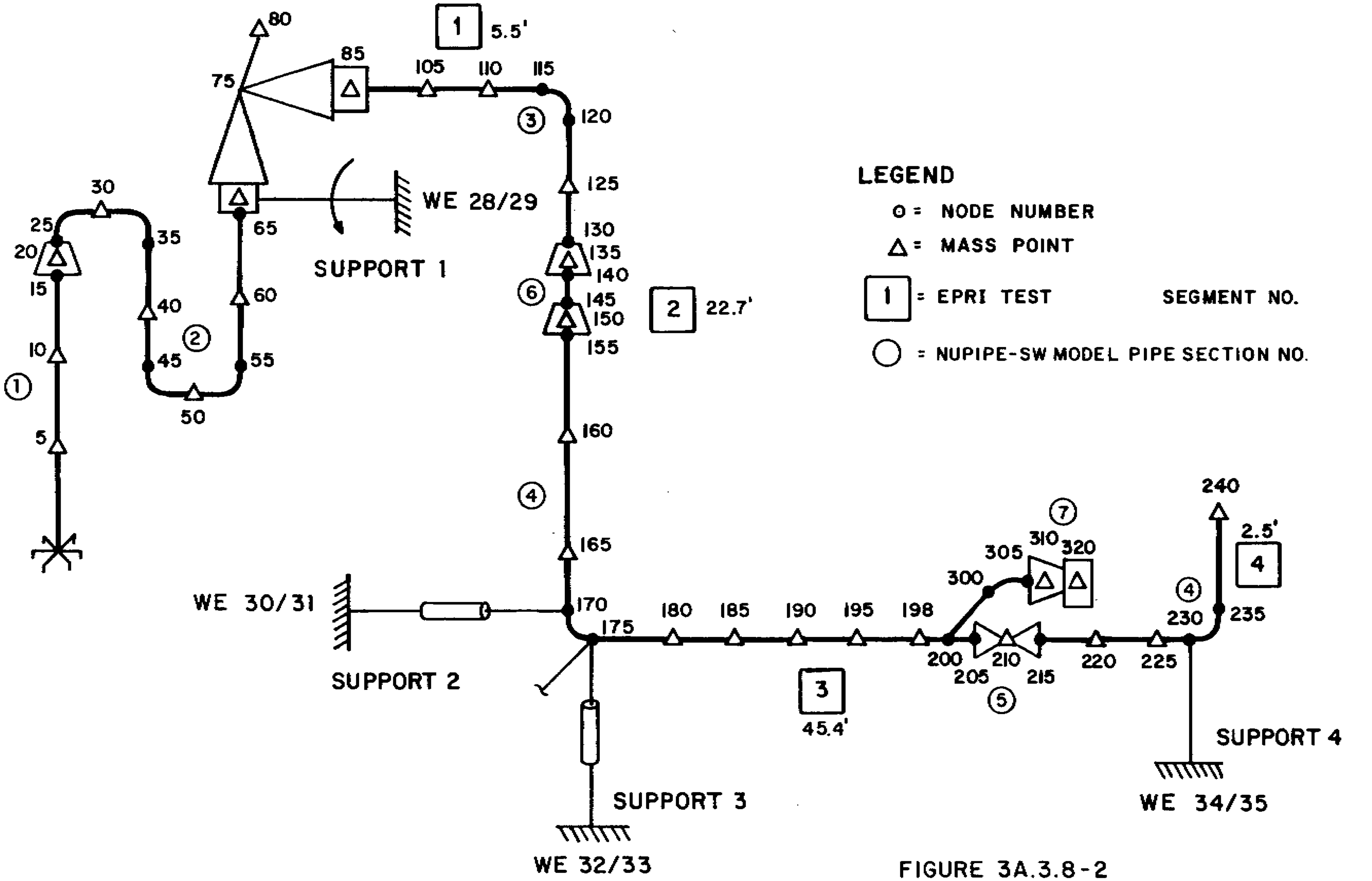


FIGURE 3A.3.8-2
NUPIPE-SW MODEL OF EPRI SAMPLE PROBLEM
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

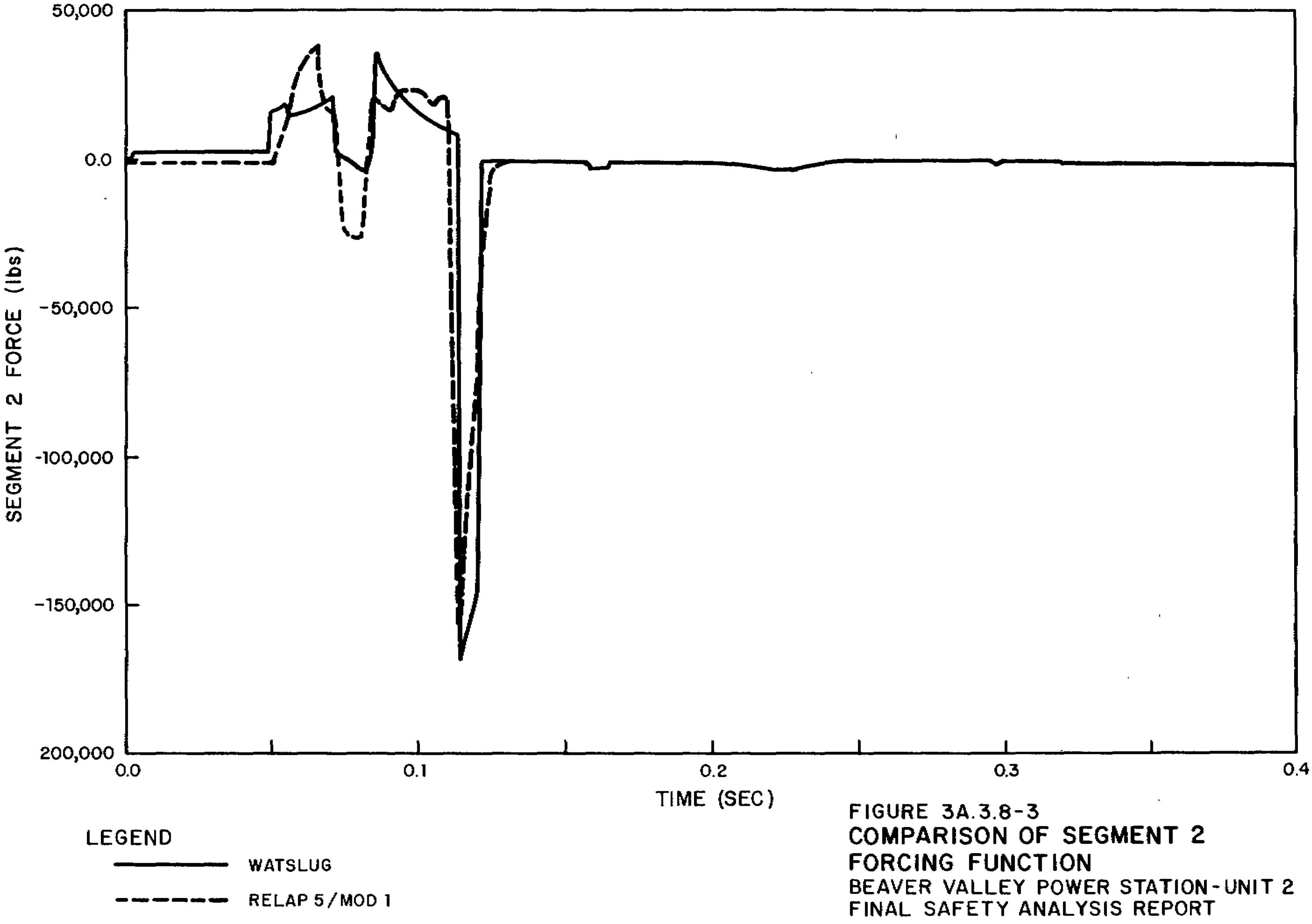


FIGURE 3A.3.8-3
COMPARISON OF SEGMENT 2
FORCING FUNCTION
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

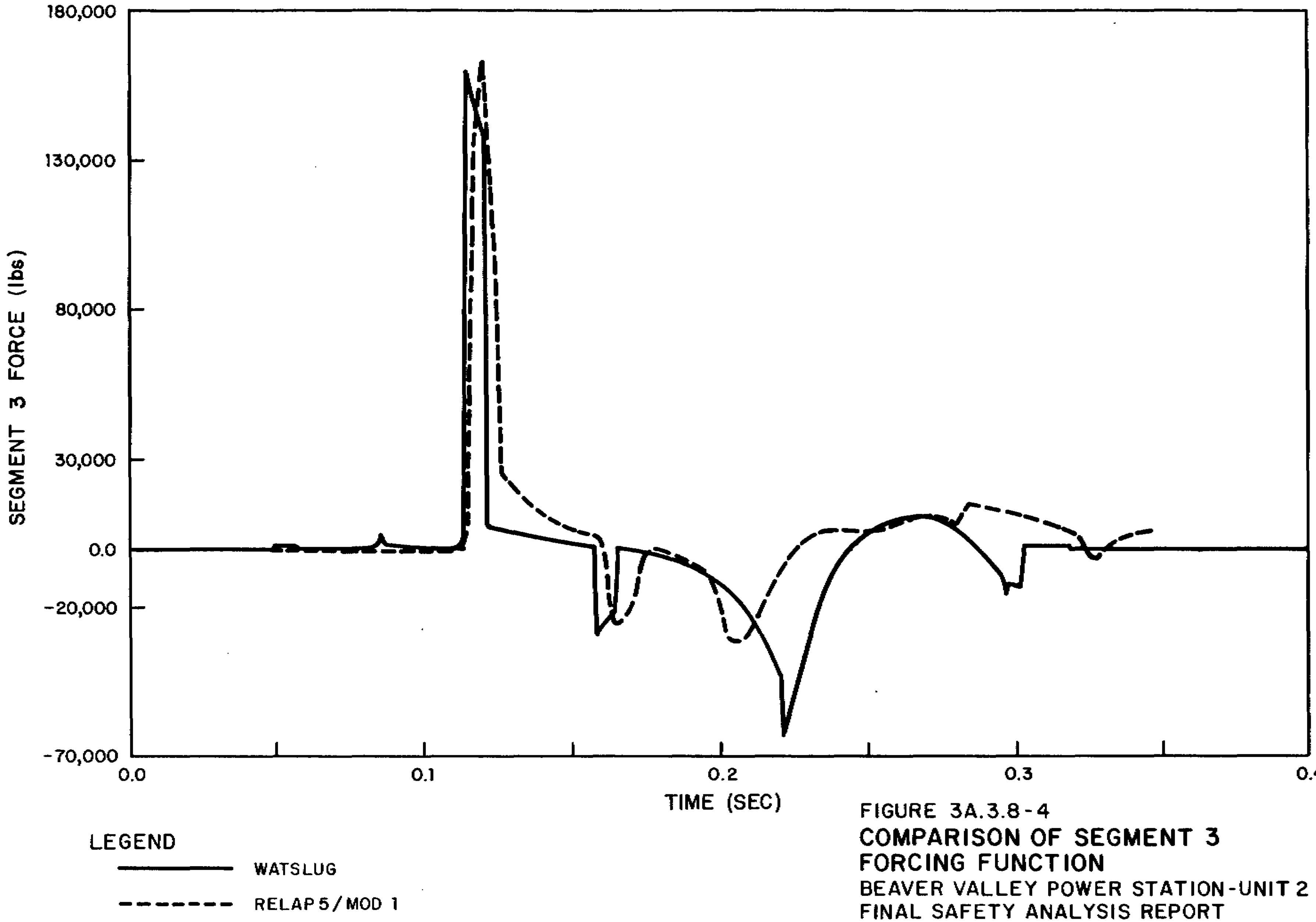
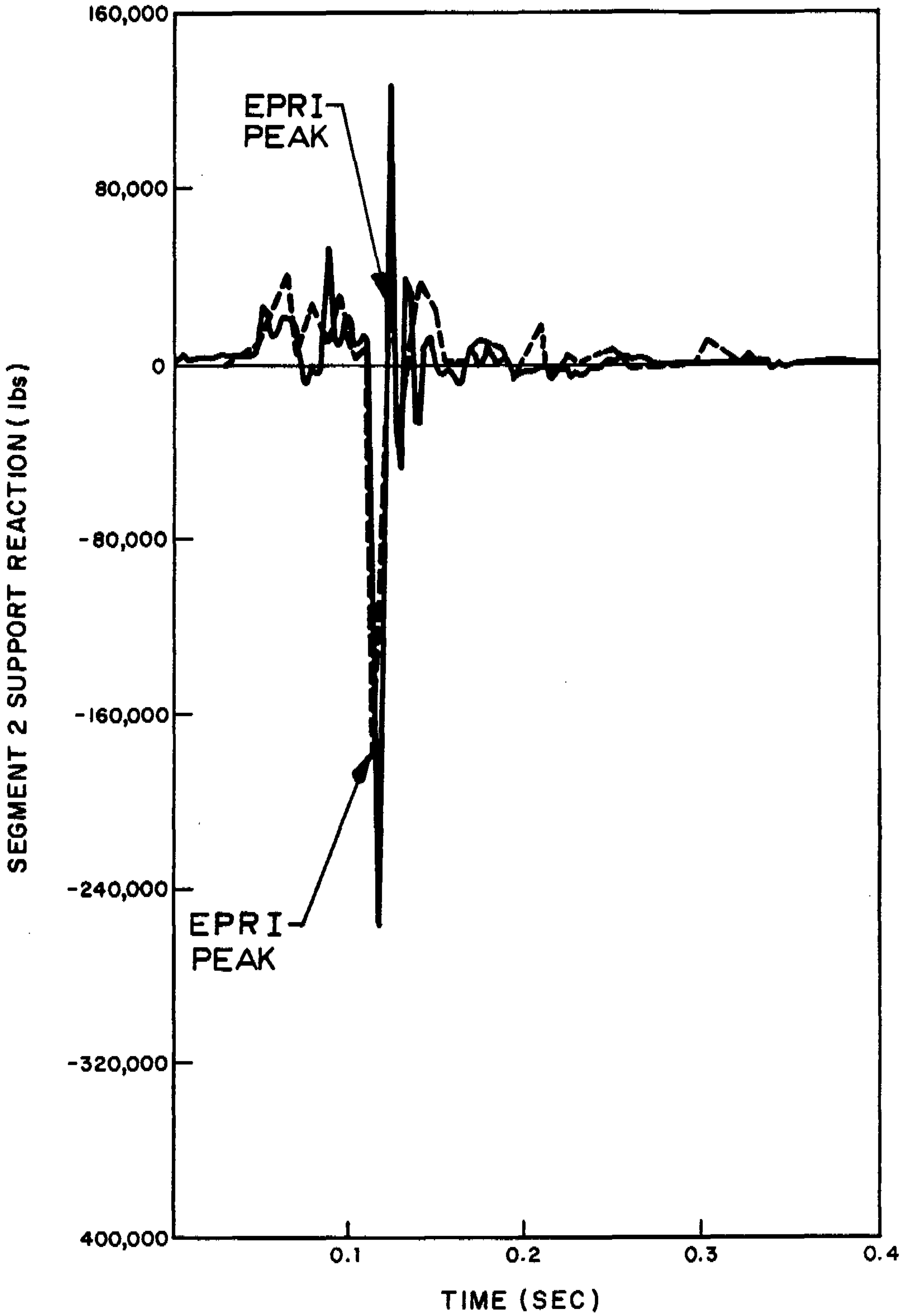


FIGURE 3A.3.8-4
COMPARISON OF SEGMENT 3 FORCING FUNCTION
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT



LEGEND

NUPIPE-SW

EPRI TEST RESULTS

FIGURE 3A.3.8-5
COMPARISON OF SEGMENT 2
SUPPORT REACTION
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

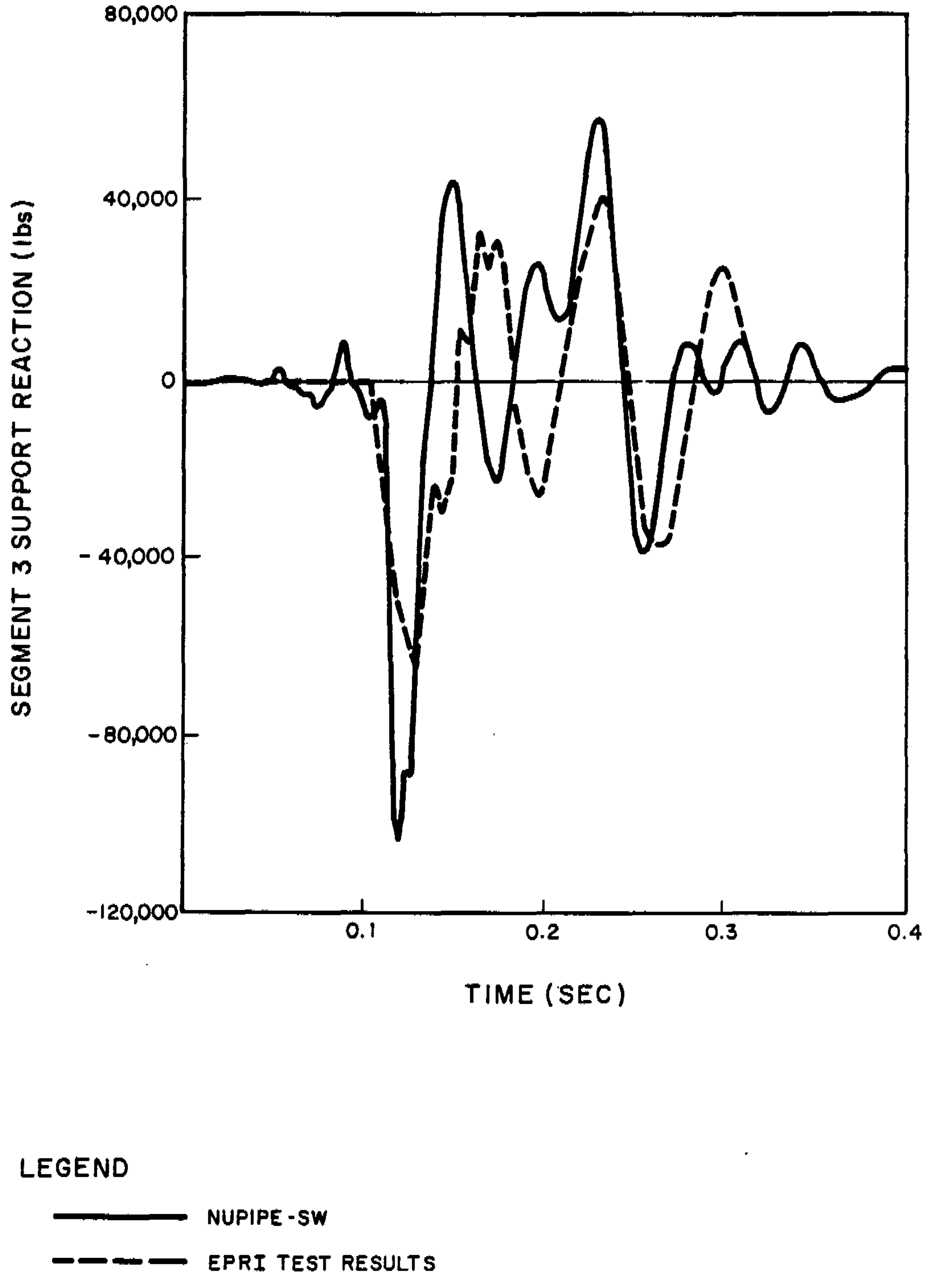
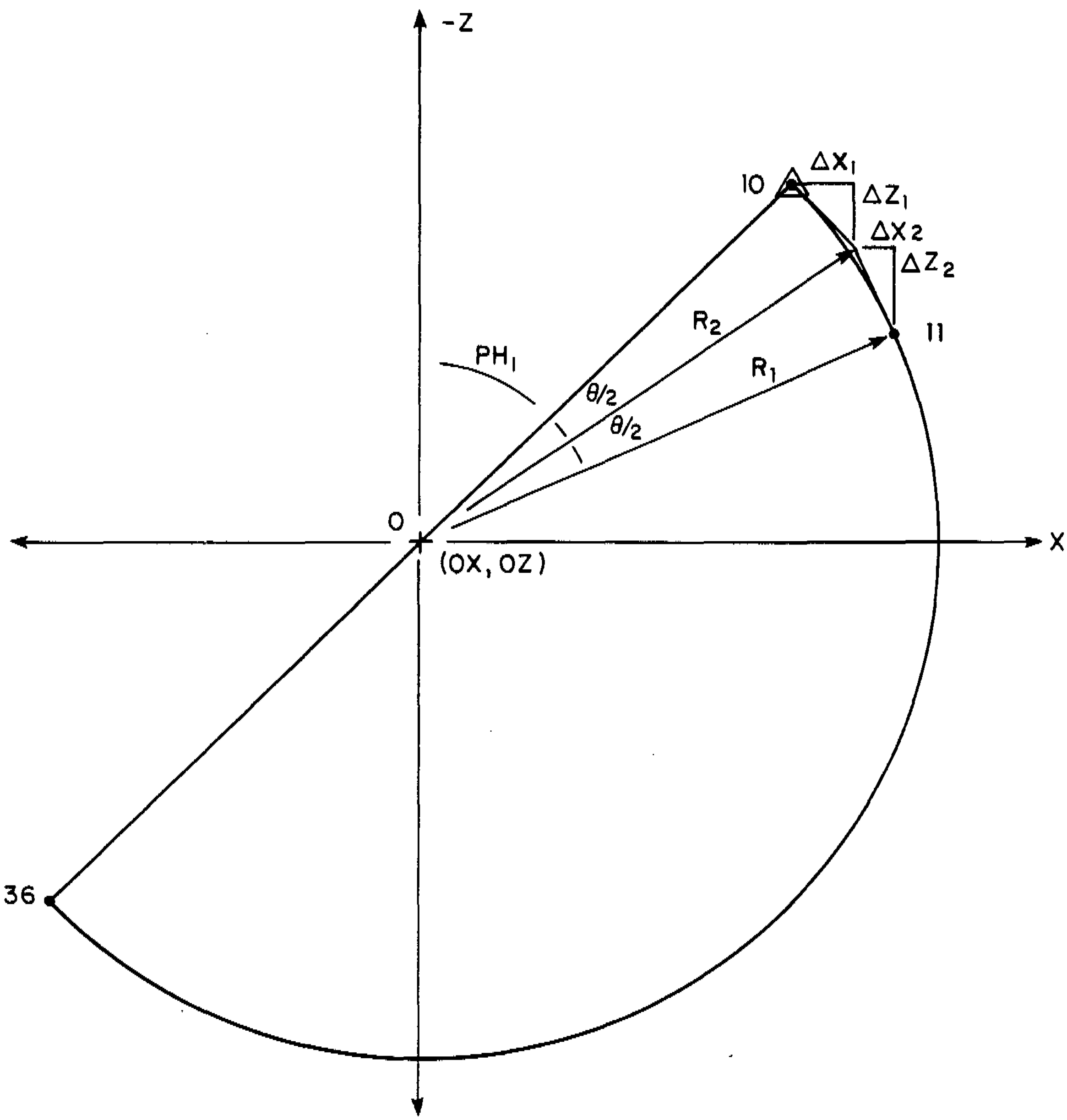


FIGURE 3A.3.8-6
COMPARISON OF SEGMENT 3 SUPPORT REACTION
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT



NOTES:

△ = DENOTES MASS POINTS

R_1 = 100 FT.

PH_1 = START ANGLE = 45°

OX = X COORDINATE OF ORIGIN OF ARC = 0.0 FT.

OZ = Z COORDINATE OF ORIGIN OF ARC = 0.0 FT.

OD = 14.0 IN.

θ = INCLUDE ANGLE BETWEEN NODES = 6.923°

NAGLE = NO. OF INCLUDED ANGLES IN ARC = 26

FIGURE 3A.3.10-1

DESIGN INPUT TO A PROBLEM

BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

APPENDIX 3B

A STUDY OF THE PROBABILITY OF AN AIRCRAFT COLLISION WITH THE BEAVER VALLEY POWER STATION - UNIT 2

BEAVER VALLEY POWER STATION

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APPENDIX 3B
LIST OF FIGURES

LIST OF FIGURES (Cont)

APPENDIX 3B

A STUDY OF THE PROBABILITY OF AN AIRCRAFT COLLISION WITH THE BEAVER VALLEY POWER STATION - UNIT 2

3B.1 INTRODUCTION

This section primarily describes the site characteristics for the Beaver Valley Power Station as they existed when the facility was licensed. As such, current site characteristics may not agree with these descriptions. This information was gathered to support or develop the original plant design bases and was accurate at the time the plant was originally licensed. It is considered historical and is not intended or expected to be updated for the life of the plant. Additionally, the operating term of the plant has been extended from 40 to 60 years by issuance of a renewed operating license. References to a 40-year plant life in this section are historical and have not been revised. Descriptions of requirements specific to the period of extended operation are contained in Chapter 19 of the UFSAR.

This appendix provides an assessment of the probability of an aircraft striking the safety-related structures of the Beaver Valley Power Station - Unit 2 (BVPS-2) in Shippingport, Pennsylvania.

To define this probability, it is necessary to identify all flight operations in the vicinity of BVPS-2 which, by their nature, involve a certain crash risk. This selection considers aircraft and airport proximity to BVPS-2, frequency of operations, and the crash history of a particular aircraft type. Airplane operations are divided into three categories: general aviation, air carrier, and military. They are further distinguished as to whether the phase of operation is airport-related (for example, some aspect of landing or takeoff) or in-flight (cruising). The previous operational breakdown leads to four separate cases for analysis.

Assuming an airplane crash to be a random phenomenon, mathematical models are constructed to compute the likelihood of an airplane impacting the plant structures. These models are based on the following:

1. The effective target area that the critical plant structures present to a specific aircraft type.

2. The number of aircraft movements (landings, takeoffs, or overflights) per year.

3. An aircraft-specific crash rate.

4. A spatial probability distribution function which measures the frequency of occurrence of a crash location with respect to a given reference mark (for example, an airport runway or an air corridor).

Information on airport facilities, operations, and area aircraft activity was obtained from airport managers, control tower chiefs, commanders of United States military units, regional Federal Aviation Administration (FAA) officials, FAA publications, and other documents (for example, airport master plan).

Data on general aviation and air carrier accidents were derived principally from aircraft accident tapes purchased from the National Transportation Safety Board (NTSB) in Washington, D.C., covering the period 1964 through 1975. The NTSB supplemental data were obtained from the Washington offices of the NTSB during an examination of the accident files for additional information not available on the tapes.

The Analysis Group at Norton Air Force Base, California, supplied data on United States Air Force (USAF) accidents (Directorate of Aerospace Safety undated).

Beaver Valley Power Station - Unit 2 is scheduled to go on line in 1986. With a 40-year operational period, it will be decommissioned in the year 2026. Probabilities are estimated for the current time period and for 2024 with the latter being a gross estimate since uncertain economic factors play a decisive role in any such calculation. Projections beyond 10 years are speculative. The analysis has not been extended from 2024 to 2026 because no significant changes are expected.

3B.2 GENERAL METHODOLOGY

The probability calculation is based on the following approximate general relationship:

$$P=NARD$$

where:

P = The probability of an aircraft collision with the target (per year).

N = The number of operations per year (landings or takeoffs, overflights).

A = The effective target area of BVPS-2.

R = A crash rate which is specific to aircraft type and operation.

D = A density distribution function which characterizes the location of crashes with respect to the airport or the air corridor.

Each term is defined and evaluated for the type of aircraft (general aviation, air carrier, and military) and the type of operation (airport-related or overflight). Using U.S. aircraft statistics, the following steps are performed:

1. Computing N: The number of current operations, and of those projected over the life span of BVPS-2, is estimated for those airports under consideration. The number of overflights during this same time period is computed for each air corridor.

2. Computing A: The effective target area presented by BVPS-2 is composed of three parts:

a. The plan area,

b. The shadow area, based on the projection onto the horizontal or the vertical face of the structures, and

c. The skidding area.

Representative aircraft impact angles used to determine shadow areas and skidding distances are computed for four cases:

a. Airport-related air carrier accidents,

b. In-flight air carrier accidents,

c. Airport-related general aviation accidents, and

d. Airport-related military aircraft accidents.

The shadow area is computed graphically for each impact angle. The skidding area is simply the product of the skidding distance and the length of the side of BVPS-2 exposed to the aircraft. Adding these contributions to the plan area, which is constant in all cases, yields the total target area.

3. Computing R: Crash data for each type of aircraft are analyzed to yield crash rates. General aviation accidents are treated as a whole. In-flight air carrier accidents are differentiated from air carrier accidents in general. Separate accident rates are generated for each type of military aircraft. The value, R, for airport-related accidents is expressed in units of crashes per operation. For in-flight accidents, the value is expressed in units of crashes per mile flown.

4. Computing D: By analyzing the distribution of accident locations relative to the end of the runway in terms of polar coordinates (r,θ) for each aircraft type, a distribution function can be calculated for airport-related accidents. Separate analyses are conducted for those accidents occurring within 5 miles of the airport and those beyond 5 miles of the airport. Military and air carrier operations in the area originate at Greater Pittsburgh Airport only. Because this airport is located more than 10 miles from BVPS-2, only one analysis is required for these operations.

A similar analysis is applied to air carrier in-flight accidents.

A distribution function is calculated based on perpendicular distance from the crash location to the centerline of the air route.

For the airport-related accidents, the value, D, is expressed in units of $(mi)^{-2}$. For in-flight accidents, this value is expressed in units of $(mi)^{-1}$.

3B.3 RESULTS

Table 3B-1 presents probability estimates of aircraft colliding with the critical structures of BVPS-2 for the present and future levels of aircraft activities in the area.

3B.4 NEARBY AIRPORTS AND AIR ROUTES

Five airports in the vicinity of the BVPS-2 could require individual accident analyses according to criteria set forth in Regulatory Guide 1.70. Greater Pittsburgh International Airport, Beaver County Airport, Fino Airport (all in Pennslyvania), Herron Airport (West Virginia), and Columbiana Airport (Ohio). Fino Airport qualifies solely on its proximity to BVPS-2. Greater Pittsburgh, Beaver County and Herron airports are considered due to present or projected levels of activity. Columbiana Airport, located more than 10 miles from the site, does not presently qualify for consideration nor will it in the future (the airport manager states that there has been a drain of area business to the Beaver County Airport in recent years and he does not foresee a reversal in the trend (Porter 1977).

Figure 3B.4-1 shows the location of the pertinent airports in relation to the BVPS-2. The facilities and operations at each airport are described in Sections 3B.4.1 through 3.B.4.4.

There are low level (V) and high level (J) flight routes in the vicinity of BVPS-2 but none that directly overlies the Shippingport area. The low level (federal) routes are 8 nautical miles in width; V12 lies east to west, and V37 is generally north to south. The high level (jet) routes are J53, J64, J145, J152, and J518; jet routes do not have a specified width. These routes are depicted schematically on Figure 3B.4-2. Details about distance from BVPS-2, frequency of overflights, etc, are presented in Section 3.B.4.5.

There are no low or high level military training routes or bombing ranges in the Pittsburgh area.

3B.4.1 Beaver County Airport

The master plan for Beaver County Airport (Baker 1974) contains current and projected data on airport facilities and operations. This airport is located 10 miles north of the Beaver Valley Power Station (BVPS). It has one paved runway (10-28) 4,500 feet long and 100 feet wide. It is a medium intensity runway, generally in good condition, with taxiway lights, runway markings, and a full-length paved parallel taxiway. It has a nonprecision instrument approach called very-high-frequency omni range (VOR) utilizing the Ellwood City VORTAC. All types of general aviation activity (for example, personal, business, commercial, instructional, and general purpose)

are represented at the airport. Recently, it has been designated as a reliever airport for Greater Pittsburgh International Airport (that is, it relieves congestion at a major airport, which has a high density of scheduled certificated airline traffic, by diverting general aviation traffic from the major airport to the reliever airport). The Beaver Valley Expressway (Route 60) will contribute considerably to this function when completed. Table 3B-2 gives the flight services offered at Beaver County Airport. Operations can be summarized as follows: (Pennsylvania Department of Transportation 1974).

Operation	Percentage of Aviation Traffic
Pleasure	10
Training	50
Business	10
Charter (Passengers)	25
Charter (Cargo)	5

Table 3B-3 presents estimates of the capacity of the single runway. There are no plans to expand this particular runway. With respect to flight patterns, Beaver County Airport experiences its peak demand levels in visual flight rules (VFR) conditions. A single VOR instrument approach procedure provides limited capability to operate within the instrument flight rules (IFR) system. Any VFR aircraft desiring to penetrate the Pittsburgh Terminal Control Area (TCA) must obtain an Air Traffic Control (ATC) clearance prior to operating within this area. In the vicinity of the Beaver County Airport, this designated airspace begins at 4,000 feet mean sea level (msl) (2,748 feet above ground level (agl) at Beaver County Airport) and terminates at 8,000 feet msl. Aircraft at the airport utilize a standard traffic pattern at an altitude of 800 feet agl for light single-engine aircraft and at 1,000 feet agl for multi-engine aircraft. In contrast, the IFR airspace is structured by navigational aids and airways. Figure 3B.4-3 illustrates the low-altitude IFR environment in the vicinity of the Beaver County Airport and Figure 3B.4-4 shows the approach procedures to Beaver County. These airways and navigational aids allow aircraft to transit the area and define instrument approach procedures for the various local airports capable of handling IFR aircraft.

Aircraft activity at Beaver County Airport is expected to increase in future years (Table 3B-4). Extrapolating this trend to the year 2024 yields 131,920 movements in that year. The increase corresponds to an annual average increase of 6 percent over the base (starting) figure.

For the immediate future, a previously used sod runway (2-20) is proposed to be reopened to accommodate small aircraft at those times when winds do not favor operations on the main runway. This secondary, or crosswind, runway is 3,000 feet long and 75 feet wide.

In addition, the installation of a localizer-type nonprecision instrument approach to Runway 10 is recommended.

3B.4.2 Greater Pittsburgh International Airport

Information on the Greater Pittsburgh Airport was supplied primarily by the Chief of Operations at the Greater Pittsburgh Tower (Bernhard 1977). The runway at Greater Pittsburgh Airport closest to BVPS-2 is 28R-10L approximately 11.2 miles away. It is 10,500 feet in length. The other three runways, 28L-10R, 32-14, and 23, have respective lengths of 9,500, 8,100, and 3,936 feet. The airport has an FAA tower and a surveillance radar approach to all runways. Air carrier, military, and general aviation aircraft utilize the facilities. The military operations include an interceptor squadron, an aerial refueling squadron, and a troop carrier group. The 911th Tactical Airlift Group of the USAF Reserve, the 147th Air Refueling Wing, and the 112th Tactical Fighter Group of the Pennslyvania Air National Guard are stationed at Greater Pittsburgh. A breakdown of the aircraft is presented in Table 3B-5 with air taxi operations separated out from general aviation.

No specific record is maintained of aircraft types utilizing a particular runway. However, when landings are to the west, most air carrier and military traffic use runway 28L, with air taxi and general aviation using runway 32 or 23, while departures are on runway 28R. When landings are to the east, air carrier and military traffic use runway 10L, with air taxi and general aviation using runway 10R and departures are on runway 14. Table 3B-6 gives the distribution of traffic by runways, as provided by the facility chief. Figures 3B.4-5 and 3B.4-6 depict the routes and holding patterns in the Greater Pittsburgh airspace. All traffic, whether air carrier, air taxi, general aviation, or military, uses the same arrival/departure route structure. Where the departure and arrival tracks intersect, altitude separation is maintained. The departure is tunneled at 5,000 feet and the arrival maintains 6,000 feet until intersection is accomplished. In the immediate area of the BVPS-2, most traffic is at 5,000 feet msl or higher.

The purposes for which general aviation operations are conducted are summarized as follows:

Operation	Percentage of Aviation Traffic
Business	63
Commercial charter (Passengers)	20
Mail service	10
Commercial charter (Cargo)	5
Pleasure	2

In 1976, there were 116,480 general aviation operations, including air taxi movements. Because Beaver County Airport has been assigned the function of a reliever airport for Greater Pittsburgh

International, the rate of growth of general aviation traffic at this airport will probably be less than the national average. However, in projecting activity over the life of BVPS-2, an annual increase of 6 percent of the base year's figure is invoked. This approach, which is consistent with national statistics, (FAA 1976a), leads to a figure in year 2024 of 452,000 operations. National predictions extend only to 1988 and several factors could intervene to alter the predictions.

In 1976, there were 182,610 air carrier operations. The FAA (1976a) predictions point to a 3.4 percent annual average increase over the base year's figure. In other words, applying the given fractional increase to 182,610 and then multiplying by the number of years (48) yields the total increase between 1976 and 2024. The final result is about 481,000 operations in 2024. A new runway, to be labeled runway 28L, is being constructed 1,500 feet south of the present 28L. The present 28L will be retained and designated 28 center. The new runway will be 200 feet in width and approximately 12,500 feet in length. Completion may not be for 3 or 4 years. This runway will be suitable for any type of aircraft.

Data on military aircraft movements were provided by the individual Air Force units stationed at Greater Pittsburgh International (Prave 1977; Bright 1977, and Bitonti 1977). The l12th Tactical Fighter Group flies the A7 aircraft in 2,600 to 2,800 sorties per year (a maximum of 5,600 operations per year). The 911th Tactical Airlift Group flies C-123 aircraft in 1,200 sorties per year (2,400 operations per year). The 147th Air Refueling Group flies the KC 135 aircraft in 12 to 15 sorties per week (a maximum of 1,560 operations per year). Over the past 10 years, the level of military operations has declined at Greater Pittsburgh Airport. According to the commanders of the various units, operations are expected to remain stable in the near future and local long range forecasts are not available. A national forecast for military air activity (FAA 1976b) up to 1988 indicates zero growth.

Beaver Valley Power Station - Unit 2 does not fall within the TCA of the Greater Pittsburgh Tower. Military aircraft entering or exiting from the TCA are controlled by the Pittsburgh Tower in the same manner as general aviation or air carrier planes. They do not lock into any training routes until they are well away from this area. The area training routes, SR815,816,817,818 (Bitonti 1977), which are low level, low speed VFR routes, begin at distances not less than 18 miles from BVPS-2. The C-123 aircraft generally fly these routes 1,000 feet above ground level (Bitonti 1977).

The KC-135 aircraft fly their air refueling tracks always above 12,000 feet (Bright 1977) with most of their activity concentrated in the Columbus, Ohio, area. The A-7 aircraft, once outside the control of the Greater Pittsburgh Tower, fly high speed direct routes to Warren Grove, New Jersey (60 percent of the time), and Atterbury, Indiana (25 percent of the time). The remaining 15 percent of the flights is divided among Watertown (New York), Grailing (Michigan),

Columbus (Ohio), and Harrisburg (Pennsylvania). The closest lateral approach of the aircraft to BVPS is 5 miles (Prave 1977). No accident analysis for the in-flight (training route) phase of USAF activity was deemed necessary.

3B.4.3 Herron Airport

Information on Herron Airport was obtained from the Airport Directory (1977) and from the airport manager (McVay 1977). This public airport, which is located 8.2 miles southwest of the BVPS-2 site, has three runways: 4-22, which is 2,018 feet long by 25 feet wide with an asphalt surface; 1-19, which is 1,400 feet long by 30 feet wide; and 14-32, which is 800 feet long by 30 feet wide. The latter two have grass compositions. No quantitative estimate is available as to the relative frequency of use of the runways, although the asphalt runway does experience the heaviest traffic. There are runway lights but no instrument approach systems. Fifty-three aircraft are based at Herron. A breakdown of the aircraft is given in Table 3B-7. The majority are light single-engine planes weighing less than 5,000 pounds. The Cessna 310 and Beechcraft Baron have respective gross weights of 5,300 and 5,400 pounds.

In lieu of any records, it has been estimated (Scheff 1977) that local plus itinerant operations average 500 per year per based airplane, resulting in approximately 26,500 operations per year at Herron.

The airport manager does not anticipate any substantial change in the level of activities in the near future and cannot make any projections beyond that time. There are no plans to expand the facilities.

The national trend is toward a steady rise in general aviation movements in the years to come (FAA 1976a), barring any major economic setbacks.

For purposes of analysis, Herron Airport is conservatively assumed to remain competitive with Beaver County Airport and to grow at approximately the same rate.

This corresponds to an annual average increase of 6 percent of the base (initial) year's figure (26,500), resulting in 102,800 operations in 2024.

3B.4.4 Fino Airport

Information on Fino Airport was provided by the airport owner/manager (Gowley 1977). It is a private landing strip located approximately 1.5 miles southwest of the BVPS-2. Its single runway (12-30), composed of grass, is 1,750 feet long and 100 feet wide. It can accommodate any single engine general aviation aircraft. There is one airplane based at the airport, a Cessna Cardinal RG (2,800 pounds maximum weight), which is flown solely by the owner. There is no

instrumentation at the airport and all operations are VFR. There is some runway lighting.

The owner estimates that when conditions are favorable, he may take off as many as 3 to 4 times per day and that, on an annual basis, he averages two movements (one take-off and one landing) per day or approximately 700 operations per year. No transient aircraft use the airport, principally because Fino has no facilities.

Power lines adjacent to runway 30 do not present a hazard, according to the owner, because of the steep climb slope associated with take-offs. Runway 12 is not used for landing. Flight patterns connected with landing or take-off are somewhat dependent on wind direction but can generally be classified as standard left-oriented. The owner states that he deliberately steers clear of the BVPS-2, his closest approach being 3/4 to 1 mile away laterally, 2,500 feet vertically.

In the foreseeable future, the owner has no plans to expand either activities or facilities. The operation at Fino may eventually be phased out because it is a small, private airport.

In the probability calculation, the angle between the runway centerline and the line joining the airport and the power station is incorporated. A conservative choice was runway 12 which led to an angle of 75 degrees rather than the 105 degrees for runway 30.

3B.4.5 Low and High Altitude Routes

The federal (V) airways and high altitude jet (J) air corridors in the vicinity of Shippingport are identified in Section 3B.4. The V airways (V12 and V37) experience a maximum of 100 total overflights per day, distributed about equally between them. Only about 2 percent of the flights are air carriers, the rest being general aviation (air taxi, cargo charter, business, etc) (Aber 1977). The Cleveland Air Route Traffic Control Center (Norris 1977) provided statistics on the J airways, specifically the Peak Day Traffic Count for 1976. It is assumed that the national forecast for air carrier operational increases can be directly applied to all air corridor operations. The data are summarized in Table 3B-8.

3B.5 CRITICAL STRUCTURES

The following structures of BVPS-2 comprise the critical targets for any incoming aircraft:

1. Control room extension,

2. Electric cable tunnel,

3. Auxiliary building,

4. Fuel and decontamination buildings,

5. Main steam and cable vault buildings,

6. Service building,

7. Reactor containment,

8. Safeguards area, and

9. Diesel generator building.

The intake structure is not considered a critical target because BVPS-2 has an alternate cooling system to provide sufficient cooling and heat dissipation in the event of the loss of the intake structure (Section 3.8.4).

The plant configuration is shown on Figure 3B.5-1. The total plan area of these structures is computed to be 76,770 square feet (0.0028 square mile).

The critical structures are exposed the most to aircraft approaching from the southeast, when viewed in terms of potential shadow and skidding areas. Some noncritical buildings provide substantial shielding to the south (for example, the turbine, waste handling, and condensate polishing buildings). There is one cooling tower to the northeast and one to the south-southeast. To the north, in addition to part of the cooling tower, shielding is afforded by an office building. The maximum skidding distance to the critical structures is from the southeast, even though Route 168 to the east, because of its elevation, would limit the maximum possible sliding distance of any aircraft. A comparable skidding area is seen for aircraft coming in from the northwest, but the accompanying shadow area would be smaller for this case.

3B.5.1 Skidding Distance

A representative skidding distance is defined or computed for each of the following aircraft operations:

1. General aviation,

2. Air carrier, airport-related,

3. Air carrier, in-flight,

4. Military, fighter aircraft, and

5. Military, transport, or tanker aircraft

The aircraft accident tapes of the NTSB list skidding distance information for 3,508 fatal general aviation accidents during the period 1964 through 1975. Table 3B-9 presents the distribution of skidding distances. A plot of the cumulative fraction of accidents with stopping distances greater than or equal to a specific value,

S_D, is shown on Figure 3B.5-2. An exponential curve is fit to the data points in piecewise fashion. This function, $F(S_D)$, is

$$F(S_D) = e^{-(S_D/72)}, S_D \leq 75'$$

$$0.56e^{-(S_D/180)}, 75' < S_D \leq 375'$$

$$0.25e^{-(S_D/239)}, 375' < S_D \leq 925'$$

$$0.45e^{-(S_D/624)}, 925' < S_D \leq 3{,}000'$$

Integrating $F(S_D)$ between 0 and 3,000 yields a representative skidding distance of 126 feet.

The NTSB data tapes and the accident files in Washington (Miller 1978) together yielded skidding distances for 54 fatal air carrier accidents between 1964 and 1975. Of these, 12 were in-flight accidents and 42 airport-related. The skidding distances and their averages for each case are presented in Table 3B-10.

Information on typical skidding distances for military aircraft is divided into fighter and tanker/transport categories (Strategic Air Command undated).

	Skidding Distance	
Type of Aircraft	Landing	Takeoff
Fighter	1,017	1,437
Tanker/Transport	1,143	712

Averaging the figures for landings and takeoffs, the representative skidding distances for fighter and tanker/transport aircraft are 1,227 and 928 feet, respectively.

3B.5.2 Impact Angle

The NTSB data tapes contained impact angles for 1,121 fatal general aviation accidents. The distribution of angles is given as follows:

Range of Angles (degrees)	Number of Occurrences
0-10	103
11-20	51
21-30	134
31-40	46
41-50	209
51-60	185
61-70	51
71-80	96
81-90	246

A plot of this distribution is shown on Figure 3B.5-3. One vertical scale is the number of occurrences in a given 10-degree interval, the other is a probability density function (fractional number of occurrences divided by the interval width). A weighted impact angle is computed by applying a weighting factor to the probability density distribution. This factor is the inverse of the tangent of the impact angle, 1/tan ϕ, which measures the projection of the vertical aspect of a structure onto the horizontal surface. The integral of the product of the weighting factor and the probability density function yields the ratio of the length of the horizontal projection to the height of the structure. The integration is performed piecewise with the numerical result of 1.91. The corresponding impact angle is

$$\phi = \tan^{-1}\left(\frac{1}{1.91}\right) = 28^{\circ}$$

A representative impact angle for air carriers is computed based on 40 accident cases (National Transportation Safety Board 1977 and Miller 1978), 12 in-flight accidents, and 28 airport-related. The average values of the impact angles listed in Table 3B-11 are 58 degrees and 35 degrees, respectively, for the two types of accidents. Data on impact angles for military aircraft are not as available. In the absence of any definitive information, it is conservatively assumed that airport-related military aircraft accidents have the same characteristic impact angle as general aviation accidents.

3B.5.3 Shadow and Skidding Areas

The shadow areas associated with the impact angles (Table 3B-11) were calculated graphically. The direction of approach of the aircraft was chosen to be southeast, the direction of maximum potential skidding distance. The length of the side exposed to the aircraft was 205 feet. Added to this figure, there was a contribution due to the length of the aircraft wing span (one-half the wing span added to each end of the BVPS-2's exposure).

Conservative estimates are made for the wing span for each aircraft category. A value of 50 feet was chosen for general aviation (the majority of general aviation aircraft have wing spans less than this) and a value of 150 feet for air carriers (the wing spans of the B-747 and B-727 are 195 feet and 108 feet, respectively). Of the three military aircraft (the A-7, KC-135, and C-123), the KC-135 has the largest wing span (142 feet). This number is adopted for the KC-135 and C-123 and one-half this value for the A-7.

The input data and the results of the target area calculations are presented in Table 3B-12.

3B.6 ACCIDENT RATES

National general aviation accident statistics for the period 1964 through 1975 indicate a total of 2,750 fatal accidents with airport data (airport-related). Of these, 84 accidents occurred beyond 5 miles of an airport (National Transportation Safety Board 1977). The total number of operations for this time period is estimated from several sources (FAA 1976a; FAA 1976b; and Mercer 1978). Table 3B-13 summarizes the data. The accident rates corresponding to these two cases are:

2.15×10^{-6} crashes per operation,
all accidents with airport data

6.56×10^{-8} crashes per operation,
accidents beyond 5 miles with airport data

For air carriers, two accident rates were computed:

1. Airport-related accidents beyond 5 miles of an airport,

2. Accidents in-flight.

Table 3B-14 indicates that there were, between 1964 and 1975, 50 fatal accidents with airport data, of which 14 were beyond 5 miles. Fifty-six in-flight accidents were identified for this time period. Air carrier activities (number of operations and total miles flown) are presented in Table 3B-15 (FAA 1976b).

The following crash rates were calculated:

1.22×10^{-7} crashes per operation,
accidents beyond 5 miles with airport data

2.30×10^{-9} crashes per mile flown, for
in-flight accidents.

For USAF aircraft, the crash rates were supplied in terms of accidents per 100,000 hours flown for the types of aircraft stationed at Greater Pittsburgh International Airport (Table 3B-16) (Crewse 1977). To convert these figures to accidents per operation, an average flight time of 4 hours was chosen (the higher this number, the higher the number of crashes per operation). Choosing those accidents that were airport-related, Table 3B-17 on crash rates was compiled.

3B.7 SPATIAL DISTRIBUTION FUNCTION

Table 3B-17 (National Transportation Safety Board 1977) presents the distribution of fatal air carrier accidents as a function of airport proximity for 1964 through 1975. Of the 14 accidents beyond 5 miles with airport data, specific information on nine of these was obtained (Miller 1978). The data for computing a probability density function, f(r), where r is the distance from the runway to the crash location, are

r(mi) = 9.7, 25.0, 9.7, 8.0, 8.2,
11.3, 9.0, 18.5, 8.9.

The cumulative frequency of events with radius, r, greater than or equal to specified values is plotted on Figure 3B.7-1. Fitting an exponential function to this curve yields

$$e^{-(r-5)/6.74}$$

f(r) is the derivative of one minus this function or

$$f(r) = 0.148^{-(r-5)/6.74}$$

The density function, f(r), is assumed uniform, where (θ) is the angle from the airport runway centerline to the crash location.

The joint probability density function is the product f(r) f(fθ). Integrating this expression over a differential area and then dividing by this same area (θ in radians) generates a probability density (per square mile) which is designated D (r,θ), the distribution function

$$D(r,\theta) = \frac{0.024e}{r}^{-(r-5)/6.74} \qquad (3B\text{-}2)$$

The function D(r,θ) was derived for general aviation accidents with airport data both within and beyond 5 miles of the airport. Table 3B-19 (National Transportation Safety Board 1977) gives the distribution of fatal general aviation accidents as a function of airport proximity (1964 through 1975).

These data are plotted on Figure 3B.7-2. An exponential fit to these points yields the function

$$e^{-r/1.57}$$

so that the probability density function is:

$$f(r) = 0.637e^{-r/1.57} \tag{3B-3}$$

In determining $f(\theta)$, 69 accidents were reviewed (Miller 1978 and Scribner and Chang 1974). The following data were analyzed:

Angle from Runway Centerline, θ	Cumulative Fraction of Accidents with Angular Deviation $\geq \theta$
0°	69/69 = 1.00
10°	50/69 = 0.72
20°	37/69 = 0.54
30°	30/69 = 0.43
40°	29/69 = 0.42
50°	25/69 = 0.36
60°	22/69 = 0.32
70°	22/69 = 0.32
80°	19/69 = 0.28
90°	17/69 = 0.25

The values are plotted on Figure 3B.7-3. The best fit exponential is $0.02e^{-\theta/49.6}$ so that the function $f(\theta)$ is

$$0.020^{-\theta/49.6}$$

Because θ has a maximum value of 180°, a constant factor is multiplied to $f(\theta)$ so that its integration between 0° and 180° equals 0.5. The function $D(r,\theta)$ for accidents within 5 miles of the airport becomes

$$D(r,\theta) = \frac{0.37e^{-r/1.57}}{r} e^{-\theta/49.6} \tag{3B-4}$$

For accidents beyond 5 miles, a uniform distribution in θ is assumed. $f(r)$ was derived based on 51 accidents (Miller 1978 and Scribner and Chang 1974) with radii distributed as follows:

r	Cumulative Fraction $\geq$ r
5	51/51 = 1.00
6	47/51 = 0.92
7	38/51 = 0.75

8	28/51 = 0.55
9	22/51 = 0.43
10	20/51 = 0.39
11	16/51 = 0.31
12	14/51 = 0.27
13	11/51 = 0.22
14	7/51 = 0.14
15	6/51 = 0.12
20	3/51 = 0.06
21	2/51 = 0.04

These data are plotted on Figure 3B.7-4.

The best fit exponential has the form

$$e^{-(r-5)/4.56}$$

and the function D(r,θ) is given by

$$\frac{0.034e^{-(r-5)/4.56}}{r}$$

The data base for (r,θ) values for USAF accidents were obtained from two sources (Strategic Air Command undated and Directorate of Aerospace Safety undated). The distribution of accidents within 5 and 10 mile zones of the airport are shown on Figures 3B.7-5 and 3B.7-6. The cumulative distribution functions for r and θ are plotted on Figures 3B.7-7 and 3B.7-8 and are described, respectively, by the exponentials

$$e^{-r/1.91} \text{ and } e^{-\theta/29.6}$$

The distribution function D(r,θ) becomes

$$D(r,\theta) = \frac{0.51e^{-r/1.91}}{r} e^{-\theta/29.6} \qquad (3B\text{-}5)$$

This formulation is employed (conservatively) even though the Greater Pittsburgh Airport is more than 10 miles from BVPS-2.

For the V and J airways, data on inflight crash locations with respect to the centerline of the flight path were secured from two sources (Miller 1978 Lowe and Robbins 1976) for the period 1964 through 1974. The data are displayed below with Figure 3B.7-9 being a plot of this data.

Perpendicular Distance from Flight Path, R miles	Number of Accidents
0	6
$0<R\leq 1/2$	3
$1/2\leq R1$	3
$1<R\leq 2$	2
$2<R\leq 3$	4
$3<R\leq 4$	1
$4<R\leq 5$	1
$5<R\leq 6$	0
$6<R\leq 7$	0
$7<R\leq 8$	1
$8<R\leq 9$	0
Total	21

An exponential fit yields the cumulative distribution function f(R) (USNRC 1975 and Miller 1978). The resulting probability density function was halved since only one side of an airway is considered;

$$f(R) = 0.23e^{-R/2.17} \tag{3B-6}$$

This is the distribution function, D(R), (density function per mile) for the in-flight probability analysis.

3B.8 DISCUSSION

The results of the probability analysis for the present and for the year 2024 have been presented in Section 3B.3. The analysis is regarded as conservative in several respects.

1. The skidding distance of an aircraft, as reported by accident investigators, is often the total length over which the wreckage is spread. A large fraction of the aircraft's energy would probably be dissipated over an appreciably shorter distance.

2. The direction of approach of an aircraft to BVPS-2 was chosen to maximize the target area (no weighting of the various possible directions was performed).

3. For the direction of aircraft approach chosen, the maximum possible skidding distance was less than 1,000 feet (as imposed by local topography). For the A-7 aircraft, a greater value was used. In addition, the southeast corner of BVPS-2 is shielded by a cooling tower from this approach so that the extra contribution to the skidding area due to half a wing span is conservative. Furthermore, the representative wing spans are considered somewhat high.

4. In general, the statistics were chosen to reflect conservatism. (For example, an average crash rate for air

carrier accidents over several years was chosen even though the trend in recent years has been toward a decreasing rate).

5. The pilot can exert some control over an aircraft in distress and, to the extent possible, would attempt to avoid hitting a solid structure such as a power plant.

The previous qualitative arguments offer substantial justification for reduction of the probabilities cited herein below 10^{-7} per year.

3B.9 ACCIDENTS AT NEARBY AIRPORTS FROM 1964 TO 1977

Information on aircraft accidents in the vicinity of the four airports described in Section 3B.4 for the period 1964 to 1977 was obtained from the NTSB aircraft accident data tapes (information for 1978 was not available from this source at the time of the analysis). During this period, 58 airport-related accidents were associated with these airports. Table 3B-19 presents the pertinent information including aircraft data and accident summaries. Typically, the aircraft involved was general aviation, less than 12,500 pounds in weight, sustained "substantial" damage in an accident involving no injuries, and occurred on or near the airport during the landing phase. Greater Pittsburgh, Beaver County, and Herron were the airports of record in 26, 20, and 12 cited accidents, respectively, between 1964 and 1977.

3B.10 REFERENCES FOR APPENDIX 3B

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Tables for Section 3B

TABLE 3B-1

PROBABILITY ESTIMATES OF AIRCRAFT COLLIDING WITH CRITICAL BVPS-2 STRUCTURES

Category	Year	Flight Operations	Probability (collisions/year)
General Aviation			
Greater Pittsburgh International Airport	1976-1977	116,480	1.72×10^{-8}
	2024	452,000	6.60×10^{-8}
Beaver County Airport	1976-1977	34,000	1.32×10^{-8}
	2024	131,920	5.12×10^{-8}
Herron Airport	1976-1977	26,500	1.89×10^{-8}
	2024	102,800	7.33×10^{-8}
Fino Airport	1976-1977	700	1.64x10-7
	2024	700	1.64×10^{-7}
Military (Greater Pittsburgh)			
A-7	1976-1977	5,600	2.05×10^{-8}
	2024	5,600	2.05×10^{-8}
KC-135	1976-1977	1,560	4.19×10^{-9}
	2024	1,560	4.19×10^{-9}
C-123	1976-1977	2,400	5.82×10^{-8}
	2024	2,400	5.82×10^{-8}
Air Carrier			
Greater Pittsburgh International Airport	1976-1977	182,610	1.80×10^{-7}
	2024	478,800	4.72×10^{-7}
Air Routes			
V12	1976-1977	18,250	1.24×10^{-12}
	2024	48,180	3.27×10^{-12}
V37	1976-1977	18,250	3.77×10^{-10}
	2024	48,180	9.95×10^{-10}

TABLE 3B-1 (Cont)

Category	Year	Flight Operations	Probability (collisions/year)
J518	1976-1977	25,185	3.96×10^{-8}
	2024	66,430	1.04×10^{-7}
J152	1976-1977	36,500	4.13×10^{-11}
	2024	95,995	1.09×10^{-10}
J145	1976-1977	8,760	2.38×10^{-10}
	2024	22,995	6.24×10^{-10}
J53	1976-1977	11,680	4.81×10^{-10}
	2024	30,660	1.26×10^{-9}
J64	1976-1977	16,790	1.58×10^{-10}
	2024	44,165	4.16×10^{-10}

TABLE 3B-2

BEAVER COUNTY AIRPORT
FIXED BASE OPERATOR'S FACILITIES AND SERVICES

Item	Beaver Aviation Service	Skyline Motors Aviation	Moore Aviation Service	Schreck's Flying Service
Year Established	1967	1970	1966	Not Available
Services Offered	1. Training 2. Aircraft rental 3. Charter -Passenger -Cargo 4. Aircraft sales 5. Maintenance	1. Training 2. Aircraft rental 3. Charter -Passenger -Cargo 4. Aircraft sales	1. Training 2. Aircraft rental 3. Aircraft sales	1. Training 2. Aircraft rental
Number of Aircraft Utilized	13	10	3	2

TABLE 3B-3

BEAVER COUNTY AIRPORT
ANNUAL AND HOURLY AIRFIELD CAPACITIES*

Runway	Capacity	Number of Movements
10-28	VFR Hourly	92
10-28	IFR Hourly**	36
10-28	Annual	151,130
10-28	Weighted Hourly	87

NOTES:

*Michael Baker, Jr., Inc. 1974
**Based upon VOR approach with radar monitoring.

TABLE 3B-4

BEAVER COUNTY AIRPORT,
TRENDS IN AIRCRAFT ACTIVITY

Activity	1974	1979	1984	1994
Based Aircraft	86	101	122	181
Single-Engine	71	83	101	149
Multi-Engine	15	18	21	32
Aircraft Operations	24,861	34,000	43,200	69,800
Local	9,444	13,600	17,300	27,900
Itinerant	15,417	20,400	25,900	41,900
Instrument Approaches	804	1,072	1,500	2,400

TABLE 3B-5

GREATER PITTSBURGH INTERNATIONAL AIRPORT,
TYPES OF AIRCRAFT AND CARRIER UTILIZATION

Category	Make	Model	Number	Gross Weight (lb)
Military	Boeing	KC-135	8	362,100
(Based at Airport)	Fairchild-Republic	C-123	16	60,000
	Ling-Temco-Vought	A-7	20	42,500
General Aviation	Agusta Helicopters	A-109	2	5,402
(Based at Airport)	Grumman	G-1159	4	62,500
	Hawker-Siddeley	DH-25	2	22,500
	Lockheed	L-329	1	40,921
	Rockwell	AC-68	2	10,300
	Rockwell	NA-265	3	23,000
Air Carrier				
USAir	Nords	--		
(Home Base)		BAC-111		
	Douglas	DC-9		
	Boeing	B-727		
American Airlines	Boeing	B-707		
(Transient)	Boeing	B-727		
	Boeing	B-747		
Eastern Airlines	Boeing	B-727		
(Transient)	Douglas	DC-9		
	Lockheed	L-1011		
Nordair (Transient)	Boeing	B-737		
Northwest Airlines	Boeing	B-707		
(Transient)	Boeing	B-727		
Trans-World Air-	Boeing	B-707		
lines(Transient)	Boeing	B-727		
	Douglas	DC-9		
	Lockheed	L-1011		
United Airlines	Boeing	B-727		
(Transient)	Boeing	B-737		
	Boeing	B-747		
	Douglas	DC-8		
	Douglas	DC-10		

TABLE 3B-5 (Cont)

Category	Make	Model	Number	Gross Weight (lb)
Air Taxi	Beech	BE-18	1	9,900
	Douglas	DC-3	3	26,900
Other				
4 Commuter Airlines (Transient)	Beechcraft	BE-99		

TABLE 3B-6

GREATER PITTSBURGH INTERNATIONAL AIRPORT, DISTRIBUTION OF TRAFFIC BY RUNWAY *, **, ***

Runway	Percent of Annual Arrivals (IFR & VFR) for Each Runway	Percent of Annual Instrument Approaches for Each Runway
05	Closed to Landings and Takeoffs	
10L	20	40
10R	5	1
14	1	
23	1	
28L	55	58
32	14	****
28R	4	1

NOTES:

*From table provided by James J. Hanten, Facility Chief, Greater Pittsburgh International Airport, March 7, 1977.
**Automatic Terminal Information Service (ATIS) is provided.
***Both arrival service and departure service are provided.
****ILS System was commissioned January 20, 1977. The percentage of instrument use is conjecture at this time, but it is expected to be 18 to 10 percent after 1 year.

TABLE 3B-7

GENERAL AVIATION AIRCRAFT BASED AT HERRON AIRPORT

Make and Model	Number	Make and Model	Number
Beech		Luscomb	1
Beechaft Baron	1	Navion	
Bonanza	5	Rangemaster	3
Cessna		Piper	
Skymaster	1	Apache	2
140	3	Cherokee	2
150	5	Comanche 400	3
170	4	Twin Comanche	2
172	4	Tripacer	4
175	2	J-3	1
182	3	J-4	1
190	4		
310	1		
1210	1		

TABLE 3B-8

AIR CARRIER OPERATIONAL INCREASE FORECAST FOR
V AND J AIRWAYS IN THE VICINITY OF SHIPPINGPORT

Airway	Perpendicular Distance From Power Station (mi)	Peak Day Count (1976)	Projection for Operations/yr
V12	24.3	50 (18,250/yr)	48,180
V37	11.9	50 (18,250/yr)	48,180
J53	10.4	32 (11,680/yr)	30,660
J64	13.6	46 (16,790/yr)	44,165
J145	11.3	24 (8,760/yr)	22,995
J152	18.2	100 (36,500/yr)	95,995
J518	2.5	69 (25,185/yr)	66,430

TABLE 3B-9

DISTRIBUTION OF SKIDDING DISTANCES
IN GENERAL AVIATION ACCIDENTS
1964 THROUGH 1975

Range of Distance (ft) S_D	Number of Occurrences	Range of Distance (ft) S_D	Number of Occurrences
0-50	2,129	1,201-1,300	4
51-100	391	1,301-1,400	2
101-150	262	1,401-1,500	4
151-200	175	1,501-1,600	1
201-250	136	1,601-1,700	0
251-300	109	1,701-1,800	0
301-350	58	1,801-1,900	1
351-400	40	1,901-2,000	2
401-450	40	2,001-2,100	2
451-500	33	2,101-2,200	1
501-550	10	2,201-2,300	1
551-600	32	2,301-2,400	0
601-650	9	2,401-2,500	1
651-700	14	2,501-2,600	0
701-750	8	2,601-2,700	0
751-800	8	2,701-2,800	1
801-850	2	2,801-2,900	0
851-900	10	2,901-3,000	10
901-950	6	-	1
951-1000	5	-	0
1,001-1,100	3	-	
1,101-1,200	7	5,801-5,900	1
		Total	3,508

TABLE 3B-10

SKIDDING DISTANCES IN AIR CARRIER ACCIDENTS
1964 THROUGH 1975

Skidding Distance
(ft)

In-Flight Accidents	Airport-Related Accidents			
0	610	360	3,800	995
1,050	375	1,050	800	775
300	600	440	270	1,525
55	1,200	1,840	350	248
800	340	1,020	107	400
0	0	1,400	250	267
1,000	750	580	1,600	
950	1,200	200	3,200	
150	1,015	0	262	
175	700	700	1,200	
180	1,080	279	790	
1,100	3,600	250	850	
Average value = 480	Average value = 888			

TABLE 3B-11

REPRESENTATIVE IMPACT ANGLES
IN AIR CARRIER ACCIDENTS

Impact Angles

In-Flight Accidents (Degrees)	Airport-Related Accidents (Degrees)		
90	22	39	45
90	60	90	13
35	70	90	
8	90	5	
60	4	70	
60	9	90	
90	90	70	
11	14	5	
30	5	6	
40	8	1	
90	6	4	
90	45	6	
	12	5	

TABLE 3B-12

TARGET AREA INFORMATION

Aircraft Type or Operation	Plan Area (ft^2)	Shadow Area (ft^2)	Frontal Exposure of Plant (ft)	Skidding Distance (ft)	Skidding Area (ft^2)	Total Target Area (ft^2)	Total Target Area (mi^2)
General aviation	76,770	38,000	255	126	32,130	146,900	0.0053
Air carrier, airport-related	76,770	29,650	355	888	315,240	421,660	0.0151
Air carrier, in-flight	76,770	14,685	355	480	170,400	261,855	0.0094
Military - KC-135, C-123	76,770	38,000	347	928	322,016	436,786	0.0157
Military - A-7	76,770	38,000	276	1,227	338,655	453,425	0.0163

TABLE 3B-13

GENERAL AVIATION OPERATIONS
1964 through 1975

Calendar Year	Operations at Towered Airports	Total Operations at Towered and Nontowered Airports
1964	23,020,000	63,945,000*
1965	26,573,000	73,814,000*
1966	33,445,000	92,903,000*
1967	37,223,000	103,397,000*
1968	41,567,000	115,464,000*
1969	41,957,000	116,547,000*
1970	41,384,000	114,956,000*
1971	40,401,000	112,225,000*
1972	38,172,000	115,400,000**
1973	41,363,000	114,897,000*
1974	43,124,000	125,700,000**
1975	45,297,000	130,700,000**
	Total 453,526,000	Total 1,279,948,000

NOTES:

* In the period 1972 through 1977, nontowered general aviation operations constituted conservatively 64 percent of all operations (FAA 1976a; Mercer 1978).

** Federal Aviation Administration 1976a.

TABLE 3B-14

AIR CARRIER ACCIDENTS VERSUS AIRPORT PROXIMITY
1964 THROUGH 1975

	Fatal Accidents		
Proximity Class	Total	With Airport Data	Without Airport Data
In traffic pattern	9	0	9
Within 1/4 mile	1	0	1
Within 1/2 mile	2	1	3
Within 3/4 mile	2	0	2
Within 1 mile	4	0	4
Within 2 miles	8	0	8
Within 3 miles	3	0	3
Within 4 miles	4	0	4
Within 5 miles	3	3	0
Total within 5 miles	37	36	1
Total beyond 5 miles	57	14	43
Total	94	50	44

TABLE 3B-15

AIR CARRIER ACTIVITIES
1964 through 1975

Year	Number of Operations	Aircraft Miles Flown (x 1,000)
1964	7,600,000*	1,278,000*
1965	8,100,000*	1,363,000*
1966	8,702,900	1,464,200
1967	9,845,200	1,813,000
1968	10,649,400	2,124,000
1969	10,707,700	2,359,700
1970	10,138,200	2,383,400
1971	9,946,300	2,343,600
1972	10,238,600	2,336,900
1973	10,204,600	2,401,900
1974	9,389,300	2,213,500
1975	9,350,400	2,202,600
	Total 114,672,000	Total 24,282,000

NOTE:

*Estimated

TABLE 3B-16

MILITARY AIRCRAFT (A-7, C-123, KC-135) ACCIDENT RATES
1968 THROUGH 1976

Type of Aircraft	Year	Phase of Operation	Number	Rate*
A-7	1970	Landing, lost control	1	
	1971	Takeoff, climb, initial	1	2.7
	1971	Landing, approach	1	2.7
	1972	In flight, normal	1	1.6
	1972	In flight, climb, prolonged	1	1.6
	1973	In flight, normal	5	5.7
	1975	In flight, normal	3	3.4
	1975	Landing, approach	3	3.4
	1976	In flight, normal	3	3.0
C-123	1968	Takeoff, climb, initial	1	.8
	1969	Takeoff, climb, initial	1	.8
	1970	In flight, normal	1	.9
	1970	Landing, approach	2	1.8
	1971	Landing, approach	1	1.4
	1972	Landing, approach	1	3.2
KC-135	1968	Takeoff, climb, initial	1	.2
	1968	In flight, normal	1	.2
	1969	In flight, normal	2	.5
	1971	In flight, normal	1	.3
	1971	Landing, approach	1	.3
	1974	Takeoff, climb, initial	1	.3
	1974	In flight, normal	1	.3
	1975	Takeoff, climb, initial	1	.4

NOTE:

*Per 100,000 flying hours.

TABLE 3B-17

AIRPORT-RELATED MILITARY AIRCRAFT CRASH RATES
1968 THROUGH 1976

	Accidents/Operation		
Year	A-7 Crash Rate	KC-135 Crash Rate	C-1234 Crash Rate
1968	0	$1.6x10^{-5}$	$0.4x10^{-5}$
1969	0	$1.6x10^{-5}$	0
1970	$5.4x10^{-5}$	$3.6x10^{-5}$	0
1971	$10.8x10^{-5}$	$2.8x10^{-5}$	$0.6x10^{-5}$
1972	0	$6.4x10^{-5}$	0
1973	0	0	0
1974	0	0	$0.6x10^{-5}$
1975	$6.8x10^{-5}$	0	$0.6x10^{-5}$
1976	0	0	0
	Average = $2.6x10^{-5}$	Average = $1.8x10^{-5}$	Average = $0.2x10^{-5}$

TABLE 3B-18

GENERAL AVIATION ACCIDENTS VERSUS AIRPORT PROXIMITY
1964 THROUGH 1975

	Fatal Accidents		
Proximity Class	Total	With Airport Data	Without Airport Data
In traffic pattern*	685	673	12
Within 1/4 mile	302	296	6
Within 1/2 mile	242	237	5
Within 3/4 mile	90	89	1
Within 1 mile	242	242	0
Within 2 miles	438	427	11
Within 3 miles	328	324	4
Within 4 miles	267	263	4
Within 5 miles	117	115	2
Total within 5 miles	2,711	2,666	45
Total beyond 5 miles	3,640	84	3,556
Total	6,351	2,750	3,601

NOTE:

* For purposes of analysis, those accidents in the traffic pattern are distributed among other categories (as their precise location is unknown). This is accomplished as follows (accidents with airport data): There are 8 other categories:

673/8 = (84 x 8) + 1
within 1/4 mi = 85 + 296 = 381
within 1/2 mi = 84 + 237 = 321
within 3/4 mi = 84 + 89 = 173
within 1 mi = 84 + 242 = 326
within 2 mi = 84 + 427 = 511
within 3 mi = 84 + 324 = 408
within 4 mi = 84 + 263 = 347
within 5 mi = 84 + 115 = 199

TABLE 3B-19

U.S. REGISTERED CIVIL AIRCRAFT ACCIDENTS
IN THE VICINITY OF THE BEAVER VALLEY POWER STATION
FROM 1964 THROUGH 1977

Date	Location	Make	Model	Weight Category (lb)	Airport	Airport Proximity	Aircraft Damage	Injury Index	Operational Phase
10/30/64	Pittsburgh, Pa.	Curtis	C-46F	>12,500	Greater Pittsburgh	On airport	Substantial	None	Forced Landing
8/12/64	Beaver Falls, Pa.	Brantly	B2	<12,500	Beaver County	On airport	Substantial	Minor	Forced Landing
5/03/64	Beaver Falls, Pa.	Mooney	M20C	<12,500	Beaver County	Within 1/4 mile	Substantial	None	Forced Landing
9/06/64	Pittsburgh, Pa.	Piper	PA24	<12,500	Greater Pittsburgh	On airport	Substantial	None	Landing
10/23/64	New Cumberland, WVa.	Champion	7FC	<12,500	Herron	On airport	Substantial	None	Landing
11/09/65	Coraopolis, Pa.	Douglas	DC3	>12,500	Greater Pittsburgh	Within 2 Miles	Destroyed	Serious	Forced Landing
6/15/65	Beaver Falls, Pa.	Cessna	120	<12,500	Beaver County	On airport	Substantial	None	Taxiing after landing - nose over
8/07/65	Beaver Falls, Pa.	Beechcraft	C-18S	<12,500	Beaver County	On airport	Substantial	None	Taxiing after landing- collided with ditches
2/10/66	Greater Pittsburgh Airport	Fairchild	F-27J	>12,500	Greater Pittsburgh	On airport	Substantial	None	Forced Landing
3/19/66	Coraopolis, Pa.	Cessna	172G	<12,500	Greater Pittsburgh	On airport	Substantial	None	Taxiing to takeoff - nose over
7/03/66	Pittsburgh, Pa.	Mooney	20E	<12,500	Greater Pittsburgh	On airport	Substantial	None	Landing - wheels up
7/11/66	New Cumberland, WVa.	Aeronca	7FC	<12,500	Herron	On airport	Substantial	None	Landing - gear collapsed

TABLE 3B-19 (Cont)

Date	Location	Make	Model	Weight Category (lb)	Airport	Airport Proximity	Aircraft Damage	Injury Index	Operational Phase
7/23/66	New Cumberland, WVa.	Cessna	337	<12,500	Herron	On airport	Substantial	None	Landing - collided with fence
12/30/66	Coraopolis, Pa.	Piper	PA24	<12,500	Greater Pittsburgh	On airport	Substantial	None	Landing - gear collapsed
5/15/67	McKees Rocks, Pa.	Mooney	M20C	<12,500	Greater Pittsburgh	Within 4 miles	Substantial	None	Forced Landing
11/21/67	New Cumberland, WVa.	Cessna	172F	<12,500	Herron	On airport	Substantial	None	Overshoot on landing - collided with dirt bank
12/23/67	Beaver Falls, Pa.	Piper	PA25	<12,500	Beaver County	Within 3 miles	Substantial	None	Landing, final approach - collided with trees
3/09/68	Pittsburgh, Pa.	Mitsubishi	MU-2B	<12,500	Greater Pittsburgh	On airport	Substantial	None	Landing - wheels up
7/13/68	Beaver Falls, Pa.	Piper	J3	<12,500	Beaver County	On airport	Destroyed	Serious	In flight acrobatics - stall
9/01/68	Beaver Falls, Pa.	Cessna	180	<12,500	Beaver County	On airport	Substantial	None	Landing - gear collapsed
12/28/68	Coraopolis, Pa.	Cessna	150	<12,500	Greater Pittsburgh	On airport	Substantial	None	Taxiing from landing - nose over
4/13/69	Coraopolis, Pa.	Beech	T34	<12,500	Greater Pittsburgh	On airport	Substantial	None	Landing - wheels up
3/06/69	Coraopolis, Pa.	Beech	D18S	<12,500	Greater Pittsburgh	On airport	Substantial	None	Landing - gear collapsed
8/20/69	Coraopolis, Pa.	Cessna	310N	<12,500	Greater Pittsburgh	In traffic pattern	Substantial	Serious	Landing, final approach - collided with wires
9/10/70	Coraopolis, Pa.	Piper	PA-28R	<12,500	Greater Pittsburgh	On airport	Substantial	None	Taxiing from landing - collided with dirt bank

TABLE 3B-19 (Cont)

Date	Location	Make	Model	Weight Category (lb)	Airport	Airport Proximity	Aircraft Damage	Injury Index	Operational Phase
10/24/70	New Cumberland, WVa.	Lake	LA4	<12,500	Herron	Within 3 miles	Destroyed	Fatal	In flight - collision with ground
11/12/70	Clinton, Pa.	Beech	G18S	<12,500	Greater Pittsburgh	Within 3 miles	Destroyed	Serious	Landing, final approach - collided with ground
7/08/70	Coraopolis, Pa.	Handly-Page	137	<12,500	Greater Pittsburgh	On airport	Substantial	None	Crashed on landing
8/20/71	Pittsburgh, Pa.	Convair	580	>12,500	Greater Pittsburgh	On airport	Substantial	None	Landing - gear collapsed
6/12/71	Beaver Falls, Pa.	Cessna	150J	<12,500	Beaver County	On airport	Substantial	Minor	Taxiing from landing - went over step bank, became inverted
5/01/71	Beaver Falls, Pa.	Cessna	172	<12,500	Beaver County	In traffic pattern	Substantial	None	Propeller blade failed after takeoff - emergency landing
6/04/71	Coraopolis, Pa.	Ham-Flugzen	HFB320	>12,500	Greater Pittsburgh	On airport	Substantial	None	Fire - Takeoff aborted
4/21/71	Coraopolis, Pa.	Piper	PA23	<12,500	Greater Pittsburgh	On airport	Substantial	None	Gear retracted on takeoff
4/28/72	New Cumberland, WVa.	Piper	PA28	<12,500	Herron	On airport	Substantial	None	Landing - collided with trees
4/29/72	New Cumberland, WVa.	Aeronca	7AC	<12,500	Herron	On airport	Substantial	None	Landing - collided with wind tee
6/12/73	Pittsburgh, Pa.	Convair	580	>12,500	Greater Pittsburgh	On airport	Substantial	Minor	Gear collapsed on takeoff
4/10/73	Coraopolis, Pa.	Piper	PA32	<12,500	Greater Pittsburgh	Within 1/2 mile	Destroyed	Fatal	Landing, final approach - collided with trees
3/21/73	Beaver Falls, Pa.	Cessna	150J	<12,500	Beaver County	On airport	Substantial	None	Landing - wing hit runway

TABLE 3B-19 (Cont)

Date	Location	Make	Model	Weight Category (lb)	Airport	Airport Proximity	Aircraft Damage	Injury Index	Operational Phase
6/13/73	Beaver Falls, Pa.	Beech	D95A	<12,500	Beaver County	On airport	Substantial	None	Landing - gear up
7/30/73	Beaver Falls, Pa.	Piper	Pa.16	<12,500	Beaver County	On airport	Substantial	None	Takeoff - gear collapsed
12/07/73	Coraopolis, Pa.	Beech	E18S	<12,500	Greater Pittsburgh	Within 1/2 mile	Destroyed	Fatal	Takeoff - stall
7/29/74	Beaver Falls, Pa.	Cessna	182	<12,500	Beaver County	Within 1 mile	Substantial	Minor	Landing - collided with trees
8/26/74	New Cumberland, WVa.	Cessna	172	<12,500	Herron	On airport	Substantial	None	Landing, overshoot - collided with fence
7/24/74	Pittsburgh, Pa.	Piper	PA28R	<12,500	Greater Pittsburgh	On airport	Substantial	None	Taxiing from landing - gear collapsed
4/23/75	Beaver Falls, Pa.	Cessna	172K	<12,500	Beaver County	On airport	Substantial	None	Taxiing from landing - nose down
4/01/75	Beaver Falls, Pa.	Hiller	UH-12B	<12,500	Beaver County	Within 2 miles	Substantial	None	Engine failure in flight - hard landing
9/18/75	New Cumberland, WVa.	Cessna	150G	<12,500	Herron	On airport	Substantial	Minor	Landing - stall
9/02/75	New Cumberland, WVa.	Cessna	172H	<12,500	Herron	On airport	Substantial	None	Landing, overshoot - collided with ditches
9/03/75	Beaver Falls, Pa.	Hiller	UH-12B	<12,500	Beaver County	On airport	Substantial	None	Takeoff aborted - collision with ground
11/30/75	Pittsburgh, Pa.	Piper	PA-28R	<12,500	Greater Pittsburgh	On airport	Substantial	None	Landing - gear collapsed
1/22/76	Beaver Falls, Pa.	Piper	PA28	<12,500	Beaver County	On airport	Substantial	None	Landing - collided with snowbank
9/19/76	Beaver Falls, Pa.	Cessna	150L	<12,500	Beaver County	On airport	Substantial	None	Landing - gear collapsed

TABLE 3B-19 (Cont)

Date	Location	Make	Model	Weight Category (lb)	Airport	Airport Proximity	Aircraft Damage	Injury Index	Operational Phase
9/16/76	Beaver Falls, Pa.	Beech	G18S	<12,500	Beaver County	On airport	Substantial	None	Landing, overshoot - collided with ditches
5/26/76	Sewickley, Pa.	Bellanca	17-30A	<12,500	Greater Pittsburgh	Within 4 miles	Destroyed	Fatal	In flight - stall
7/13/76	New Cumberland, WVa.	Grumman American	AA-1B	<12,500	Herron	On airport	Substantial	None	Landing - gear collapsed
5/7/77	Beaver Falls, Pa.	Alon	A-2	<12,500	Beaver County	On airport	Substantial	None	Landing - gear collapsed
8/17/77	Pittsburgh, Pa.	Beech	G18S	<12,500	Greater Pittsburgh	On airport	Substantial	Serious	Takeoff, initial climb - stall, forced landing
5/7/77	New Cumberland, WVa.	Cessna	172	<12,500	Herron	Within 1/4 mile	Substantial	None	Landing - collided with trees

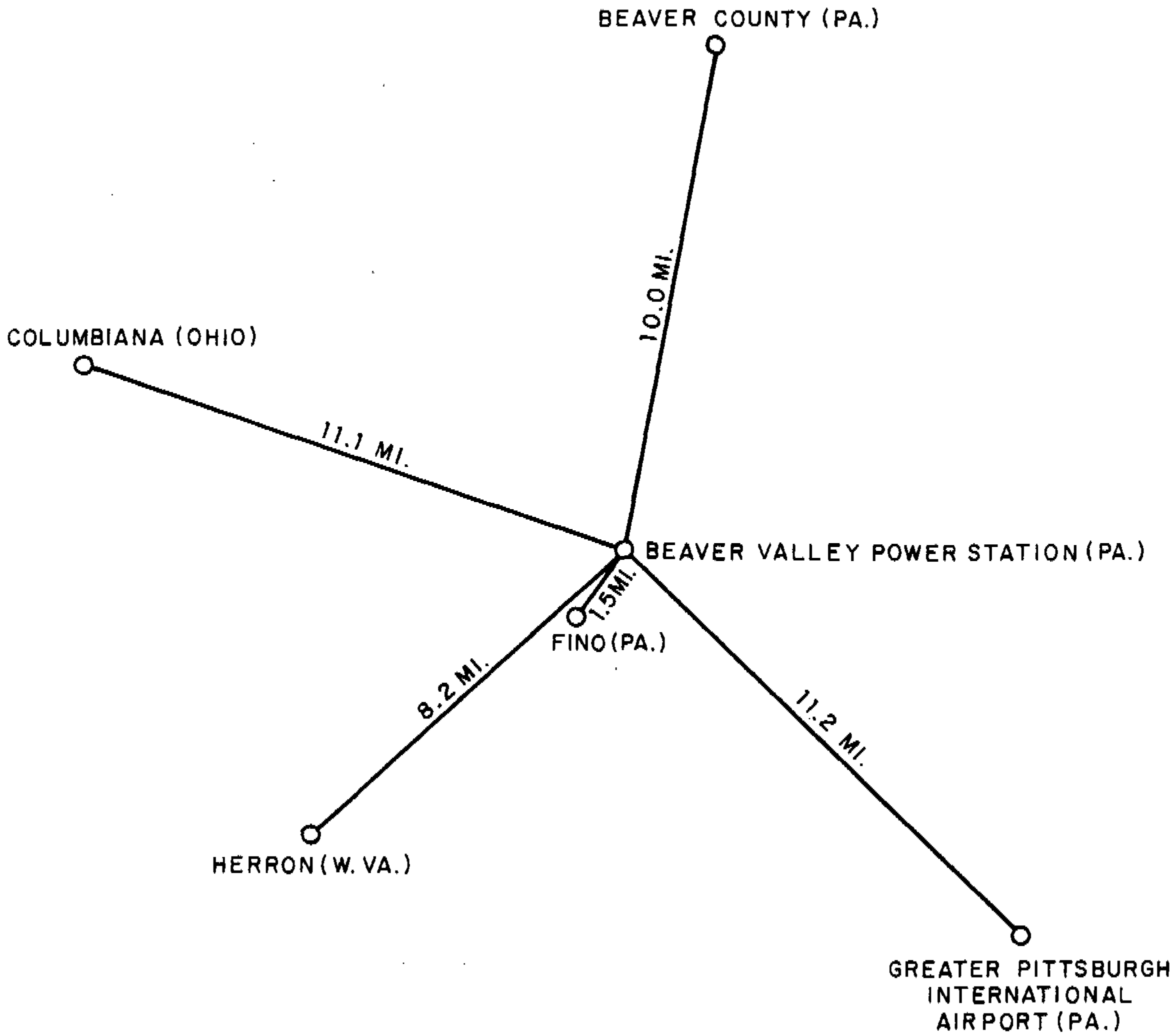


FIGURE 3B.4-1

AIRPORTS IN THE VICINITY OF THE BEAVER VALLEY POWER STATION

BEAVER VALLEY POWER STATION – UNIT 2
FINAL SAFETY ANALYSIS REPORT

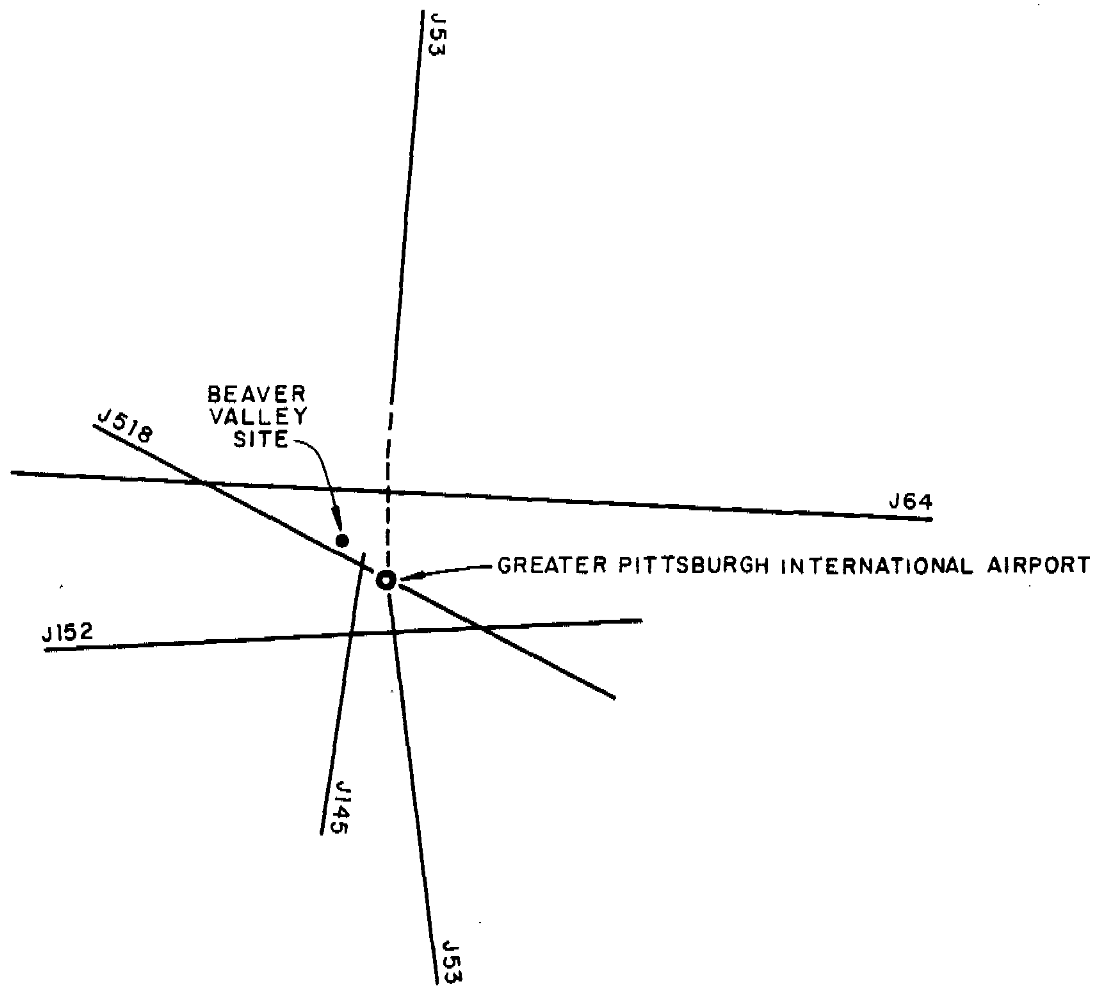


FIGURE 3B.4-2
JET AIR ROUTES IN THE
BEAVER VALLEY AREA
BEAVER VALLEY POWER STATION – UNIT 2
FINAL SAFETY ANALYSIS REPORT

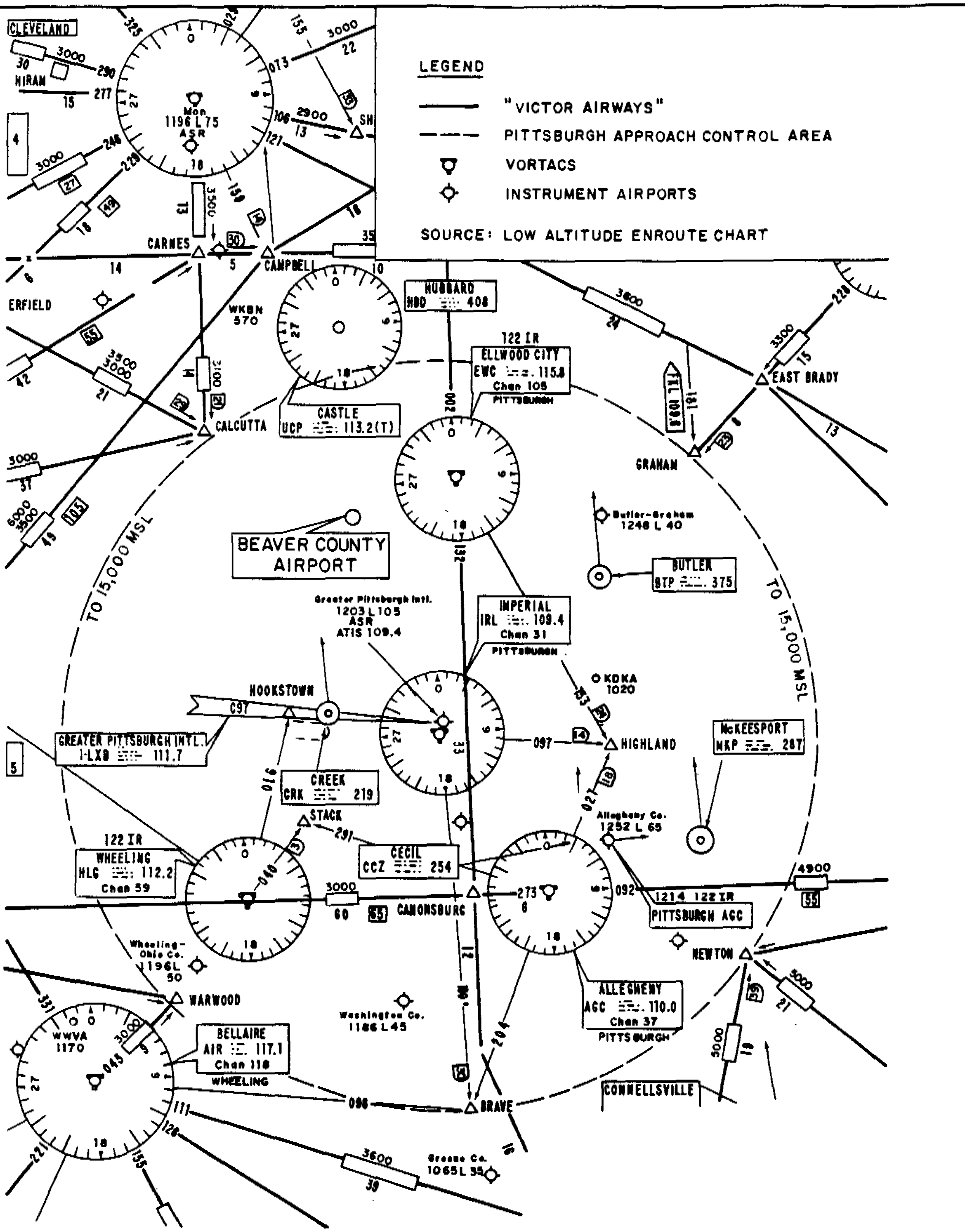
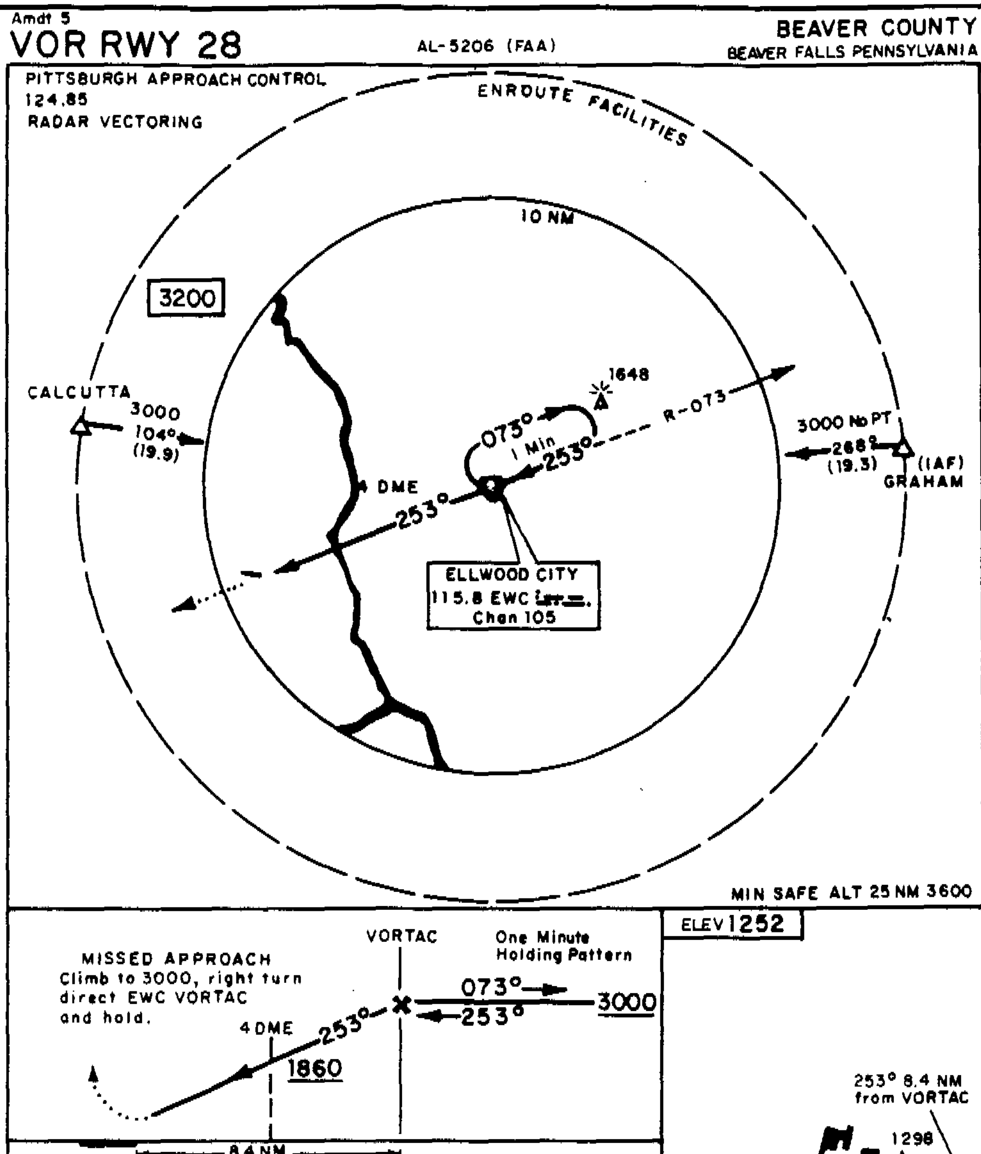


FIGURE 3B.4-3
BEAVER COUNTY AIRPORT
IFR ROUTE STRUCTURE
PITTSBURGH AREA

BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT



CATEGORY	A	B	C	D
S-28	1860-1 608 (700-1)			1860-1¼ 608 (700-1¼)
CIRCLING	1920-1 668 (700-1)		1920-1½ 668 (700-1½)	1920-2 668 (700-2)
	DME MINIMA			
S-28	1740-1 (500-1)			
CIRCLING	1800-1 548 (600-1)		1800-1½ 548 (600-1½)	1820-2 568 (600-2)

Use greater Pittsburg, Pennsylvania altimeter setting

ΔNA

FA to MAP 8.4 NM					
Knots	60	90	120	150	180
Min Sec	8.24	5.36	4.12	3.22	2.48

VOR RWY 28
6 JAN. 1977

40°46'N - 80°24'W

PUBLISHED BY NOS, NOAA TO IACC SPECIFICATIONS

BEAVER FALLS PENNSYLVANIA
BEAVER COUNTY

FIGURE 3B.4-4
BEAVER COUNTY AIRPORT
PUBLISHED INSTRUMENT APPROACH
PROCEDURE CHART

BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

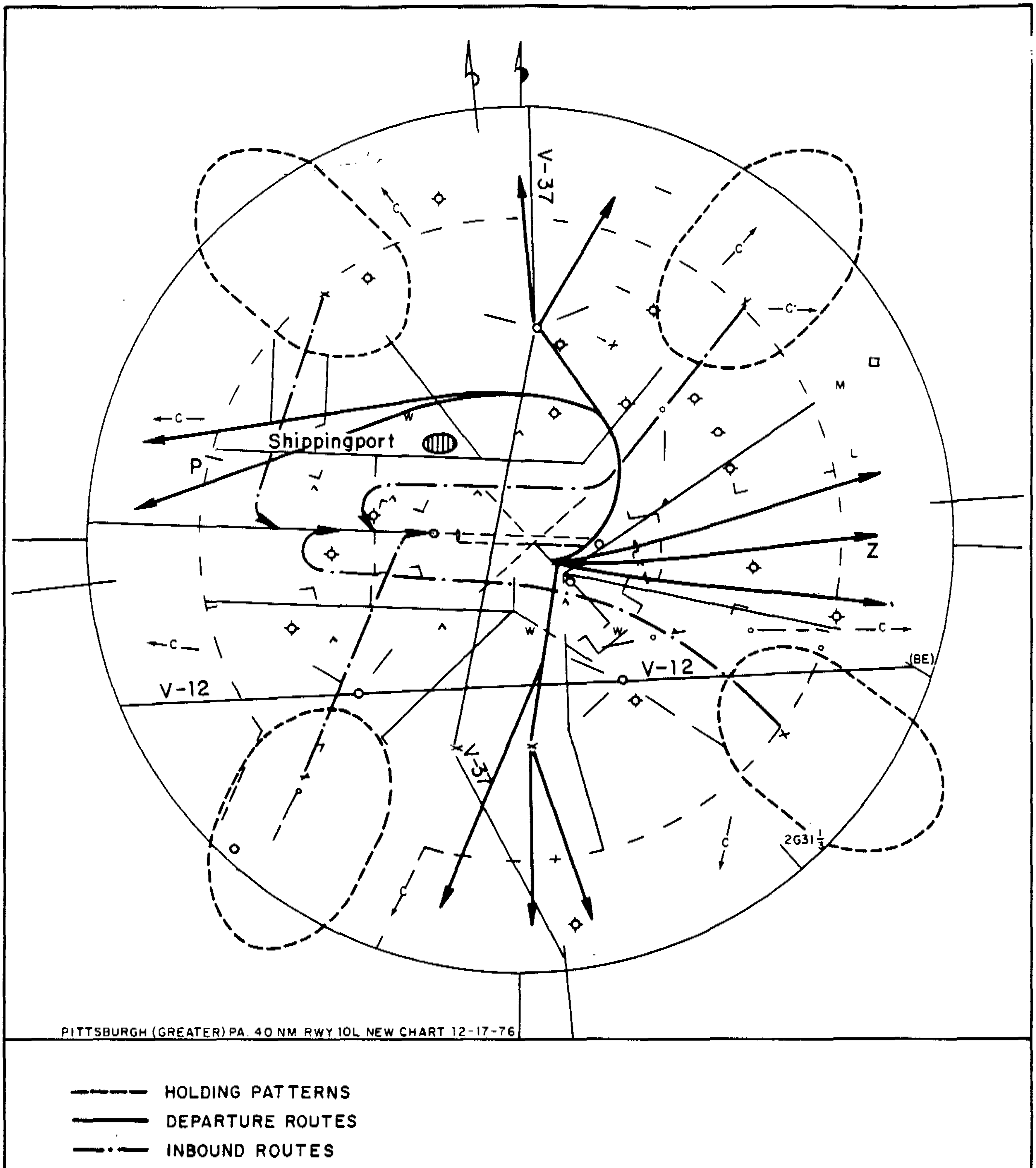
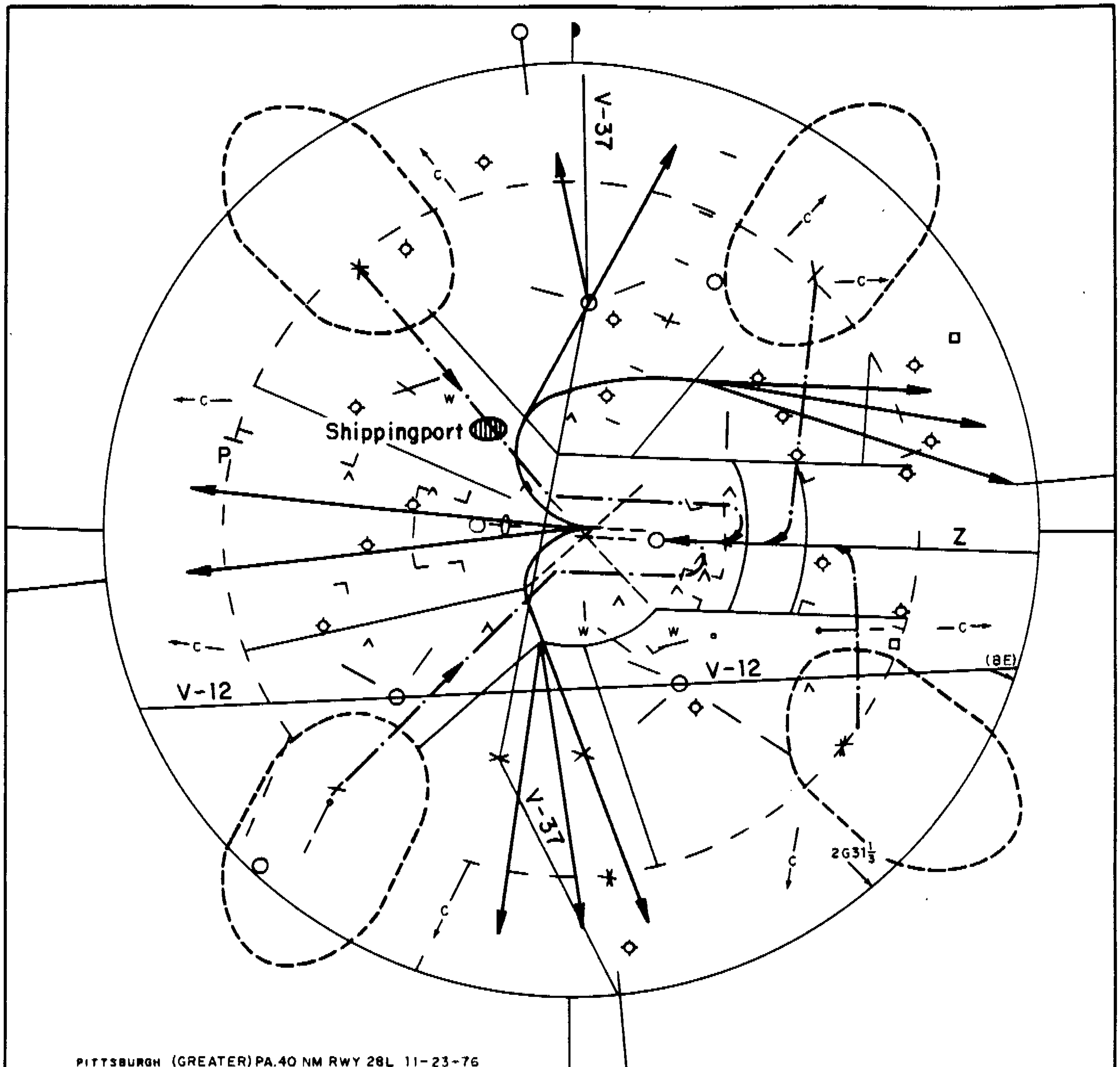


FIGURE 3B.4-5
ROUTES AND OPERATING PATTERNS, TRAFFIC TO THE EAST, GREATER PITTSBURGH AREA

BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT



HOLDING PATTERNS
DEPARTURE ROUTES
INBOUND ROUTES

FIGURE 38.4-6
ROUTES AND OPERATING PATTERNS, TRAFFIC TO THE WEST, GREATER PITTSBURGH AREA
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

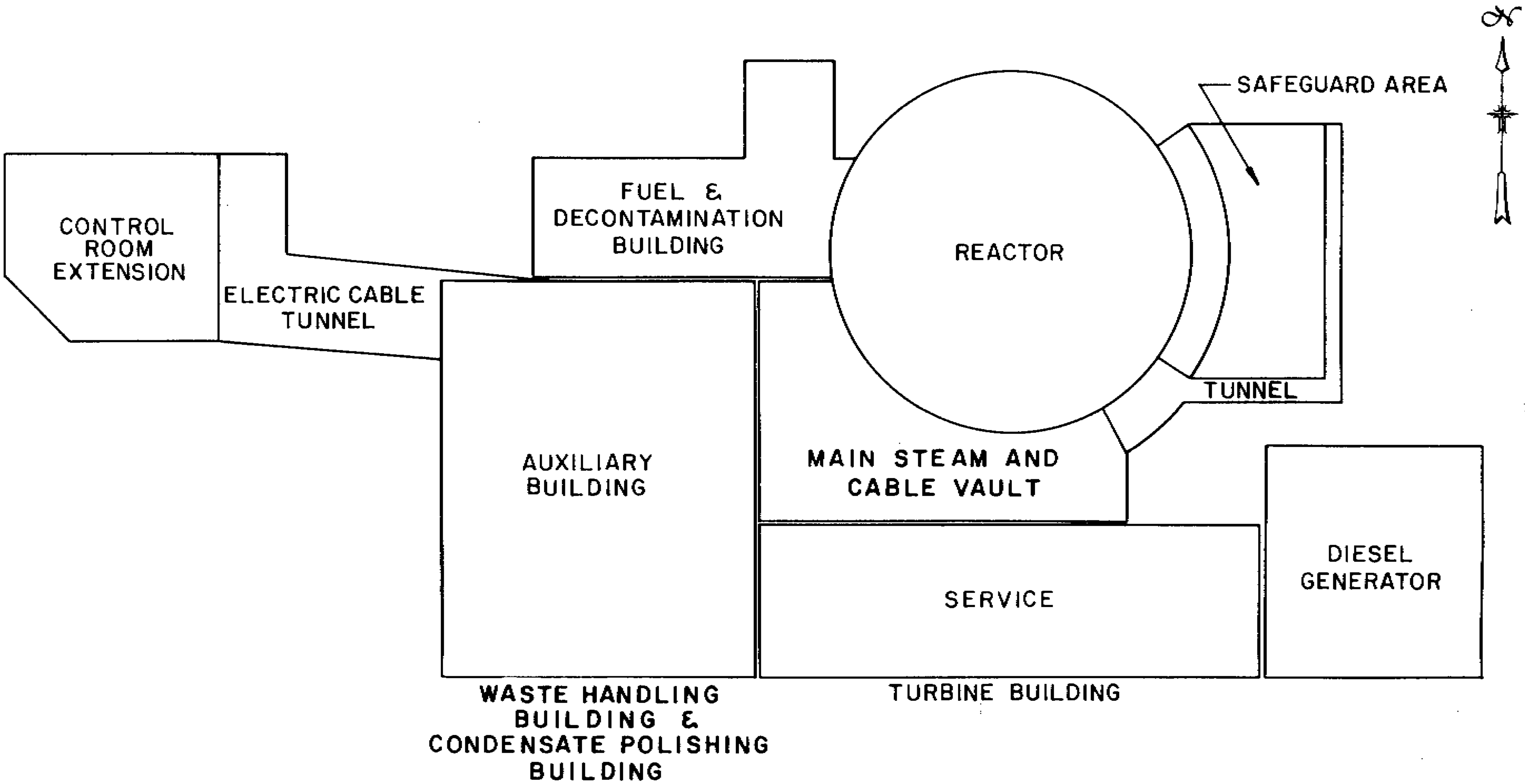


FIGURE 3B.5-1
BEAVER VALLEY POWER STATION
UNIT No. 2
SAFETY-RELATED STRUCTURES
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

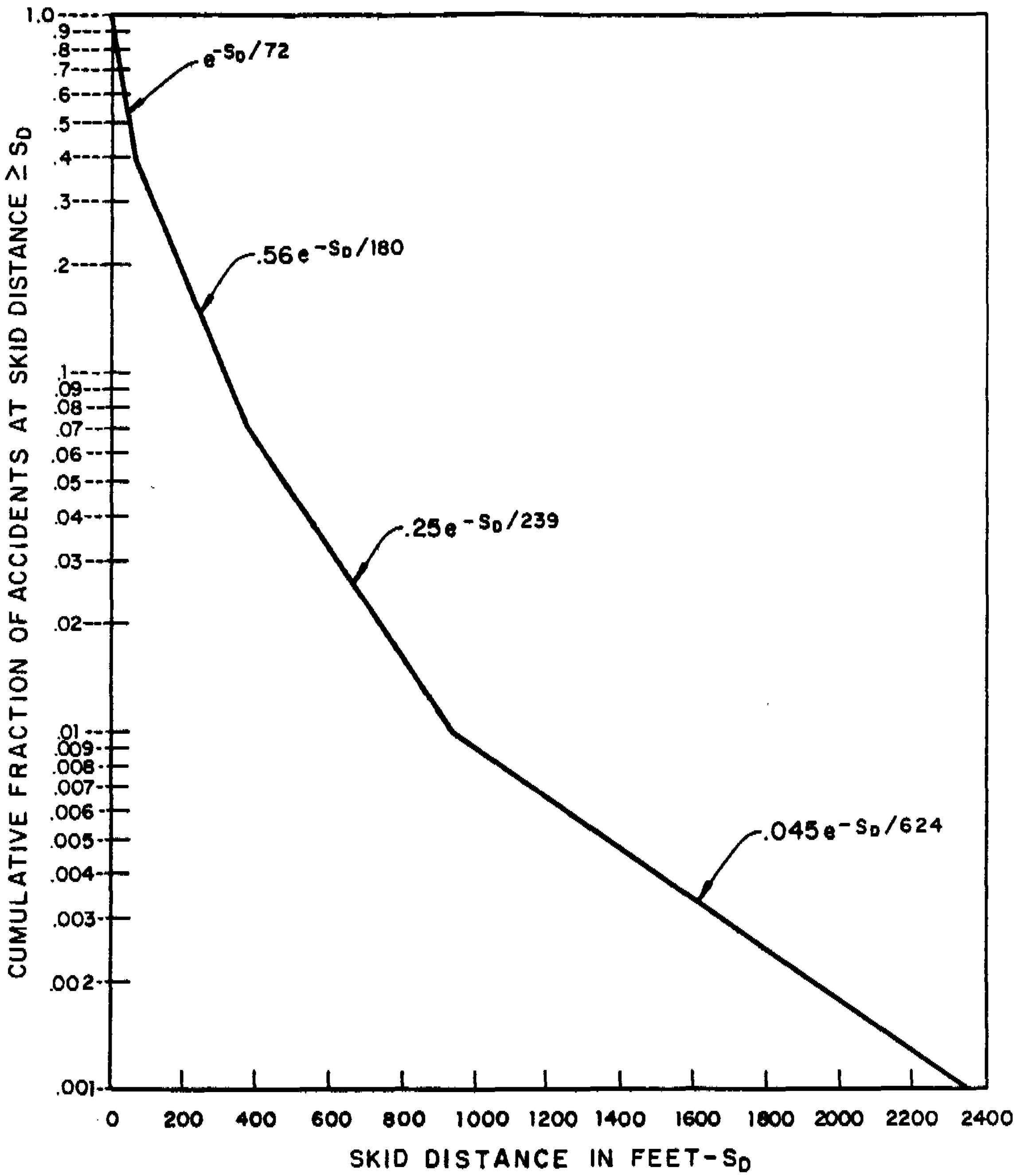


FIGURE 3B.5-2

FREQUENCY CURVE OF SKID DISTANCE FOR GENERAL AVIATION ACCIDENTS

BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

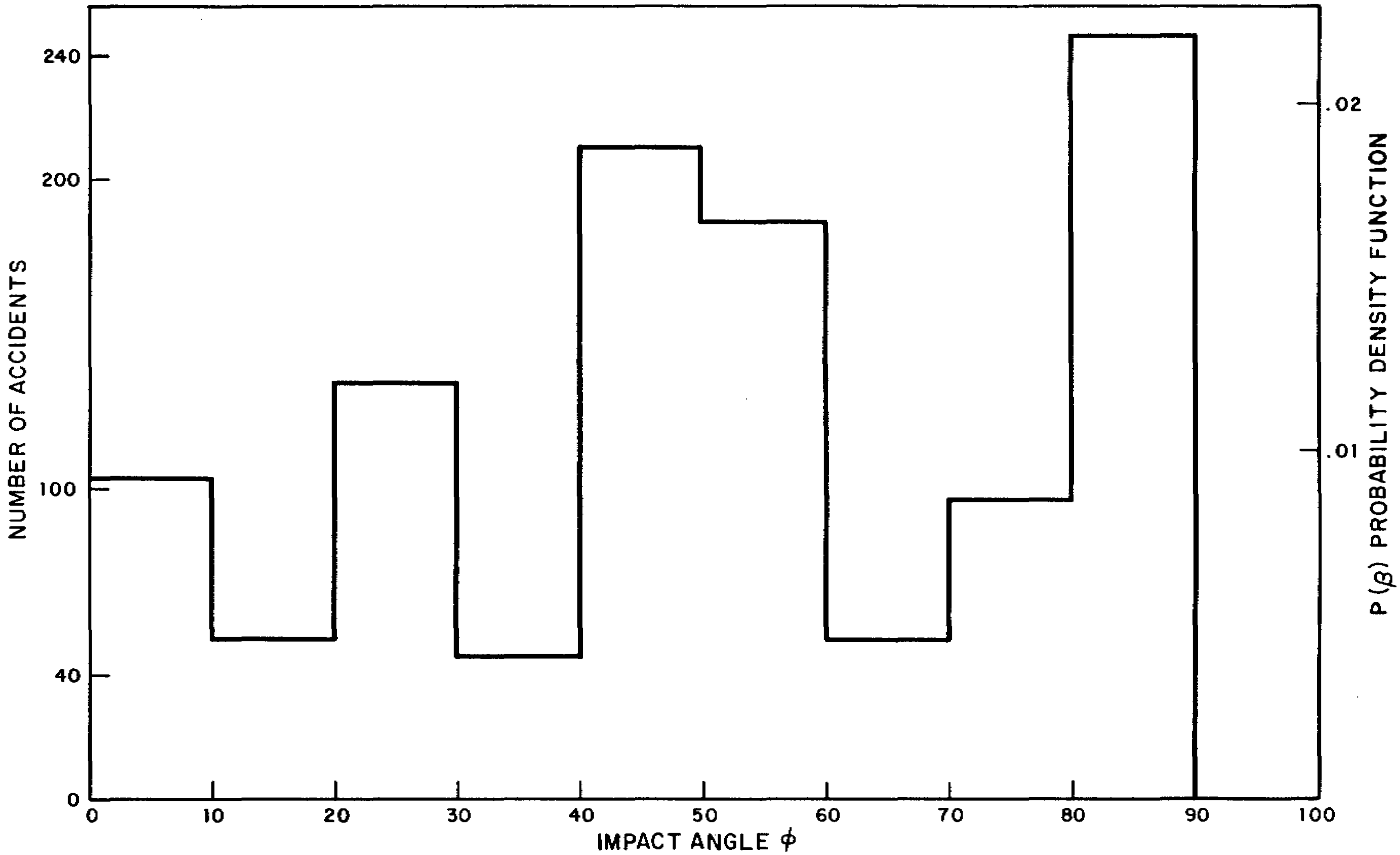


FIGURE 3B.5-3
DISTRIBUTION OF IMPACT ANGLE FOR GENERAL AVIATION ACCIDENTS
BEAVER VALLEY POWER STATION—UNIT 2
FINAL SAFETY ANALYSIS REPORT

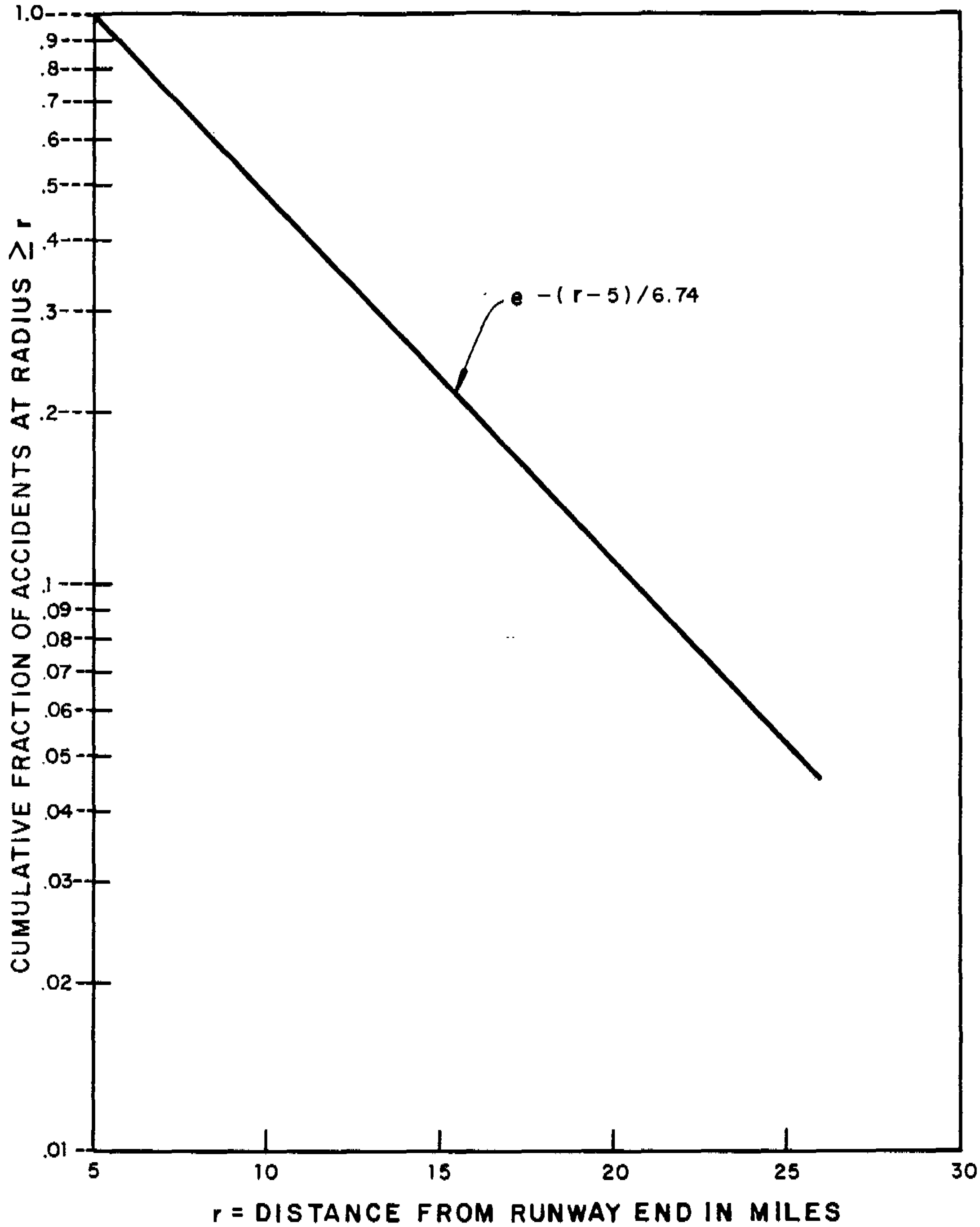


FIGURE 3B.7-1
FREQUENCY CURVE OF RADIAL LOCATION FOR AIR CARRIER ACCIDENTS
BEAVER VALLEY POWER STATION – UNIT 2
FINAL SAFETY ANALYSIS REPORT

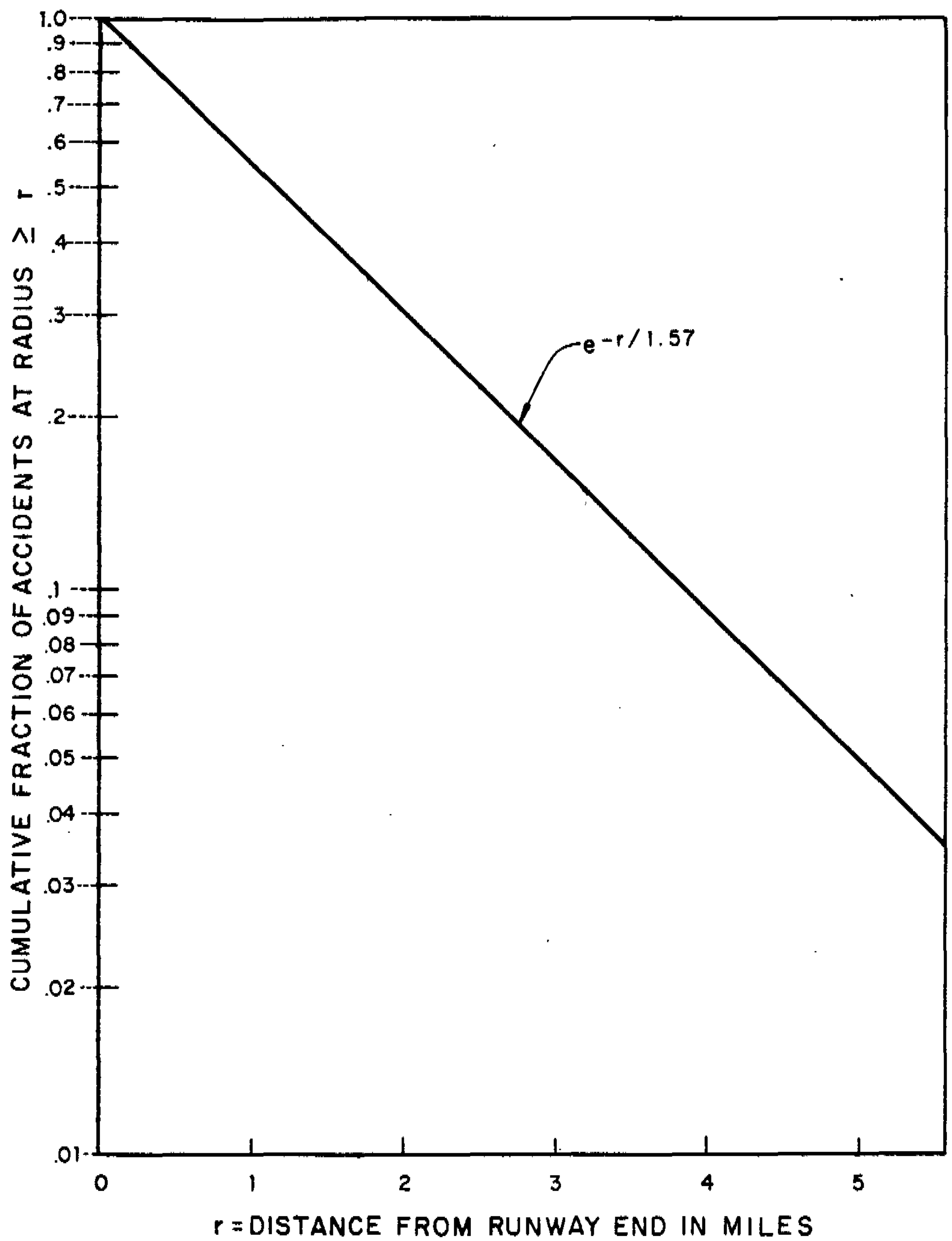


FIGURE 3B.7-2
FREQUENCY CURVE OF RADIAL LOCATION FOR GENERAL AVIATION ACCIDENTS
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

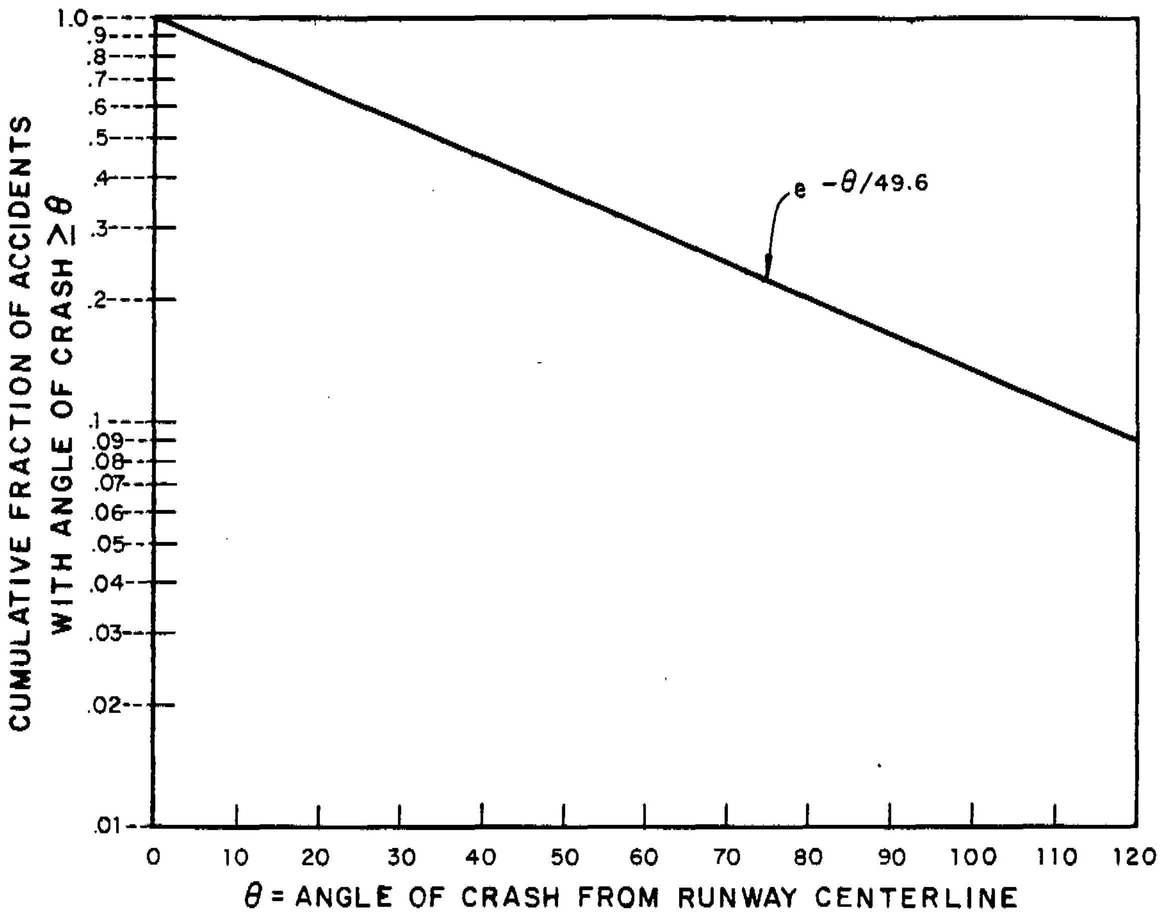


FIGURE 3B.7-3

FREQUENCY CURVE OF ANGULAR LOCATION FOR GENERAL AVIATION ACCIDENTS

BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

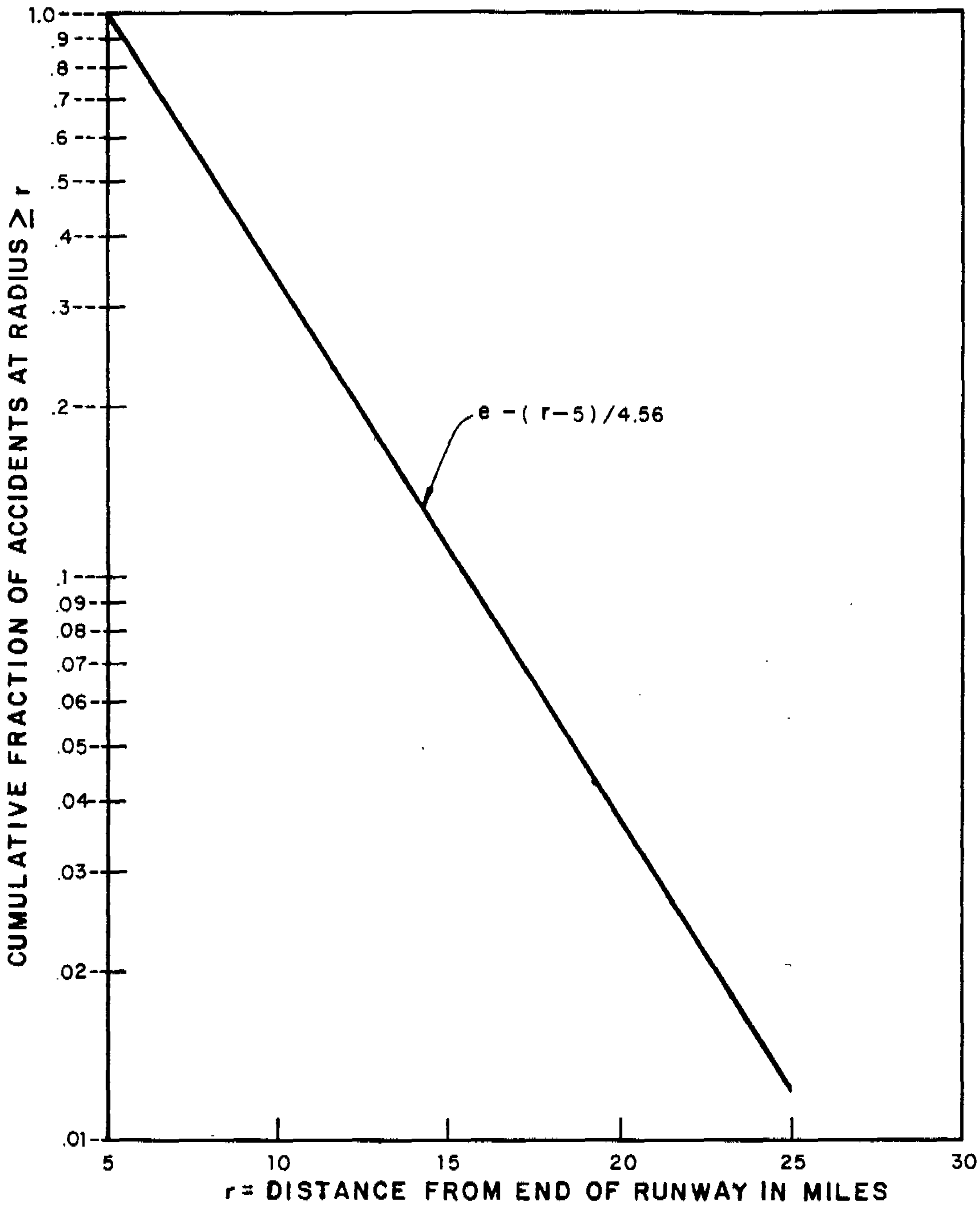


FIGURE 3B.7-4

FREQUENCY CURVE OF RADIAL LOCATION FOR GENERAL AVIATION ACCIDENTS BEYOND 5 MILES

BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

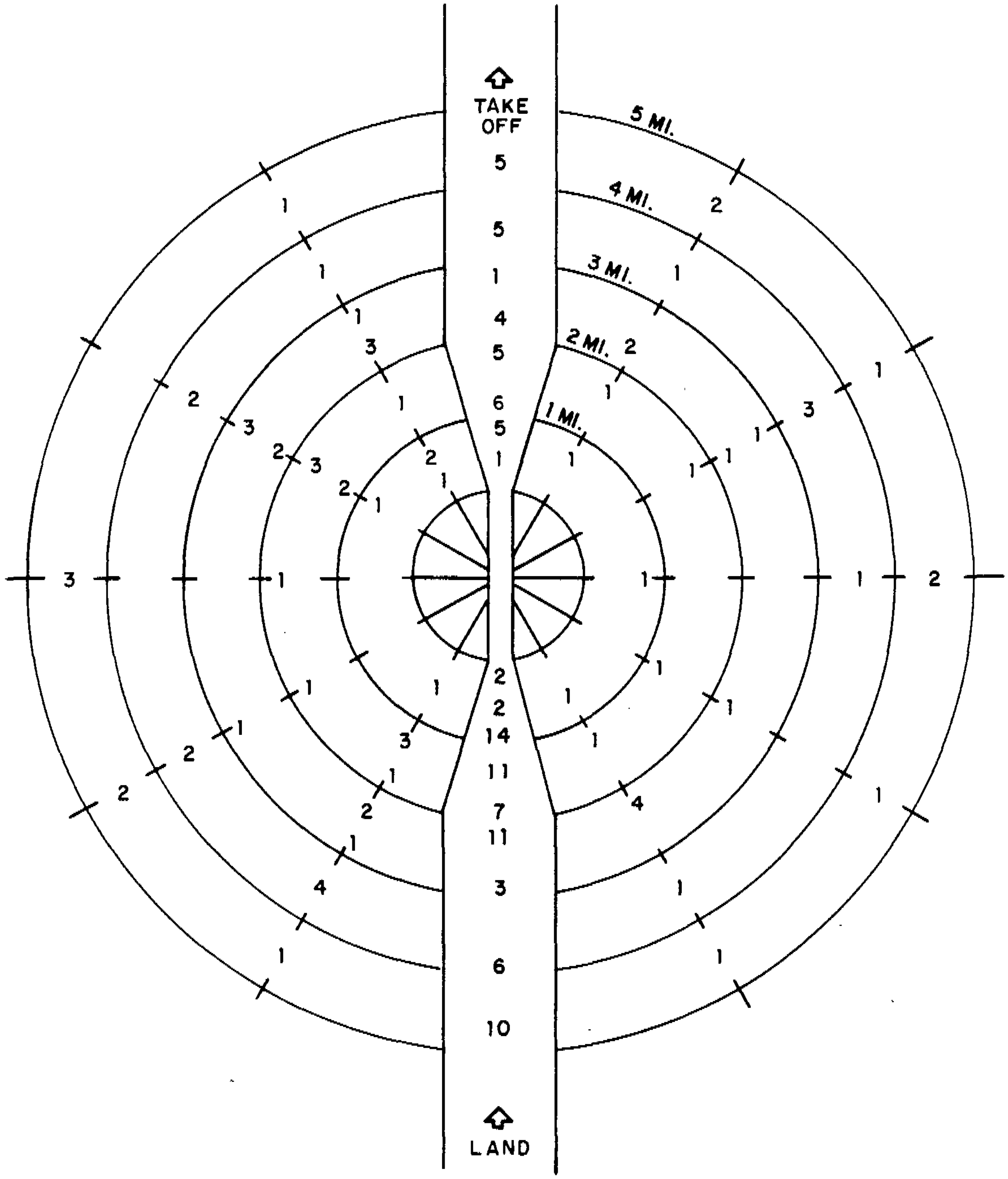


FIGURE 3B.7-5
LOCATION OF 174 USAF MAJOR AND MINOR AIRCRAFT ACCIDENTS IN THE VICINITY OF AIRFIELDS
5 MILE ZONE 1960-1964
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

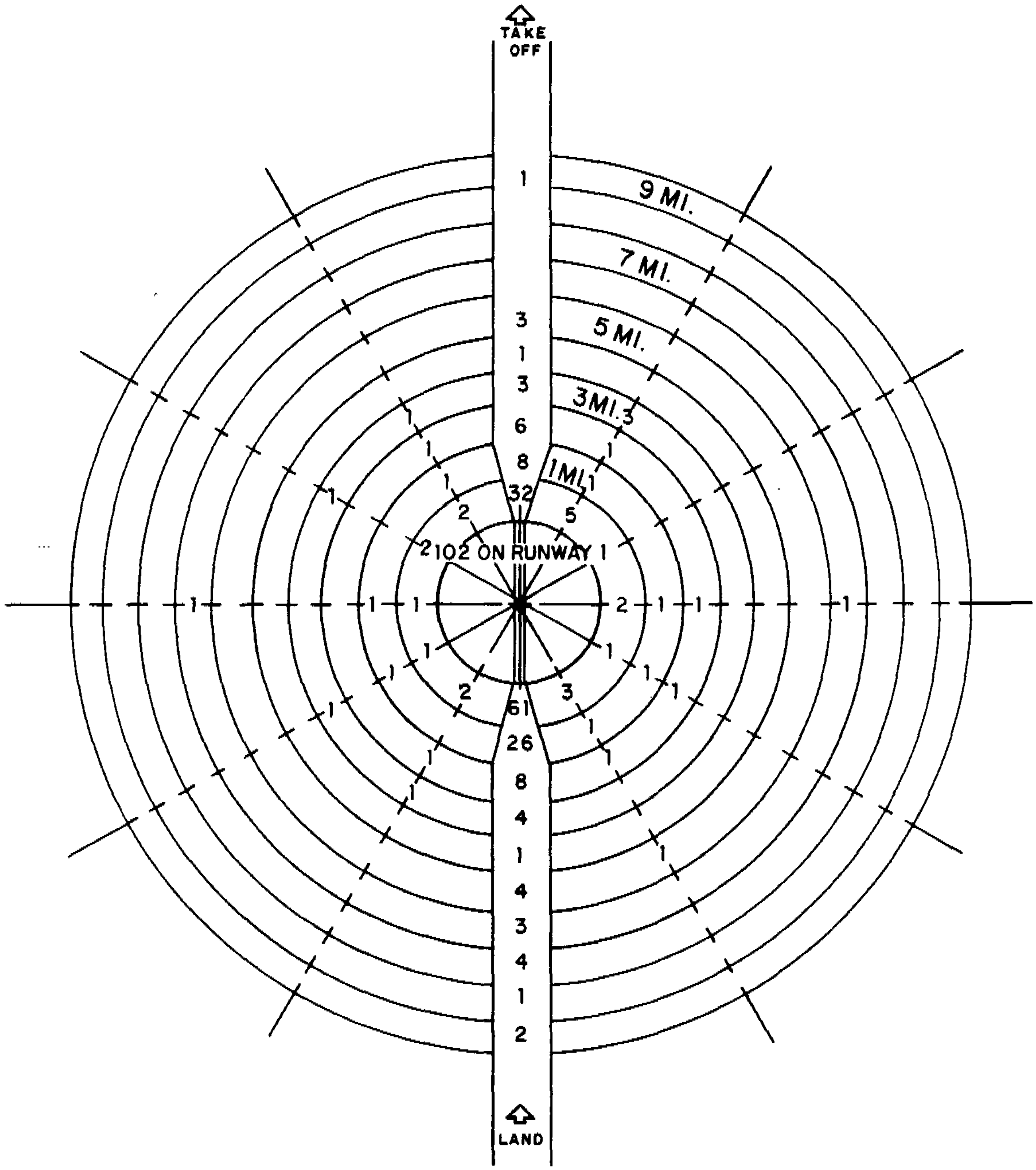


FIGURE 3B.7-6

LOCATION OF 314 USAF MAJOR AIRCRAFT ACCIDENTS IN THE VICINITY OF AIRFIELDS 10 MILE ZONE 1968-1972

BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

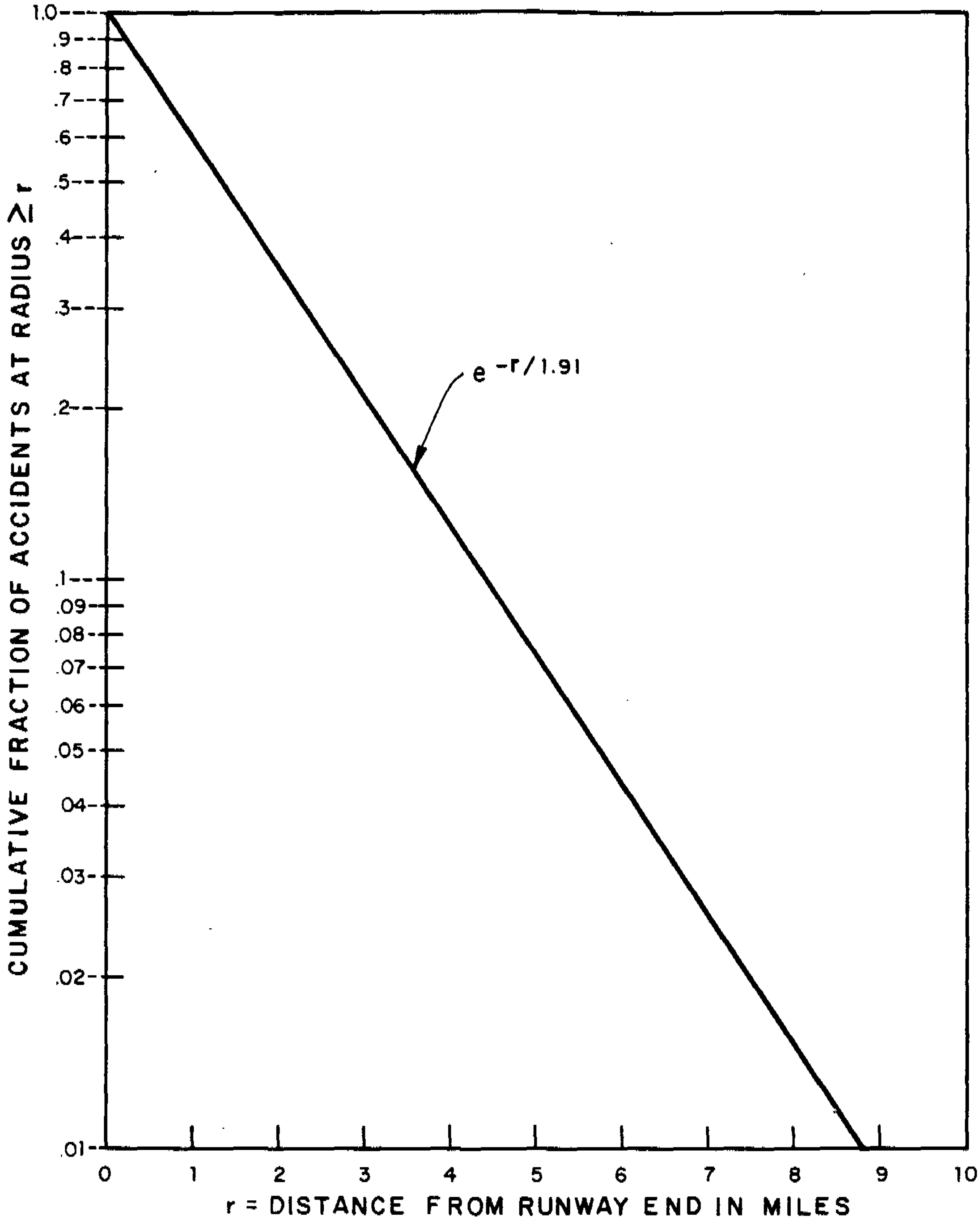


FIGURE 3B.7-7
FREQUENCY CURVE OF RADIAL LOCATION FOR MILITARY AIRCRAFT ACCIDENTS
BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

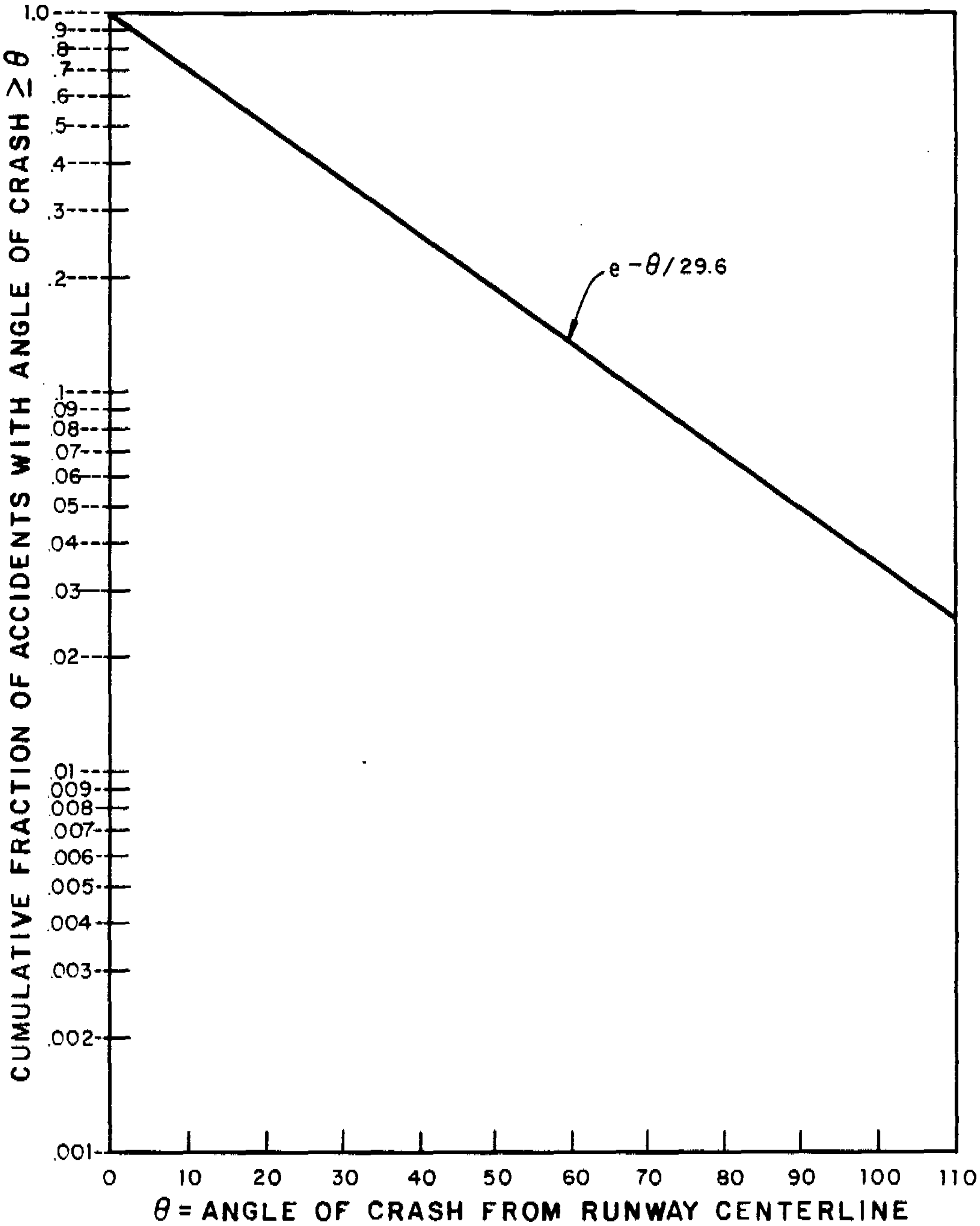


FIGURE 3B.7-8
FREQUENCY CURVE OF ANGULAR LOCATION FOR MILITARY AIRCRAFT ACCIDENTS
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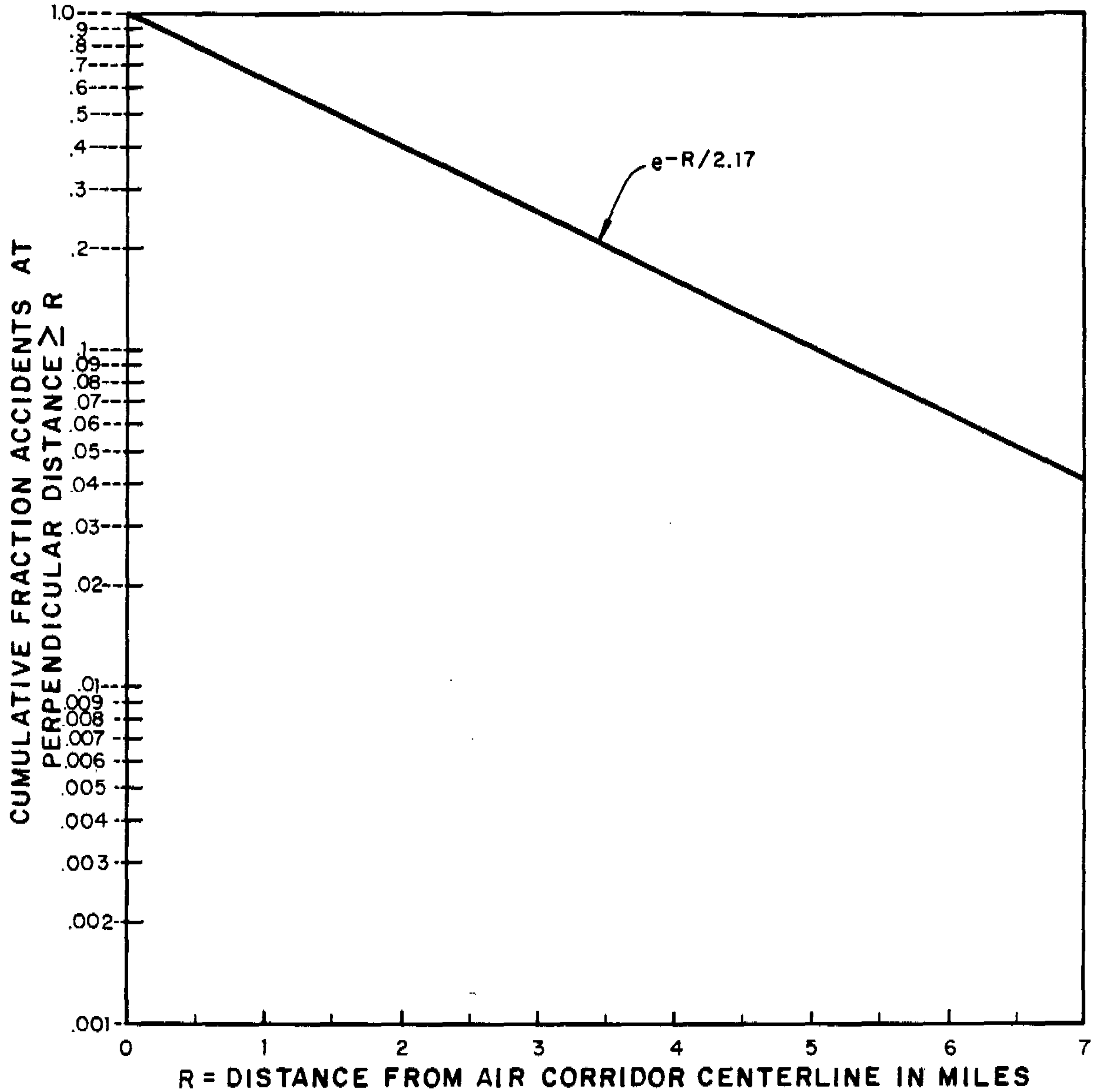

FIGURE 3B.7-9

FREQUENCY CURVE OF CRASH LOCATION FOR IN-FLIGHT AIR CARRIERS

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