PALISADES PLANT

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STEAM GENERATOR REPAIR REPORT

Consumers Power Company

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PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

1.0 INTRODUCTION, SUMMARY, AND CONCLUSIONS

During the operating history of the Palisades Plant, the steam generators have been afflicted by a number of corrosion-related phenomena. (The problems associated with wastage, intergranular attack, and "denting" at the Palisades Plant have been well documented elsewhere and will not be discussed in this report.) As a consequence of those problems, a substantial portion of the excess heat transfer capacity of the Palisades Plant steam generators has been removed. Although major progress has been made toward arresting or retarding the various corrosion mechanisms through changes in plant equipment, secondary chemistry, and operating procedures, denting remains an unresolved issue and threatens to result in additional steam generator problems.

With the uncertainty that exists with regard to future plugging requirements and the expected useful lives of the existing steam generators, Consumers Power Company has evaluated those practical major repair alternatives that are available to restore the steam generators to their original condition and regain the initial heat transfer capability. Complete replacement of the steam generators has been determined to be the preferred method of repair if major repair becomes necessary.

Recognizing the length of the procurement lead times associated with repair components and the potential time required to obtain regulatory approval for a program of major repair, Consumers Power Company has embarked upon a contingency planning effort designed to mitigate possible adverse consequences of continued corrosion. The major elements of this contingency plan include: procurement of replacement steam generators and other long lead time materials, engineering studies and detailed engineering necessary to support the construction aspects of repair, preparation of work plans associated with repair activities, and preparation of applications for permits needed from various regulatory bodies to conduct the repairs.

At this time, no date has been established for commencing the repair effort described in this report and, although Consumers Power Company is encouraged by the results of recent eddy current testing of the existing units, it is considered only prudent to proceed with contingency planning while doubt exists about the continued degradation of the tubing. Sleeving, plugging, and other qualified methods will continue to be employed in the future to extend the lives of the original units as long as practical. The decision to ultimately replace the steam generators will be based upon system reserve margins, inspection requirements, the condition of the old units with respect to derating, and other relevant factors.

The following report discusses the elements of a steam generator repair program, based upon complete replacement of the existing units, which is considered to offer the optimum solution for the Palisades Plant if major repairs become necessary. The primary emphasis of the report is on the safety-related aspects of the repair; however, other important aspects of the program are also addressed.

The information presented in this report reflects the most current design information at the time of preparation. Since the design work for the program is currently underway, it may not be possible to present detailed information for all phases of the project. For those cases where engineering or design information is not available, the basis and criteria are presented. If there are alternatives under consideration, a summary of these is presented.

1.1 SUMMARY OF STEAM GENERATOR REPAIR PROGRAM

1.1.1 REPAIR ALTERNATIVES

Of the current options available to restore full steam-generating capability of the Palisades Plant, replacement of the existing steam generators with complete new units is the economic as well as the technical choice. The other major repair alternatives considered were the retubing of units in situ and repair limited to the replacement of the evaporator portions of the original units, including the primary heads (see Section 8.0). The key decision variables for selection between alternatives were the man-rem exposure associated with the repair activities and the plant outage requirement; complete replacement is the clear choice on the basis of both of these criteria.

1.1.2 REMOVAL AND REPLACEMENT OF THE STEAM GENERATORS FROM CONTAINMENT

The feasibility of various schemes for the removal and replacement of the steam generators has been examined. Three schemes were evaluated in detail:

a. Steam generator removal and replacement through the southeast containment wall

b. Steam generator removal and replacement through the northeast containment wall

c. Steam generator removal and replacement through the containment dome

The three schemes are equally feasible; however, from the standpoint of schedule and accident considerations, the first scheme is the most desirable. Therefore, the construction-related evaluations addressed in this report are only for the pathway through the southeast containment wall.

A 1/2-inch to 1-foot scale model of the containment, including the construction opening, has been constructed to aid the repair program (see Section 4.9). The model has been used extensively to detail the removal and replacement sequences of the steam generators from the containment.

1.1.3 STEAM GENERATOR CHARACTERISTICS

The replacement steam generators are designed to physically match the essential parameters of the existing steam generators and be compatible with the performance characteristics utilized in the Palisades Plant Final Safety Analysis Report (FSAR) and the license for operation at 2530 MWt. Although the plant safety analysis is now done for a power level of 2530 MWt, the replacement steam generators are designed for operation at 2650 MWt.

Improved design features are being incorporated in the replacement steam generator design to increase long-term integrity and reliability. These improved features will have no significant adverse impact on the plant safety analysis. The shop-fabricated replacement steam generators will be designed and manufactured to updated manufacturing techniques and American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) editions.

1.1.4 SAFETY-RELATED CONSIDERATIONS

The potential impacts of the replacement steam generators on the design basis events, analyzed in the FSAR, have been evaluated and are described in Section 6.1. Those evaluations conclude that the replacement steam generators will have no significant adverse impact on the design basis events.

Construction-related incidents pertaining to the transportation and handling of the steam generators have been evaluated and are described in Sections 4.1.4 and 6.2. Those evaluations conclude that the construction activities will have no significant adverse impact on the safety of the Palisades Plant.

The fire prevention and protection program to be in operation during the steam generator repair program is described in Section 6.3. It is concluded that these measures provide reasonable assurance that the Palisades Plant is protected from significant damage due to fire during the repair activities.

1.1.5 ALARA CONSIDERATIONS

Personnel exposure will be maintained "as low as is reasonably achievable" (ALARA) throughout the steam generator repair program.

Estimates of the exposures to personnel involved in the various repair alternatives have been developed using projections of work activity durations, manpower levels, and expected radiation levels.

1.1.6 OFFSITE RADIOLOGICAL CONSEQUENCES

Radiological evaluations of the gaseous and liquid releases attributable to the steam generator repair have been conducted. The effects of the releases are less than those associated with normal operation of the facility on the basis of the discussion in Section 6.2.2.

1.1.7 UNIQUE ASPECTS OF PROGRAM

As presently contemplated, there are no unique engineering or construction aspects of the Palisades Plant steam generator repair program. The repair program, including the fabrication of replacement units, will utilize conventional nuclear industry manufacturing and construction methods. The shop fabrication of the steam generators will be conducted in accordance with standard shop practices. The closure of the temporary construction opening in the containment will be performed in a manner similar to that used to close the original containment construction opening. The transport and rigging of the steam generator will utilize proven techniques. In short, the repair program will rely on fabrication and construction practices or techniques which have been previously qualified for similar applications.

1.1.8 STEAM GENERATOR DISPOSAL

The repair activity and ultimate disposal of the existing steam generators are separate issues. This report discusses the various means by which the steam generators can be disposed of to demonstrate the feasibility of disposal. Economic and ALARA considerations will determine the method to be utilized. Because of the uncertainty of the timing of the repair and availability of offsite disposal facilities, the ultimate disposition of the old units cannot be finalized at this time.

1.2 IDENTIFICATION OF PRINCIPAL AGENTS

Consumers Power Company, hereinafter called Consumers, is a public utility incorporated to do business under the laws of the State of Michigan. Consumers is the sole owner and operator of the Palisades Nuclear Plant. Consumers will have overall responsibility for the steam generator repair program.

Bechtel Power Corporation, a Nevada corporation (hereinafter called "Bechtel" except in respect to the performance of services of a professional engineer), and Bechtel Associates Professional Corporation, a professional corporation authorized to practice professional engineering in the State of Michigan (hereinafter called "Bechtel" in respect only to the performance of services of a professional engineer), have been retained by Consumers to provide construction planning, engineering, procurement, and cost and scheduling services for the steam generator repair program.

Combustion Engineering, Inc., a Delaware corporation authorized to do business in the State of Michigan (hereinafter called Combustion), will manufacture the replacement steam generators. Combustion also provided the existing steam generators.

Because of the uncertainty of the timing of the repair, the principal agent for the installation portion of the repair program cannot be identified at this time.

1.3 10 CFR 50.59 CONSIDERATIONS

Repair or replacement of equipment at a power plant, performed in accordance with appropriate procedures, is a maintenance activity that is routinely conducted. Because of the scope of the steam generator repair, it was considered prudent to evaluate this activity to determine:

a. If the proposed repair activity would involve a change in the technical specifications incorporated in the licence.

b. If the proposed repair activity would involve an unreviewed safety question per the requirements of 10 CFR 50.59(a)(2).

Each design basis event FSAR accident analysis has been evaluated, and it has been concluded that the replacement steam generators would not alter the conclusions reached in the FSAR. The evaluation indicates that the steam generator repair does not involve an unreviewed safety question. A change in the plant technical specifications is required to incorporate the main steam line isolation on high containment pressure discussed in Section 3.6 of this report.

The construction incident potential has been evaluated to determine the presence of any new or unique accidents and the potential impact on cooling spent fuel. The evaluation indicates that the steam generator repair activity does not involve an unreviewed safety question.

Additionally, before replacement of the Steam generators, the effective Palisades Plant Technical Specifications will be reviewed again and revised as necessary.

1.4 CONCLUSIONS

The fundamental conclusions reached are that the steam generator repair can be conducted utilizing proven manufacturing and construction techniques and that the repair program does not result in any significant adverse impact on the plant safety analysis and the ability to maintain a safe configuration and cool stored fuel. Additionally, current FSAR safety analyses are applicable to the replaced steam generators. The detailed bases supporting these conclusions are provided in the report that follows.

2.0 REPLACEMENT COMPONENT DESIGN

Combustion is designing and shop-fabricating two replacement steam generator units shown in Figure 2.2-1. The design and performance of the replacement steam generators will match that of the steam generators being replaced such that:

a. The replacement steam generators' physical parameters are essentially equal to those of the original units.

b. The replacement steam generators' operating characteristics and parameters are compatible with the plant safety analysis at 2530 MWt.

Although the plant safety analysis is for a power level of 2530 MWt, the replacement steam generators are designed for operation at 2650 MWt (see Table 2.1-2).

The replacement units incorporate many improved design features derived from both plant operating experience and development programs directed toward design of steam generators for long-term integrity and reliability. This section discusses the design and manufacture of these replacement units.

2.1 COMPARISON WITH EXISTING COMPONENT DESIGN

2.1.1 PARAMETRIC COMPARISON

The replacement steam generators for the Palisades Plant will have physical and mechanical characteristics similar to the original design documented in the FSAR (see Table 2.1-1). These characteristics provide the replacement steam generators with a thermal performance consistent with the original steam generators.

The original steam generators were fabricated to the 1965 Edition of the ASME Code, including all addenda through Winter 1966. The replacement steam generators are being designed and fabricated to the 1977 edition of the ASME Code. The stress report for the replacement steam generators will be based on the 1977 edition of the ASME Code. The fabrication and analysis requirements for the replacement steam generators will be at least equivalent to those utilized for the original steam generators.

The replacement steam generators will include a number of design improvements which are discussed in Section 2.2. Many of the advanced design features incorporated on the replacement steam generators are similar to features included on Combustion's System 80 steam generators.

Data for the replacement steam generators at 2450 MWt is presented in Table 2.1-1, allowing comparison of these parameters between the original steam generators and the replacement steam generators. The design data for the replacement steam generators at 2530 MWt and 2650 MWt is shown on Table 2.1-2.

Materials used in the fabrication of the replacement steam generators are procured to the requirements of the 1977 edition of the ASME Code. These materials are identical to those used in the original steam generators except where specific design improvements have been incorporated or the fabrication practice has been improved. Table 2.2-1 enumerates applications of materials for the original steam generators and the replacement steam generators.

2.1.2 PHYSICAL COMPATIBILITY WITH EXISTING STEAM GENERATORS AND SYSTEMS

The replacement steam generators (see Figure 2.2-1) are designed and fabricated, to the extent possible within the constraints of physical dimensions and design requirements, to preserve the existing plant mechanical interfaces, including the support structures and loadings. Interfaces between the steam generators and other plant components and systems are maintained.

2.1.3 ASME CODE APPLICATION

The present operating steam generators were designed and fabricated to the requirements of the 1965 edition of ASME Code, Section III, including all addenda through Winter 1966. The replacement steam generators will be fabricated to the requirements of the 1977 Edition of the ASME Code. Design of the replacement steam generators is consistent with the design of the primary coolant system. Materials used in fabrication are being procured to the requirements of current codes. All material certification tests will be performed and recorded as required by the ASME Code.

2.1.4 REGULATORY GUIDE APPLICATION

The compilation below addresses regulatory guides considered applicable to the fabrication of the replacement steam generators. It must be noted that these guides were issued after construction of this facility. The intent is to accommodate the guidance of these regulatory guides insofar as they provide an acceptable method to allow the licensee to comply with the requirements of 10 CFR 50.

a. Regulatory Guide 1.29, Seismic Design Classification (February 1976)

   Combustion's design of the replacement steam generators is consistent with design to withstand the effects of the safe shutdown earthquake (SSE) and the classification guidance of this regulatory guide.

b. Regulatory Guide 1.31, Control of Stainless Steel Welding (May 1977)

   Combustion's shop fabrication welding quality assurance is described in CENPD-210, Quality Assurance Program - A Description of the Combustion Engineering Nuclear Steam Supply System Quality Assurance Program.

c. Regulatory Guide 1.34, Control of Electroslag Weld Properties (December 1972)

   Where electroslag welding is utilized, Combustion requires its suppliers to follow the recommendations of this guide.

d. Regulatory Guide 1.43, Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components (May 1973)

   Combustion's shop fabrication weld cladding of the replacement steam generators utilizes materials which are not susceptible to underclad cracking. These materials do not require the controls listed in this guide.

e. Regulatory Guide 1.44, Control of the Use of Sensitized Stainless Steel (May 1973)

All of the unstabilized austenitic steels used for component parts of the primary coolant pressure boundary are utilized in the final heat treated condition required by the ASME Code, Section II material specification for that particular type or grade of alloy. Processing and fabrication are performed utilizing established techniques to avoid sensitization.

f. Regulatory Guide 1.48, Design Limits and Loading Combinations for Seismic Category I Fluid System Components (May 1973)

Combustion's design of the replacement steam generators meets the requirements of general design Criterion 2 and the ASME Code. The loading combinations and design limits utilized in the stress report for the steam generators are consistent with those considered in the FSAR.

g. Regulatory Guide 1.50, Control of Preheat Temperature for Welding of Low-Alloy Steel (May 1973)

Combustion shop fabrication practices are in agreement with Regulatory Positions C.1.a, C.3, and C.4. The soundness of all welds is verified by an acceptable examination procedure which per Regulatory Position C.4 meets all requirements of this regulatory guide.

h. Regulatory Guide 1.71, Welder Qualification for Areas of Limited Accessibility (December 1973)

Combustion's shop fabrication welding complies with the fabrication requirements specified in ASME Code, Sections III and IX.

i. Regulatory Guide 1.83, Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes (Rev 1, July 1975)

Combustion has designed the replacement steam generators to allow access to the tubes for inspection and plugging.

j. Regulatory Guide 1.84, Code Case Acceptability - ASME III Design and Fabrication (August 1977)

k. Regulatory Guide 1.85, Code Case Acceptability - ASME III Materials (August 1977)

2.2 COMPONENT DESIGN IMPROVEMENTS

The replacement steam generator design incorporates the traditional Combustion design features and design improvements that have evolved through several generations of steam generator designs in response to the operational steam generator problems that have occurred in the nuclear industry.

The replacement steam generators will essentially duplicate the physical, thermal, and hydraulic characteristics of the original units while incorporating a combination of features proven in field operation and design improvements to mitigate operational problems.

The heating surface has been selected to provide thermal performance which would match that presently installed and to respond to plant thermal transients in the same manner as does the existing unit. The design will provide improvements in thermal/hydraulics, notably in secondary flow distribution. These improvements are intended to minimize flow stagnation, steam blanketing, and harmful solids accumulations. It is important to avoid harmful solids deposits in contact with heat transfer tubing in the steam generators. The blowdown arrangements are designed to take advantage of the improved flow distribution, making significant improvements in the effectiveness of blowdown in removing harmful solids deposits.

The tube support system utilizes the traditional Combustion eggcrate tube support, with its low flow resistance, support against vibration and wear, and resistance to tube denting or lateral tube deformation.

The bend region tube support system also uses the standard Combustion approach, with double 90 degree bends and support assemblies of interlocking strips. The design provides positive restraint against vibration and resistance to tube deformations during LOCA, steam line break, and seismic events. The tube support system provides rugged, positive support while minimizing flow resistances and the possibility of local dryout of steam blanketed regions.

Access openings and inspection ports are provided to enable inspection of tubes, tubesheet, and support surfaces within the tube bundle as well as at the periphery of the bundle.

2.2.1 DESIGN FEATURES TO IMPROVE PERFORMANCE

2.2.1.1 Thermal Performance

In order to minimize the effect on plant transient performance, heat transfer tubes of 3/4 inch outside diameter and .042 inch average wall thickness (consistent with Combustion's System 80 design) will be provided in such quantities and lengths on the replacement units that the product of their area (A) and heat transfer coefficient (U) equals the product of the original area and original coefficient, i.e., (UA) new = (UA) original. The total cross-sectional flow area of the heat transfer tubes will be equal to the cross-sectional flow area of the tubes in the original units. The control of these parameters on the replacement units allows the hydraulic impedance to primary flow to essentially correspond with that of the original steam generators.

2.2.1.2 Deleted

2.2.1.3 Blowdown Capability

The potential blowdown capabilities for the replacement steam generators are designed to take advantage of the improvements incorporated, which contribute to the improved secondary flow distribution and hydraulics within the operating steam generator. The recirculating fluid exits from the downcomer and flows radially across the tube bundle. The lowest fluid velocities occur near the center open region of the tube bundle where dropout of solid particles may occur.

In this region, free of heat transfer tubes, blowdown duct that takes suction in a circular pattern adjacent to the innermost tubes is provided. Figure 2.2-4 shows the schematic arrangement of the blowdown duct and its relationship to the tube bundle. At the end of each circumferential section of the blowdown duct, a transport duct (with no blowdown openings) carries the blowdown fluid across the divider lane discharging through intersecting holes drilled in the tubesheet to a 6-inch Schedule 80 blowdown nozzle. See Section 3.1 for a description of the connecting blowdown system. The internal blowdown duct and tubesheet blowdown connection for the replacement units have been sized to accommodate future higher blowdown capabilities than those available on the original steam generators.

The blowdown capability provided on the replacement steam generators is able to provide effective continuous blowdown and potential periodic blowdown at high flowrates.

2.2.1.4 Tube Supports

There are essentially three types of structures within the Palisades Plant replacement steam generators that provide support to the tubes. In addition, the tubesheet, into which the tubes are expanded through the full thickness, provides fixed support to the extremities of all tubes. The three types are (see Figure 2.2-5):

a. H - Horizontal Grid or "Eggcrate"

   This grid may be a full circular structure or a partial circle bounded by the circumference and a chord. It is composed of slotted strips intersecting at an angle of 60 degrees and joined together at the outer and inner perimeters with a pair of square bars, top and bottom. Normally, the strips alternate between a 2-inch slotted one and a 1-inch unslotted one, both .090-inch thick. The structure is fabricated in a fixture and thereafter handled as a plate (see Figure 2.2-6).

b. V - Vertical Grid

   Unlike the eggcrate support, this structure is assembled during the installation operation concurrently with the tubing operation. It is composed of vertical, slotted 2-inch strips intersecting with horizontal .5-inch strips, both .090-inch thick. The assembly is bounded about the periphery by either square bars or custom-shaped plates depending on location (see Figures 2.2-7 and 2.2-8).

c. B - Batwing Strips

   These strips provide out-of-plane support to the bend corners which would otherwise be quite flexible. The strips are 2-inch deep and .090-inch thick. They rest in a slotted T-bar at their lower center and are joined together at their extremities by a wrapper strip which takes a sinuous shape when completely installed (see Figure 2.2-7).

The tube support system (described above) selected for the Palisades Plant replacement units is particularly advantageous in that it provides sufficient strength while minimizing resistance to flow.

The large open flow area available in this support system avoids the accumulation of boiler water deposits by eliminating local flow eddies and flat surfaces present in other commonly used tube bundle support systems. Avoiding the accumulation of corrosion products avoids the concentration of acid-producing chloride cells, which is believed to cause accelerated carbon steel support corrosion and subsequent tube denting. Further, if magnetite growth due to the chloride salt concentration does occur within the eggcrate, the geometry should not produce the denting phenomena. The thin strips which make up the support matrix do not have sufficient rigidity to collapse the tube wall.

Both the horizontal and vertical grids are made from 409 ferritic stainless steel. This material was selected because of its high resistance to general corrosion and thinning. The ferritic stainless is preferable to austenitic stainless because the coefficient of thermal expansion is more compatible with carbon steel and Inconel material.

2.2.1.5 Feedwater Ring and Feedwater Nozzle Liner

The feedwater distribution ring for the replacement steam generators will consist of two half-ring pipes connected by a "goose neck" elbow arrangement to a header at the feedwater nozzle. A gap of approximately 20 inches will be provided at a location 180 degrees from the feedwater nozzle to allow access to the area below the feed ring. Support for the ring is provided by attachments welded to the steam generator pressure shell. Distribution of feedwater to the downcomer will be accomplished by discharge nozzles spaced around the circumference of the ring. These nozzles will be 90 degree elbows (J-tubes) mounted on top of the ring. This design, as opposed to underside discharge ports, prevents immediate feed ring draining in the event of loss of feed flow, thereby minimizing the possibility of water hammer.

Thermal fatigue protection for the feedwater nozzle will be provided by a nozzle thermal liner. It is the purpose of this liner to insulate the nozzle material from cold water transients. The liner also acts as the piping link between the feed nozzle and the feedwater ring. The design of the thermal liner also serves to minimize the leakage at the liner/header joint, thereby minimizing the potential for water hammer following feedwater flow interruptions.

2.2.1.6 Tube Lane Divider Plate

On the replacement units, a divider plate is mounted in the tube lane between the hot and cold leg sides of the tube bundle beneath the flow distribution baffle. The divider plate performs a dual function of preventing preferential bypass of the tube bundle by recirculating water exiting from the downcomer along the tube lane and providing a positive support arrangement for the blowdown duct. The divider plate is attached to the lower shell by tongue and groove joints so that there is no structural interaction with the secondary shell under pressure and thermal deflections.

2.2.1.7 Integral Flow Restrictor Steam Outlet Nozzle

The steam outlet nozzles on the replacement steam generators will incorporate an integral flow restriction equivalent to 70% of total flow area (see Figure 2.2-10). These nozzles are similar to those installed on Combustion's System 80 steam generator and will serve to limit the generator blowdown rate during a main steam line break accident.

2.2.1.8 Water Sampling Provisions

Each replacement steam generator will be provided with 2 secondary water sampling connections (see Figure 2.2-10). They consist of a 3/8-inch diameter internal water sampling line which exits through a 3/4-inch instrument nozzle in the steam drum shell. Internally the sample line provides water samples from a sampling cup on the separator deck on the basis that recirculating water from the steam separators is most nearly representative of boiler water in contact with the heat transfer tubes. See Section 3.3 for a description of the connecting sampling system.

Water sampling may also be taken from the bottom blowdown line external to the steam generator.

2.2.2 DESIGN FEATURES TO IMPROVE MAINTENANCE AND INSPECTION

2.2.2.1 Handholes

The replacement steam generators will include four 6-inch handhole openings on the lower and intermediate shells to facilitate inspections. The lower two handholes will be positioned just above the tubesheet and have provisions for viewing through the tube bundle shroud.

The upper handholes are located just above the eggcrate in the tube lane and are adjacent to the bend region of the tube bundle. These handholes will also incorporate the provision for viewing through the tube bundle shroud (see Figures 2.2-1 and 2.2-10).

2.2.2.2 Inspection Ports

Two 2-inch inspection ports will be added to the replacement units just above the tubesheet secondary face to provide accessibility to the tubesheet surface and to allow use of an inspection device such as a boroscope to observe tubes on either side of an opening between two particular tube rows.

2.2.2.3 Deleted

2.2.2.4 Primary Head Drains

To facilitate draining of the steam generator primary head before maintenance or inspection activities in this area, the replacement units include a drain nozzle (see Figure 2.2-11) on both the inlet and outlet plenums of the primary head. See Section 3.4 for a description of the connecting drain system.

2.2.2.5 Recirculation and Chemical Cleaning Nozzle

The replacement units will include a provision to allow recirculation of steam generator secondary water during standby or shutdown periods or for circulation of chemical cleaning fluids if chemical cleaning of the generator is undertaken. This capability is provided by a nozzle of 6-inch nominal pipe size in the steam drum shell to which is attached a piping system penetrating the steam separator deck and terminating in a sparger ring discharging in the riser space above the tube bundle. See Section 3.2 for a description of the connecting recirculation system.

2.2.2.6 Manways

The replacement steam generators will have larger primary and secondary manways to improve access for personnel and equipment during inspection and maintenance activities. The manways will be 18-inch inside diameter and provide access to both the inlet and outlet plenums of the primary head, as well as the secondary steam drum area.

2.3 SHOP TESTS AND INSPECTIONS

Combustion will perform all tests and inspection required by ASME Code Section III during the fabrication of the replacement steam generators. Both primary and secondary side hydrostatic tests will be performed in Combustion's manufacturing facility in accordance with ASME Code Section III. The replacement steam generators will have an N-Code stamp upon delivery at the plant site. In conjunction with the ASME Code hydrostatic tests, a cyclic leak test of the secondary side will be conducted to ensure the integrity of the tube-to-tubesheet seal welds. Consumers will arrange source inspection and perform audit functions related to fabrication and shop testing.

2.4 STORAGE CRITERIA FOR NEW STEAM GENERATORS

The following criteria are provided for use in the event that the new steam generators are not installed upon arrival at the site. Application of these criteria should maintain the integrity of the steam generators during storage.

a. A strippable protective coating will be applied to the replacement steam generators before shipment. The component surface will be protected by maintaining the integrity of this coating during storage.

b. Suitable supports should be provided for the replacement steam generators during horizontal storage.

c. An internal positive inert gas atmosphere should be maintained in the primary and secondary sides of the replacement steam generators during storage and verified through periodic testing.

d. The ambient temperature of the steam generators in storage should be maintained a minimum of 20F above the measured dew point of the internal gas.

PALISADES PLANT SGRR

TABLE 2.1-1

STEAM GENERATOR COMPARISON DATA [1]

	Original Steam Generators	Replacement Steam Generators
A. Primary Side		
1. Thermal power, MWt	2450	2450
2. Design pressure, psi	2500	2500
3. Design temperature, °F	650	650
4. Cold leg temperature, °F	547.8	547.8
5. Hot leg temperature, °F	598.5	598.5
6. Coolant flow, 10^6 lb/hr	62.25	62.25
7. Calculated pressure drop, psid	30.5	29.5
8. Normal operating pressure, psi	2100	2100
B. Secondary Side		
1. Design pressure, psi	1000	1000
2. Design temperature, °F	550	550
3. Flow rate, 10^6 lb/hr	5.281	5.281
4. Steam outlet pressure, psi	770	770
5. Feedwater temperature, °F	429.1	429.1
C. Dimensions		
1. Evaporator outside diameter, in	164	164
2. Steam drum outside diameter, in	239-3/4	239-3/4
3. Overall length, in	709.78	742.00
4. Tubing outside diameter, in	0.750	0.750
5. Tubing wall thickness, in	.048	.042
D. Hydrostatic Pressure		
1. Primary, psia	3125	3125
2. Secondary, psia	1250	1250
E. Weights and Volumes		
1. Complete vessel dry, lb	924,596	934,637
2. Vessel C. G. dry, in	345.32	344.02
3. Secondary fluid 0% power, lb	209,180	207,771
4. Secondary fluid 100% power, lb	129,164	147,288

Note:
[1] Values are per steam generator, except Item A.1.

TABLE 2.1-2

REPLACEMENT STEAM GENERATOR DATA[1]

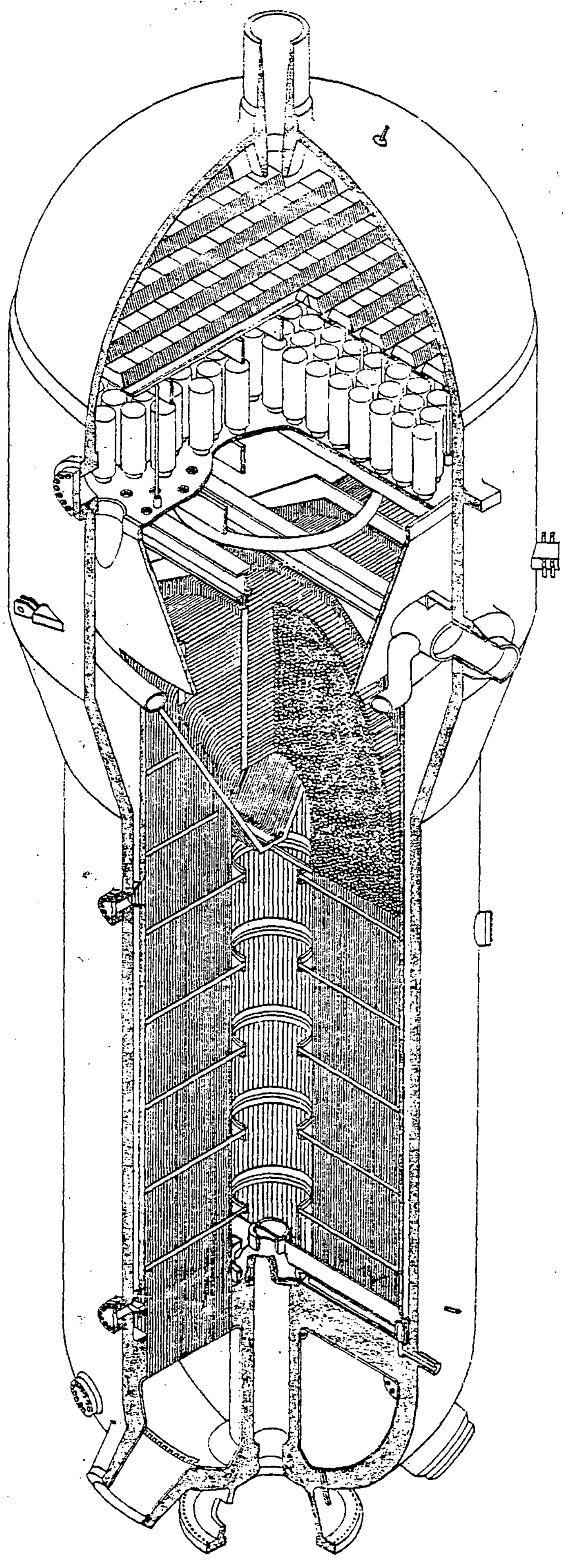
		Safety Analysis	Design
A. Primary Side			
1.	Thermal power, MWt	2530	2650
2.	Design pressure, psi	2500	2500
3.	Design temperature, °F	650	650
4.	Cold leg temperature, °F	542.5	547.8
5.	Hot leg temperature, °F	595.4	598.5
6.	Coolant flow, 10^6 lb/hr	62.5	70.0
7.	Calculated pressure drop, psid	29.6	36.1
9.	Normal operating pressure, psi	2100	2250
B. Secondary Side			
1.	Design pressure, psi	1000	1000
2.	Design temperature, °F	550	550
3.	Flowrate, 10^6 lb/hr	5.491	5.786
4.	Steam outlet pressure, psi	770	770
5.	Feedwater temperature, °F	435	438
C. Weights and Volumes			
1.	Complete vessel dry, lb	934,637	934,637
2.	Vessel C.G. dry, in	344.02	344.02
3.	Secondary fluid 0% power lb	207,771	207,771
4.	Secondary fluid 100% power, lb	145,969	143,934

NOTE:
(1) Values are per steam generator, except Item A.1.

TABLE 2.2-1

STEAM GENERATOR MATERIALS

	Original Steam Generators	Replacement Steam Generators
Upper, intermediate, and cone shells	SA-302, Grade B alloy steel	SA-533, Grade A, Class I alloy steel
Lower shell	SA-516, Grade 70 carbon steel	SA-533, Grade A, Class I alloy steel
Tubesheet forging	SA-508, Class II alloy steel	SA-508, Class III alloy steel
Tube support plates	SA-36 carbon steel	--
Eggcrate tube supports	A-570, Grade D/ A-303-64, Grade D carbon steel	A-176, Type 409 stainless steel
Primary head	SA-302, Grade B alloy steel	SA-533, Grade B, Class I alloy steel
Primary head clad	Stainless steel	Stainless steel
Tubesheet clad	Inconel	Inconel
Heat transfer tubing	SB-163 Inconel	SB-163 Inconel
Secondary head	SA-516, Grade 70/ SA-302, Grade B carbon steel/ alloy steel	SA-516, Grade 70 carbon steel
Nozzles/primary stay	SA-508, Class II alloy steel	SA-508, Class III alloy steel



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

REPLACEMENT
STEAM GENERATORS

Figure 2.2-1

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FIGURE 2.2-2 DELETED

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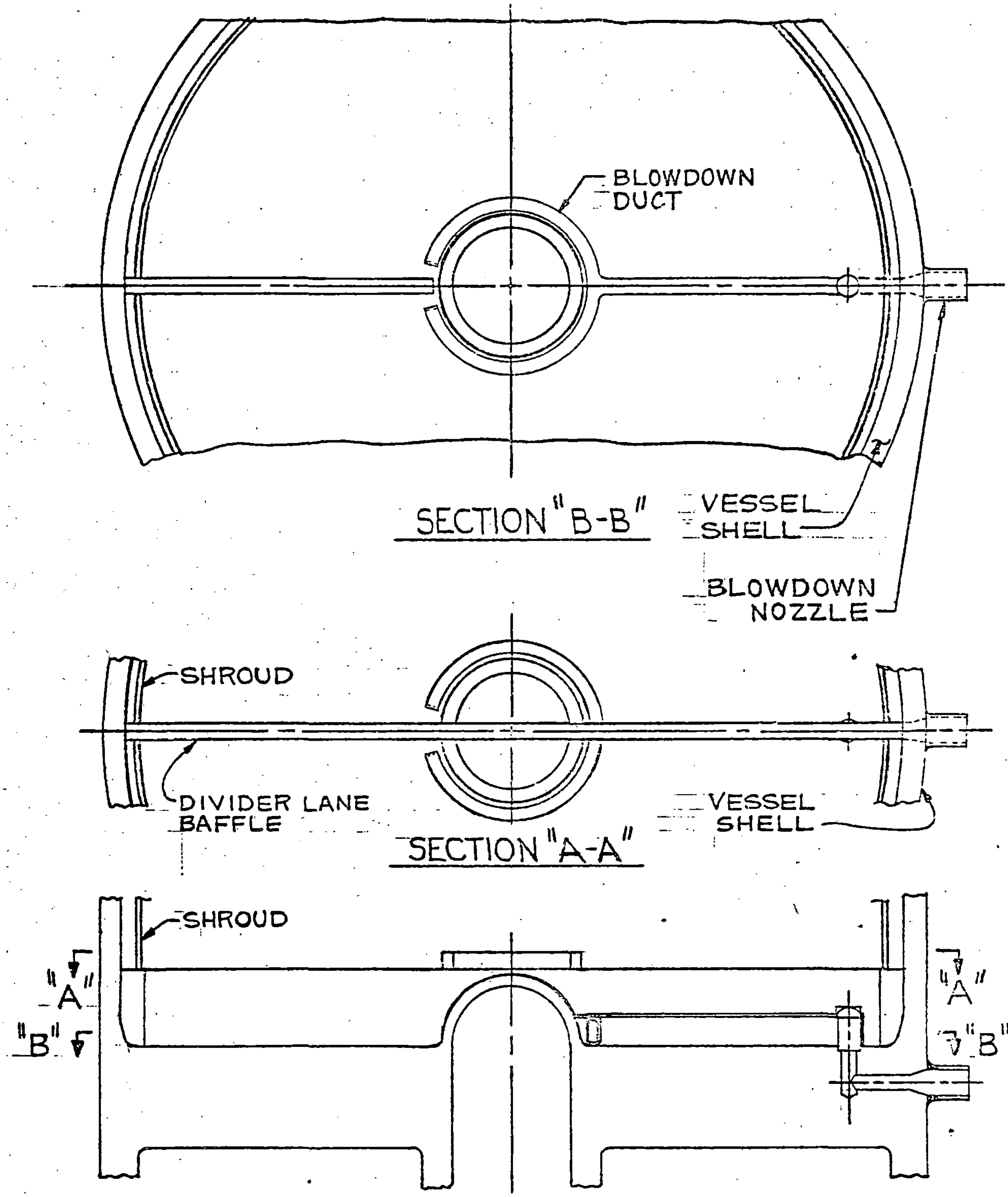
REV. 1

FIGURE 2.2-3 DELETED



SECONDARY BLOWDOWN ASSY

PALISADES PLANT STEAM GENERATOR REPAIR REPORT
STEAM GENERATOR SECONDARY BLOWDOWN ASSEMBLY Figure 2.2-4

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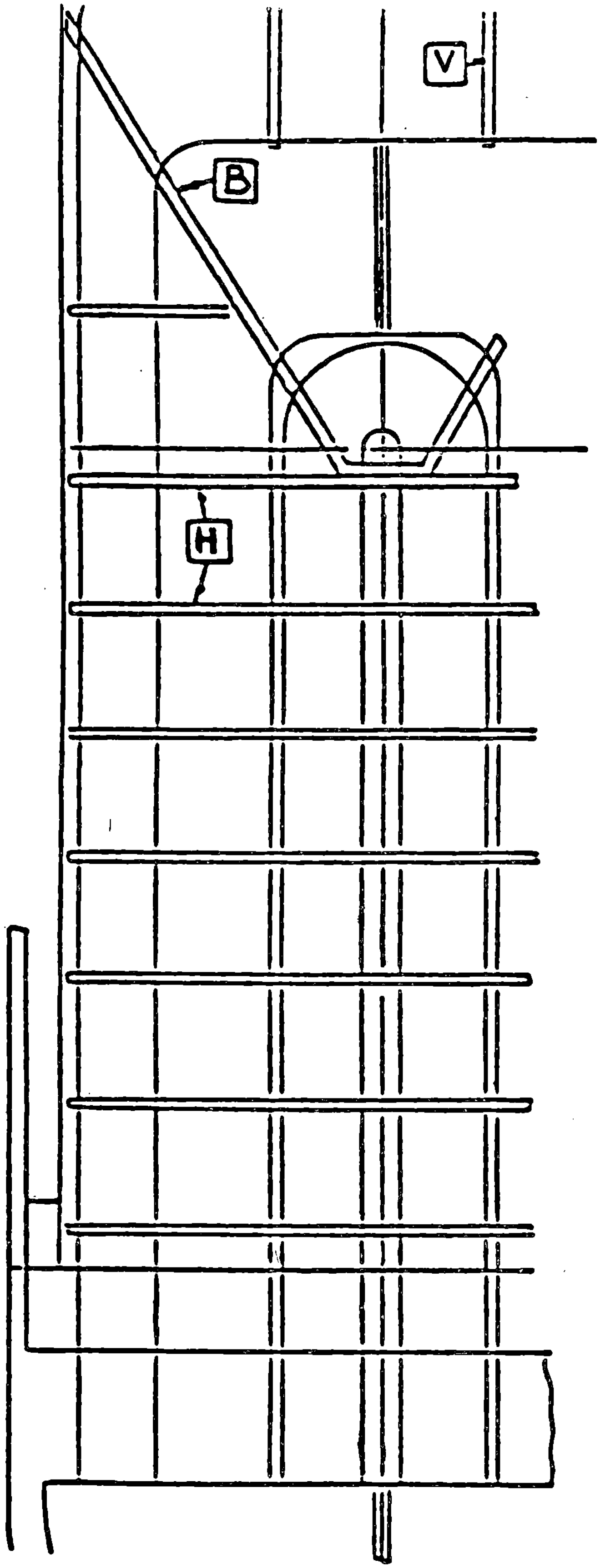


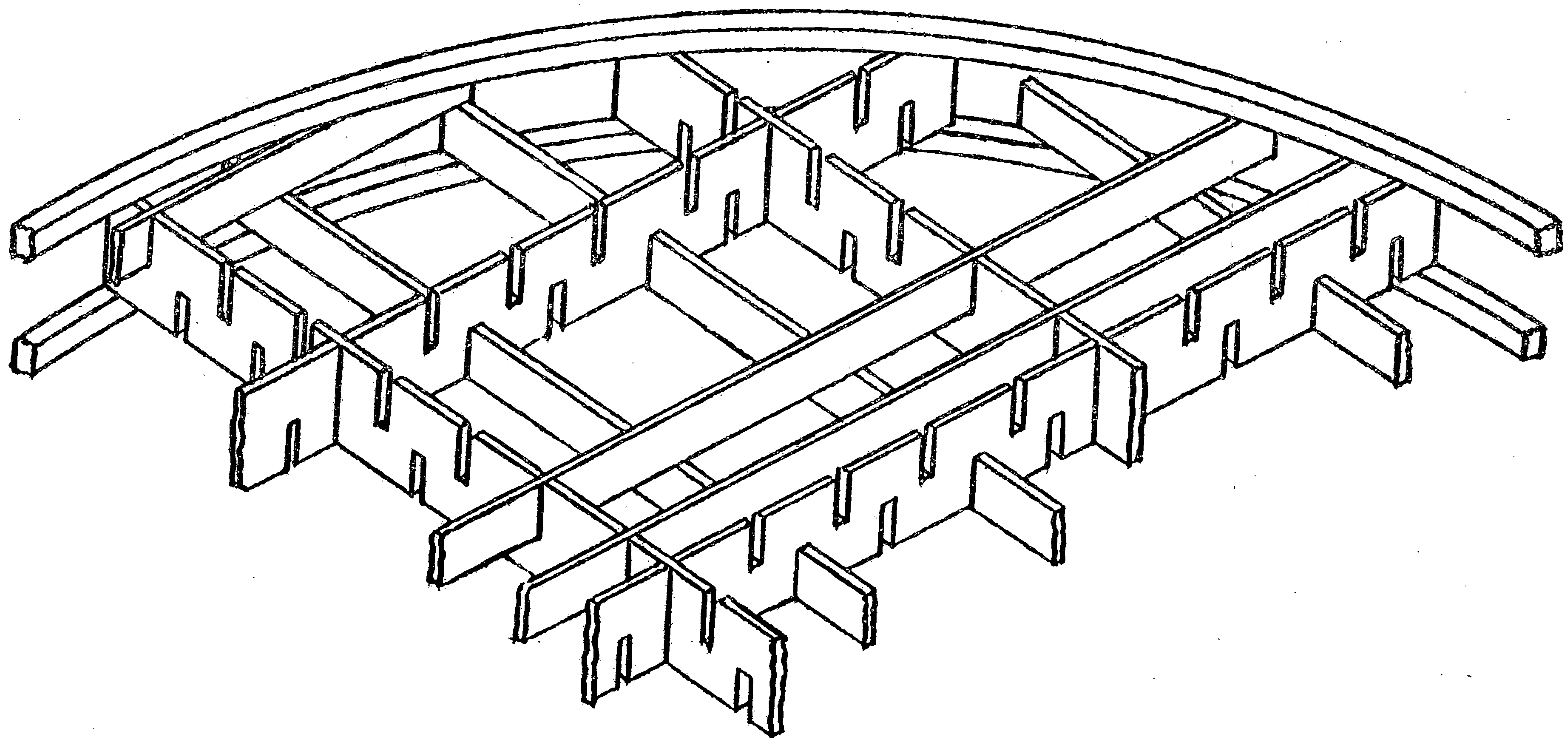
SUPPORT TYPES

H HORIZONTAL GRID (EGGCRATE)
V VERTICAL GRID
B BATWING STRIPS

PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

TUBE SUPPORT TYPES

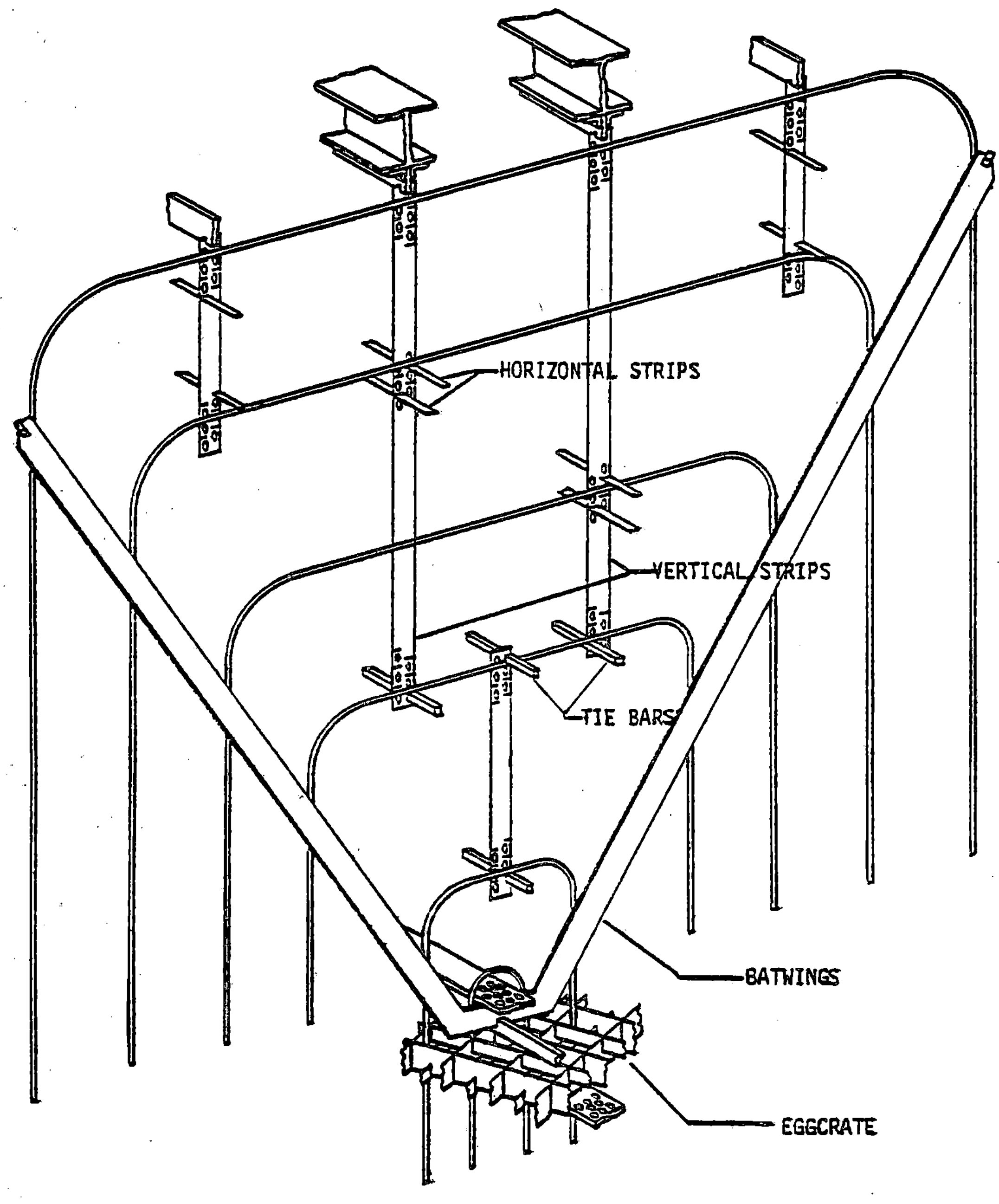
Figure 2.2-5



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

EGGCRATE ASSEMBLY

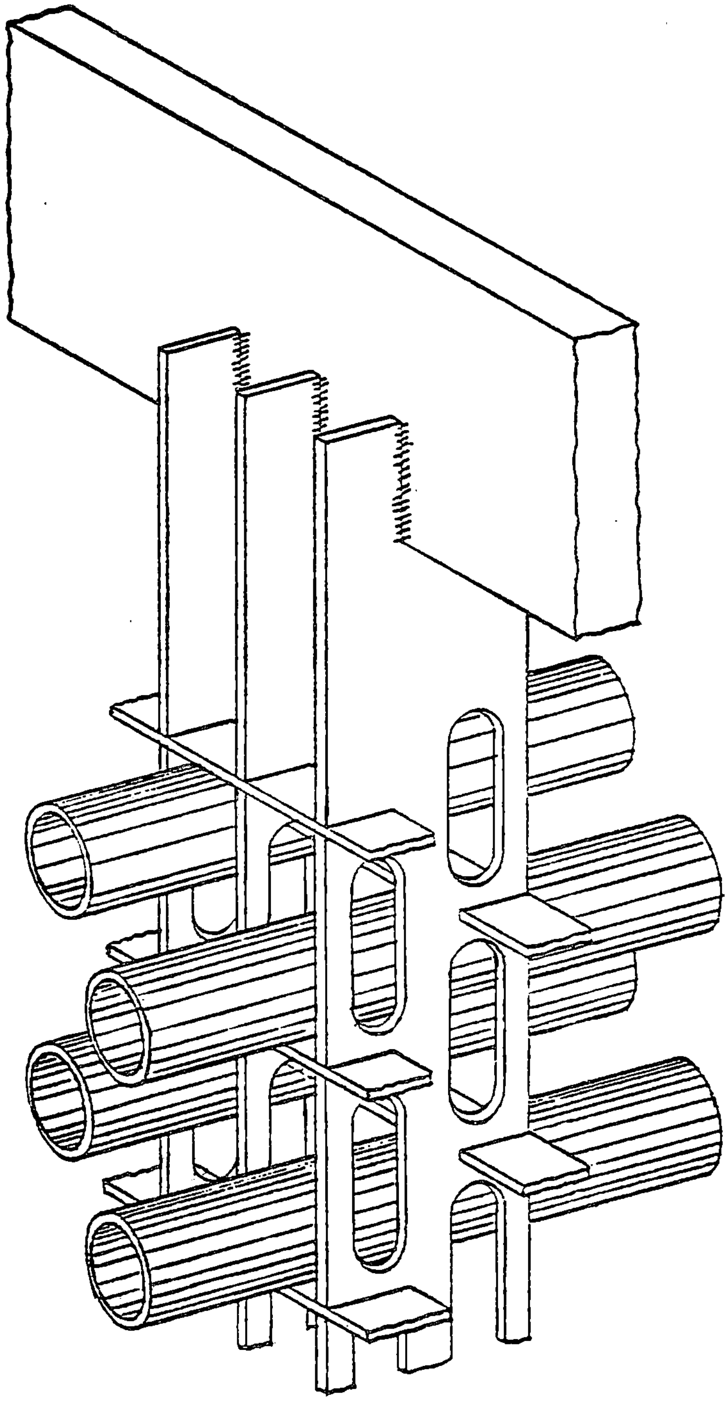
Figure 2.2-6



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

BEND REGION TUBE SUPPORTS

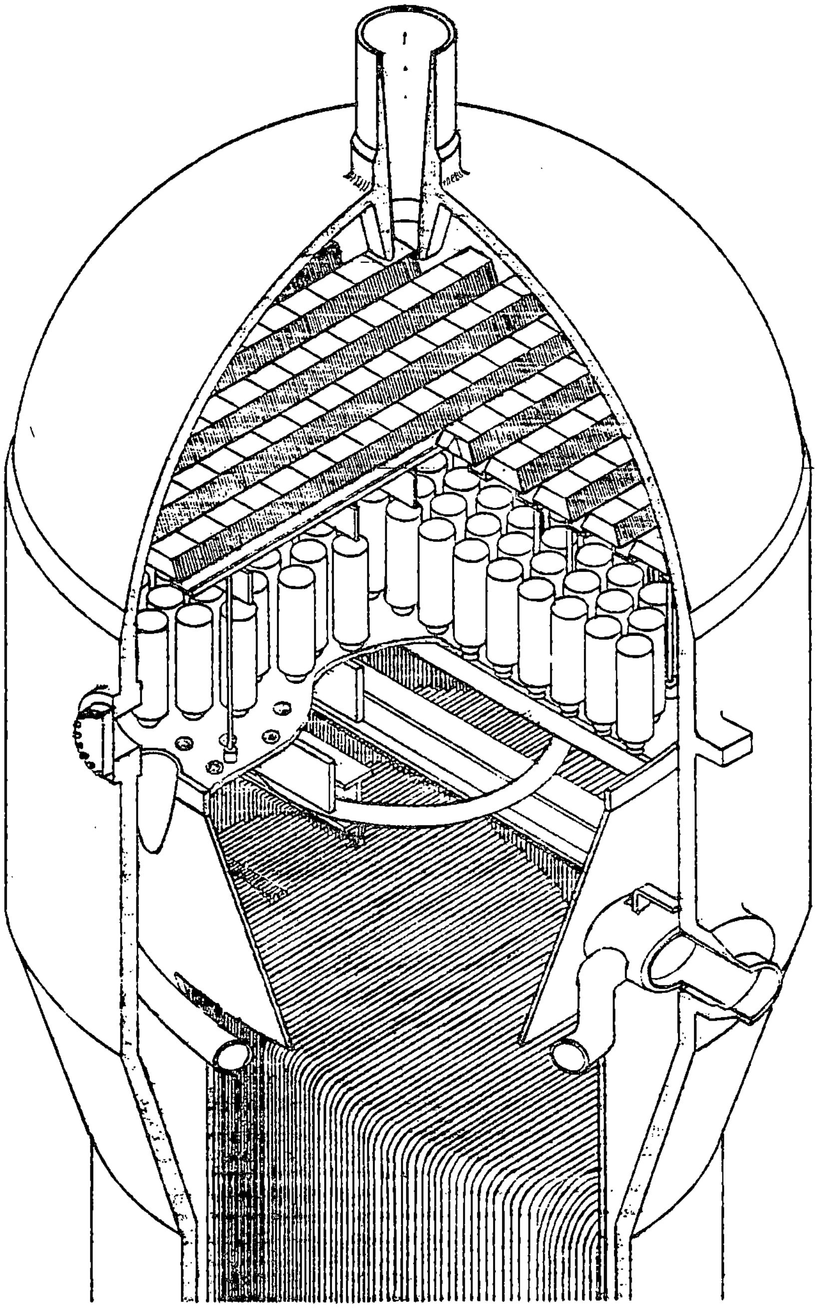
Figure 2.2-7



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

TUBE SUPPORT

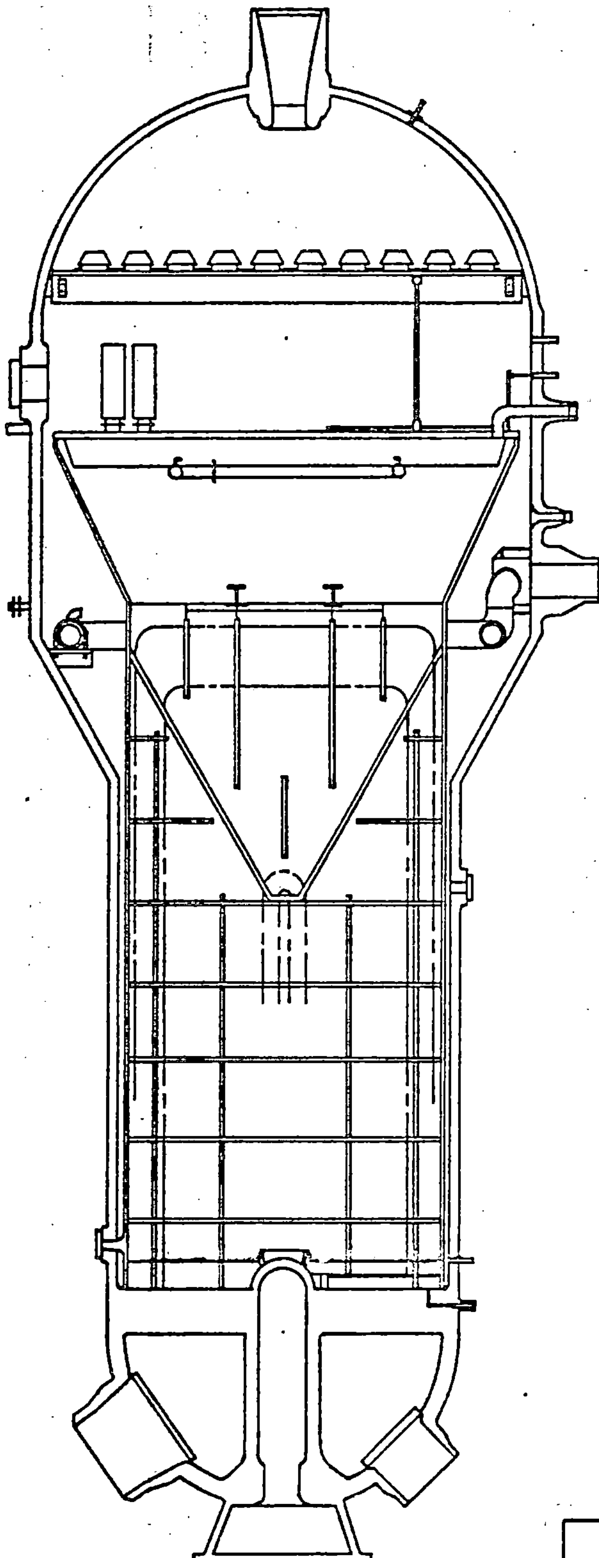
Figure 2.2-8



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

UPPER ASSEMBLY

Figure 2.2-9



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STEAM GENERATOR
FLOW RESTRICTOR NOZZLE

Figure 2.2-10

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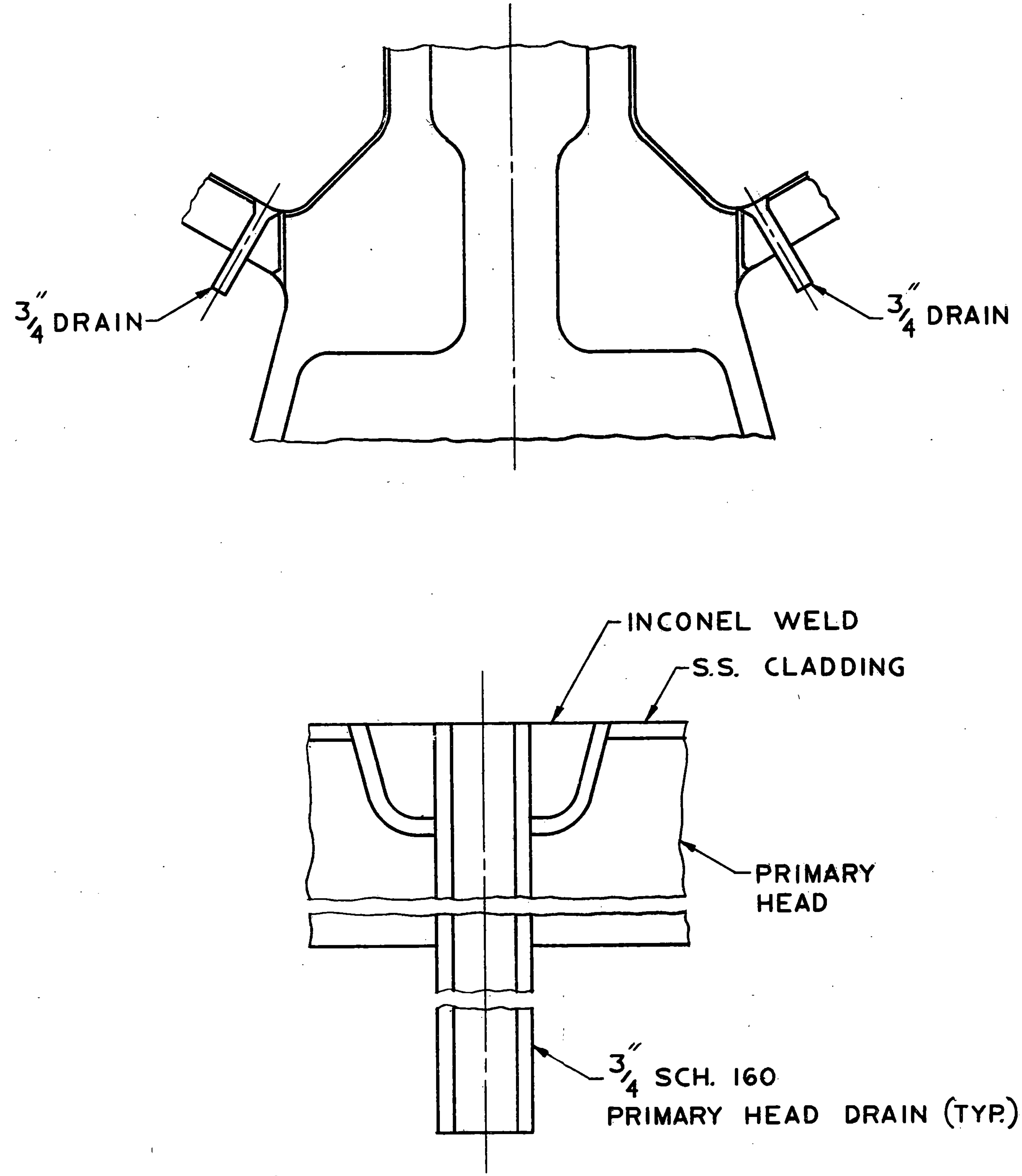


PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

PRIMARY HEAD DRAINS

Figure 2.2-11

3.0 BALANCE-OF-PLANT SYSTEM MODIFICATIONS

3.1 BLOWDOWN SYSTEM

The existing steam generator blowdown system is designed for a maximum capability of 100,000 lb/hr. Each existing steam generator has two blowdown connections, one for bottom blowdown and one for surface blowdown. Controls are provided to discharge both the surface and bottom blowdown to the flash tank for further processing in the blowdown system. Figure 3.1-1 shows a schematic representation of the existing blowdown system.

The new steam generator design utilizes only bottom blowdown; a 6-inch diameter nozzle is provided for this purpose (See Section 2.2.1.3). The current intent is that the existing piping for the bottom blowdown will be connected to the new steam generator nozzle, and the present system blowdown capability will be maintained. The modified blowdown system is shown in Figure 3.1-2. The blowdown system is a nonsafety system, except the portion from the steam generator to the second isolation valve (CV-0771 for steam generator E-50A and CV-0770 for steam generator E-50B), which consists of seismic Catagory 1 and ASME Code Section III, Class 2 piping and valves.

3.2 RECIRCULATION SYSTEM

The existing steam generator recirculation system is used as needed to maintain appropriate water chemistry while the steam generators are in wet layup or similar conditions. The system takes its suction from the bottom blowdown and discharges into the steam generator through the surface blowdown connections. The piping for the recirculation system is integrated with the blowdown system outside the containment. The steam generator blowdown pumps are used for maintaining the recirculation flow through the steam generator internals, and have a circulating capability of about 100 gpm. Figure 3.1-1 shows the existing recirculation system.

In the new steam generator, a 6-inch nominal pipe nozzle is provided in the upper steam drum shell for steam generator recirculation and other purposes (See Section 2.2.2.5). The existing recirculation piping will be connected to the new steam generator recirculation nozzle for discharging water into the steam generator. The present connection for the recirculation suction line on the bottom blowdown line will be maintained. Figure 3.1-2 shows the piping arrangement for the modified recirculation system for the new steam generator. An additional pump will be provided in parallel with the existing pumps to increase the recirculation flow capability to approximately 150 gpm, which would provide a 6.4 hour steam generator turnover time. The recirculation system is a nonsafety system.

3.3 SAMPLING SYSTEM

Figure 3.3-1 indicates the parameters being sampled for the existing steam generator at the turbine analyzer panel. These samples are drawn from the secondary side of the steam generator for monitoring the water chemistry. Additions to the existing secondary sampling system will be made to increase the capability of sampling the steam generators. The additional sample points will utilize the new steam generator sampling nozzles (See Section 2.2.1.8) and will continuously monitor pH, conductivity (specific and cation), and sodium. The modifications to the sampling systems are shown in Figure 3.3-2. The piping from the steam generator sampling nozzles will pass through the containment and will be provided with automatically controlled containment isolation valves. The sampling system is a nonsafety system, except the portion from the steam generator to the containment isolation valve, which consists of Seismic Catagory 1 and ASME Code Section III, Class 2 piping and valves.

3.4 PRIMARY HEAD DRAINS

The existing steam generators do not have primary head drains to enable complete draining before entry for maintenance activities. The new steam generators have two 3/4-inch drain nozzles located in each primary head as described in Section 2.2.2.4. Each drain will be double valved as close as is practical to the steam generator. The drains will be connected to the drain collection header, which discharges into the primary drain tank as shown on Figure 3.4-1. The portion of the primary head drain system from the steam generators to the second isolation valves consists of seismic Category 1 and ASME Code Section III, Class 1 piping and valves. The remainder of the system will be either Class 3 or nonsafety, with a transition to Class 3 at the drain header.

3.5 WIDE RANGE LEVEL INDICATION

A differential pressure type level transmitter will be added to each new steam generator to provide steam generator wide range level indication of about 44 feet. The top head of the new steam generators will have a 1-inch nozzle at el 671', which will be used for the low-pressure sensing line connection. The high-pressure sensing line will be connected to pressure taps at el 627' (see Figure 3.5-1). With this addition, an operator can determine the water level in the secondary side of each steam generator during wet layup (or similar operations) beyond the range measurable with the present level indicators (about 15 feet).

The new level indicating system is functionally independent, both electrically and mechanically, of any safety-related systems. The sensing lines will be in accordance with ASME Code Section III, Class 2 and seismic Category I classifications. The transmitter will be located in a low radiation zone. The transmitter output will electrically connect to a new indicator in the main control room and will not be used to automatically initiate or terminate any action.

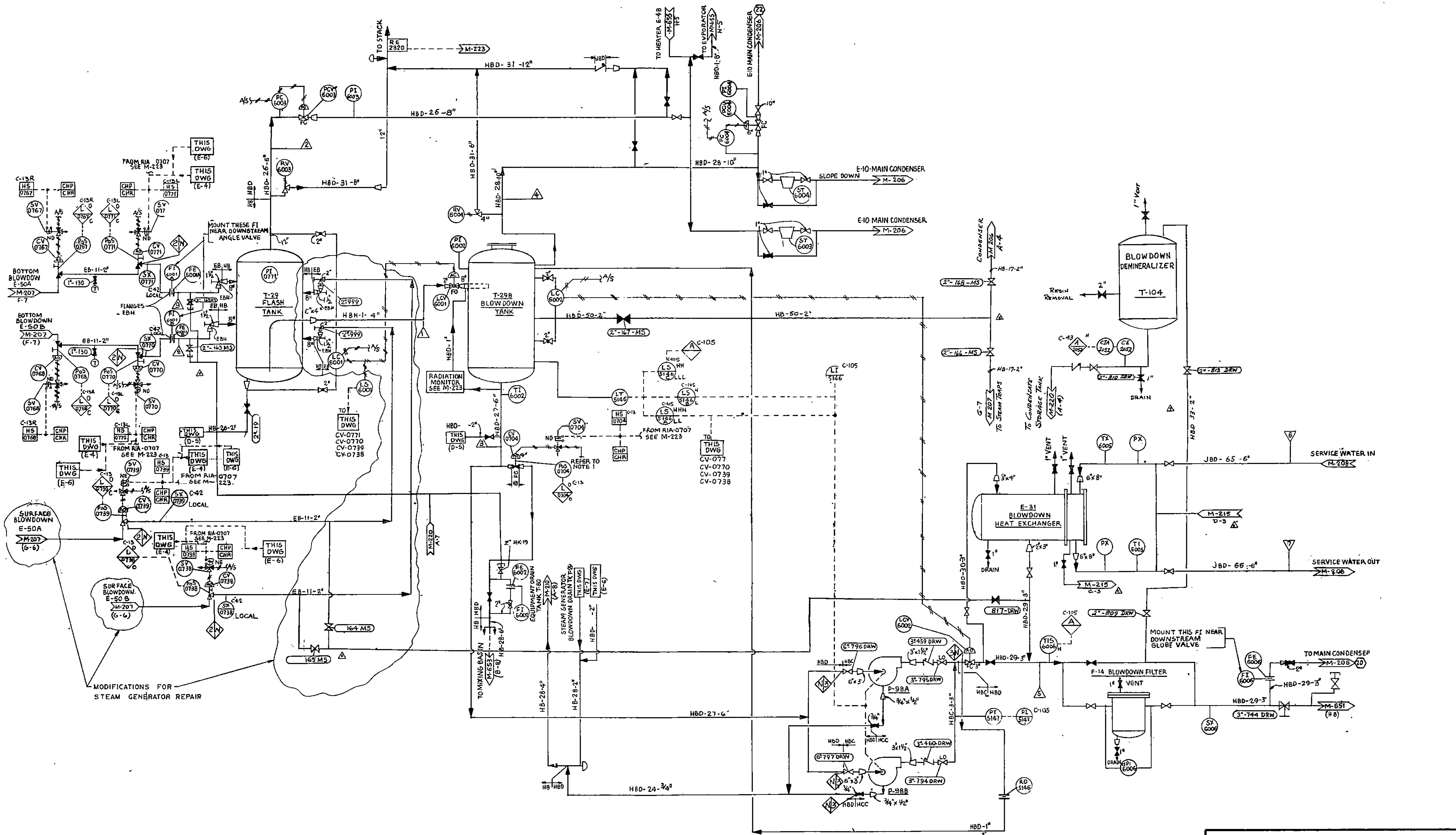
3.6 MAIN STEAM ISOLATION VALVE CLOSURE SIGNAL

The existing circuitry provides for closure of the Main Steam Isolation Valves based on low steam generator pressure. This closure signal is provided to protect against excessively high releases of steam to the containment as a result of a steam line break.

The replacement steam generators have steam nozzle flow restrictors to restrict the blowdown rate following a steam line break. The effect of the flow restrictors is to reduce the rate of steam generator pressure change during blowdown. Hence, following a design basis steam line break with the replacement steam generators, the high containment pressure trip setpoint will be reached before the low steam generator pressure trip is reached.

The Main Steam Isolation Valve Closure signal will be modified to be actuated from high containment pressure as well as low steam generator pressure. This will reduce the mass/energy release following a steam line break and result in lower containment peak pressures.

The containment pressure instrumentation and circuitry are safety grade.

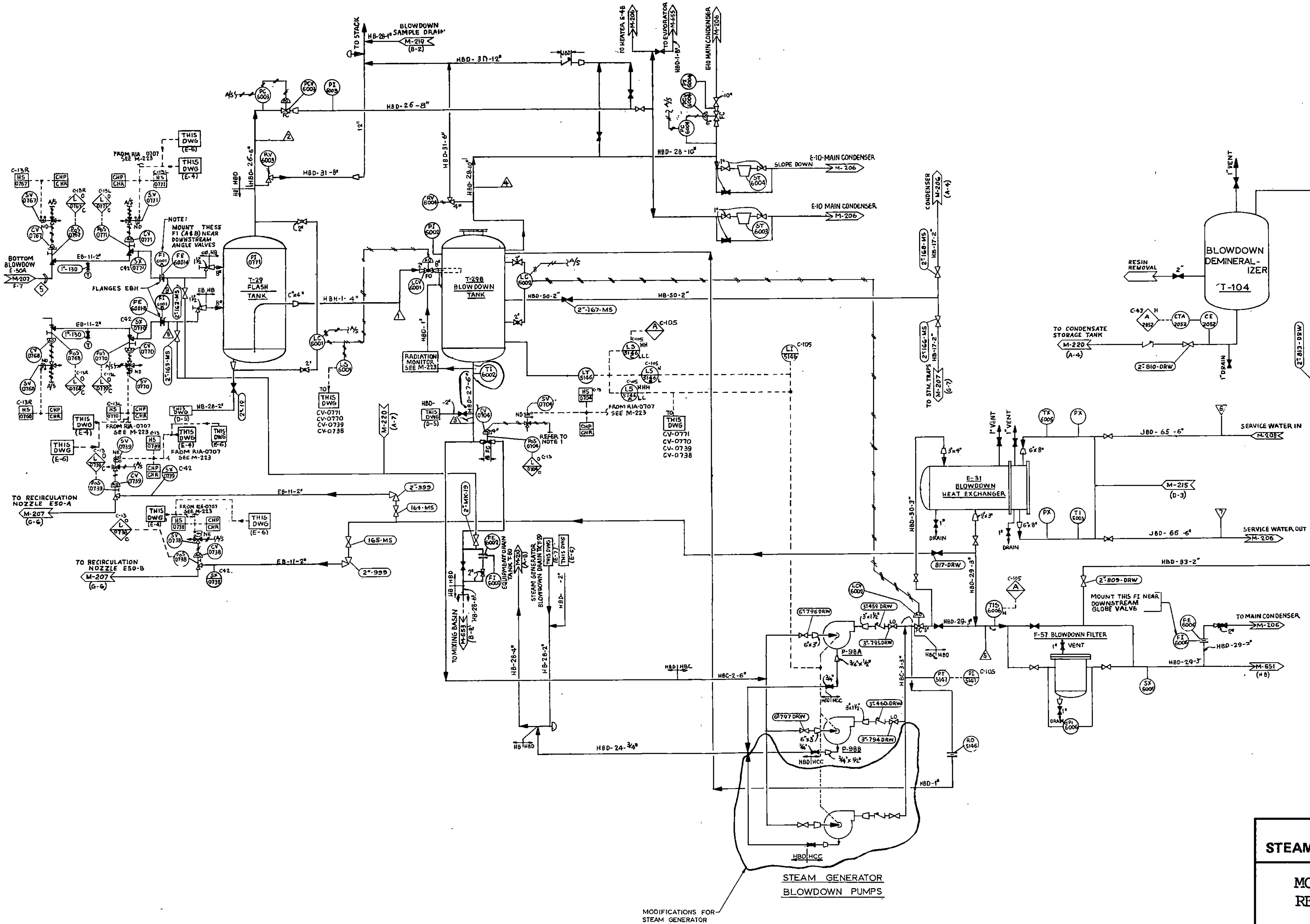


STEAM GENERATOR BLOWDOWN PUMPS

PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

EXISTING BLOWDOWN AND
RECIRCULATION SYSTEM

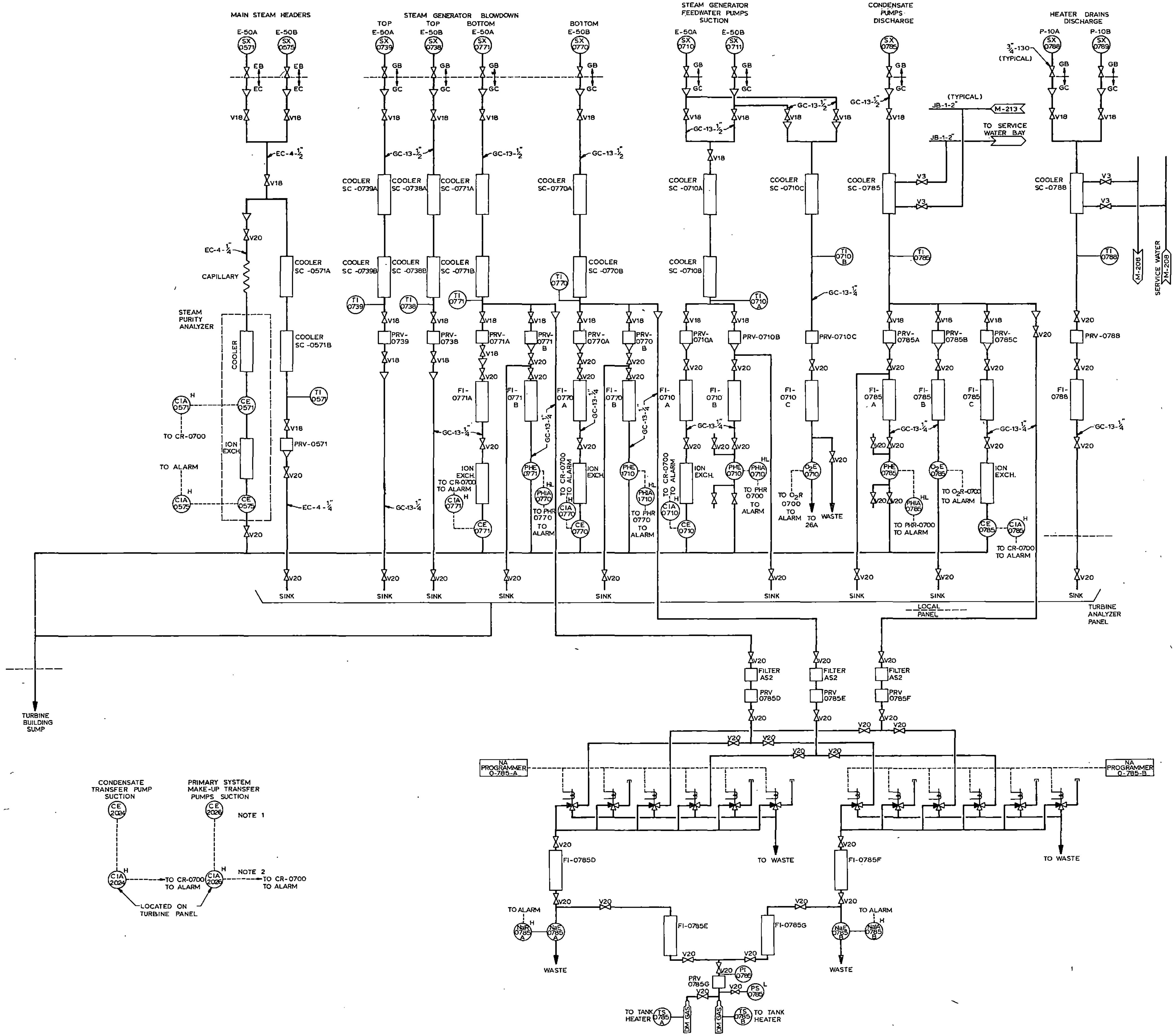
Figure 3.1-1



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

MODIFIED BLOWDOWN AND RECIRCULATION SYSTEM

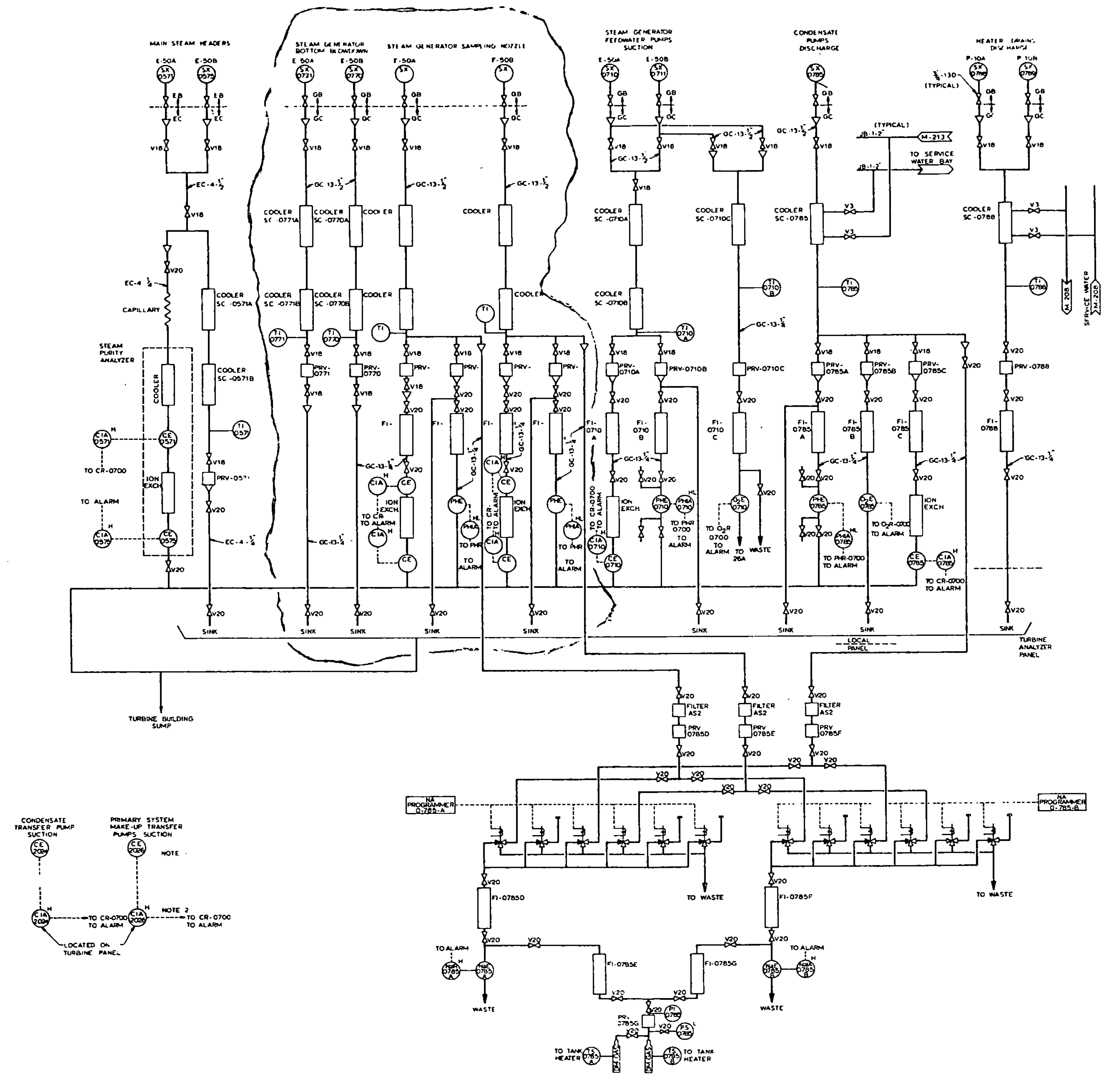
Figure 3.1-2



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

EXISTING TURBINE ANALYZER PANEL FOR SAMPLING SYSTEM

Figure 3.3-1



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

MODIFIED TURBINE ANALYZER PANEL FOR SAMPLING SYSTEM

Figure 3.3-2

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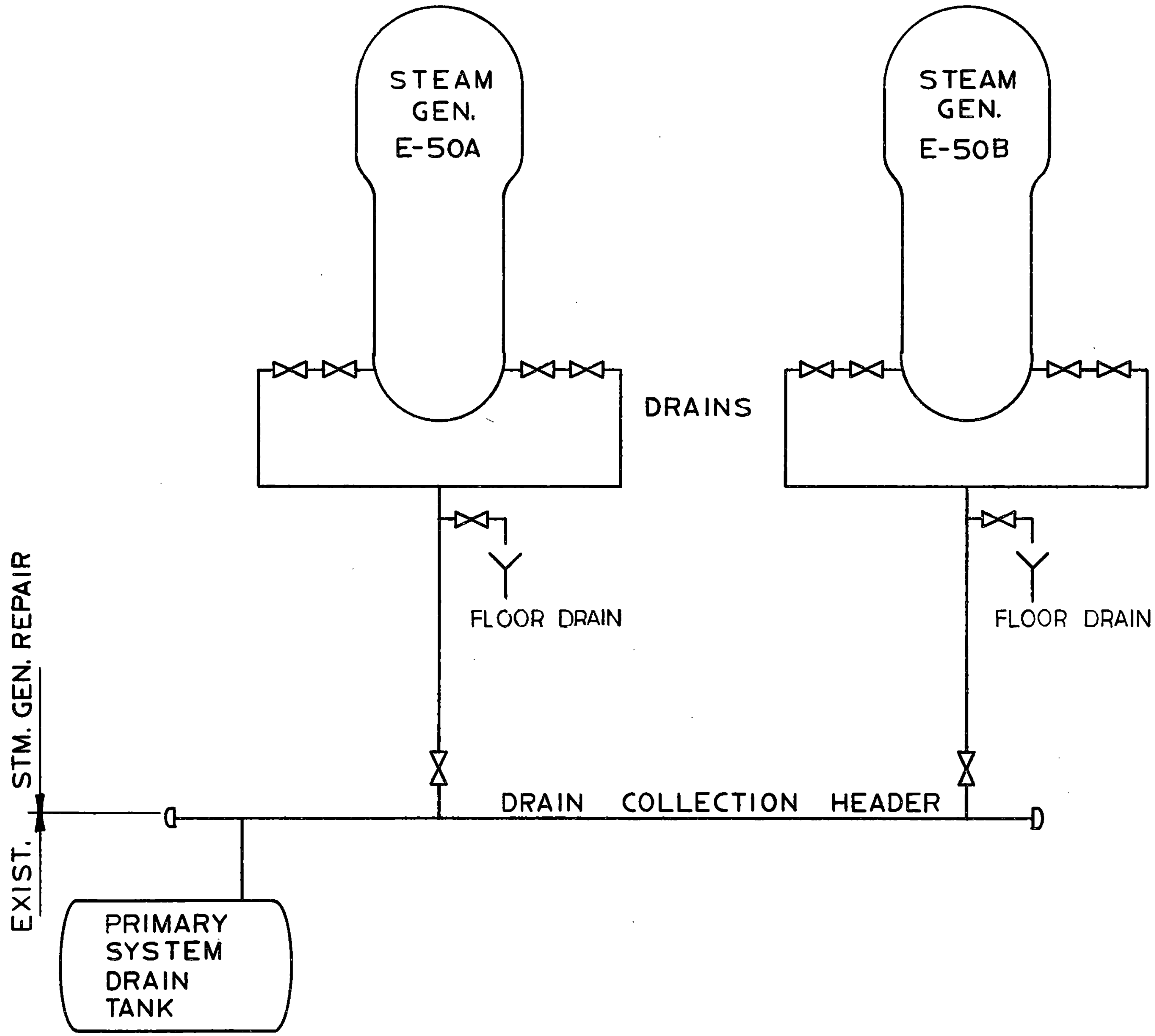
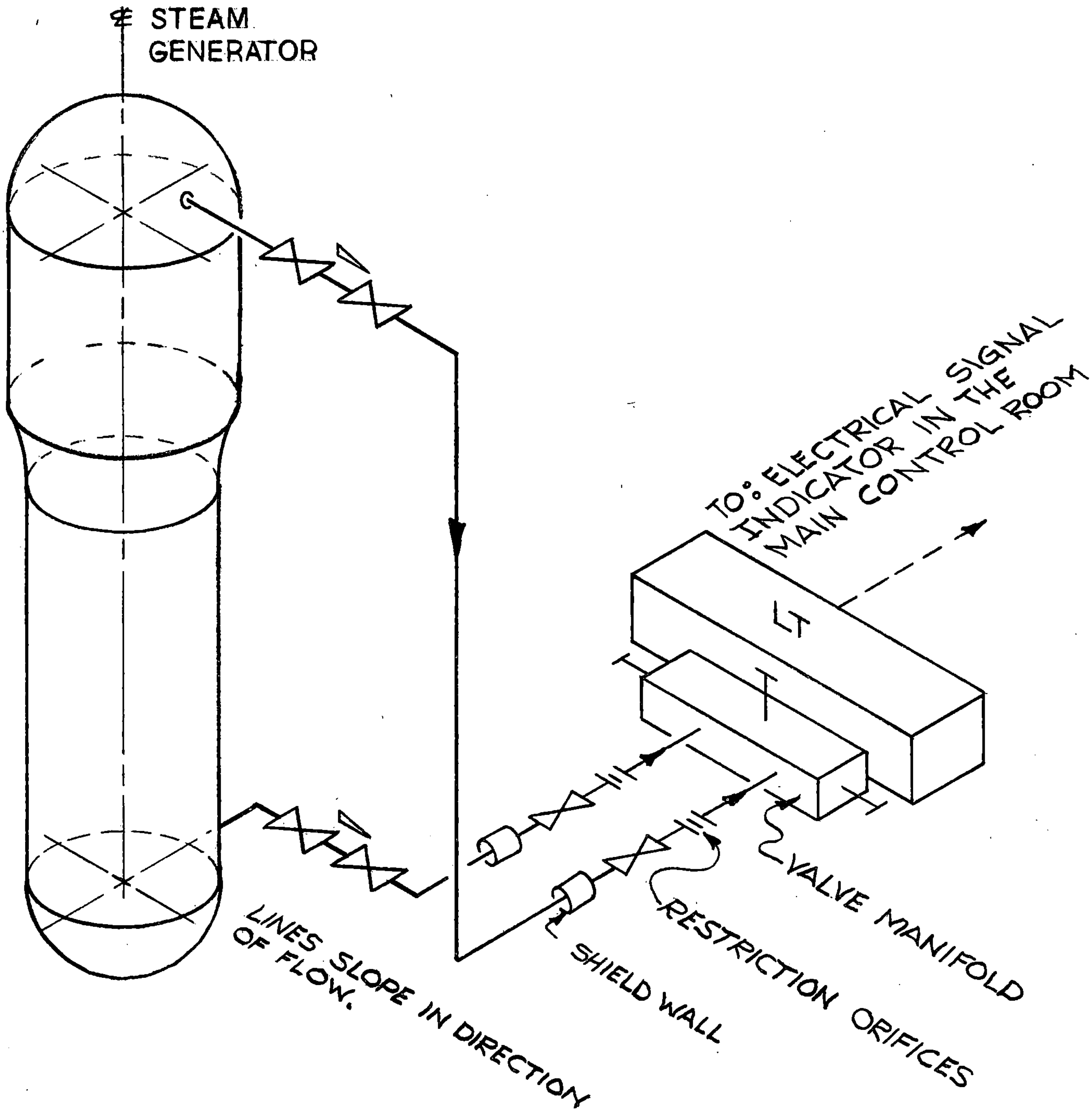


PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

PRIMARY HEAD DRAIN SYSTEM

Figure 3.4-1



PALISADES PLANT STEAM GENERATOR REPAIR REPORT
WIDE RANGE LEVEL TRANSMITTER Figure 3.5-1

4.0 REPLACEMENT PROGRAM AND PROCEDURES

This section discusses the engineering evaluation of various activities required to implement the steam generator repair. It should be noted that implementation methods and procedures may vary from that described below as engineering is finalized. The methods below are provided to demonstrate feasibility of implementation.

4.1 CONSTRUCTION CONSIDERATIONS

4.1.1 SITE PREPARATION

The plant site will be prepared as necessary for the following activities pertaining to the replacement of the existing two steam generators. As described, none of these activities will impinge on safety-related underground piping, conduits, or electrical duct banks.

a. Receipt of Two New Steam Generators

The new steam generators will be offloaded using a newly constructed barge slip which is described in Section 4.1.1.1.

b. Storage of New Steam Generators

If the new steam generators are not installed upon arrival at the site, they will be stored in compliance with the storage criteria listed in Section 2.4.

c. Preparation of Outside Area Adjacent to Containment Building Construction Opening

As shown in Figure 4.1-1, the foundations will be installed in this area to support the rigging equipment used for transporting the steam generators through the containment wall construction opening. The rigging equipment description is given in Section 4.1.2.2. In order to accommodate the rigging equipment and to provide adequate space for handling the steam generators, an additional area at el 625'-0" will be built-up with a structural backfill and held by a retaining wall. The containment building wall has been evaluated for the additional loads imposed due to the rigging equipment that is loaded with a steam generator to ensure that there are no adverse structural effects.

d. Transportation of Steam Generators

The steam generators will be transported between the steam generator storage area and the containment building along the access route as shown in Figure 4.1-1. The plant access road will be widened to accommodate the width required for the steam generator transporters. Borings will be taken along the access road to confirm that the soil bearing capacity is adequate before steam generator transportation. There are no safety-related underground pipes, cables, etc, involved along the route, except for a fire protection line which is adequately protected by a minimum overburden of 5'-6".

e. Temporary Construction Facilities

Adequate construction facilities will be provided as necessary for the labor force and material storage. Controlled access areas will be designated. Such facilities are shown in Figure 4.3-1 and are described in Section 4.1.1.3. Supplemental access control is described in Section 4.3.1. Temporary construction fences will be installed to delineate construction areas and to provide construction security.

4.1.1.1 Barge Slip Facilities

A temporary barge slip will be constructed within the existing plant area as shown in Figure 4.1-2. See Section 7.2.3 for dredging and construction details.

A hopper type barge will be used to transport the new steam generators to the Palisades Plant jobsite. As presently contemplated, the steam generators will be offloaded from the barge by lifting them with a mobile jacking frame. The jacking frame loaded with a steam generator will be brought to the dock at el 588'-0" ($\pm$). At this point the steam generators will be transferred to the waiting crawler transporters. The new steam generators will then be transported to the steam generator storage area pending installation.

4.1.1.2 Steam Generator Storage Facilities

The storage requirements are different for the new and old steam generators.

4.1.1.2.1 Storage of New Steam Generators

If storage of the new steam generators is needed, the storage facilities will conform to the criteria set forth in Section 2.4.

4.1.1.2.2 Storage of Old Steam Generators

The requirements and design criteria for a storage building for the old steam generators are described in Section 4.4.2. If this building is necessary, adequate biological shielding will be provided in the form of 18-inch thick normal weight (150 lb/ft^3) concrete walls all around its periphery (see Section 4.4.6). The steam generator storage building will have a removable metal deck roof to facilitate rigging and handling of the steam generators (see Figures 4.1-3 and 4.1-4). This building is located in a plant area where there are no safety-related structures or equipment.

4.1.1.3 Temporary Construction Facilities

Existing plant warehouse and offices will be utilized, as far as possible, to support the construction force and store equipment and tools during the steam generator repair effort. Exact areas and details relating to temporary construction facilities will be developed before the construction phase.

4.1.1.4 Containment Preparation

4.1.1.4.1 Defueling of the Reactor

All fuel assemblies will be removed from the reactor and stored in the spent fuel pool before the commencement of the repair activities that could affect plant safety.

4.1.1.4.2 Equipment and Material Modifications

Equipment and material will be temporarily relocated to provide rigging clearances as discussed in Section 4.2. Rigging equipment inside the containment is discussed in Section 4.1.2.

4.1.1.4.3 Laydown Areas

Laydown areas for the major equipment to be relocated inside the containment will be provided as shown in Figure 4.1-5 as follows:

a. Reactor Vessel Thermal Shield

   The reactor vessel thermal shield which is presently stored at el 649'-0" will be removed from the containment through the construction opening.

b. Main Steam Line

   The laydown area for the main steam lines will be located on the northwest side of the 649-foot level.

c. Missile Shields

   The missile shields over the control rod drive mechanisms may be stored in their normal laydown areas on the 649-foot level or left in place over the reactor vessel.

It is noted that these laydown areas are provided to demonstrate laydown feasibility; the final design may alter the configuration and scope.

4.1.1.4.4 HVAC During Steam Generator Repair Program

During the steam generator repair program, the containment will be kept heated, ventilated, and air conditioned as required by seasonal conditions. The existing air purge supply and exhaust system, containment air cooler recirculation fans, and unit heaters will be kept operational. Additional construction-related ventilation equipment, including portable fans, hoods and filters, will be used to remove fumes associated with welding and cutting operations. A temporary construction covering will be provided over the construction opening to reduce the infiltration of dust, sand, and water into the containment. Control of airborne radioactivity during the steam generator repair program is discussed in Section 4.3.3.

4.1.1.4.5 Floor Drains and Sumps

Floor drains and sumps will be protected during the repair outage to prevent their becoming clogged with construction debris. On completion of the repair the drains will be inspected and cleaned as required.

4.1.2 RIGGING

A rigging scheme has been developed to replace the steam generators which would avoid hazard to the fuel stored in the spent fuel pool. This scheme will also not adversely affect safety-related equipment, systems, and structures necessary to maintain the spent fuel pool cooling and makeup capability. This rigging scheme utilizes existing containment building structural supports and is described in the following sections. The rigging scheme is subject to improvement or modification as the design progresses and/or to suit the equipment requirements of the selected rigging subcontractor; however, future modifications of the rigging scheme will not adversely affect the spent fuel pool cooling and makeup capability or the safety-related structures or equipment.

4.1.2.1 Rigging Inside Containment

As presently contemplated, the rigging scheme developed for inside the containment to replace the steam generators is shown in Figures 4.1-6 through 4.1-10. The major pieces of the rigging equipment, as discussed in Section 4.1.2.3, will be brought into the containment through the construction opening. The center pole section is erected on a steel platform which is supported on the biological shield walls of the reactor cavity at el 649'-0". The rotating temporary semi-gantry crane is supported by the center pole section and the existing polar crane rail. The rotating temporary semi-gantry crane header beam with its lifting hardware is positioned over the old steam generator, and the lifting collars will be installed on the steam generator's trunnions.

The steam generator is freed from all piping and restraints as discussed in Sections 4.2.4 and 4.2.5. The steam generator is then lifted clear of all obstruction to approximately el 660'-6" as shown in Figure 4.1-8. The semi-gantry crane is rotated about the center line of the containment building approximately 90 degrees until the steam generator is located directly over the radial center line of the construction opening. The steam generator is rotated approximately 47 degrees about its vertical axis and then lowered horizontally onto the tilt-up sled. The rigging tilt-up hardware is attached to the lower nozzle of the steam generator.

The tilt-up sled is assembled with a vertical yoke which pins up to the ears on the steam generator's tilt-up hardware. As shown in Detail 2 of Figure 4.1-10, the tilting sleds are equipped with machinery dollies and travel on a runway system. As the steam generator is lowered, the tilt-up sled is progressively winched out of the construction opening along the runway as shown in the same figure.

The steam generator down-ending procedure sequence is shown in Figures 4.1-9 and 4.1-10. In Sequence 1, two additional tilting sleds are positioned on the runway behind the No. 1 tilt-up sled. In Sequence 2, the lower section of the steam generator has been seated on the No. 2 sled and the lower nozzle tilting mechanism is free of the No. 1 sled. In Sequence 3, the lowering procedure has been completed and the large diameter portion of the steam generator now rests on the self-adjusting saddles of the No. 3 sled.

At this point the steam generator is supported completely by the runway system and portion of the elevator beam outside the containment. The steam generator is freed from the semi-gantry crane hoists by removing the lifting collars from the steam generator trunnions, and the No. 1 tilt-up sled is also removed from the runway. Then the steam generator is slowly winched out of the containment onto the elevator beam portion of the runway. From this point the steam generator is lowered by the jacking frame onto transporters as described in Section 4.1.2.2 and moved to the steam generator storage laydown area.

Removal of the old steam generators will serve to satisfy the load test requirements of rigging equipment inside the containment building. The old steam generators will be loaded to achieve the required test load.

Installation of the new steam generators will follow a reverse procedure similar to that given above for the removal of old steam generators.

4.1.2.2 Rigging Outside Containment

The rigging scheme outside the containment as presently being considered is shown in Sequences 1 through 4 of Figure 4.1-11. The old steam generator is rolled out on the tilting sleds along the runway as described in Section 4.1.2.1. The old steam generator is positioned on the elevator beams supported by the jacking frames so that the lift points of the steam generator align with the jacking bars of the jacking frame as shown in Sequence 1. Once in position, the steam generator, together with elevator beam and sleds, is secured with lifting slings to the jacking system. Holddowns between steam generator and sleds and between sleds and elevator beams are attached, securing the steam generator in place as shown in Sequence 2. The lifting slings are raised until snug, allowing the elevator beams to be disconnected from the runway.

In Sequences 3 and 4 the steam generators, sleds, and elevator beam assembly is then lowered and secured into the transporters that have been positioned under the jacking frame assembly. The slings attaching the steam generator assembly to the jacking system are removed. The steam generator is then ready for transport to the steam generator storage laydown area as described in Section 4.1.2.4. Installation into the containment of the new steam generators will follow a reverse procedure similar to that given above for removal of the old steam generators.

4.1.2.3 Rigging Equipment

As presently contemplated and as shown in Figures 4.1-6 through 4.1-17, the following major pieces of rigging equipment will be utilized for handling, transporting, and removing/installing the steam generators.

4.1.2.3.1 Inside Containment Building

a. Rotating semi-gantry crane consisting of pole section, box girders, header beam assembly, sill beams mounted on trucks, load blocks and hoisting lines, including swivel spreader beam

b. One downending and 2 tilting sleds mounted on machinery dollies

c. Runway system

4.1.2.3.2 Outside Containment Building

a. Jacking system and elevator beams, including sleds

b. Transporter system

4.1.2.3.3 At Barge Slip Area

a. Travelling jacking system, including elevator beam

b. Transporter system

4.1.2.3.4 At Steam Generator Storage Area

a. Jacking frame, including elevator beams

b. Transporter system

4.1.2.3.5 At Access Road

a. Transporter system

b. Tilting sleds

4.1.2.4 Steam Generator Transportation

Before transporting the steam generators between the containment building and the steam generator storage area, the access route will be completely surveyed and all obstructions removed. Soil borings will be taken to confirm that the access path has sufficient strength. The maximum soil bearing load under the crawler tracks is expected to be 5 ksf. This access path has been used frequently by heavy construction traffic for miscellaneous work within the plant area. The maximum grade is 7% with a minimum turning radius of 42'-0" feet along the access route.

After removal from the containment building, the old steam generator will be moved by transporters to its laydown area near the steam generator storage area. The path taken by the transporters is shown in Figure 4.1-1. The new steam generator will be loaded on transporters and then transported along the north access road to the containment building for installation. The second old and new steam generators will be transported in a similar manner.

The types of transporters that are presently being considered for transportation of steam generators are shown on Figures 4.1-13 and 4.1-14.

There are existing underground fire lines, storm drains, and culverts involved at different places along the route. Adequate earth cover or other temporary protection will be provided to ensure safe transportation of steam generators over these underground utilities.

4.1.2.5 Rigging Controls

Rigging operations and equipment associated with the steam generator repair program as described in Sections 4.1.2.1 and 4.1.2.2 will be closely monitored and prequalified to ensure the safe and efficient handling of steam generators. All rigging operations and procedures shall be developed and performed by experienced personnel. The rigging equipment will be load tested or prequalified in accordance with established industry codes and standards before the handling and installation of new steam generators.

Throughout every phase of the rigging project detailed inspection and work procedures shall be used to ensure the timely and proper performance of all the rigging operations. Stringent administrative controls will also be employed to ensure overall safety and to preclude damage to equipment or structures during transportation, handling, and installation/removal of steam generators. A scale model of the containment and rigging equipment as described in Section 4.9 will be utilized to further verify the rigging sequence pertaining to rigging operations inside the containment building.

4.1.3 RIGGING LOAD SUPPORTS

All design loads imposed by rigging equipment over existing containment building walls, floors, and other members are identified in Figure 4.1-18.

4.1.4 CONSTRUCTION-RELATED INCIDENTS

The following unlikely incidents have been postulated during handling of steam generators, including the load testing:

a. Dropping new steam generators while being offloaded from the barge at the barge slip area

b. Dropping old/new steam generators during transportation

c. Dropping old/new steam generators during rigging operations at the steam generator storage area

d. Dropping old/new steam generators during rigging operations adjacent to the containment building

e. Dropping old/new steam generators, rigging, or other equipment during construction activities inside the containment

The locations of hypothetical incidents a. through c. are physically far enough removed so as not to affect any safety-related structures or equipment. Hypothetical incidents d. and e. could result in damage to the containment structure; however, they would not present a hazard to fuel pool cooling and makeup capability. None of the fuel cooling and makeup system equipment is located inside the containment nor in areas which could be exposed to any of the postulated incidents. Since the reactor will be defueled, any damage to safety-related equipment inside the containment as a result of the unlikely drop of steam generator, rigging, or other equipment during construction activities will not involve any safety considerations relative to the safe shutdown condition of the plant. Even though the above postulated events are not likely to occur, precautions will be taken to further minimize any damage to the containment building or safety-related equipment. In addition to the rigging controls described in Section 4.1.2.5, construction equipment, such as cranes required for erection of rigging equipment outside the containment, will be positioned or rigged to preclude any possibility of their having a significant impact on Class I structures or safety-related equipment needed for fuel pool cooling.

4.1.5 CONTAINMENT STRUCTURAL CONSIDERATIONS

The new steam generators will be similar in size and weight to the existing steam generators. The small changes in the centers of gravity and weights (see Table 2.1-1) for the new steam generators have an insignificant effect on previously calculated loads; therefore, no modifications to the support system will be necessary. The only structural considerations involved are those related to the support of temporary rigging loads for the removal and replacement of the steam generators, the construction opening in the containment building wall, and the laydown areas required by the construction activities inside the containment; however, because of new elevations of main steam lines, the main steam line supports will be modified.

Preliminary structural analyses have been made for the containment shell and internals to ensure that structural integrity of the containment building will remain essentially the same as that of the existing containment structure. The design criteria for the containment building structural repairs will be based on the same criteria as outlined in Appendix B of the Palisades Plant FSAR for Class 1 structures.

After the steam generator repairs are completed, an integrated leak rate test will be performed in accordance with the Palisades Plant Technical Specifications to ensure the leaktight integrity of the containment building. A modified structural integrity test, similar to that described in the FSAR, will be performed for the area affected by the containment construction opening.

4.1.5.1 Construction Opening in Containment Wall

For purposes of rigging and handling the steam generators, a preliminary construction opening size was chosen as shown by dotted lines in Figures 4.1-19 and 4.1-20. The centerline of this opening is located at azimuth N 118° 40'. The bottom and top elevations are at el 650'-0" and 688'-0", respectively. The width of the opening is 37 feet along the centerline. The containment structural shell analysis has been based on this opening size; however, the rigging scheme presently under consideration permits an opening smaller in size than the preliminary opening originally selected. The configuration and size of the latter opening is shown by solid lines in Figures 4.1-19 and 4.1-20. The bottom and top elevations of the new opening are at el 649' and el 679', respectively. The width of the smaller opening is 32 feet along the horizontal centerline. Results of the shell analysis based on the larger preliminary opening will not be significantly altered by the smaller opening.

4.1.5.2 Tendon Detensioning and Removal

4.1.5.2.1 Criteria for Detensioning/Removing Tendons

Before the construction opening is made in the containment, the hoop and vertical tendons passing through the opening must be detensioned and removed. Moreover, additional vertical and hoop tendons must be detensioned in order to ensure (see Figure 4.1-20):

a. That flexural and membrane stresses everywhere in the containment shell shall be within the allowable limits throughout the construction period

b. That the prestress state at the opening area shall be approximately zero

c. That the prestress distribution around the opening shall be small after the opening is made

4.1.5.2.2 Method for Detensioning/Removing Tendons

Based upon the above considerations, the following scheme for tendon detensioning and removal has been developed.

a. Detensioning and/or removing vertical tendons between azimuth N 48° 40' and N 188° 40' (70 vertical tendons)

b. Detensioning and/or removing hoop tendons between buttresses located at N 85° and N 145° and between elevation 644' and 684' (54 hoop tendons)

Vertical and hoop tendon detensioning sequences will start from the center and proceed symmetrically with respect to the centerlines of the construction opening.

4.1.5.2.3 Finite Element Modelling of the Containment Shell

The BSAP computer program (Reference 1) was used to estimate the load combination of dead load and the prestress redistribution due to tendon detensioning and/or removal. Shell elements were used to simulate the buttresses, wall and dome, and beam elements were used to simulate the ring girder. A total of 1,401 nodal points and 1,508 elements are used in the finite element modelling of the containment shell structure as shown in Figure 4.1-21. The isometric computer plot is shown in Figure 4.1-22. The sustained

Young's modulus of elasticity, $E = 2.1 \times 10^6$ psi, and Poisson's ratio, 0.17, were the concrete properties used in the analyses of the containment shell. The effective prestress forces, which were obtained after considering all the prestress losses, are 370 Kips/ft per group, 290 Kips/ft, and 665 Kips/ft for dome, vertical, and hoop tendons, respectively. The tendon force was represented by the anchorage force applied at buttresses and the ring girder and a distributed pressure due to the curvature of the tendon. The wind and earthquake loads provided in the uniform building code are insignificant compared to the prestress forces.

4.1.5.2.4 Results of Containment Shell Analysis

The membrane hoop and meridian forces as well as the meridian and hoop stresses at the inside and outside surface of the containment shell have been investigated. Results along vertical and horizontal cross sections through the center of the construction opening area have been presented in Figures 4.1-23 to 4.1-28.

The compressive strength of the concrete f'_c is 5,000 psi. The allowable stresses for the concrete are as follows:

Membrane tensile stress	$\sqrt{f'_c} = 70.7$ psi
Flexural tensile stress	$3\sqrt{f'_c} = 212$ psi
Membrane compressive stress	$.3f'_c = 1,500$ psi
Flexural compressive stress	$.6f'_c = 3,000$ psi

Results show that prestress levels anywhere in the containment shell are within allowable limits. The prestress level at the construction opening area is approximately zero.

4.1.5.3 <u>Removal of Concrete and Liner Plate</u>

4.1.5.3.1 Sequence of Material Removal

After the necessary tendons are detensioned and/or removed, the construction opening will be made as follows:

a. Chip the concrete in accordance with the opening size requirements

b. Cut the rebar

c. Cut and cap the tendon sheathing

d. Cut the liner plate

The cutting patterns of steel reinforcement, tendon sheathing, and liner plate are shown in Figures 4.1-19 and 4.1-20. These cutting patterns have been chosen in order to facilitate the cadwelding of rebar, splicing of tendon sheathing, and welding of liner plate for closing the opening.

4.1.5.3.2 Finite Element Analysis of Containment Shell With Opening

As shown in Figure 4.1-21, shell elements in the construction opening area have been removed in the finite element analysis. The isometric computer plot is shown in Figure 4.1-29.

Results of this computer analysis for estimating the stress re-distribution along the horizontal and vertical cross sections of the containment wall due to the presence of the construction opening are shown in Figures 4.1-30 through 4.1-35. These sections are taken along the centerlines of the construction opening.

Results show that the stress levels anywhere in the containment shell are within the allowable limits and the stress level around the opening area is low.

4.1.5.4 Closing the Construction Opening

4.1.5.4.1 Creep Effect Consideration

After the opening is closed, the replaced concrete will undergo creep. The maximum differential creep strain between the construction opening and other area of the containment can be derived from the FSAR as 0.22×10^{-6} in./in./psi. This value is obtained by assuming that the replaced concrete will undergo the largest creep strain, while the creep strain in the other area is zero. This uniform differential creep strain can then be represented by the equivalent thermal load as follows:

$$-\alpha\Delta T = 0.22 \times 10^{-6} \times 1500$$

Where α (0.000005 in./in./F) is the thermal coefficient of the concrete and 1500 psi is the allowable compressive stress.

This leads to the temperature difference of $\Delta T = -66F$. To predict the creep effect, a factor of 1.05 has been imposed, resulting in the temperature difference of 69.3F. The finite element modelling shown in Figure 4.1-22 has been used to estimate the maximum prestress loss due to the maximum differential creep strain. Results are shown in Figures 4.1-36 and 4.1-37. Additional reinforcement bars will be provided to carry the membrane tensile forces resulting from these creep effects. In this manner, the prestress loss due to creep effect will be avoided.

4.1.5.4.2 Sequence of Closing the Construction Opening

After the steam generator replacement is completed, the construction opening will be closed and tendons replaced and re-tensioned. The final containment shell structure will have been restored to its original structural integrity.

The general sequence to close the opening is as follows:

a. Replace the liner plate

b. Remove the sheathing caps and splice the tendon sheathing

c. Place the creep reinforcement bars

d. Restore the existing reinforcement bars

e. Replace the concrete

f. Replace and retension tendons

g. Perform the modified structural integrity test

h. Perform the integrated leak rate test

4.2 EQUIPMENT AND MATERIAL REMOVAL AND REPLACEMENT

4.2.1 MECHANICAL EQUIPMENT

As presently being considered, no major mechanical equipment will have to be relocated because it interferes with the replacement of the steam generators.

As appropriate, equipment within the containment will be covered to ensure cleanliness during the repair.

4.2.2 INSTRUMENTATION

The following instrumentation, sensing lines, and associated supports will be temporarily removed and relocated inside the containment:

a. Sensing lines on Steam Generator E-50A for Pressure Transmitters PT-0751A through D, Level Transmitters LT-0751A through D, LT-0701 and LT-0702, and Sampling Point SX-0719

b. Sensing lines on Steam Generator E-50B for Pressure Transmitters PT-0752A through D, Level Transmitters LT-0752A through D, LT-0703 and LT-0704, and Sampling Point SX-0718

c. The sensing line support structures for the sensing lines described in a. and b.

Disconnection of instrumentation cables to the above transmitters is discussed in Section 4.2.3. The open ends of lines will be capped to ensure cleanliness during the repair.

As appropriate, the instrumentation and sensing lines will be returned to service using standard procedures followed during routine plant maintenance programs.

4.2.3 ELECTRICAL EQUIPMENT

4.2.3.1 Temporary Removal and Relocation of Electrical Equipment Inside the Containment

Table 4.2-1 provides a list of electrical equipment and instruments which will be temporarily removed and/or relocated inside the containment because their existing location would lead to interference with the steam generator repair operation.

In addition, the removal of 480 V Motor Control Center B09 will result in the interruption of power supply to the loads normally supplied from this motor control center. These loads are listed on Table 4.2-2.

4.2.3.2 Temporary and Permanent Electrical Loads Associated With Steam Generator Repair and the Corresponding Power Source

Table 4.2-3 provides a list of the temporary and permanent electrical loads associated with the steam generator repair and identifies the normal and temporary power sources for each load.

It is to be noted that the table does not include other plant loads not directly or indirectly associated with the steam generator repair program even though such loads may continue to remain energized for the duration of the repair. As an exception, Table 4.2-3 includes the fuel pool cooling system pumps and associated auxiliary loads even though the fuel pool cooling system is not directly associated with the steam generator repair.

4.2.3.3 Temporary Alternate Electrical Power Supplies

Temporary alternate electrical power supplies inside the containment will be arranged as follows:

a. One or more 480 V/277 V power distribution panels

b. One or more 208 V/120 V lighting and power distribution panels

A temporary 2400 V or 480 V feeder will be installed to supply temporary load centers inside the containment as shown on the single lines, Alternates 1 and 2, Figures 4.2-1 and 4.2-2.

The temporary alternate power supplies will be used to supply the following categories of loads:

a. Miscellaneous rigging loads inside the containment during the repair

b. Permanent loads required to be operational during the repair and supplied originally from the 480 V Motor Control Center B09. Motor Control Center B09 will be removed during the repair program (see Table 4.2-1).

c. Permanent loads required to be operational during the repair but requiring temporary relocation and supplied originally from power sources outside the containment. Such loads will be powered from temporary alternate power supplies only in those cases where it is not practical to recable to the original penetrations connected to the normal power supply.

4.2.3.4 <u>Relocation and Recabling of Permanent Loads, Instruments, and Devices Inside the Containment Required to be Operational During the Repair Program</u>

4.2.3.4.1 Permanent Loads Normally Supplied from 480 V Motor Control Center B09 and Required to be Operational During the Repair Program

As shown on Table 4.2-1, 480 V Motor Control Center B09 inside the containment will be deenergized and removed during the repair work. Therefore, permanent loads normally supplied from MCC B09 which are required to be operational during the repair program will be supplied from the temporary power or lighting distribution panels. Where required, temporary local starters will also be provided.

The actual procedure will involve disconnecting all cabling to the motor control center, removing and relocating the motor control center, and recabling to the required loads with temporary cables from the temporary power or lighting distribution panels.

4.2.3.4.2 Permanent Loads and Instruments Connected to Power Sources Outside the Containment and Required to be Operational during the Repair Program

The permanent loads and instruments are connected to power sources outside the containment and are required to be operational during the repair program. They fall into three categories:

a. Devices which do not interfere with the steam generator repair. In addition, the associated cabling is routed via trays and conduits which do not interfere with the repair program.

b. Devices which directly interfere with the steam generator repair and have to be removed and relocated

c. Devices which do not directly interfere with the repair. However, the associated cables are installed in trays or conduits that interfere with the repair work and, therefore, will have to be removed.

Category a. devices will not be affected by the repair operation and will continue to function from their normal power sources.

In the case of category b. devices, the associated cables will be disconnected and coiled back in the trays. The associated conduits, if any, will also be moved away. After the relocation of the load out of the interference path, the disconnected cable may be temporarily reconnected to the device, if possible, or a new cable may be installed from the penetration to the device. During the repair program, these loads will continue to draw power from the normal source.

After the conclusion of the repair work, the devices will be moved back to the original location and reconnected using the old or new cable as the case may be.

In the case of category c. devices, the cables will be disconnected and removed from the trays or conduits causing interference, and then the trays or conduits will be dismantled. The existing cables will be temporarily connected back to the devices via alternate routes or new temporary cables will be installed from the penetrations to the devices. During the repair program, these loads will also continue to draw power from the normal source.

After the conclusion of the repair work, the trays or conduits will be reinstalled in the original position and the existing or new cables will be re-installed as per the original routing.

4.2.3.5 Power Sources

4.2.3.5.1 Availability of Class 1E Electrical Systems

Before starting major repair activities within the containment, the reactor core will be offloaded and transferred to the fuel storage facility. Therefore, during the steam generator repair, it is not necessary to maintain the availability of the Class 1E electrical systems required to provide the capability to shut down the reactor.

However, the full core of fuel elements will be stored in the spent fuel pool during the repair. To ensure safe storage of fuel under all foreseeable conditions and to protect against radiation release from irradiated fuel, the fuel pool cooling system and all associated support systems will be kept fully operational.

During the repair program, the plant electrical system alignment will be such as to ensure at all times reliable and redundant power supplies to the fuel pool cooling system and all associated support systems.

4.2.3.5.2 Configuration of Offsite and Onsite Power Sources During the Repair Program

During the repair program, the "quick disconnect" links between the main generator terminals and the isolated phase bus will be removed. This will enable continued power supply to the plant auxiliary power system buses from the 345 kV switchyard via the main transformer and station Power Transformers 1-1 and 1-2 (see Figure 4.2-3).

In addition, the 345 - 4.16 kV and 345 - 2.4 kV Startup Transformers 1-1 and 1-2, respectively, will also be available to provide the second source of offsite power during the repair.

Both 345 kV circuits, one to the main transformer and the other to the startup transformers, may be deenergized during certain construction phases of the repair program, during which time the diesels or an alternate 345 kV circuit will be utilized.

2400 V emergency diesel Generators 1-1 and 1-2 will be maintained in the "ready for operation" status to provide two sources of onsite power supply during the repair; 125 V dc Batteries D01 and D02 and associated battery chargers will also be maintained operational to provide the 125 V dc supplies and the preferred 120 V ac supplies through the inverters.

4.2.3.5.3 Operation of Systems Related to Plant Safety During Steam Generator Repair

The following major systems related to plant safety will be maintained in fully operational status during the repair:

a. Spent fuel pool cooling system

b. Service water system, both critical and noncritical

c. Containment air circulation and cooling system

d. Compressed air system

e. Component cooling system

f. Fire protection system

g. Radwaste systems

h. Radiation protection and monitoring systems

Reliable operation of these systems related to plant safety is enhanced by ensuring that the availability of electric power sources is not allowed to degrade to levels lower than that required for normal plant operation. This requirement will apply to offsite ac power sources, onsite standby ac power sources, and onsite dc power supplies.

The actual configuration of offsite and onsite power sources during the repair has been discussed under Section 4.2.3.5.2.

It is therefore concluded that the operational capability of systems related to plant safety is not compromised to any degree during the repair program.

4.2.4 PIPING

In order to accomplish the steam generator repair it will be necessary to cut portions of the following major piping systems:

a. Primary coolant piping

b. Main steam piping

c. Main feedwater piping

d. Steam generator blowdown piping

Location of cut areas for reactor coolant system, main steam system, main feedwater system, and blowdown system piping are shown in Figures 4.2-4, 4.2-5, 4.2-6, and 4.2-7, respectively. As appropriate, the open ends of cut piping will be covered to ensure cleanliness during the repair.

The piping will be re-installed in accordance with FSAR criteria as close as possible to the original installation, except for changes required by the systems modifications discussed in Section 3.0 and below. Piping weld end preparations, welding, and nondestructive examination for the installation will be in accordance with the latest edition of the ASME Code.

The horizontal portion of the main steam line will be raised 30.2 inches to accommodate the new steam generator flow restrictor nozzle. A spool piece of the same length will be used to connect the raised portion of the new main steam line with the existing steam pipe. Figure 4.2-5 shows the location of the new main steam line. The upward relocation of the steam line will have no significant effects on the stresses in the system. This conclusion is based on the conservative data used in the original calculations and the method of raising the main steam line by adding the additional height in a vertical line above a rigid support. The spool piece will be seismic Category 1 and ASME Code, Section III, Class 2.

4.2.5 CONCRETE AND STRUCTURAL STEEL

The following structures or portions of structures within the containment will be removed and replaced to facilitate replacement of the steam generators.

a. Steam generator upper guide steel supports (see Figure 4.2-8)

b. Grating platform at el 657'-0" including supporting members

c. Portion of peripheral concrete curb at el 649'-0" around the proposed opening will be removed as necessary to remove interference with temporary runway supports.

4.2.6 COATINGS

All coatings removed during the repair program will be replaced with equivalent or better coatings utilizing Consumers' approved procedures.

4.3 RADIOLOGICAL PROTECTION PROGRAM

The radiological protection program to be implemented for the repair effort will be in accordance with 10 CFR 20, the Palisades Plant Health Physics Procedures, and the ALARA practices described in Section 4.3.5 of this report.

4.3.1 SUPPLEMENTAL ACCESS CONTROL

Facilities will be provided for the repair effort to accommodate the personnel involved. (See Figures 4.3-1 and 4.3-2). These facilities include:

a. Outside Access Control Point

   1. Radiological protection training facility

   2. Craft change area

   3. Locker area

   4. Toilet

   5. Protective clothing pickup area

   6. Protective clothing dress-out area

b. Inside Access Control Point

1. Radiation control point

2. Protective clothing undressing area

3. Storage area for protective clothing

4. Health physics area

5. Laundry area

The following is a brief description of the access control procedure currently contemplated for entering and exiting the containment.

Personnel will enter the locker area, disrobe, pick up their protective clothing, and dress before proceeding through access control to the personnel air lock or the equipment hatch. The requirements for protective clothing are specified in the Palisades Plant Health Physics Procedures.

Personnel leaving the containment will remove their shoe covers, gloves, and other protective clothing at the step-off pad and be frisked for residual contamination. They will then exit through the radiation control point and return to the locker area for their street clothes. If an individual has been contaminated, he or she will be directed to don a clean set of protective clothing, as necessary, following removal of the contaminated clothing in the undressing area, and be escorted to the decontamination showers in the permanent health physics area in the auxiliary building. Decontamination methods and requirements will be in accordance with the Palisades Plant Health Physics Procedures.

Additional health physics support will be provided to the Palisades Plant health physics organization in order to implement health physics related activities. Health physics technicians will be utilized for monitoring and assistance at the steam generators, personnel air lock, equipment hatch, and access control. In addition, television monitors will be utilized for reduction of personnel exposures when visual observations are required for work in high radiation areas.

Personnel involved in work areas with a potential for high-level contamination will wear 2 sets of protective clothing.

The outer set of protective clothing will be removed when leaving the work area and deposited in a container. The second set will be removed at the step-off pads discussed above.

4.3.2 LAUNDRY

The existing laundry is appropriately sized to accommodate the additional volume during the repair effort. Disposable protective clothing will be utilized when effective. Laundering of protective clothing and cleaning and sanitizing of respiratory equipment will be in accordance with the Palisades Plant Health Physics Procedures.

4.3.3 CONTROL OF AIRBORNE RADIOACTIVITY AND SURFACE CONTAMINATION

Airborne radioactivity inside containment during the steam generator repair effort will be controlled, monitored, and ultimately released via the plant vent stack. Air will be drawn through the hatches and construction opening and exhausted by the purge system via the plant ventilation stack, thus precluding airborne radioactive particles or gases from leaving containment openings utilized for construction activities. This air will be conditioned, if necessary, for removal of airborne radioactivity by use of two installed recirculation filters (HEPA plus charcoal absorber) rated at 6,000 cfm each. The air being exhausted from the plant will be monitored as it passes the existing sampling station located within the ventilation stack. It should be noted that even if credit were not taken for the purge system that the potential offsite dose would still be less than the 10 CFR 50, Appendix I, limits.

In addition to bulk containment atmosphere control of airborne radioactivity, appropriate localized control will also be provided. Radioactivity generated during the cutting of the primary coolant pipes will be contained within specially designed contamination control envelopes, which will provide local high efficiency filtration. Personnel working inside these control envelopes will wear respiratory protection equipment, as required, described and implemented by the Palisades Health Physics Procedures.

Section 4.3.1 describes the method of controlling the spread of surface contamination by personnel removing their outer set of protective clothing when leaving the control envelope.

The radioactive release and dose assessment associated with cutting the primary coolant loop are provided in Sections 4.3.7 and 6.2.2.

4.3.4 SUPPLEMENTAL PERSONNEL MONITORING REQUIREMENTS

4.3.4.1 Monitoring of Airborne Radioactivity

Mobile air monitors will be used, as required, to monitor the airborne radioactivity inside the contamination control enclosures and in other work areas inside containment. Airborne radioactivity samplers coupled with laboratory analyses will also be employed.

4.3.4.2 Monitoring of Workers for Ingested Radioactivity

Workers who are planning to enter airborne radioactivity or contamination areas will be given an initial whole body count at the start of their employment. Subsequently, workers will be given whole body counts or bioassays, as necessary, to comply with requirements set forth in Palisades Plant Health Physics Procedures.

4.3.4.3 Personnel Monitoring

All personnel entering the radiation controlled area will be provided with personnel dosimetry in accordance with the Palisades Health Physics Procedures.

4.3.4.4 Radiation and Contamination Surveys

Detailed surveys which provide proper control of radiation and contamination will be performed, as required, throughout the repair effort. These surveys will be performed in accordance with the Palisades Plant Health Physics Procedures.

4.3.4.5 Portable Survey Instruments

A description of typical portable survey instruments used at the Palisades Plant is included in Table 4.3-1.

4.3.5 GENERAL ALARA CONSIDERATIONS

Personnel exposures will be maintained as low as is reasonably achievable (ALARA) in accordance with 10 CFR 20.1(c), Regulatory Guide 8.8 (Revision 2), and as defined in the Palisades Plant Health Physics Procedures.

4.3.5.1 Use of Scale Model in Radiological Protection Program

A scale model of the Palisades Plant containment has been constructed in order to better assess structural and operational parameters that give rise to occupational radiation exposure (see Section 4.9). The model will provide a valuable tool in the continuing study of methods for reducing doses. Among the most important considerations are:

a. Shielding or Equipment Removal

The model, in conjunction with actual field survey data, will be used to study radiation fields to determine temporary shielding requirements. Where shielding may prove difficult or ineffective, the source of exposure will be considered for removal. The model will aid decisions on shielding or removal of radiation sources including, but not limited to, heat exchangers, drain lines, tanks, and primary coolant pipe segments.

b. Man-Rem Assessment

The model aids in predicting the expected man-rem dose for activities in high radiation areas. Decisions related to radiation exposure, such as employing local decontamination or determining the number of people required for an activity, can be made early in the design phase of the project in order to incorporate the most effective solutions to the reduction of exposures.

c. Work Planning

The model will be used to develop construction work plans to establish the most efficient procedures for performing work in high radiation areas.

d. Craft Training

The model will be used for the orientation and training of supervisory and key craft personnel to supplement construction work plans to achieve the most efficient utilization.

4.3.5.2 Temporary Shielding

Shielding will be used, as necessary, to reduce the dose rates from components such as heat exchangers, valves, and temporary storage areas for contaminated pieces of pipe, cleanup materials, and tools. Temporary shielding will be used, as necessary, for the steam generator and associated piping while it is being cut out of the primary coolant loop and during weld-back operations. The steam generator shell will also help shield the more contaminated parts of the old steam generators. Openings in the old steam generators created by cutting the connecting pipes will be closed by welding plates over the openings. These plates will be supplemented, as required, with lead or other shielding to provide a minimum shielding value that is no less than that afforded by the original piping.

4.3.5.3 Local Decontamination

Decontamination of localized areas within the steam generators and primary coolant piping may be performed. Decontamination of other work areas will be performed periodically, depending on the contamination levels. Paper and plastic sheeting will be used to facilitate collection and cleanup of contamination. In all cases, the Palisades Health Physics Procedures will be followed.

4.3.5.4 Low Background Radiation Waiting Areas

Low background radiation waiting areas will be established where workers must wait between tasks. Special signs, tape, or rope-off areas will be utilized to designate these areas. This technique will result in minimum stay times in high background radiation areas, yet provide for higher work efficiency than would the technique of waiting in an area outside of containment. Health physics personnel will work with the job supervisors to ensure that personnel not required in the work area remain in the waiting area. Television camera monitoring will provide supplemental coverage of high radiation areas, and it is anticipated that clothing and/or hard hat (where applicable) color codes will be utilized to provide rapid discrimination of the various work groups. Such coding will allow early detection of individuals entering areas not directly applicable to their work function and thereby reduce unnecessary exposures.

4.3.5.5 Training of Craft Personnel

Selected craft personnel will be given a comprehensive course in radiological protection. This course will consist of instruction and demonstrations covering, in detail, the basic theory and practice of radiation protection principles, emergency planning, radiological protection program, and decontamination activity. Additional instructions and training will be provided for those individuals requiring the use of respiratory protective equipment and/or scheduled to work in high radiation areas. This training will involve system familiarization through review of the scale model and practice with mockup equipment while wearing respiratory protective equipment, as applicable. The minimum training required for all personnel will be successful completion of the orientation course described in the Palisades Plant Health Physics Procedures.

4.3.6 MISCELLANEOUS WASTE DISPOSAL

4.3.6.1 Concrete Disposal

Concrete will be removed from the containment external walls before liner plate removal and will be disposed of as nonradioactive material. This concrete has an insignificant amount of transferable contamination (transferable contamination is considered insignificant if it is less than 2200 dpm/100 cm^2 per 49 CFR 173.397) without surface decontamination. The small amount of concrete removed from areas internal to containment will be considered contaminated and may either be decontaminated before cutting by vacuuming and/or scrubbing with detergent and water to reduce the amount of transferable contamination to as low as is reasonably achievable below 2200 dpm/100 cm^2, or may be appropriately packaged for shipment. Following removal from the containment, contaminated concrete will be shipped as "low specific activity" (LSA) material to a licensed land burial site.

4.3.6.2 Miscellaneous Dry Waste Disposal

Contaminated metal shavings from the various cutting operations and miscellaneous dry contaminated waste, such as paper and rags, will be put in standard shipping containers and shipped as LSA material to a licensed land burial site.

4.3.6.3 Liquid Radwaste Disposal

There are three potential sources of radioactive liquid to be disposed of. These sources are:

a. Water drained from the reactor coolant system

b. Laundry wastewater

c. Local decontamination waste fluids

The radioactive releases associated with these sources are discussed in Subsection 6.2.2.4.

The primary coolant will be processed by the chemical and volume control system as described in Section 4.3 of the Palisades Plant FSAR. After appropriate sampling, the laundry wastewater may be discharged without processing through the radwaste system. The estimated activity level for laundry waste effluent is low, as indicated by the effluent total given in Table 6.2-3. Laundry wastes are treated by passage of waste through a series of filters (20 and 5 microns), followed by dilution and release. The filtration process serves to remove a major portion of the radioactivity adhering to laundered garments, particularly the cobalt, manganese, and iron isotopes which normally are present in insoluble particulate form.

The small amount of liquid waste generated as a result of local decontamination will be treated either as part of the normal liquid radwaste processing scheme or solidified directly into a sodium silicate matrix on a batch basis if the solutions are incompatible with plant systems. The compatibility of decontamination solutions with existing processing equipment will be determined prior to their use. Sodium silicate solidification methods have been tested at the Palisades Plant for a broad spectrum of waste types, including oils, boric acid, and miscellaneous dirty wastes.

4.3.7 MAN-REM ASSESSMENT

4.3.7.1 Man-Rem Assessment for Continuing Operation

Assuming that the replacement steam generator tubes maintain their integrity during the remaining operating lifetime of the plant, radiation exposure attributed to steam generator work will be reduced. It is not expected to exceed 25 to 50 man-rem per year for a tube inspection operation in accordance with Regulatory Guide 1.83. It has been estimated that approximately 250 man-rem could be saved each year following the steam generator repair.

4.3.7.2 Man-Rem Assessment for the Repair Effort

Health physics survey data have been reviewed for the period from November 1976 through March 1978 at various times after shutdown to determine trends in dose rates and radionuclide contributors that affect operations in the vicinity of the steam generators. (Typical survey results and data are shown in Figures 4.3-3 through 4.3-5.) It is believed that this plant survey data is representative of conditions expected at the start of the repair activity, provided that appropriate dose rate increases due to activity buildup within the steam generators are considered (Reference 3). Survey data in the vicinity of the steam generators and primary coolant piping were averaged and used to project general field and dose rate estimates at various times after shutdown (see Figures 4.3-6 and 4.3-7). The man-rem assessment for the repair effort is shown in Table 4.3-2.

4.3.7.2.1 Radiation Field Uncertainties

a. Radiation fields were taken from actual Palisades Plant surveys and adjusted for activity increase as a function of time before steam generator removal. In developing the man-rem predictions, it was assumed that the radiation fields would not decay throughout the repair effort. Actual radiation fields will decrease with time. Therefore, the actual total job man-rem are expected to be lower than the calculated values.

b. The effectiveness of temporary shielding or local decontamination will be further defined as dose rate survey data and primary system samples are gathered during future outages. Reduction factors as now estimated are indicated in Table 4.3-2.

4.3.7.2.2 Assumptions Used to Estimate Manhours by Area for Dose Calculations

a. Nonwelding Operations in Radiation Area

1. 50% of manhours in radiation area

2. 30% of manhours checking in and out through Health Physics and Security

3. 20% of manhours in lower radiation area of containment

b. Welding Operations in Radiation Area

Welding operations in the radiation area are based on rotating welder and helper between work area and lower radiation area, as work operations dictate, to minimize welders' exposure.

1. 35% of manhours in radiation area

2. 30% of manhours checking in and out through Health Physics and Security

3. 35% of manhours in lower radiation area of containment

c. Outside Work

All manhours are outside of the containment.

d. Welding of Primary Coolant Pipe

The following is based on rotating welder and welder's helper between work area and lower radiation area as work operations dictate to minimize welders' exposure.

1. 35% of manhours at primary coolant pipe broken down as follows:

   Outside pipe 35% X 68% = 24%
   <u>Inside pipe* 35% X 32% = 11%</u>
   Total in place = 35%

   *Inside pipe manhours required to grind and clad inside of primary coolant pipe by conventional manual methods

2. 35% of manhours in low radiation area of containment

3. 30% of manhours checking in and out through Health Physics and Security

e. Stress Relieving of Primary Coolant Pipe

1. 10% of manhours inside pipe

2. 30% of manhours within 6 feet of pipe

3. 30% of manhours in lower radiation area of containment

4. 30% of manhours checking in and out through Health Physics and Security

f. X-Ray and NDT of Primary Coolant Pipe

A total of 2,600 hours is allowed to x-ray the primary coolant pipe. This is based on 4 hours availability per day for x-raying inside containment.

1. 37% of manhours checking in and out through Health Physics and Security

2. 20% of manhours inside of pipe

3. 30% of manhours within 6 feet of outside pipe

4. 13% of manhours in lower radiation area of containment

g. Approximately 780 additional hours of time for x-ray technicians are included to x-ray the main steam and feedwater lines.

h. Rigging

1. The following work operations were considered to be outside of the power plant building but inside of the security fence:

   (a) Set up equipment to handle the new steam generators from the barge slip to the containment

   (b) Set up equipment external to the containment to handle the new and existing steam generators

   (c) Offload, move to storage, and later transport steam generators to containment

   (d) Remove all external rigging equipment from the site

   (e) Decontaminate and remove all internal rigging from the site

   (f) 21% of manhours allowed to install new steam generators

2. The following work operations were considered to be at the containment operating floor level or higher:

   (a) Install rigging equipment inside containment

   (b) Decontaminate and remove rigging equipment from inside of containment

   (c) 60% of the manhours allowed to install new steam generators

3. Seventy percent of the manhours associated with 2. above were considered to be at the operating floor while 30% were required for checking in and out through Health Physics and Security.

4. Forty-nine percent of the manhours associated with removing the existing steam generators from the containment were considered to be within 6 feet of the primary coolant pipe or bottom of the steam generator, 21% were required to check in and out through Health Physics and Security, and 30% were next to the existing steam generators outside of the containment.

5. The manhours associated with moving the existing steam generators to storage were considered to be adjacent to the existing steam generators but outside of the containment.

i. An assumption was made that 50% of the manhours required to cut the primary coolant pipe would be spent within 6 feet of the outside of the primary coolant pipe or bottom of the steam generators, with partial exposure to the inside of the primary coolant pipe before the steam generator removal. Thirty percent of the remainder would be spent checking in and out through Health Physics and Security, and 20% would be spent in an area of low radiation.

j. It was also assumed that 50% of the manhours required to level, line up, and tack the primary coolant pipe would be spent within 6 feet of the outside of the primary coolant pipe, with partial exposure to the inside of the primary coolant pipe after steam generator removal. Thirty percent of the remainder would be spent checking in and out through Health Physics and Security, and 20% would be spent in an area of low radiation.

k. The manhours associated with miscellaneous piping operations were spread in accordance with the piping hours spent in each area.

l. Distributables

1. Welder tests and miscellaneous services were considered to be outside the plant buildings but inside the security fence.

2. Startup, cleanup, and scaffolding distributables were prorated on the basis of total manual manhours at each location, excluding the outside work.

m. Nonmanual Labor

The following assumptions were made on the nonmanual labor manhours:

1. All manhours, except for superintendents and engineers, would be expended outside plant buildings but inside the plant security fence (office work).

2. Fifty percent of the superintendents' and engineers' manhours for piping, electrical, civil, rigging, and safety would be outside the plant building but inside the security fence (office work).

3. The remaining 50% of the superintendents' and engineers' manhours were prorated to each of the work areas based on the total manual hours spent in each area by discipline. For instance, the piping superintendents' and engineers' manhours were prorated on the basis of the total of manual hours spent in performing the piping operations in each area.

4.3.7.2.3 Technique for Estimating Radiation Dose

The total dose is dependent on the following factors:

a. Dose rates (rem/hr) before shielding or decontamination

b. Shielding or decontamination effectiveness

c. Duration of tasks (hours)

d. Manhours required to complete tasks

e. Fraction of time the task is in radiation field of interest

The entire repair program has been divided into discrete areas. The total personnel exposure in an area is the product of the dose rate and the manhours required to complete all tasks involved. The total exposure for the entire job is a summation of the exposure for all areas.

Therefore:

$$Ei = Di \text{ (rem/hr)} \times Mi \text{ (manhours)}$$
$$E = \sum Ei \text{ (man-rem)}$$

where:

Ei = Total personnel exposure for area i (man-rem)
Di = Average dose rate in area i (rem/hr)
Mi = Manhours to complete all tasks in area i (manhour)
E = Total personnel exposure for all areas (man-rem)

4.3.7.2.4 Confirmation of Man-Rem Estimate

Daily radiation dose logs will be maintained for each worker stationed within the higher dose rate ($\geq$10 mrem/hr) areas of the containment. Weekly, monthly, or quarterly records will be maintained for those working outside the containment and in other specially designated low dose rate areas. These actual doses will be tabulated by task category for confirmation of estimated doses provided in this report.

4.4 DISPOSITION OF OLD STEAM GENERATORS

The disposal effort is independent of the repair and is evaluated on that basis. Because of the uncertainty of the timing of the repair and the availability of the offsite disposal facilities, the ultimate disposition of the old units cannot be finalized at this time; however, a variety of disposition alternatives has been investigated.

The steam generators to be removed represent the single largest source of solid radioactive waste to be disposed of during the repair effort. The primary side internal surfaces of the steam generators are contaminated by a tenacious film of deposited radioactive corrosion products made up primarily of cobalt, manganese, and iron isotopes. Isotopic analyses obtained from uncleaned 2-inch long sections of steam generator tubing indicate that at the time the steam generators are removed, each will contain approximately 30 curies of deposited gamma activity (see Table 4.4.1 and References 2 and 3). The activity will decrease to approximately 2.8 curies per steam generator 2 years after shutdown, then continue to decay with the 5.6 year half-life of Cobalt-60.

4.4.1 OBJECTIVES OF HANDLING/DISPOSAL OPERATIONS

The objectives of handling/disposal operations are as follows:

a. To dispose of the steam generators safely and economically

b. To provide the means to handle/dispose of the steam generators so that radiation exposures to plant and contract personnel are as low as is reasonably achievable

c. To minimize the release of radioactivity to the environment so as to keep radiation exposure to the public as low as is reasonably achievable and within the limitations of 10 CFR 20

4.4.2 ONSITE STORAGE

If it is decided that the old steam generators will be stored onsite, a storage facility will be necessary (see Section 4.1.1.2.2).

Before removal from the containment, the openings in the steam generators will be sealed to prevent the release of radioactivity during transfer and subsequent onsite storage (see Section 4.3.5.2). Sealing will be performed by welding plates of steel over each pipe opening. The steel plates will be thick enough or supplemented by lead shielding, if required, so that external dose rates at the sealed opening are not higher than adjacent surface areas.

The only significant radiological consideration associated with storage is the direct radiation from the steam generators (see Section 4.4.6). Shielding will be provided to ensure acceptable radiation levels external to the storage facility. Section 4.4.7 demonstrates that there are no credible accident considerations associated with onsite storage of the sealed steam generators that result in the release of radioactivity from the steam generators.

Based on the above considerations, the required storage facility design criteria are:

a. Appropriate shielding for direct dose

b. Access for periodic surveillance of steam generator seal integrity using portable monitors

c. Environmental protection in a weathertight and restricted-entry shelter

4.4.3 OFFSITE DISPOSAL

The following three methods were investigated as alternative means of shipping the removed steam generators to a licensed land burial site:

a. Shipment by barge in one piece

b. Shipment by truck cut up

c. Shipment by rail cut up

4.4.3.1 Preparation for Shipment by Barge

Barge shipment of the old steam generators is determined to be the most acceptable method from both environmental and occupational dose standpoints provided that routing and handling capabilities remain available at the time of shipment. The steam generators will be sealed before removal from the containment so that the radioactivity will be contained within a strong, tight package as required by 49 CFR 173. When the steam generators are to be shipped to a licensed land burial site, each one will be transported to the barge facility intact and shipped as low specific activity (LSA) material in accordance with applicable state and federal regulations.

4.4.3.2 Preparation for Shipment by Rail and/or Truck

In preparation for shipment of the steam generators by rail or truck to a licensed land burial site, the generators would be cut into sections suitably sized for shipment. The cutup sections would then be packaged in strong, tight packages and shipped with appropriate shielding in accordance with applicable state and federal regulations.

Cutting operations on the steam generators would be performed in enclosure envelopes, as required, to minimize the spread of airborne radioactivity. The enclosure envelopes will be provided with a HEPA filtration system to reduce the potential release of radioactivity to the environment and will be designed to allow the use of remote cutting techniques to reduce personnel exposure to radiation during cutting. Temporary shielding will also be provided, as required, to further reduce personnel radiation exposure. Radiation detection and measurement during cutting operations will be in accordance with the Palisades Plant Health Physics Procedures.

4.4.3.3 Shipment

If shipped by truck, potential disposal sites would be Sheffield, Illinois; Morehead, Kentucky; Barnwell, South Carolina; Beatty, Nevada; and Richland, Washington if in service at the time. Further, if shipped by truck, 14-foot maximum width dimension limits exist, and then only for escorted shipments. This width includes shielding and overpack material. Hence, the net width would be around 12 feet, with maximum length limited to about trailer length, or 38 feet, without special hauling permits.

Shipment by rail or barge limits disposal to Richland, Washington, as this is the only site with rail and ship offloading facilities.

For rail, the maximum width, including shielding and overpack, is 10 feet, producing an effective maximum net width of the cutup material of about 8 feet. These maximum dimensions apply for rail and truck shipments only if additional large shields are constructed and receive U.S. Department of Transportation approval permits. Utilizing existing shielded casks, dimensions of the cutup material would be considerably smaller--on the order of 10 feet by 6 feet by 3 feet. If the steam generators were shipped by truck, with a net payload of 10,000 pounds per shipment, approximately 90 shipments would be required for each of the 450-ton steam generators.

Conceivably, a single shipment of both steam generators by barge can be made to Richland, Washington, through either the Illinois - Mississippi River - Panama Canal route or the St. Lawrence Seaway - Panama Canal route. The offloading facilities at Richland, Washington, can easily accommodate a single steam generator at 450 tons.

4.4.4 MAN-REM ASSESSMENTS

If the steam generators are shipped by rail and/or truck, they must be cut into suitably-sized sections before shipment. The man-rem associated with this operation will vary depending on the length of time the steam generators are in storage before the actual cutting operation.

If barge transport is employed, cutting is not required and handling of the steam generators would be minimized. Based on radiation survey data and analogous manhour estimates established in Section 4.3.7, the man-rem associated with barge transport (1 to 5) is a small fraction of the man-rem associated with rail and/or truck shipments (575 to 750) (see Table 4.4.-2).

4.4.5 RADIOACTIVE RELEASES AND DOSE ASSESSMENT ASSOCIATED WITH OFFSITE DISPOSAL

The openings in the steam generators will be sealed before the steam generator is removed from the containment building (see Section 4.3.5.2). Since the steam generators will be sealed during storage and eventual shipment by barge, no airborne or liquid radioactive releases are associated with offsite disposal.

4.4.6 RADIOACTIVE RELEASES AND DOSE ASSESSMENT ASSOCIATED WITH ONSITE STORAGE

If onsite storage is necessary, a suitable storage facility would be constructed before the removal of the old steam generators (see Section 4.1.1.2). Since all openings in the steam generators will be sealed before removal from the containment, no airborne or liquid radioactive releases are expected as a result of onsite storage.

As discussed in Section 4.4.7, the radioactivity within the steam generators is immobile. Thus, if seal integrity was lost, releases to the environment would not be likely. Nonetheless, a surveillance program will be implemented comprised of periodic visual inspection of the external surfaces of the lower assemblies, area radiation surveys, and random swipes of the welds sealing the covered openings in the lower assemblies. This surveillance program will provide further assurance that there are no unanticipated releases of radioactivity to the environment.

The only contribution, therefore, to the annual dose equivalent to any member of the public is from direct radiation emanating from the storage facility. The storage facility would be shielded, as required, in order to limit the dose rate at the outside of the storage facility to 1.0 mR/hr. The resulting dose equivalent to an individual at the NNE site boundary (@ 2,200 feet) for a full year would be approximately 1.9×10^{-3} mrem, which is considered an insignificant contribution to the offsite dose. Furthermore, it is highly unlikely that an individual would be continuously exposed for a period of 1 year at the site boundary; therefore, the actual annual dose equivalent to any individual at this location will be lower than that given above.

4.4.7 ACCIDENT CONSIDERATIONS ASSOCIATED WITH ONSITE STORAGE

The primary concern associated with accidents involving the onsite storage of the old steam generator is the remote possibility for the release of radioactivity to the environment. The majority of this radioactivity is on the primary side surfaces of the lower assembly in the form of a protective corrosive film of metal oxides which is very adherent and refractory.

As discussed in Section 4.4.6, an additional measure of radioactivity confinement will be attained by welding cover plates over all pipe connection openings in the old steam generators.

Radioactivity could conceivably be released to the environment only if both of the conditions below occurred:

a. Radioactivity is dislodged from the primary side surfaces.

b. The lower assembly primary side boundary is breeched.

There are three mechanisms which could potentially dislodge the corrosion film:

a. Thermal shock

b. Chemical/corrosive attack

c. Mechanical shock

The old steam generator storage facility would provide a weathertight environment and minimize temperature extremes so that dislodging of corrosion by thermal shock is considered unlikely. Because the steam generators will be drained and sealed against moisture, chemical and corrosive attack is not likely to occur. The possibility of mechanical shock during storage is not great since the storage building would be an engineered structure and not subjected to general use. Even if thermal or mechanical shock is assumed, the tenacious nature of the corrosive film is such that it would not dislodge a significant amount of radioactivity.

In addition to the fact that it is highly unlikely for a significant amount of radioactivity to become dislodged from a primary side internal surface, breeching the lower assembly primary side boundary is considered an extremely remote possibility because of the minimum steel thickness of approximately 4 inches. Based on the above, it is concluded that there are no realistic accident scenarios which would result in the release of radioactivity from the generators during the onsite storage interval.

4.4.8 CONCLUSIONS

The steam generators will ultimately be disposed of in a licensed land burial site or decommissioned with the plant. ALARA considerations, economics, and burial site availability will be the factors determining the storage, handling, and shipping techniques employed.

4.5 PLANT SECURITY

Appropriate security measures will be implemented to ensure that the security program currently in effect at the site is not degraded during that portion of the steam generator repair program in which nuclear fuel is in the reactor vessel. Appropriate security measures of a reduced scope will be implemented for that portion of the program during which the nuclear fuel is completely contained in the spent fuel storage pool so as to ensure security of the fuel and fuel storage pool auxiliary systems. Pursuant to Section 2.790(d) 10 CFR 2, the specific security measures to be implemented will be addressed in a separate submittal withheld from public disclosure and are not included herein.

4.6 PLANT SYSTEMS LAYUP AND STARTUP METHODS

Because of the long outage that will be required to perform the steam generator repair, it will be necessary to take measures during layup and startup to minimize the introduction of corrosion products to critical systems or components. The methods currently considered for achieving this goal are presented in the following discussion.

In the secondary system the tube side of the feedwater heaters, including the drain coolers, gland seal condenser, and inter- and after-condensers will be placed in a wet layup with condensate quality water having 50 mg/l hydrazine (N_2H_4) and ammonia to maintain a pH range of 9.0 to 9.4. The system will be sampled and analyzed at least weekly for residual hydrazine and pH. Additional hydrazine will be added if the residual decreases to 45 mg/l. The feedwater recirculating system will be used to add and circulate the chemicals only; continuous use would result in aeration of the solution in the condenser. All heater bypass line valves are to be opened during recirculation periods. The condensate polishing demineralizers will be backflushed to clean the septums before being placed in wet layup with the rest of the feed system.

The moisture separator drain tank and the shell sides of the feedwater heaters will be drained as completely as possible for dry layup. All heaters, with the exception of the E1 and E2 heaters, which are located in the neck of the condenser and are nonisolable, will be purged weekly and blanketed with nitrogen.

During startup, the layup water on the tube side of the feedwater heaters will be drained and replaced with condensate quality water. The feedwater is to be recirculated through the condensate polishing demineralizers until normal startup chemistry specifications are met before allowing any water to enter the steam generators. The shell drains of the E5 and E6 heaters will be routed to the condenser until they meet normal feedwater chemistry specifications.

The steam side of the main turbine condenser will be drained to a level compatible with the feedwater system recirculating requirements. Plastic sheeting is to be placed over the turbine exhaust as it enters the condenser to serve as a vapor barrier isolating and protecting the turbine from moisture in the condenser. On the water side, the waterboxes will be drained and opened to allow the tubes to air dry. Before the tubes dry, the stainless steel air removal section tubes will be brushed or scraped to remove deposits, typical for these tubes, that may encourage pitting corrosion during the extended outage. Before startup, the steam side of the condenser will be hand-cleaned to remove loose oxide scale and debris, with particular attention being paid to the deaerating trays.

No special layup provisions are anticipated for the primary system. The primary coolant water will become air saturated beginning with the defueling operation; as with any refueling outage it is not feasible to control the oxygen concentration below saturation. For the replacement of the steam generators it will be necessary to drain the primary coolant system to a level below the hot and cold leg nozzles; at this point existing means for recirculating or for feed and bleed of the primary coolant will not be available for control. It is anticipated that the reactor vessel head will be blocked in place above the reactor vessel and sealed with sheeting during the construction activity; this will prevent contaminants from entering the vessel. It is likewise anticipated that some means will be employed to cover the vessel legs after they have been cut. The pressurizer vessel will be drained along with the rest of the primary system; no layup requirements are planned. The quench tank will be drained as completely as possible and purged weekly with nitrogen. During startup, the normal primary coolant chemistry limits will be observed.

Unless changed at a later date, the balance of the plant systems will be placed in a layup condition according to the respective normal plant operating procedures for refueling outages. Likewise, startup of these systems will follow normal procedures.

The instrumentation sensing lines on the primary coolant system will be backflushed into the system with demineralized water after the primary coolant level has been lowered. The sensing lines are normally filled with demineralized water during power operation, but the flushing will guard against the precipitation of boric acid that may have diffused into the lines.

The instrument sensing lines on the shell sides of the feedwater system will be drained wherever possible. Instrumentation included in the wet layup of the tube side of the feedwater system will require no special provisions.

4.7 QUALITY ASSURANCE

The quality assurance programs for Consumers Power Company, Bechtel Power Corporation, and Combustion Engineering, Inc. as applied to this project are described in this section.

4.7.1 CONSUMERS POWER COMPANY QUALITY ASSURANCE PROGRAM

The Consumers Power Company quality assurance program is described in the Consumers Power Company Quality Assurance Program Topical Report (CPC-1-A). This report is an integral part of the Consumers Power Company Quality Assurance Program Manual for Nuclear Power Plants and will be invoked for those quality assurance activities within Consumers' scope of responsibilities on this project.

4.7.2 BECHTEL POWER CORPORATION QUALITY ASSURANCE PROGRAM

Bechtel Power Corporation will perform its duties in accordance with Bechtel Topical Report BQ-TOP-1, Bechtel Quality Assurance Program for Nuclear Power Plants. This topical report will be invoked for quality assurance activities within Bechtel's scope of responsibilities on this project. Responsibility for the Palisades Steam Generator Repair Project has been assigned to the Ann Arbor office.

4.7.3 COMBUSTION ENGINEERING POWER SYSTEM GROUP NUCLEAR QUALITY ASSURANCE PROGRAM

The quality assurance program used by Combustion during the design and shop fabrication of the replacement steam generators is described in CE-NPD-210, Quality Assurance Program - A Description of the CE Nuclear Steam Supply System Quality Assurance Program.

4.8 REGULATORY GUIDE APPLICABILITY TO REPAIR PROGRAM

Section 2.1.4 discusses regulatory guide compliance during the manufacture of the steam generator units. This section discusses regulatory guide applicability to repair program activities other than steam generator manufacture.

a. Regulatory Guide 1.31 (5/77), Control of Ferrite Content in Stainless Steel Weld Metal

   Control of stainless steel welding complies with interim position on Regulatory Guide 1.31 (Branch

Technical Position MTEB 5-1, dated 11/24/75) except as discussed below.

1. Reference: Paragraph B.1.b of the Regulatory Guide

   Austenitic stainless steel welding filler materials used in the fabrication and installation of ASME Section III, Class 1, 2, and 3 components are controlled to deposit from 8 to 25% delta ferrite except for 309 and 309L welding filler materials, which are controlled to deposit from 5 to 15% delta ferrite and are used only for welding carbon or low alloy steel to austenitic stainless steel. Use of 309L welding filler material is further limited to the overlay deposit on the carbon or low alloy steel component nozzles or connecting pipe when postweld heat treatment is required.

   These limits for delta ferrite in austenitic stainless steel welding materials comply with Regulatory Guide 1.31 since the upper limit of 20% delta ferrite does not apply for welds that are not heat treated after welding (Paragraph 3b), except for solution heat treatment. Solution heat treatment, although not required after welding, is permitted in order to avoid sensitization.

   The procedure for determining the amount of delta ferrite in each heat or lot of austenitic stainless steel welding material does not comply with the regulatory guide. Determination of delta ferrite is in accordance with ASME Section III, Division 1, 1974 Edition, Paragraph NB-2433, except that an undiluted weld deposit is required for each heat of bare wire used with the gas metal arc (GMA) process.

2. Reference: Paragraph B.2 of the Regulatory Guide

   This paragraph is complied with for all tests and examinations required by ASME Section III, Division 1, 1974 Edition.

3. Reference: Paragraph B.3.a of the Regulatory Guide

   This paragraph is not complied with. Magnetic measurement of production welds for delta ferrite is unnecessary when austenitic stainless steel welding materials are controlled to deposit 8 to 25% delta ferrite based on chemistry, except for 309 and 309L welding materials, which are controlled to deposit 5 to 15% delta ferrite based on chemistry.

4. Reference: Paragraph B.3.b of the Regulatory Guide

   This paragraph is complied with for welding material certification.

5. Reference: Paragraphs B.4.a, b, and c of the Regulatory Guide

   These paragraphs are not complied with since measurement of production welds for delta ferrite is not performed.

b. Regulatory Guide 1.43 (5/73), Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components

   The restrictions of Paragraph C.1.a of the regulatory guide are observed. The remaining requirements of Regulatory Guide 1.43 are complied with when high heat input welding processes such as submerged arc-welding (SAW) and gas metal arc-welding (GMA) are used by Bechtel suppliers to clad SA 508 Class 2 forgings or to plate material of equivalent composition. Regulatory Guide 1.43 is not complied with when low heat input welding processes such as shielded metal arc-welding (SMAW) and gas tungsten arc-welding (GTAW) are used in the field by Bechtel or Bechtel subcontractors to clad completed welds and adjacent SA 508, Class 2 material during installation. It is noted in Part B of the regulatory guide that underclad cracking has not been observed in SA 508, Class 2 material welded with the "low heat input" processes.

c. Regulatory Guide 1.44 (5/73), Control of the Use of Sensitized Stainless Steel

1. General

Subject to the following statements, the use of unstabilized austenitic stainless steel complies with Regulatory Guide 1.44 for components which are part of the following:

(a) The reactor coolant pressure boundary

(b) Systems required for reactor shutdown

(c) Systems required for emergency core cooling

(d) Reactor vessel internals that are required for emergency core cooling

(e) Reactor vessel internals that are relied upon to permit adequate core cooling during any mode of normal operation or postulated accident conditions

2. Reference: Paragraph C.1 of the Regulatory Guide

Contamination of austenitic stainless steels (Type 300 series) by compounds that could cause stress-corrosion cracking is avoided during all stages of fabrication and installation. Except for trichlorotrifluoroethane (TCTFE), which meets the requirements of Military Specification MIL-C-8130 2B, cleaning is limited to solutions that contain not more than 200 ppm of chlorides. Rinsing or flushing is with water containing less than 200 ppm of chlorides. Special rinsing techniques are used to ensure complete removal of TCTFE when crevices or undrainable areas are present. Foreign substances in contact with austenitic stainless steel (die lubricants, penetrant materials, marking materials, masking tape, etc) are controlled so as not to contain more than 200 ppm of chlorides, or are removed immediately following the operation in which they are used. Crevices and undrainable areas are protected before using materials containing more than 200 ppm of chlorides. All substances in contact with austenitic stainless steel are removed before any elevated temperature treatment.

In the field, austenitic stainless steel components are stored clean and dry to prevent contamination. System hydrostatic tests are performed with water which contains less than 200 ppm of chlorides. The influent water quality during final flushing or preoperational testing of the completed system is at least equivalent to the quality of demineralized water as defined in ANSI N45.2.1 - 1973, Cleaning of Fluid Systems and Associated Components During the Construction Phase of Nuclear Power Plants.

3. Reference: Paragraph C.2 of the Regulatory Guide

All grades of austenitic stainless steels (Type 300 series) are required to be furnished in the solution heat treated condition before fabrication or assembly into components or systems. The solution heat treatment varies according to the applicable ASME or ASTM material specification.

4. Reference: Paragraph C.3 of the Regulatory Guide

All austenitic stainless steels are furnished in the solution heat treated condition in accordance with the material specification. For material solution heat treated by the material manufacturer, testing to determine susceptibility to intergranular corrosion is performed only when required by the material specification. During fabrication and installation, austenitic stainless steels are not permitted to be exposed to temperatures in the range of 800 to 1500F, except for welding and hot forming. Welding practices are controlled to avoid severe sensitization, as described in 6. below, and solution heat treatment in accordance with the material specification is required following hot forming in the temperature range 800 to 1500F. Unless otherwise required by the material specification, the maximum length of time for cooling from the solution heat treat temperature to below 800F is specified in the equipment specification. Corrosion testing in accordance with ASTM A 262-70, Practice A or E, or ASTM A 393 may be required if the maximum length of time for cooling below 800F is exceeded or the solution heat treat condition is in doubt.

5. Reference: Paragraph C.4 of the Regulatory Guide

   Use of low carbon (0.03% maximum) unstabilized austenitic stainless steel is not required since the reactor coolant meets the water chemistry requirements at temperatures over 250F.

6. Reference: Paragraph C.5 of the Regulatory Guide

   Heat treating austenitic stainless steel in the temperature range 800 to 1500F is not permitted and solution heat treatment is required following hot forming. Since sensitization is avoided, testing to determine susceptibility to intergranular attack is not performed.

7. Reference: Paragraph C.6 of the Regulatory Guide

   Welding practices are controlled to avoid severe sensitization in the heat-affected zone of unstabilized austenitic stainless steel as described below. Unless otherwise stated, the position applies to both Bechtel suppliers and subcontractors.

(a) Weld Heat Input

Bechtel controls weld heat input during field installation by using shielded metal arc-welding (SMAW) and gas tungsten arc-welding (GTAW) processes only and by limiting the size of electrodes for each process to 5/32-inch and 1/8-inch diameter maximum, respectively.

In addition to these two processes, Bechtel suppliers and subcontractors are permitted to use automatic submerged arc-welding (ASAW) and gas metal arc-welding (GMAW). Hardsurfacing operations are not included. When either automatic submerged arc-welding (ASAW) or gas metal arc-welding (GMAW) is used, or shielded metal arc-welding (SMAW) or gas tungsten arc-welding (GTAW) is used with electrodes larger than those specified above, testing in accordance with ASTM A 262, Practice A or E is required unless welding is followed by solution heat treatment.

(b) Interpass Temperature

The interpass temperature is controlled so as not to exceed 350F.

(c) Carbon Content

Susceptibility to sensitization is reduced significantly by selecting material with the lowest reported carbon content.

(d) Solution Heat Treatment

Solution heat treatment in accordance with the material specification, although not required after welding, is permitted in order to avoid sensitization. Severe sensitization is avoided by not permitting heat treatment in the temperature range of 800 to 1500F following welding. This requires a special technique when welding stainless steel safe ends (transition pieces) to carbon or low-alloy steel component nozzles or piping. Specifically, a 309L stainless steel overlay or an Inconel weld overlay is deposited on the component and the component is postweld heat treated. Following final postweld heat treatment of the component, the stainless steel safe end is welded to the weld overlay using 308 or 308L austenitic stainless steel or Inconel type welding materials.

Intergranular corrosion testing is not performed on a routine basis. Performing an intergranular corrosion test for each welding procedure serves no useful purpose when welding practices and reactor coolant water chemistry are controlled as described above.

d. Regulatory Guide 1.48 (5/73), Design Limits and Loading Combinations for Seismic Category I Fluid System Components

Regulatory Guide 1.48 is applicable to the primary head drain lines. These lines shall be analyzed to the requirements of the 1977 edition of the ASME Boiler and Pressure Vessel Code, Section III. The 3/4-inch NPS Class 1 portion of these lines shall be designed in accordance to Subsection NC as per paragraph NB-3630(d)(1).

e. Regulatory Guide 1.50 (5/73), Control of Preheat Temperature for Welding of Low-Alloy Steel

Preheat for welding of low-alloy steel is controlled in accordance with Regulatory Guide 1.50, except as described below.

1. Reference: Paragraph C.1.a of the Regulatory Guide

The regulatory position is complied with when impact testing in accordance with Subarticle 2300 of Section III, Division 1, of the ASME Boiler and Pressure Vessel Code is required. The maximum interpass temperature shall be 500F unless otherwise specified. When impact testing is not required, specification of a maximum interpass temperature in the welding procedure is not necessary in order to ensure that the required mechanical properties are met. The minimum preheat temperatures of Appendix D of Section III of the ASME Boiler and Pressure Vessel Code are required to be met regardless of whether impact testing is required or not.

2. Reference: Paragraph C.1.b of the Regulatory Guide

   The regulatory position is not complied with since the welding procedure qualification requirements of Sections III and IX of the ASME Boiler and Pressure Vessel Code are considered to be more than adequate.

3. Reference: Paragraph C.2 of the Regulatory Guide

   The regulatory position is complied with for Class 1 pressure vessels with nominal thicknesses greater than 1 inch. Maintenance of preheat beyond completion of welding until postweld heat treatment (PWHT) is not required for thinner sections, since experience has indicated that delayed cracking in the weld or heat affect zone is not a problem.

   Current usage of low-alloy steel in piping, pumps, and valves is minimal and normally is limited to Class 3 construction. When low-alloy steel piping, pumps, and valves are used, preheat is maintained until welding is complete but not until postweld heat treatment (PWHT) is performed, since the conditions which cause delayed cracking in the weld or heat affected zone (HAZ) are not present.

4. Reference: Paragraph C.4 of the Regulatory Guide

   The regulatory position is complied with when the positions stated above are not met.

f. Regulatory Guide 1.71 (12/73), Welder Qualification for Areas of Limited Accessibility

1. Reference: Paragraph C.1 of the Regulatory Guide

   Performance qualifications for personnel who weld under conditions of limited access, as defined in Regulatory Position C.1, are maintained in accordance with the applicable requirements of ASME Sections III and IX. Additionally, responsible site supervisors are required to assign only the most highly skilled welders to limited access welding. Of course, welding conducted in areas of limited access is subjected to the required nondestructive testing and no waiver or relaxation of examination methods or acceptance criteria because of the limited access is permitted.

2. Reference: Paragraph C.2 of the Regulatory Guide

   Requalification is required when any of the essential variables of ASME Section IX are changed, or when any authorized inspector questions the ability of the welder to perform satisfactorily the requirements of ASME Sections III or IX.

3. Reference: Paragraph C.3 of the Regulatory Guide

   Production welding is monitored and welding qualifications are certified in accordance with a. and b. above.

4.9 SCALE MODEL OF THE PALISADES PLANT CONTAINMENT

A scale model of the containment has been developed to be utilized in the licensing, design, construction, and startup phases of the Palisades Plant steam generator repair program. The model extends the full height of the containment with detailed presentation of those areas that are directly affected by the steam generator repair and associated systems modifications. The scale is 1/2-inch to 1 foot.

The three-dimensional presentation of the model will provide many benefits to the repair program, including the following:

a. The repair scheme will be examined in detail to ensure that the proper clearances and replacement sequences are maintained.

b. The model will be used to ensure that the repair program radiation exposures are ALARA (see Section 4.3.5.1).

c. Communication will be improved on all levels.

d. The model will provide a management tool for planning and executing construction operations.

PALISADES PLANT SGRR

TABLE 4.2-1

(SHEET 1)

ELECTRICAL EQUIPMENT AND INSTRUMENTS
TO BE TEMPORARILY REMOVED

Number	Item	Description
1.	BO9	480 V motor control center
2.	HT-1812	Containment humidity transmitter
3.	TE-1815	Containment temperature element
4.	HT-1815	Containment humidity transmitter
5.	TE-1812	Containment temperature element
6.	RE-2316	Fuel handling area radiation monitor
7.	RA-2316	Fuel handling area radiation monitor
8.	Q-15	Radiation monitor speaker (count-rate)
9.	RA-2317	Fuel handling area radiation monitor
10.	RE-2317	Fuel handling area radiation monitor
11.	J-302	Combustion's refueling disconnect panel
12.	V-49A,B	Control rod drive mechanism blowers
13.	J-301	Refueling disconnect panel
14.	U-113	Evacuation siren
15.	TV-2	Security camera
16.	J-395	Junction box for security TV
17.	L-28	Lighting panel and flood light dimmer
18.	X38	Lighting transformer
19.	POS-3040	Safety Injection Tank T-82A nitrogen valve position switch
20.	POS-3067	Safety Injection Tank T-82A vent valve position switch
21.	POS-3044	Safety Injection Tank T-82B nitrogen valve position switch
22.	POS-3065	Safety Injection Tank T-82B vent valve position switch
23.	POS-3050	Safety Injection Tank T-82D nitrogen valve position switch
24.	POS-3051	Safety Injection Tank T-82D vent valve position switch
25.	CV 652	Electrical cable tray
26.	CP 652	Electrical cable tray
27.	CV 700	Electrical cable tray
28.	CP 700	Electrical cable tray
29.	CU 663	Electrical cable tray
30.	CK 663	Electrical cable tray

PALISADES PLANT SGRR

TABLE 4.2-1

(SHEET 2)

ELECTRICAL EQUIPMENT AND INSTRUMENTS
TO BE TEMPORARILY REMOVED

Number	Item	Description
31.	C 4070	Electrical conduit
32.	C 3010	Electrical conduit
33.	C 3011	Electrical conduit
34.	C 3012	Electrical conduit
35.	C 3013	Electrical conduit
36.	C 3014	Electrical conduit
37.	C 3007	Electrical conduit
38.	C 3008	Electrical conduit
39.	C 4003	Electrical conduit
40.	C 4004	Electrical conduit
41.	C 4020	Electrical conduit
42.	C 4021	Electrical conduit
43.	C 4050	Electrical conduit
44.	C 4051	Electrical conduit
45.	C 4054	Electrical conduit
46.	C 4055	Electrical conduit
47.	C 4060	Electrical conduit
48.	C 4061	Electrical conduit
49.	C 4064	Electrical conduit
50.	C 4065	Electrical conduit
51.	C 2350	Electrical conduit
52.	C 3001	Electrical conduit

PALISADES PLANT SGRR

TABLE 4.2-2

480 V MOTOR CONTROL CENTER B09 LOAD TABULATION

Number	Item	Description
1.	MO 3041	Safety Injection Tank T-82A outlet valve
2.	H6	Fuel tilting device
3.	-	Refueling machine
4.	X38	Lighting transformer
5.	MO 3045	Safety Injection Tank T-82B outlet valve
6.	SV 1043B	Pressurizer power relief valve
7.	MO 3049	Safety Injection Tank T-82C outlet valve
8.	MO 3052	Safety Injection Tank T-82D outlet valve
9.	-	Fuel transfer machine
10.	L1	Reactor building crane
11.	V66	Air room recirculation fan
12.	-	Welding receptacles
13.	V16A and V16B	Pressurizer shed cooling fans
14.	V49A	Control rod drive mechanical cooling fan

PALISADES PLANT SGRR

TABLE 4.2-3

(SHEET 1)

TEMPORARY AND PERMANENT ELECTRICAL LOADS ASSOCIATED WITH STEAM GENERATOR REPAIR AND THEIR CORRESPONDING POWER SOURCE

Load	Power Source: Normal	Power Source: During Repair Program
Miscellaneous rigging equipment inside containment	-	Temporary 480 V power distribution panels inside the containment
Welding receptacles inside the containment	480 V Motor Control Center B09	Temporary 480 V power distribution panels inside the containment
Containment cooler recirculation fans:		
Fans V1A, V2A, V3A	480 V Load Center B12	No change
Fan V4A	480 V Load Center B11	No change
Fans V1B, V2B, V3B	480 V Motor Control Center B03	No change
Containment purge exhaust Fan V-35	480 V Load Center B11	No change
Main exhaust Fan V6A	480 V Load Center B12	No change
Main exhaust Fan V6B	480 V Load Center B11	No change
Containment purge supply Fan V-5	480 V Load Center B12	No change
Air space purge Fan V-46	480 V Motor Control Center B07	No change

PALISADES PLANT SGRR

TABLE 4.2-3

(SHEET 2)

TEMPORARY AND PERMANENT ELECTRICAL LOADS
ASSOCIATED WITH STEAM GENERATOR REPAIR
AND THEIR CORRESPONDING POWER SOURCE

Radwaste area supply Fan V-10	480 V Motor Control Center B07	No change
Radwaste area exhaust Fan V-14A	480 V Motor Control Center B07	No change
Radwaste area exhaust Fan V-14B	480 V Motor Control Center B08	No change
Waste gas Compressor C50A	480 V Motor Control Center B07	No change
Waste gas Compressor C50B	480 V Motor Control Center B08	No change
Air room recirculation Fan V6	480 V Motor Control Center B09	Temporary local starter supplied from temporary 480 V power distribution panels inside the containment
Spent fuel pool cooling Pump P51A	480 V Motor Control Center B07	No change
Spent fuel pool cooling Pump P51B	480 V Motor Control Center B08	No change
Fuel pool recirc booster Pump P82	480 V Motor Control Center B08	No change
Service water Pumps P7A and P7C	2400 V Bus 1D	No change

PALISADES PLANT SGRR

TABLE 4.2-3

(SHEET 3)

TEMPORARY AND PERMANENT ELECTRICAL LOADS ASSOCIATED WITH STEAM GENERATOR REPAIR AND THEIR CORRESPONDING POWER SOURCE

Service water Pump P7B	2400 V Bus 1C	No change
Component cooling Pumps P52A and P52C	2400 V Bus 1C	No change
Component cooling Pump P52B	2400 V Bus 1D	No change
Screen wash Pump P4	480 V Bus B14	No change
Instrument and service air Compressors C2A and C2C	480 V Bus B11	No change
Instrument and service air Compressor C2B	480 V Bus B12	No change
High-pressure air Compressor C6A (to supply air-operated valves)	480 V Motor Control Center B07	No change
High-pressure air Compressor C6B (to supply air-operated valves)	480 V Motor Control Center B08	No change
Fire Pump P9A	480 V Bus B13	No change
Fire system jockey Pump P-13	480 V Motor Control Center B05	No change
Diesel-driven fire Pump P9B controls and battery charger	120/208 V lighting Panel L20 (supplied from 480 V Motor Control Center B05)	No change

PALISADES PLANT SGRR

TABLE 4.2-3

(SHEET 4)

TEMPORARY AND PERMANENT ELECTRICAL LOADS ASSOCIATED WITH STEAM GENERATOR REPAIR AND THEIR CORRESPONDING POWER SOURCE

2400 V emergency diesel Generator 1-1 auxiliaries	480 V Motor Control Center B01	No change
2400 V emergency diesel Generator 1-2 auxiliaries	480 V Motor Control Center B02	No change
2400 V emergency diesel Generators 1-1 and 1-2 controls	120 V dc Load Centers D10 and D20	No change
2400 V and 480 V load center ACBs controls	125 V dc Load Centers D10 and D20	No change
Control power supply for control panels (various)	125 V dc Load Centers D10, D20	No change
Power supply for annunciators (various)	125 V dc Load Centers D10, and D20	No change
Radiation monitoring system	120 V ac preferred supply Panels Y10, Y20, Y30, and Y40	No change
Reactor building Crane L1	480 V Motor Control Center B09	Temporary 480 V power distribution panels
Containment jib crane	480 V Motor Control Center B04	No change

PALISADES PLANT SGRR

TABLE 4.2-3

(SHEET 5)

TEMPORARY AND PERMANENT ELECTRICAL LOADS ASSOCIATED WITH STEAM GENERATOR REPAIR AND THEIR CORRESPONDING POWER SOURCE

Containment lighting	120/208 V lighting distribution Panel L28 (supplied from Motor Control Center B09)	Temporary lighting distribution panel
	120/208 V lighting distribution Panel L27 (supplied from Motor Control Center B08)	No change
	120/208 V lighting distribution Panel L29 (supplied from Motor Control Center B02)	No change
Reactor building lighting (other than containment building)	120/208 V lighting distribution Panel L24 (supplied from Motor Control Center B07)	No change
	120/240 V lighting distribution Panel L40 (supplied from Motor Control Center B04)	No change

TABLE 4.2-3

(SHEET 6)

TEMPORARY AND PERMANENT ELECTRICAL LOADS ASSOCIATED WITH STEAM GENERATOR REPAIR AND THEIR CORRESPONDING POWER SOURCE

Motor-operated valves (various)	Motor control centers (various)	No change
Solenoid valves (various)	125 V dc Load Centers D10, D20	No change
Fuel handling area supply Fan V7	480 V Motor Control Center B07	No change
Fuel handling area exhaust Fan V8A	480 V Motor Control Center B07	No change
Fuel handling area exhaust Fan V8B	480 V Motor Control Center B08	No change
125 V dc battery Chargers D15 and D18	480 V Motor Control Center B01	No change
125 V ac battery Chargers D16 and D17	480 V Motor Control Center B02	No change

TABLE 4.3-1

(SHEET 1)

TYPICAL PORTABLE SURVEY INSTRUMENT SPECIFICATIONS

I RADGUN (A68 - 10 KG-SR - VICTOREEN)

1. The radgun is a highly sensitive portable instrument designed to measure beta and gamma radiation dose rate from background levels to 10,000 R/hr.

2. The detector is a Neher-White ionization chamber filled with 10 atmospheres of argon. The greater mass of argon at pressure makes the detector more efficient than unpressurized air chambers.

3. The detector chamber has a beta window which can be covered by the beta shutter. Because of the chamber thickness needed to contain the pressurized gas, it is not very efficient for beta.

4. The radgun contains a 60 μCi Kr 85 check source to check the low scale operation.

5. The radgun has three logarithmic ranges, 0.01 to 10 mr/hr, 0.01 to 10 R/hr and 10 to 10,000 R/hr.

II MODEL E-520 (EBERLINE)

1. The E-520 is a portable beta-gamma survey instrument.

2. Two different detector tubes are utilized: both are GM tubes.

3. The external tube is located in the hand probe and used for lower range detection (0-0.2, 0-2, 0-20, 0-200 mr/hr). The other tube is located within the case and used in a range of 0 to 2,000 mr/hr.

4. The tube in the external probe has a thin wall (30 mg/cm^2), and is used with a rotating beta shield.

TABLE 4.3-1

(SHEET 2)

TYPICAL PORTABLE SURVEY INSTRUMENT SPECIFICATIONS

III MODEL R0-2 EBERLINE

1. The R0-2 is a box-shaped, top handled, portable beta-gamma survey instrument.

2. The detector is an unpressurized ion chamber.

3. The detector has a thin, one mil, beta window with a sliding beta shield. This instrument is a good choice for beta measurements.

4. The R0-2 has a basic scale of 0-5 with selector switch positions of 0-5 mr/hr, 0-50 mr/hr, 0-500 mr/hr, and 0-5,000 mr/hr.

IV MODEL PAC-1 SAGE ALPHA COUNTER (EBERLINE)

1. The PAC-1 SAGA is a box-shaped portable alpha and gamma detecting instrument used with the external AC-3 detector. This instrument is primarily for alpha contamination surveys.

2. The gamma detector is an internal, small GM tube. This is only used with the selector switch in the "2r" position reading the bottom meter scale of 0-2 r/hr. This instrument should not be used to survey gamma fields. The gamma reading is basically to determine the alpha background reading.

3. The AC-3 alpha detector has a ZnS (Ag) scintillation crystal used with a photo multiplier tube. The window thickness is 1.5 mg/cm^2 of aluminized mylar with an active area of 59 cm^2.

4. Alpha range meter has basic upper scale of 0 - 2 K cpm used with selector switch multipliers of x1.0, x10, x100, and x1K, which gives a total range of 0 - 2,000,000 cpm.

5. There is a reset button on the handle for returning the meter reading to zero for rapid

TYPICAL PORTABLE SURVEY INSTRUMENT SPECIFICATIONS

recheck of readings and decreasing recovery time when changing scales.

V MODEL PIC-6A

1. Th PIC-6A is a portable instrument which measures the dose rate from gamma radiation.

2. The detecting element is a gas-filled ionization chamber operating in the proportional (gas multiplication) region.

3. Six decades of dose rate, from 1 mr/hr to 1000 r/hr, are measured in two ranges of three decades each.

4. A beta window in the bottom of the instrument provides for the detection of energetic beta particles.

VI CD-V700 (VICTOREEN)

1. The CD-V700 is a box-style, portable survey instrument with an external GM probe. It is primarily used for low level gamma surveys or for beta contamination detection.

2. The detector is an external, plug in GM tube. The detector probe has a beta window with a rotating beta shield.

3. Instrument range - the basic scale reads 0 - 0.5 mr/hr on the top and 0 - 300 cpm on the bottom. Both scales are used with the range switch positions x1, x10, and x100.

PALISADES PLANT SGRR

TABLE 4.3.2

(SHEET 1)

MAN-REM ASSESSMENT FOR REPLACEMENT

(The manhour and man-rem estimates have been revised refer to Table C-1-1 to C-1-5.)

	Work Area	Estimated (1) Manhours in Radiation Field	Average Radiation Field (rem/hr)	Area (1) Man-Rem Dose (Man-Rem) Unshielded	Reduction Factor (Shielding and/or Decontamination)	Area Man-Rem Dose (Man-Rem)
1.	Outside of power plant building but within security fence	213,300	$.5 \times 10^{-5}$	1.06	1.0	1.06
2.	Checking in and out through security and health pysics, as well as time spent suiting up, cleaning up, and moving to and from work area for personnel working in radioactive areas	55,300	.0025	138.25	1.0	138.25
3.	Inside containment near new construction opening	3,550	0.001	3.55	1.0	3.55
4.	Within 6 feet of outside of reactor coolant pipe or bottom of steam generator before removal of steam generators	5,050	0.03	151.5	1.0	151.5
5.	Within 6 feet of outside of reactor coolant pipe after steam generator's removal	19,100 23,400 A1	0.01	191.0 234.0 A1	1.0	191.0 234.0 A1
6.	Within 6 feet of outside of reactor coolant pipe or bottom of steam generators with partial exposure to inside of reactor coolant pipe before steam generator's removal	750	1.0	750.0	0.05	37.5
7.	Within 6 feet of outside of reactor coolant pipe with partial exposure to inside of reactor coolant pipe after steam generator's removal	4,400	1.0	4,400	0.05	220.0
8.	Inside reactor coolant pipe	4,500 200 A1 300 A2	9.0	40,500 1,800 A1 2,700 A2	0.1	4070.0 180.0 A1 270.0 A2
9.	Low radiation area within containment	41,250 4,200 A2	0.001 .005 A2	41.25 21.0 A2	1.0	41.25 21.0 A2

PALISADES PLANT SGRR

TABLE 4.3.2

(SHEET 2)

MAN-REM ASSESSMENT FOR REPLACEMENT

(The manhour and man-rem estimates have been changed refer to Table C-1-1 to C-1-5.)

Work Area	Estimated (1) Manhours in Radiation Field	Average Radiation Field (rem/hr)	Area (1) Man-Rem Dose (Man-Rem) Unshielded	Reduction Factor (Shielding and/or Decontamination)	Area Man-Rem Dose (Man-Rem)
10. Within 6 feet of top half of original steam generators	1,100	0.005	5.5	1.0	5.5
11. Within 6 feet of top half of new steam generators	8,050	0.001	8.05	1.0	8.05
12. Operating floor of containment	15,800	0.005	79.0	0.2	15.8
13. Inside containment, above polar crane	1,150	0.001	1.15	1.0	1.15
14. Auxiliary building near clean resin tank and cooling water tank	750	0.001	0.75	1.0	0.75
15. Auxiliary building near blowdown tank	6,700	0.001	6.7	1.0	6.7
16. Spent fuel pool floor	2,750	0.005	13.75	1.0	13.75
17. Within 6 feet of the bottom half of new steam generators	3,700	0.010	37.0	1.0	37.0
18. Within 6 feet of the outside of the reactor vessel	50	1.0	50.0	1.0	50.0
19. Next to the existing steam generators outside of the containment	1,000	0.02	20.0	1.0	20.0
				Total	4,993
				A1	1,666
				A2	1,193

Note:

(1) The three man-rem estimates given for work area 8 (inside of reactor coolant pipe) are presented because of three welding techniques under consideration. The ALARA considerations will be the factor determining which technique is eventually used. The total time estimated for cutting, welding, and inspecting inside the reactor coolant pipes is 4,500 manhours. One alternative (A1) is a technique, presently being investigated for feasibility, utilizing manual welding from the outside of the piping. Using this method, only 200 manhours out of a total of 4,500 would be required inside primary coolant piping. The remaining 4,300 manhours would be spent within 6 feet of the reactor coolant pipe (work area 5). The second alternative (A2) is an automatic welding technique for the cladding from inside the piping utilizing remote viewing. This method required 300 manhours inside the piping, and the remaining 4,200 manhours would be spent in a low radiation area in the containment (work area 9).

PALISADES PLANT SGRR

TABLE 4.4-1

ACTIVATED CORROSION PRODUCTS AFTER SHUTDOWN
ACTIVITY IN (CURIES x 10^{-6})/(2" SAMPLE)[1] [2]

Isotope	0 days	42 days	140 days	200 days	470 days
Cr-51	5.85	2.06	0.807	0.04	0.0
Mn-54	1.33	1.23	1.00	0.86	0.47
Co-57	0.60	0.54	0.428	0.36	0.18
Co-58	341.10	228.50	117.18	48.95	3.56
Fe-59	3.08	1.70	0.733	0.14	0.0
Co-60	6.16	6.06	5.85	5.73	5.18
Nb-95	0.31	0.14	0.0594	0.01	0.0
Zr-95	0.37	0.24	0.116	0.04	0.0
Total	358.8	240.47	126.24	56.13	9.39

Notes

(1) The activities established are an approximation which assumes that the majority of the activity is concentrated in the tubesheet.

(2) The following technique was used to find approximate activity per steam generator at the time of removal (~200 days):

3.5×10^5 inches of tube/tube sheet

(3.5×10^5 inches of tube)(56.13×10^{-6} Ci/2 inches of tube) = 9.82 curries/steam genrator at 200 days

This should increase by a factor of 2-3 for additional operation of from 3 to 5 effective full power year (Reference 3).

Therefore: (9.82 curies) x 3 $\simeq$ 29.5 Ci/steam generator at ~200 days

PALISADES PLANT SGRR

TABLE 4.4-2

MAN-REM ASSESSMENT FOR OFFSITE DISPOSAL

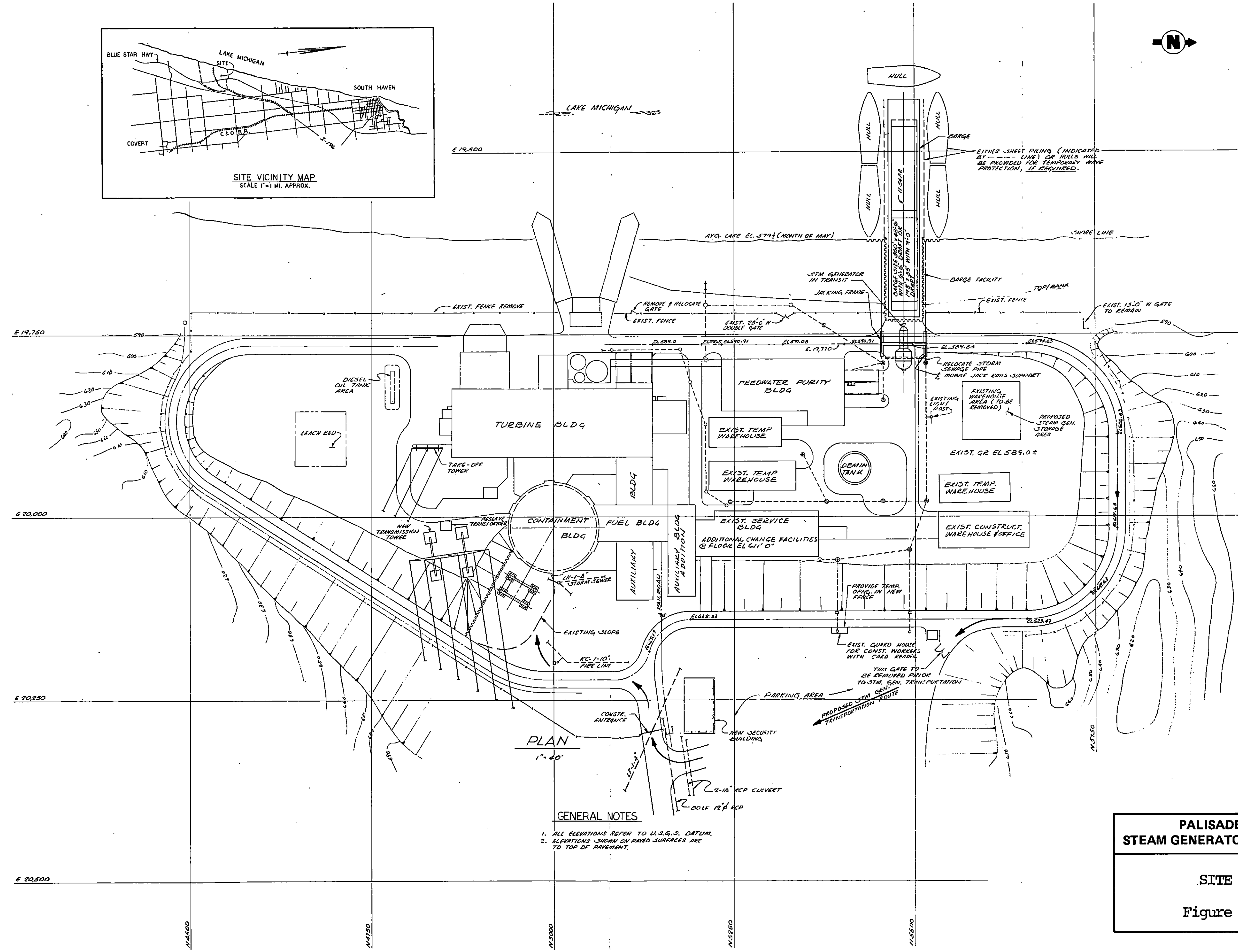
	Estimated Manhours	Radiation Field (Rem/hr)	Man-Rem (3)
Barge Shipment (near term)	220 (1)	.015-.020	1-5
Cutup and shipment by truck and/or rail (near term)	37,500 (2)	.015-020	575-750

NOTES:

(1) Assumes 10% contact with steam generators for length of trip.

(2) This estimate is based on 750 manhours required for cutting primary coolant piping (Table 4.3-2, Area 6). The manhour estimate was increased by a factor of 50 to account for the increased cutting time to reduce steam generators to the minimum of 90 pieces required for shipment.

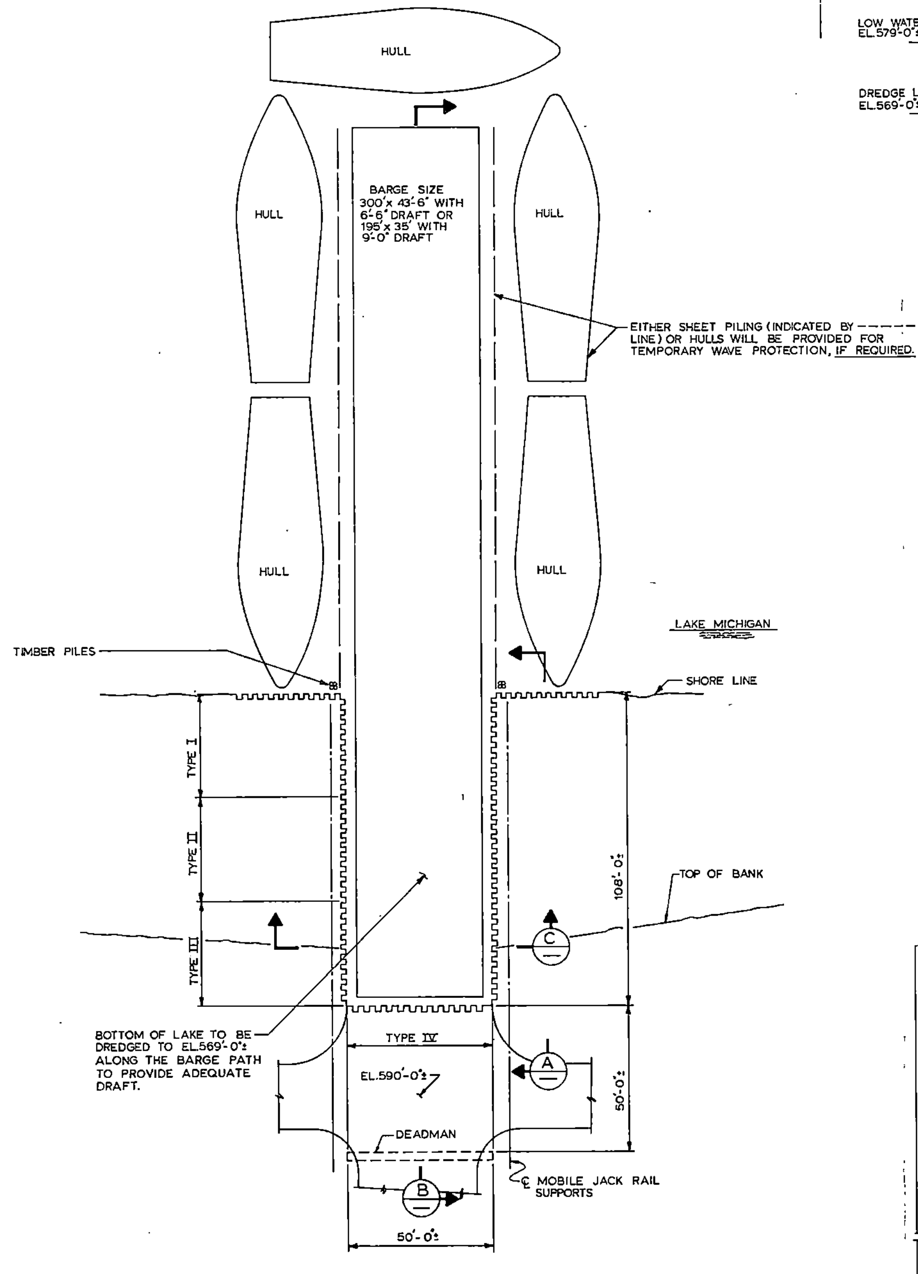
(3) Range in man-rem was included to reflect possible variations in manhour requirements and radiation field uncertainties.



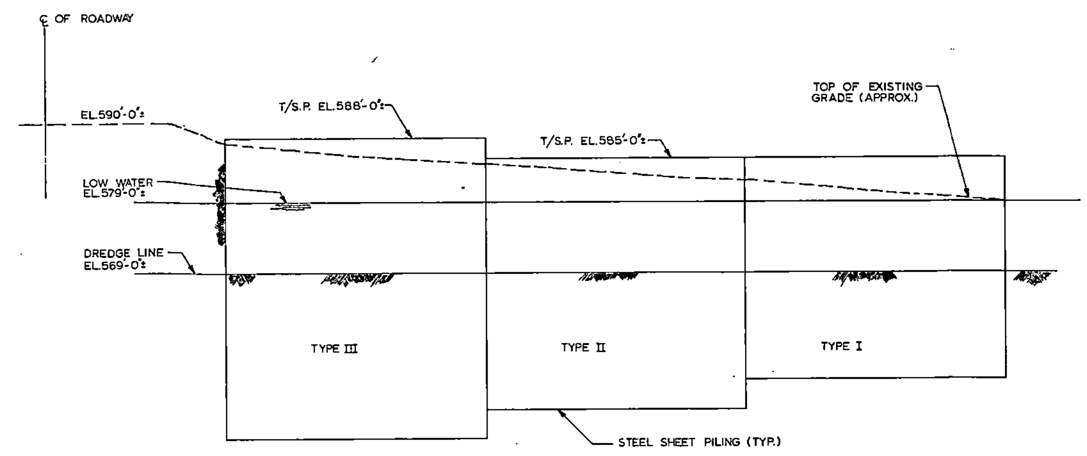
PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

SITE PLAN

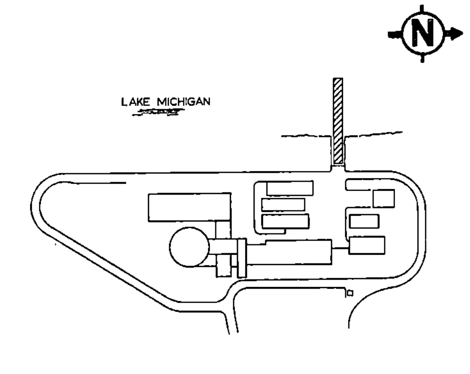
Figure 4.1-1



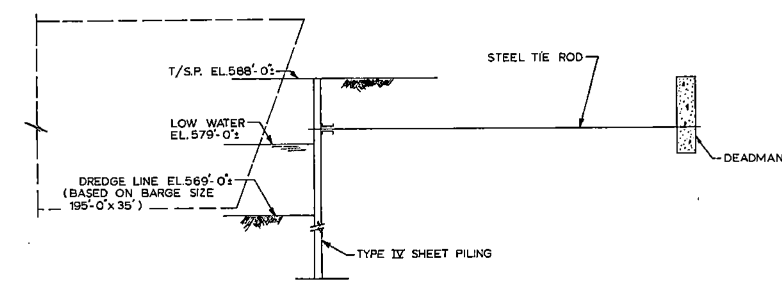
BARGE SLIP
SCALE: 1"=20'-0"



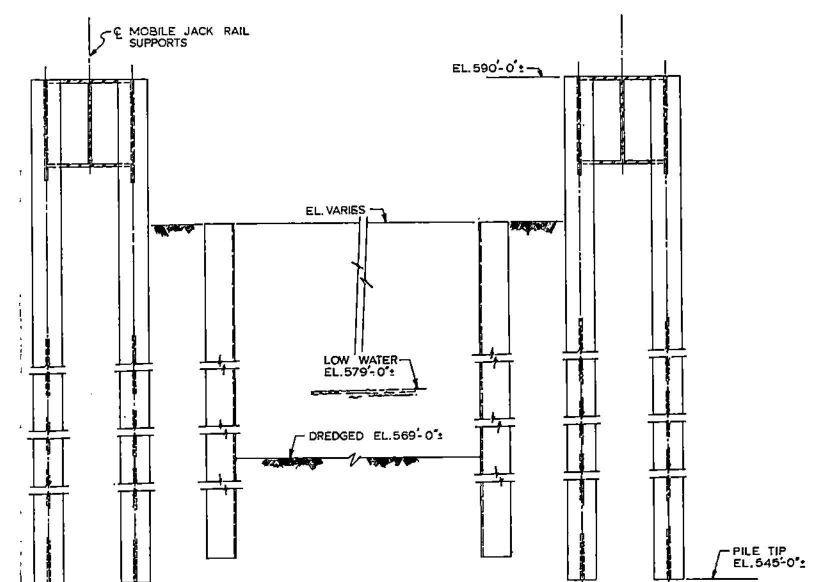
SECTION A
SCALE: 1/8"=1'-0"



KEY PLAN



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

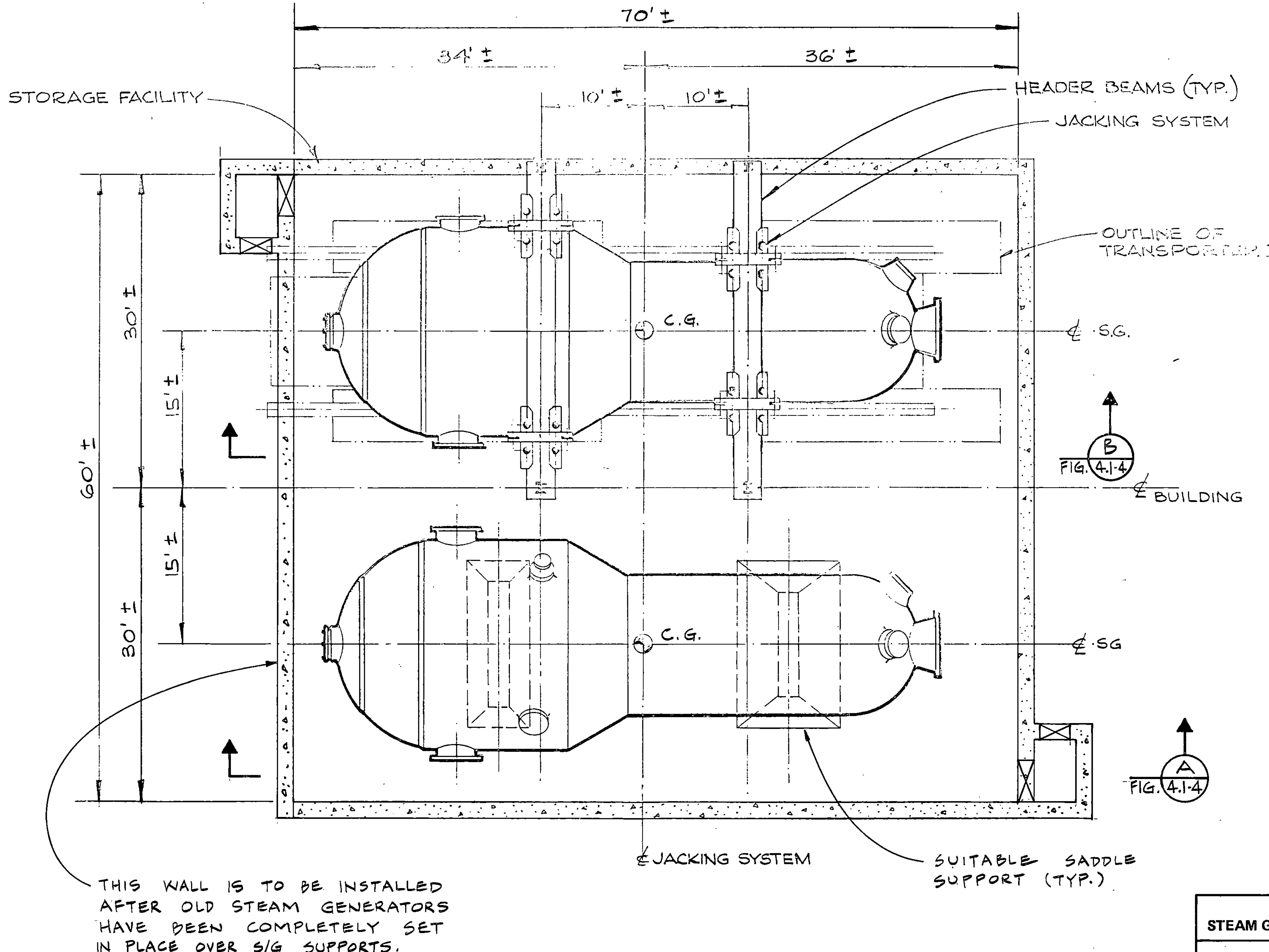


SECTION C
SCALE: 1/2"=1'-0"

PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

BARGE SLIP

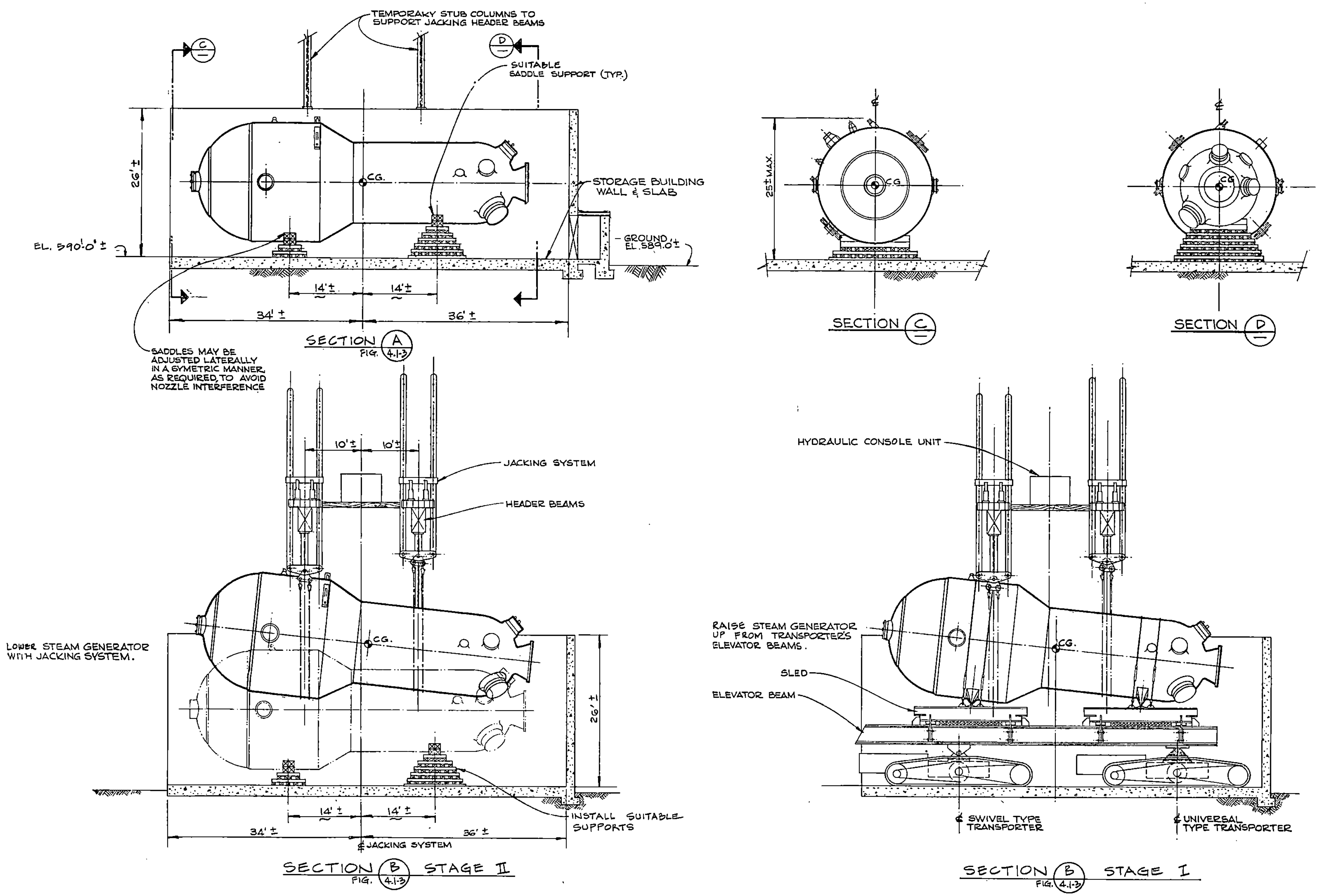
Figure 4.1-2



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

OLD STEAM GENERATOR STORAGE FACILITY PLAN VIEW

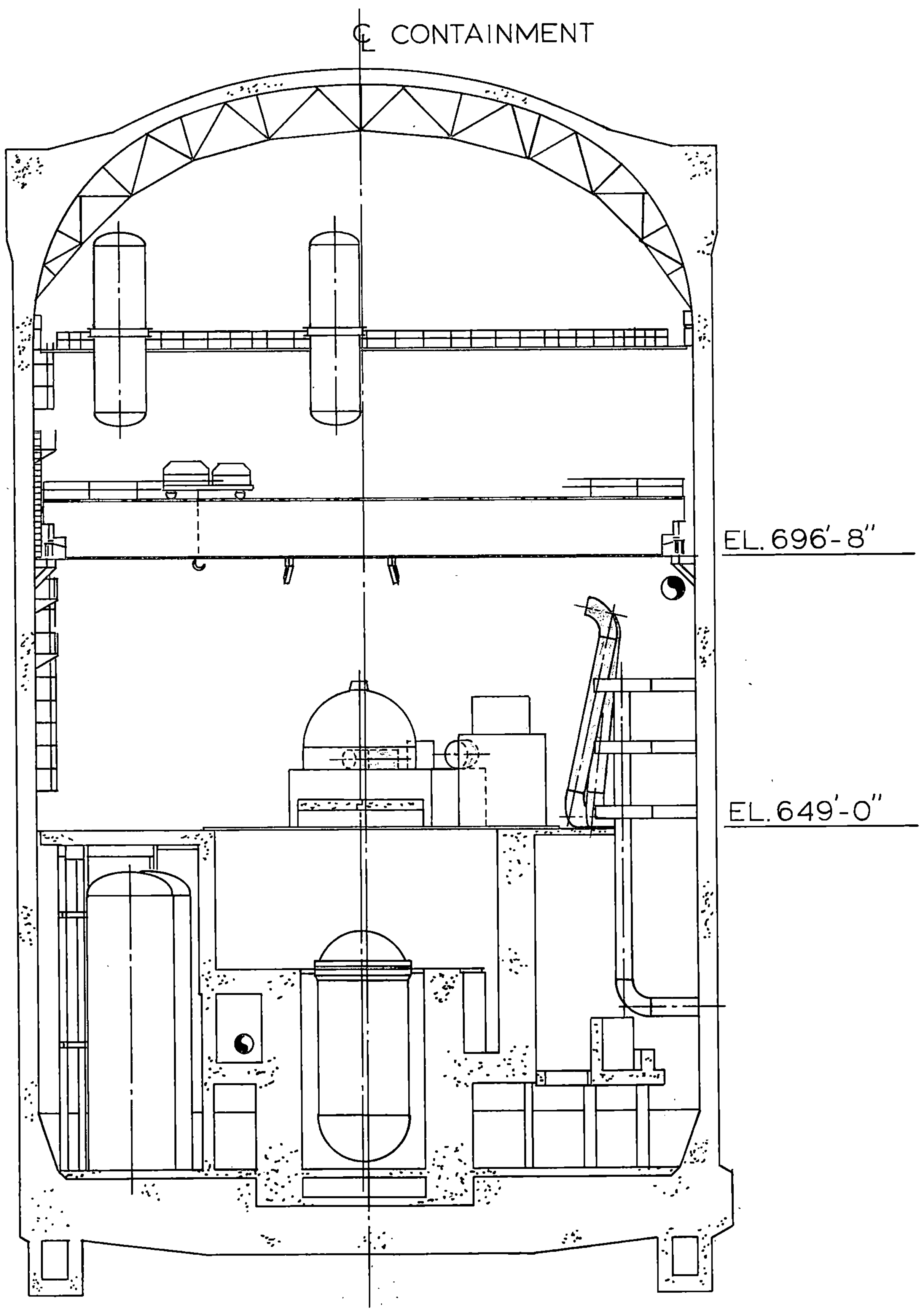
Figure 4.1-3



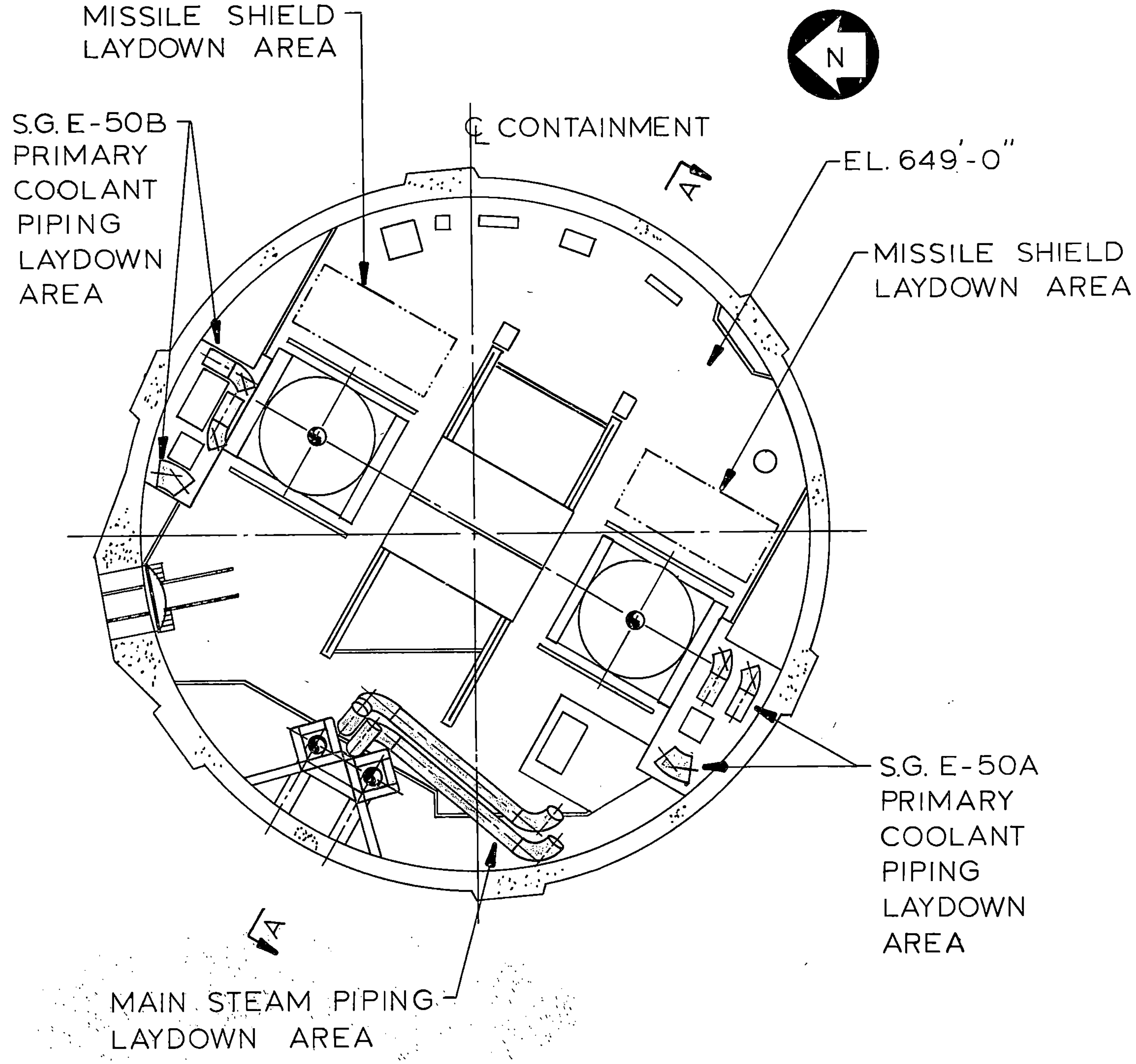
PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

OLD STEAM GENERATOR STORAGE FACILITY SECTIONS & DETAILS

Figure 4.1-4

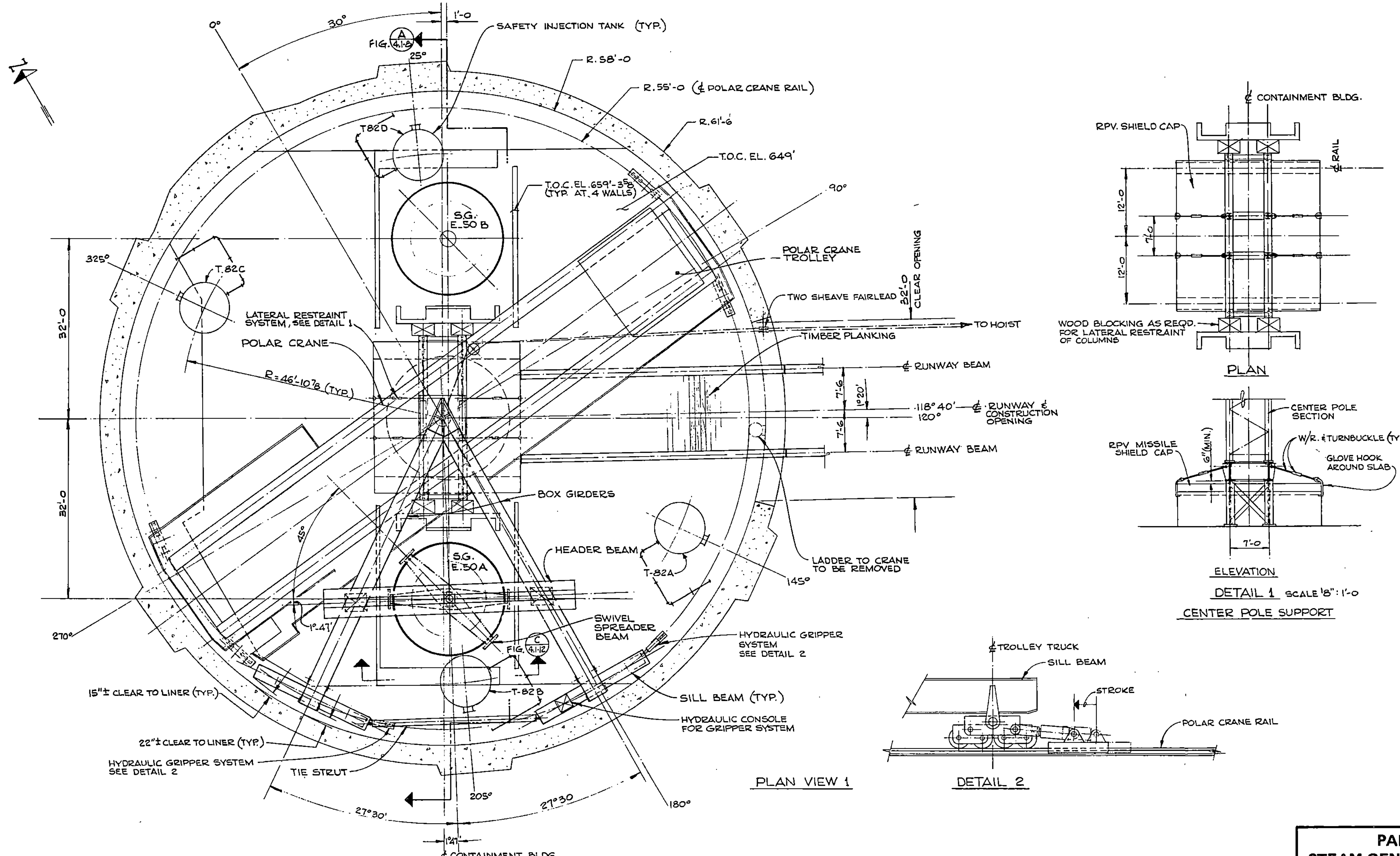


SECTION "A-A"



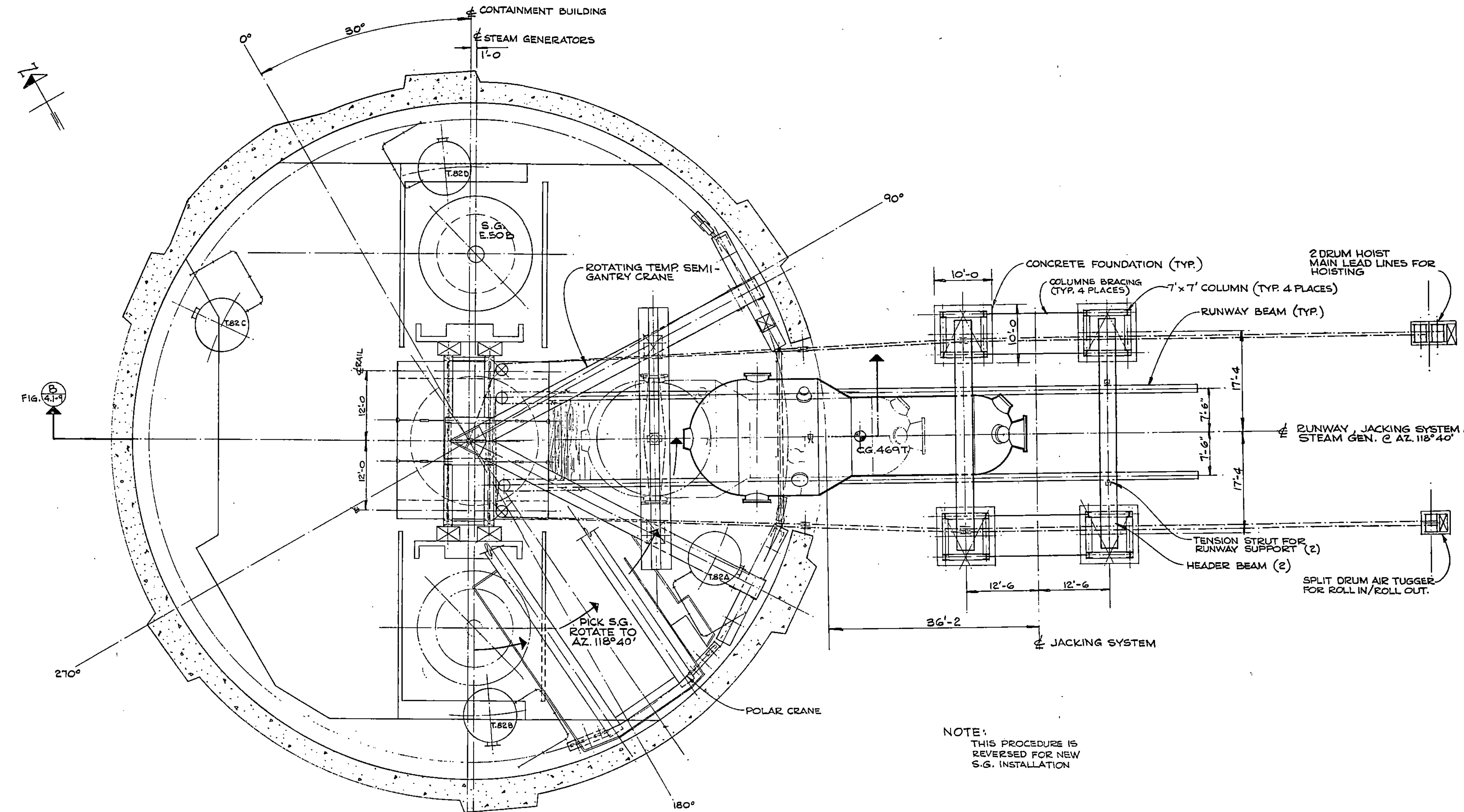
PLAN ABOVE EL. 649'-0"

PALISADES PLANT STEAM GENERATOR REPAIR REPORT
CONTAINMENT LAYDOWN AREAS
Figure 4.1-5



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

GENERAL ARRANGEMENT
PLAN VIEW, SH. 1
Figure 4.1-6

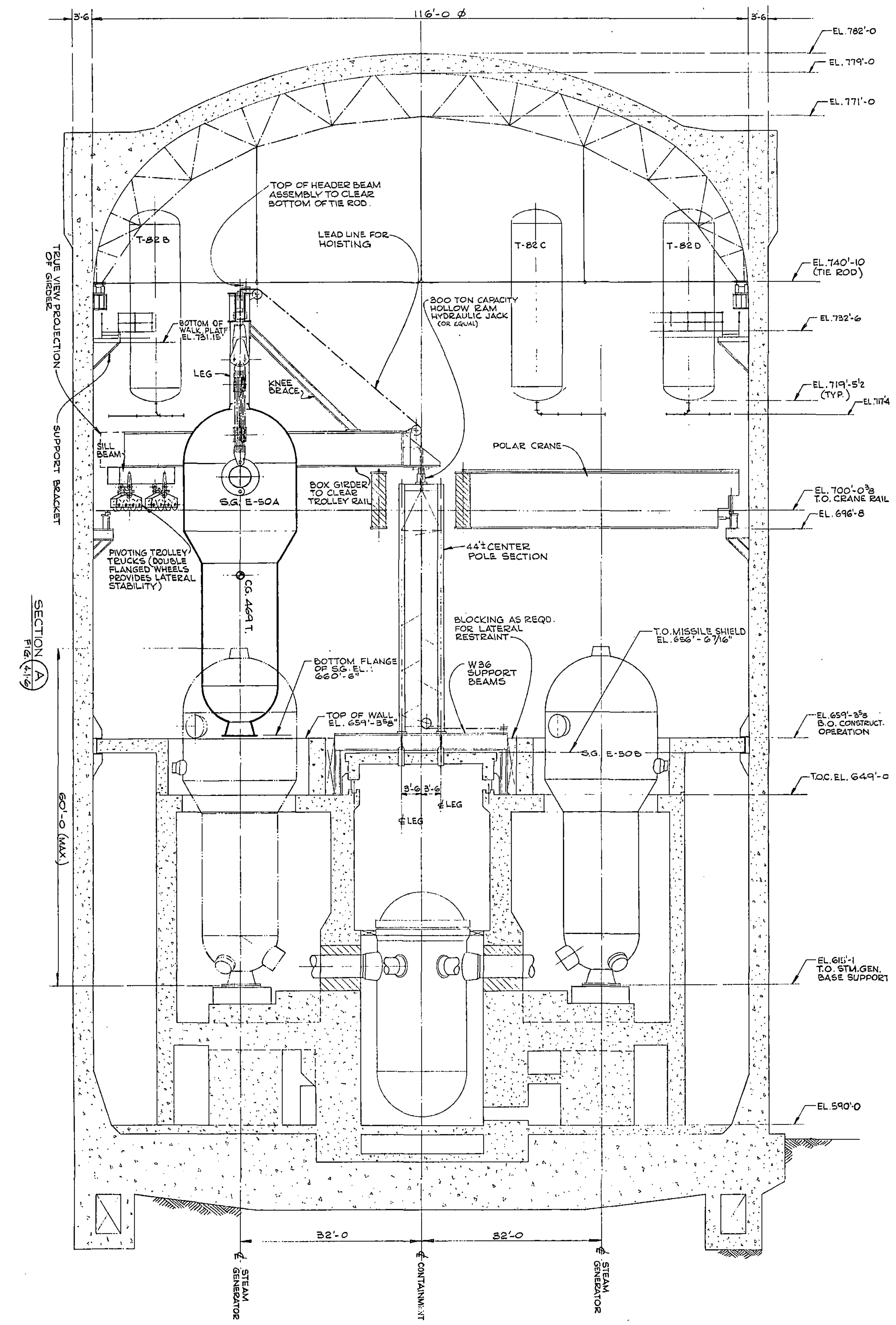


PLAN VIEW 2

PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

GENERAL ARRANGEMENT
PLAN VIEW, SH. 2

Figure 4.1-7

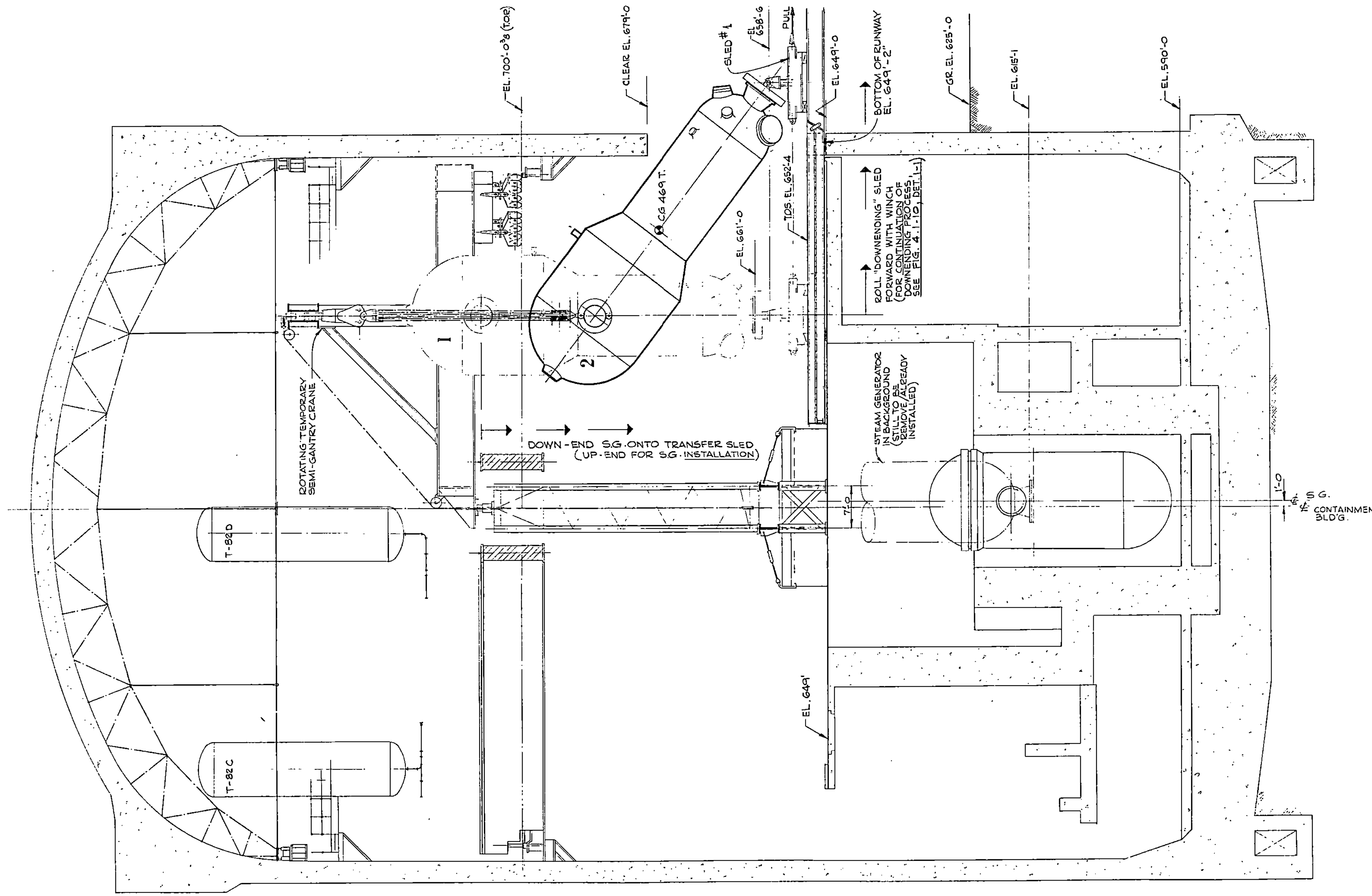


PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

GENERAL ARRANGEMENT
SECTION A-A

Figure 4.1-8

NOTE: THE PICTORIAL SEQUENCE IS REVERSED FOR NEW S.G. INSTALLATION.

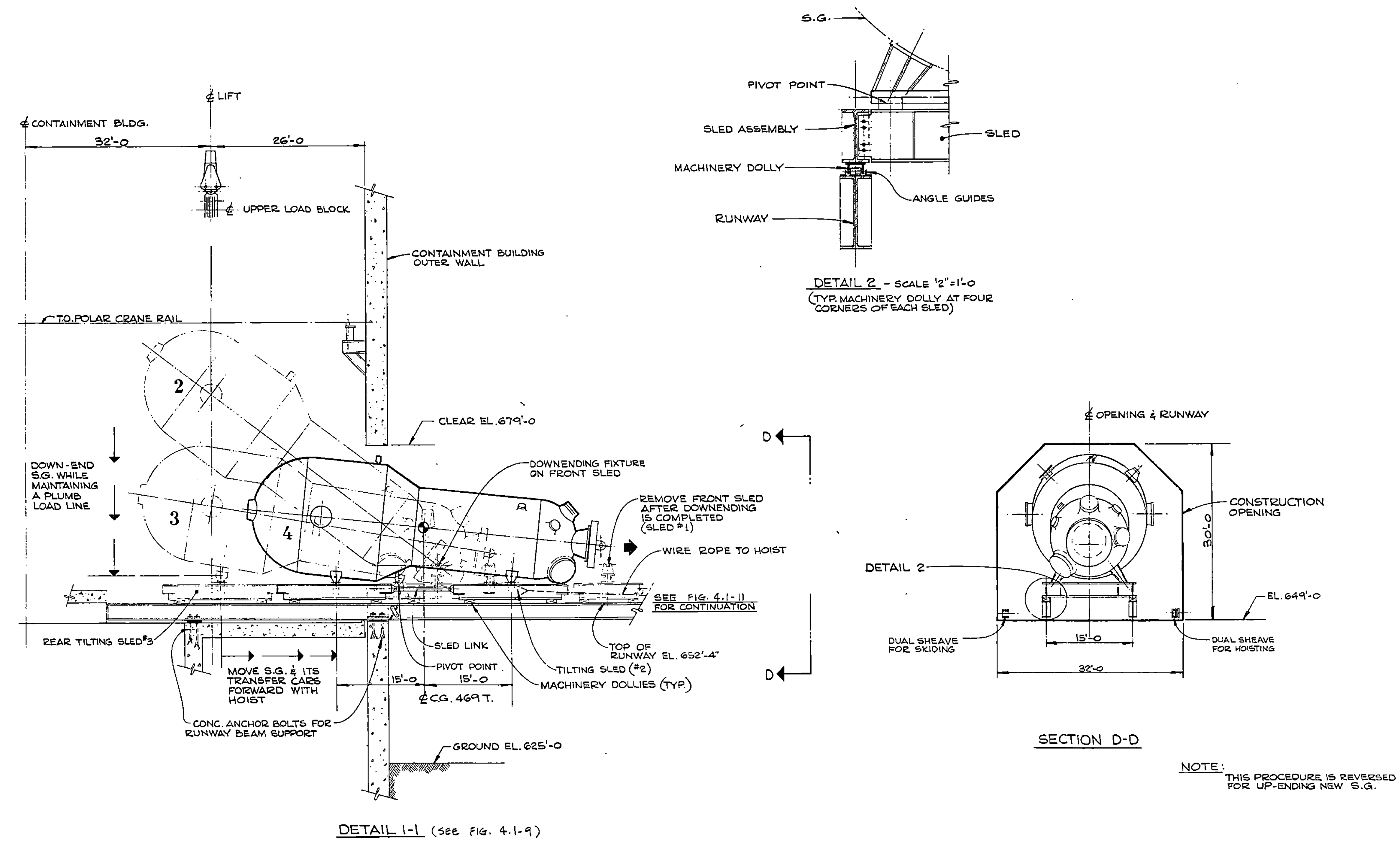


SECTION B-B (FROM FIG. 4.1-7)

PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

GENERAL ARRANGEMENT
SECTION B-B

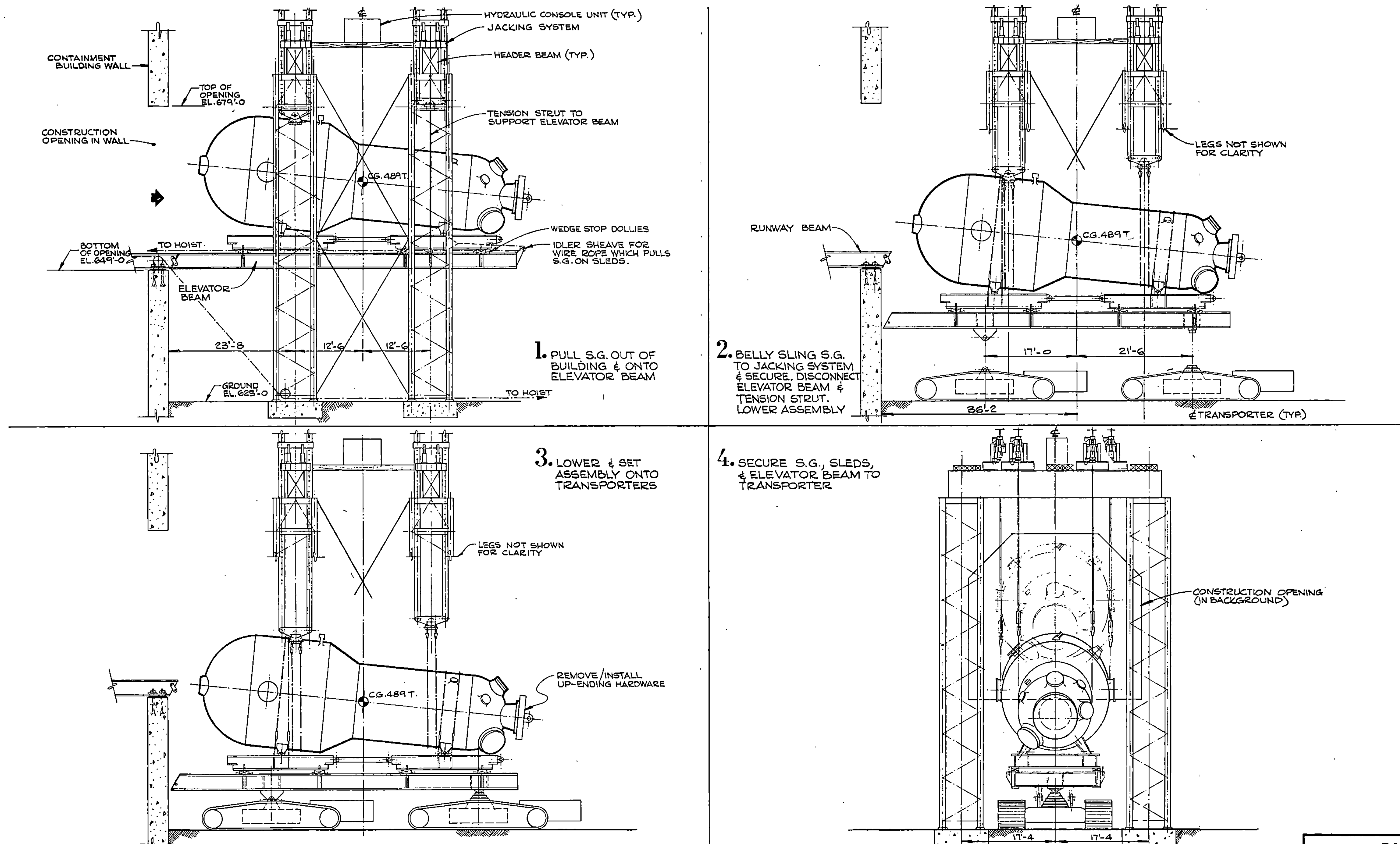
Figure 4.1-9



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

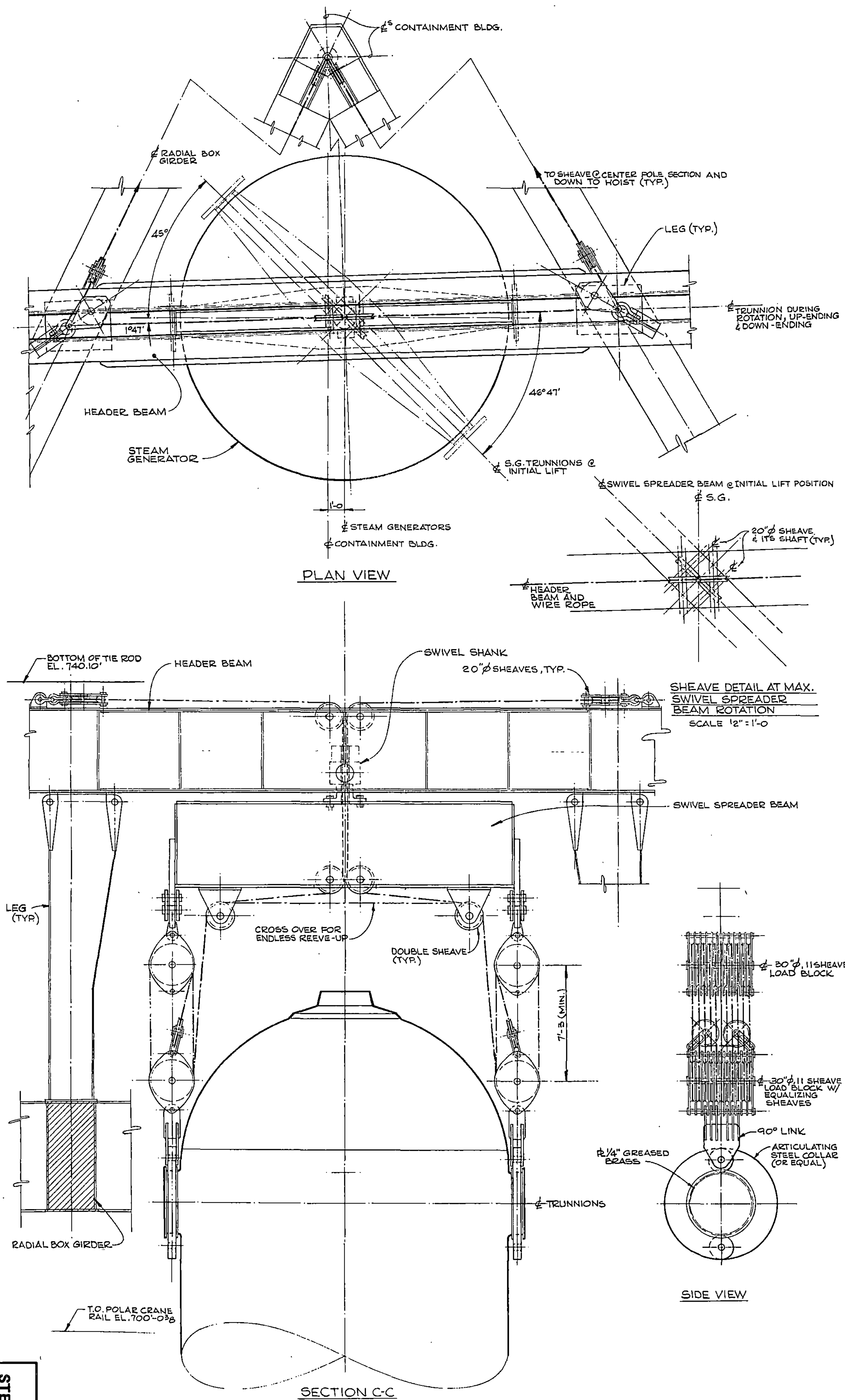
DOWN-ENDING STEAM GENERATOR ONTO SLEDS

Figure 4.1-10



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

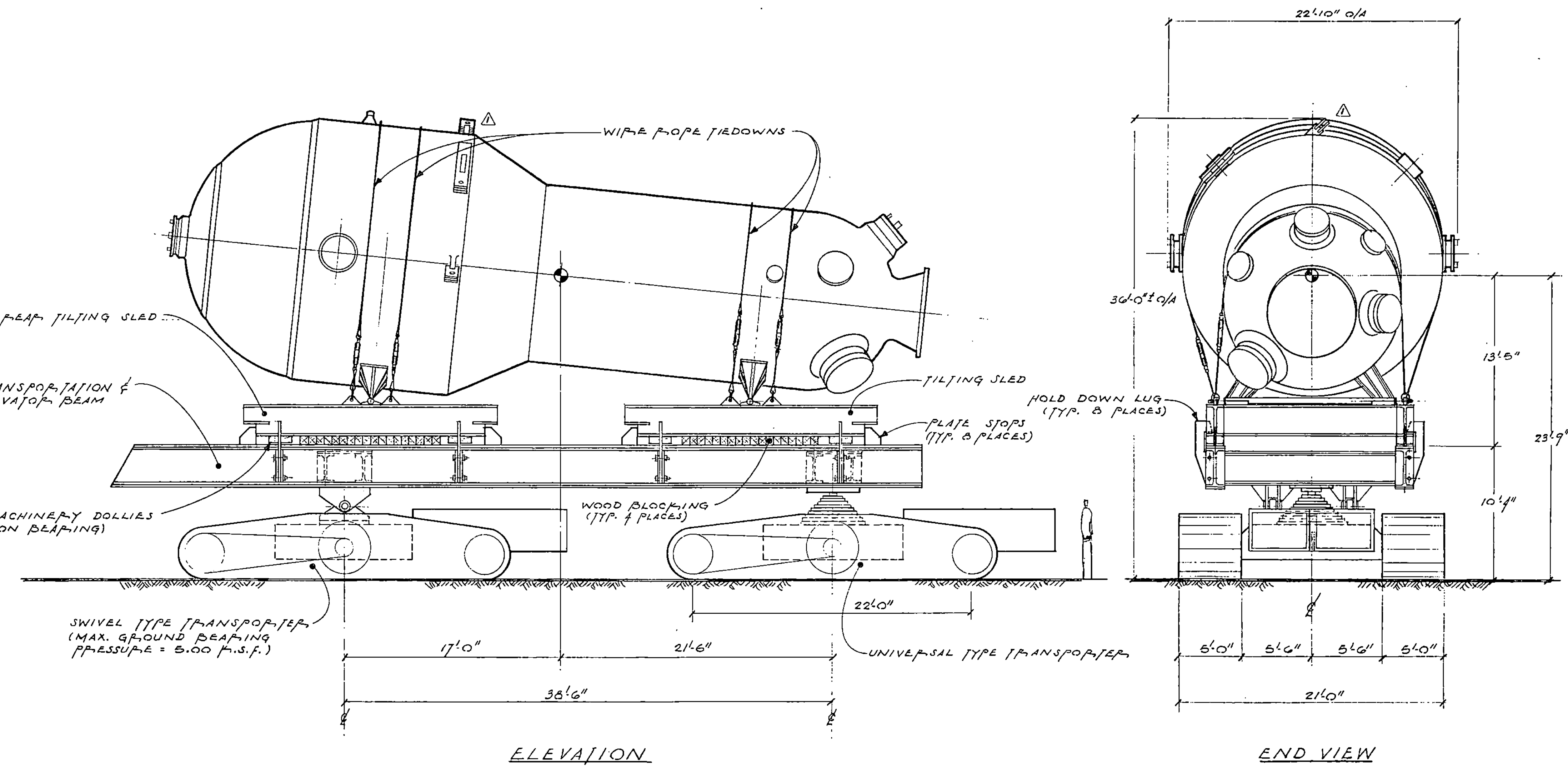
LOWERING STEAM GENERATOR
FROM ELEVATOR PLATFORM
ONTO TRANSPORTERS
Figure 4.1-11



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

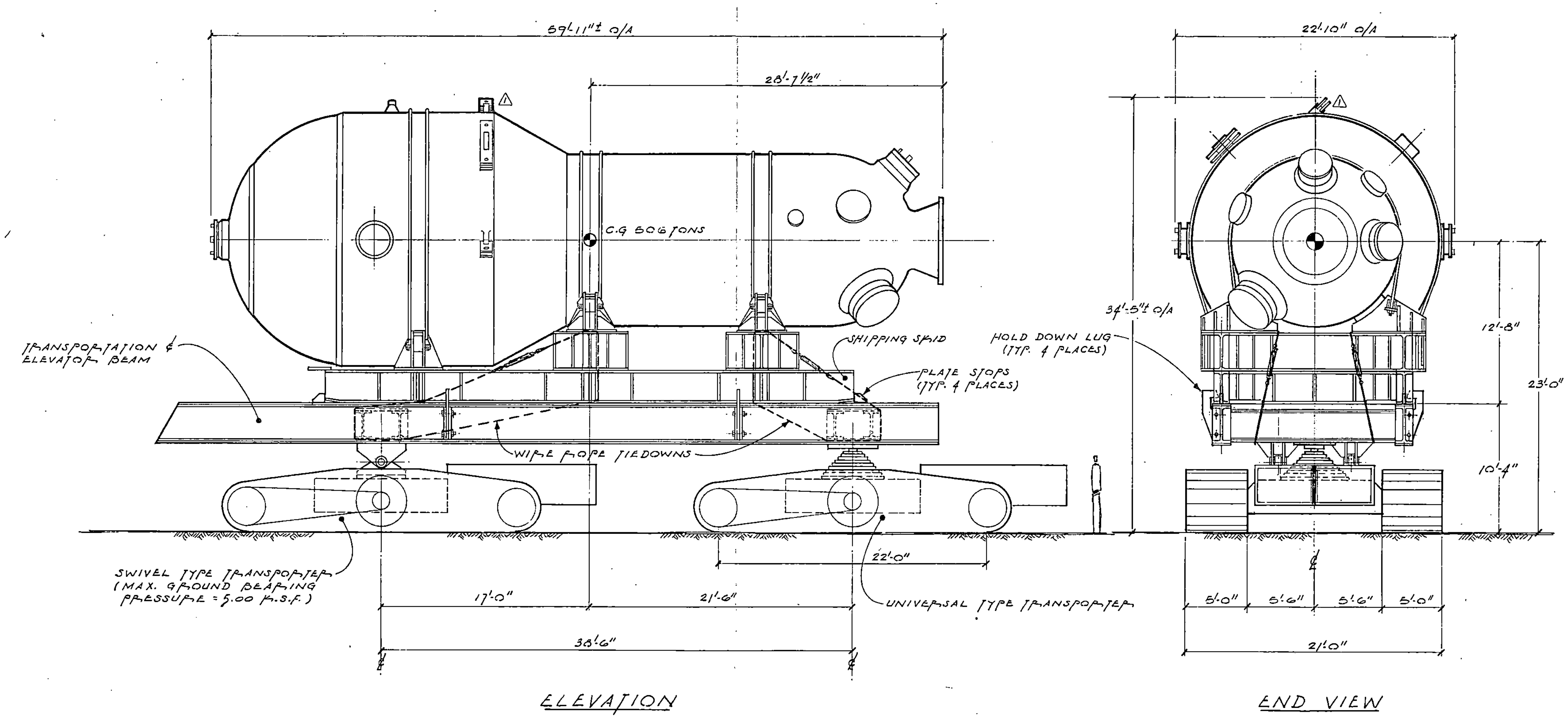
STEAM GENERATOR IN HOISTED POSITION, SECTION VIEW

Figure 4.1-12



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

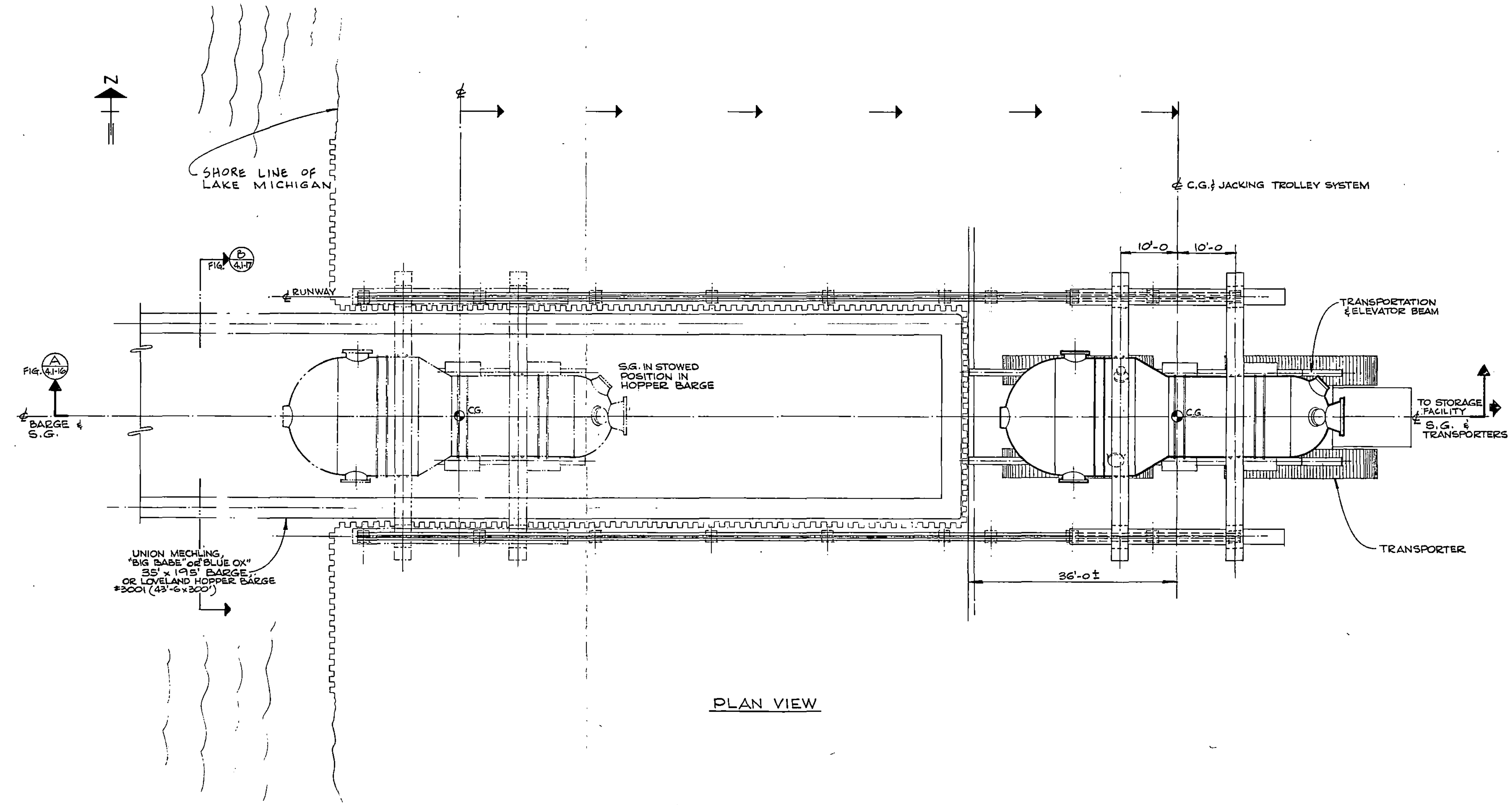
STEAM GENERATOR ON TRANSPORTER
BETWEEN STORAGE
AND CONTAINMENT
Figure 4.1-13



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

STEAM GENERATOR ON TRANSPORTER BARGE TO STORAGE

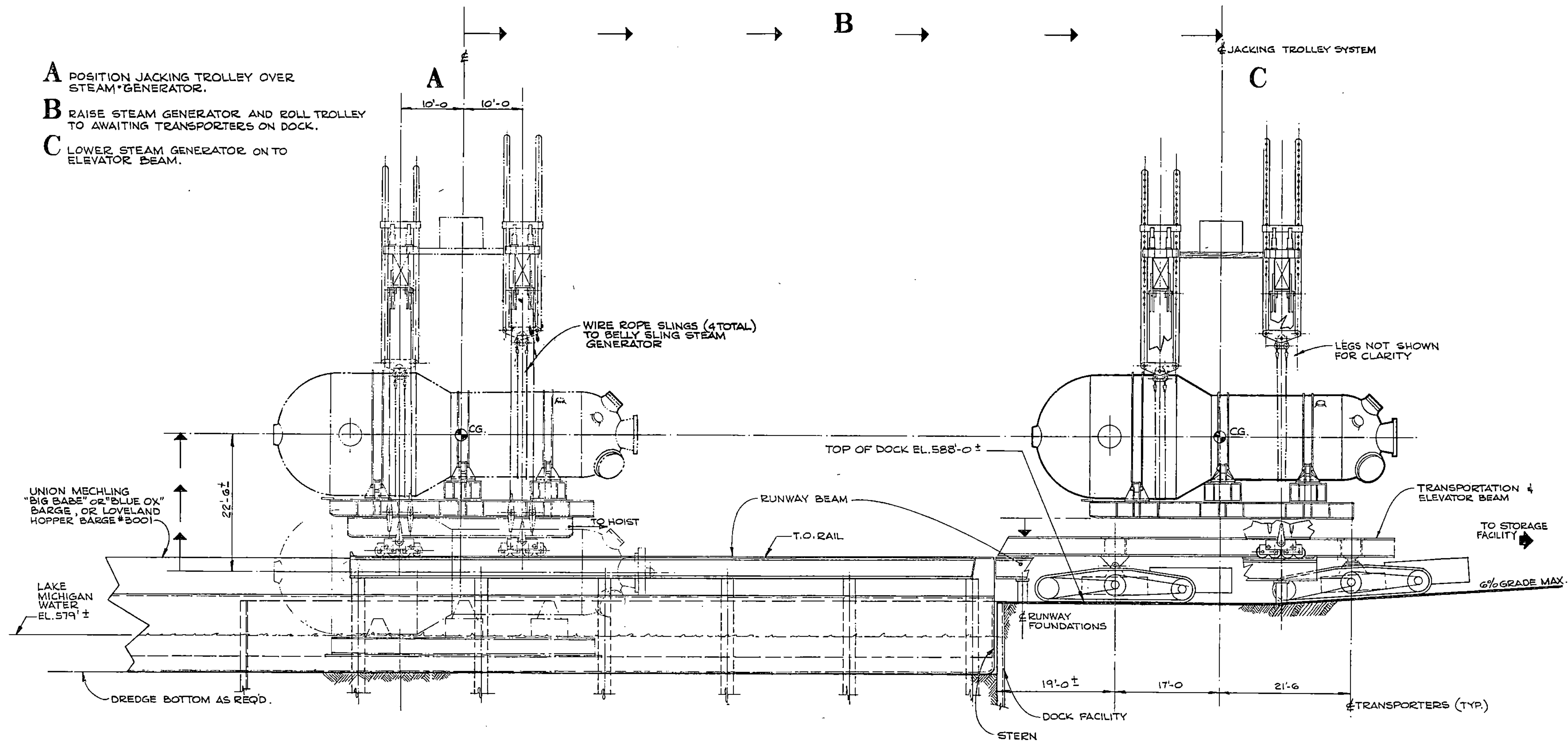
Figure 4.1-14



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

OFF LOADING STEAM GENERATOR FROM BARGE, PLAN VIEW

Figure 4.1-15



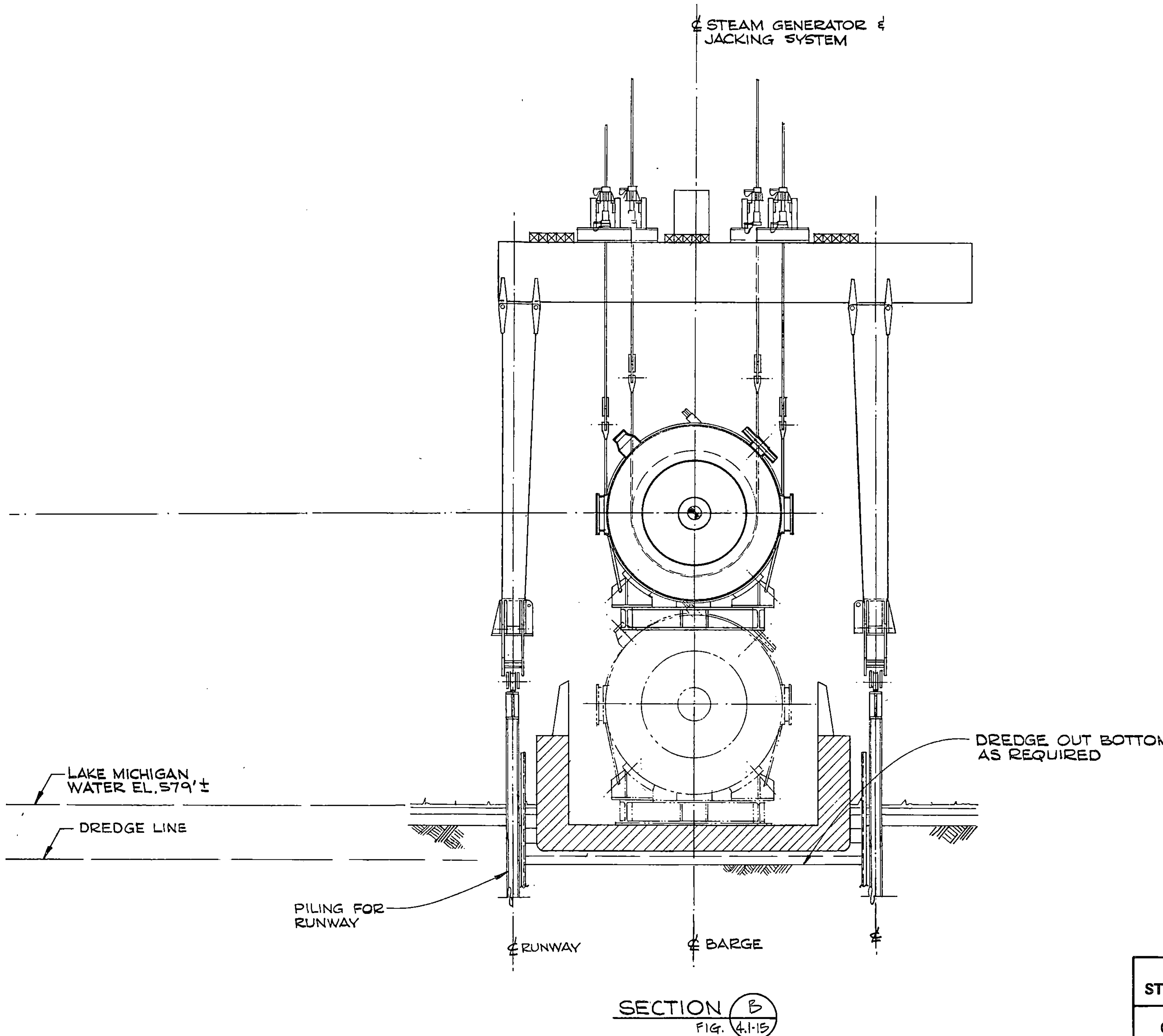
ELEVATION VIEW

SECTION A FIG. 4.1-15

PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

OFF LOADING STEAM GENERATOR FROM BARGE, ELEVATION VIEW

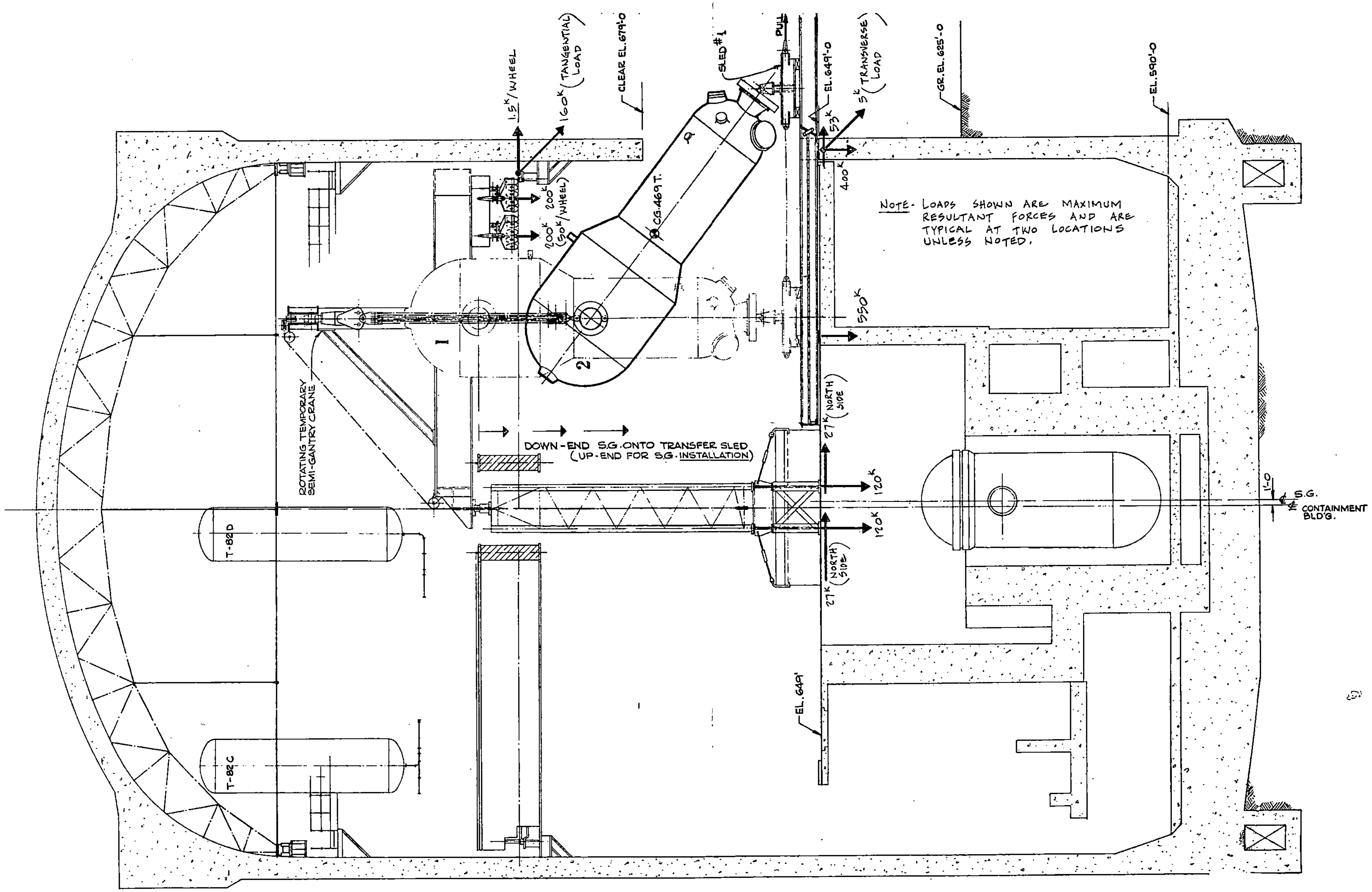
Figure 4.1-16



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

OFF LOADING STEAM GENERATOR FROM BARGE, SECTION VIEW

Figure 4.1-17

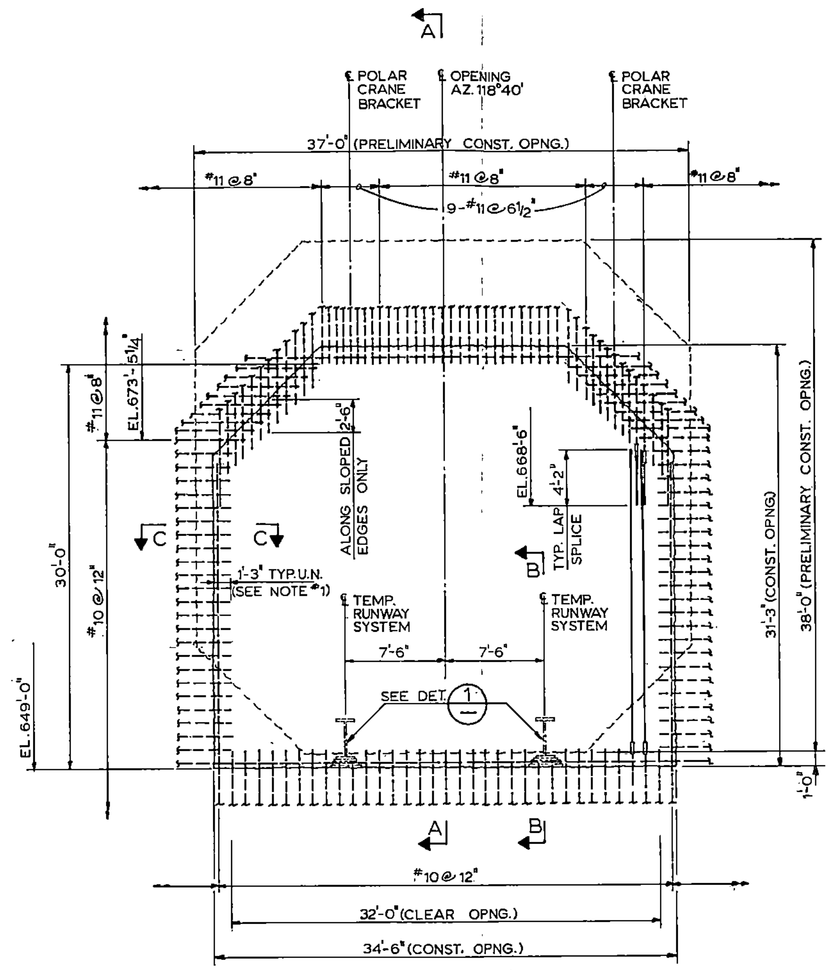


SECTION B-B (FROM FIG. 4.1-7)

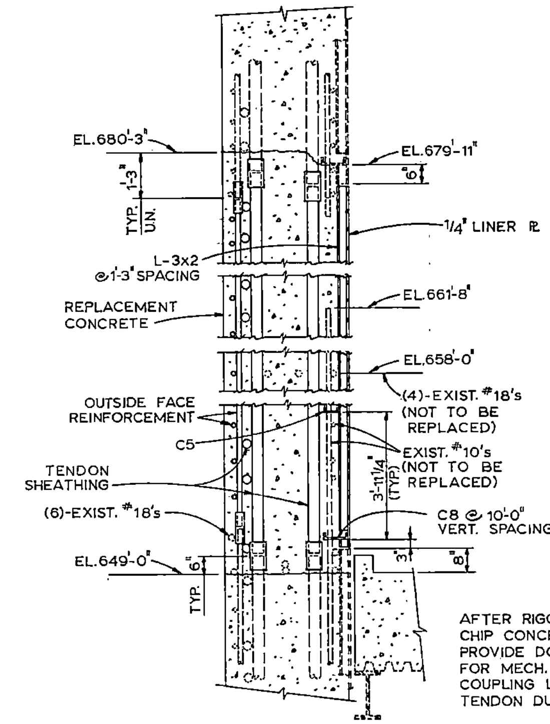
PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

CONTAINMENT INTERNALS-
RIGGING DESIGN LOADS

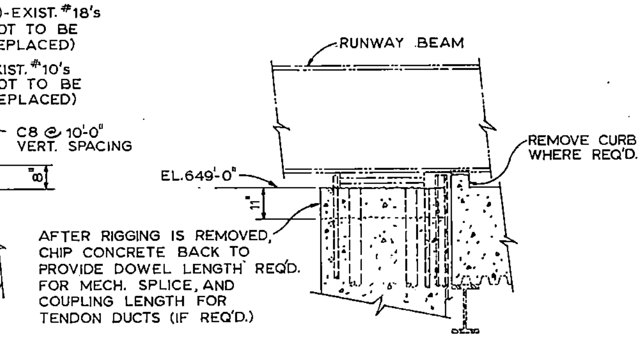
Figure 4.1-18



ELEVATION - OUTSIDE FACE REINFORCING
(LOOKING FROM OUTSIDE)
(SEE DWG. SK-C-8)
SCALE: 3/16" = 1'-0"



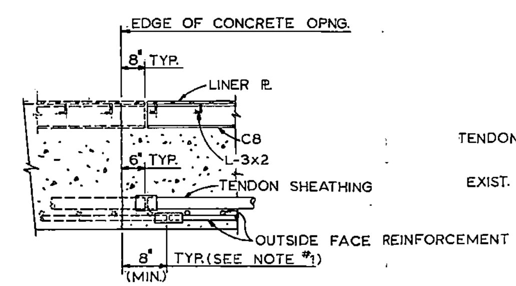
SECTION A-A
(REPLACEMENT SECTION SHOWN)
SCALE: ½" = 1'-0"



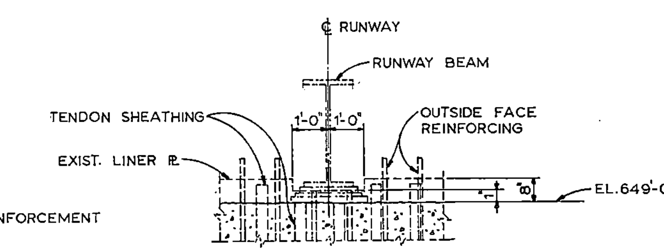
SECTION B-B
SCALE: ½" = 1'-0"

NOTES-

1. INITIAL DOWEL LENGTH AS SHOWN (1'-3" OR 2'-6") IS REQUIRED FOR ADEQUATE TOLERANCE TO AVOID INTERFERENCE WITH OTHER REINFORCEMENT WHEN LOCATING CADWELD SPLICES. 8" IS THE FINAL MINIMUM DOWEL LENGTH REQUIRED FOR CADWELD INSTALLATION AND INSPECTION.



SECTION C-C
(REPLACEMENT SECTION SHOWN)
SCALE: ½" = 1'-0"

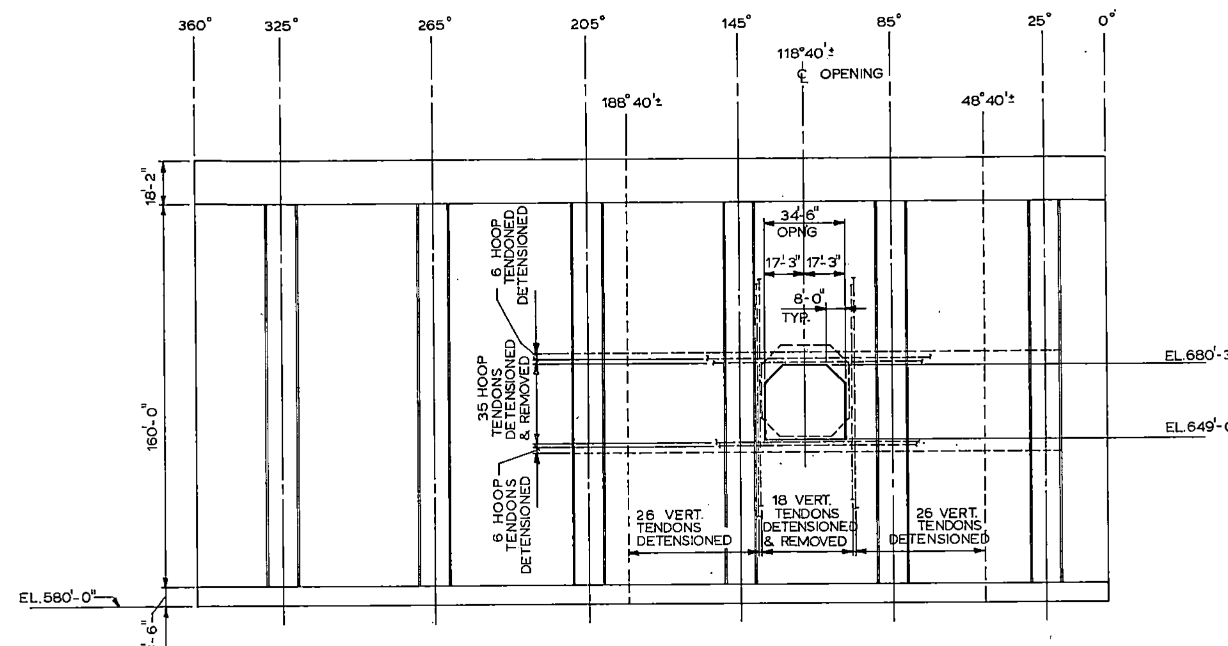


DETAIL 1
SCALE: ½" = 1'-0"

PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

CONSTRUCTION OPENING DETAILS

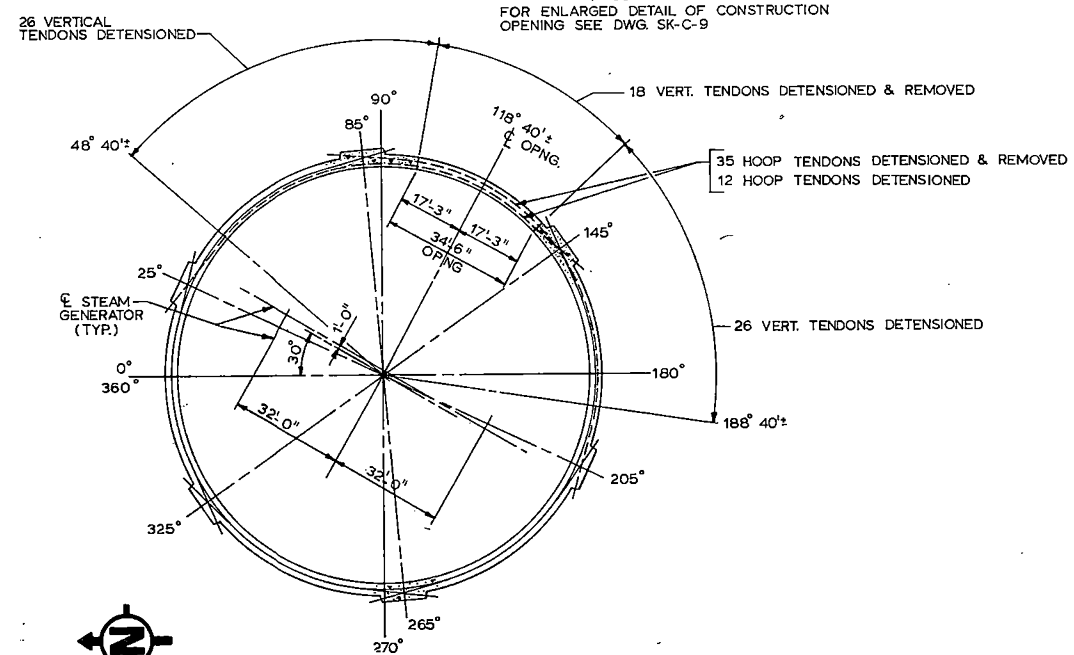
Figure 4.1-19



OUTSIDE DEVELOPED ELEVATION

1" = 30'

FOR ENLARGED DETAIL OF CONSTRUCTION OPENING SEE DWG. SK-C-9



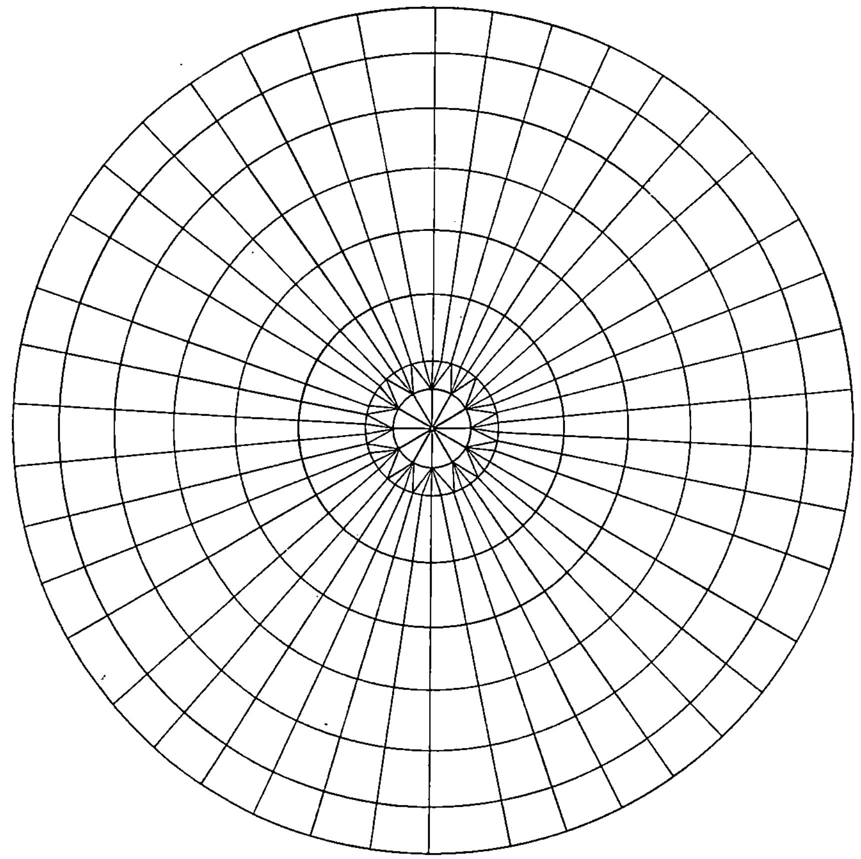
PRESTRESS REMOVAL SCHEMATIC PLAN

1" = 20'

PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

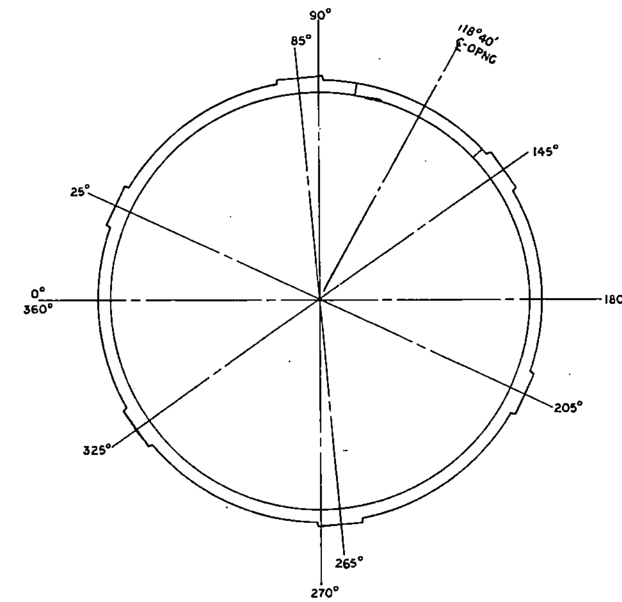
CONSTRUCTION OPENING AND TENDON DETENSION/REMOVAL

Figure 4.1-20



FINITE ELEMENT MESH AT DOME

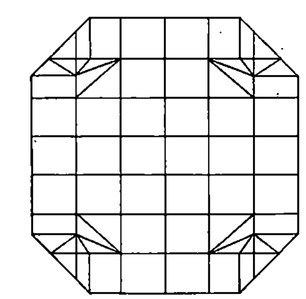
1/8" = 1'-0"



CROSS SECTION OF CONTAINMENT WALL

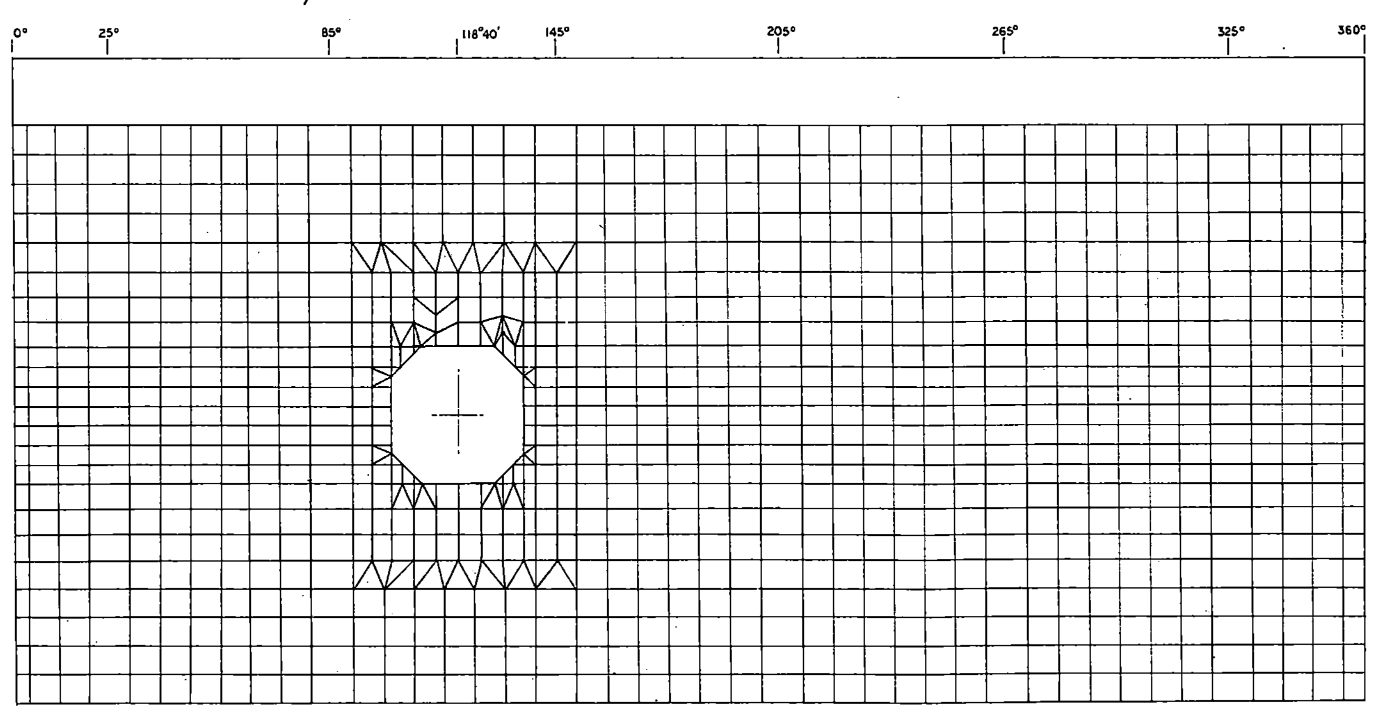
(TYPICAL)

1/16" = 1'-0"



FINITE ELEMENT MESH AT CLOSED OPENING

1/8" = 1'-0"



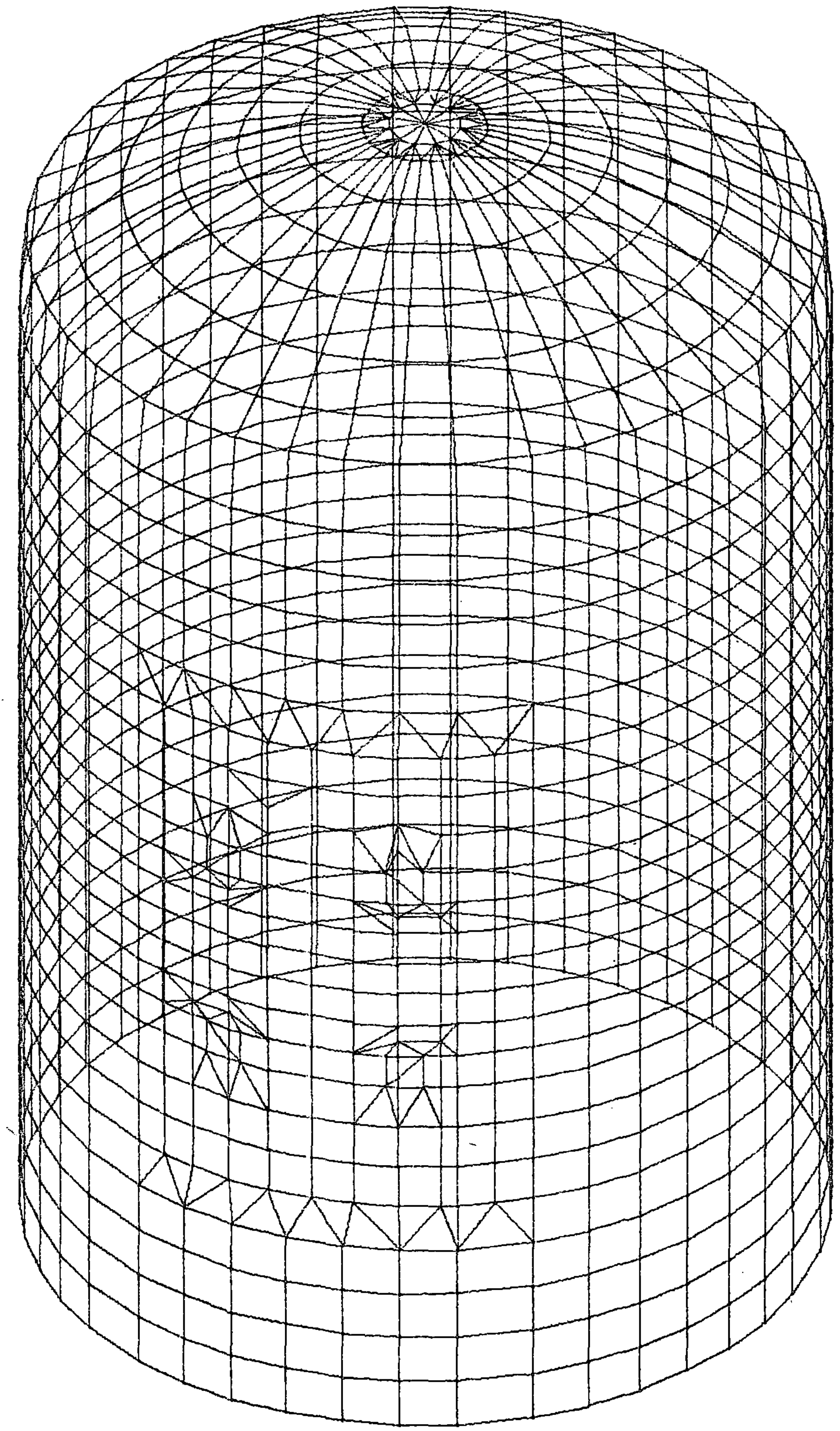
FINITE ELEMENT MESH AT WALL WITH OPENING

1/16" = 1'-0"

PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

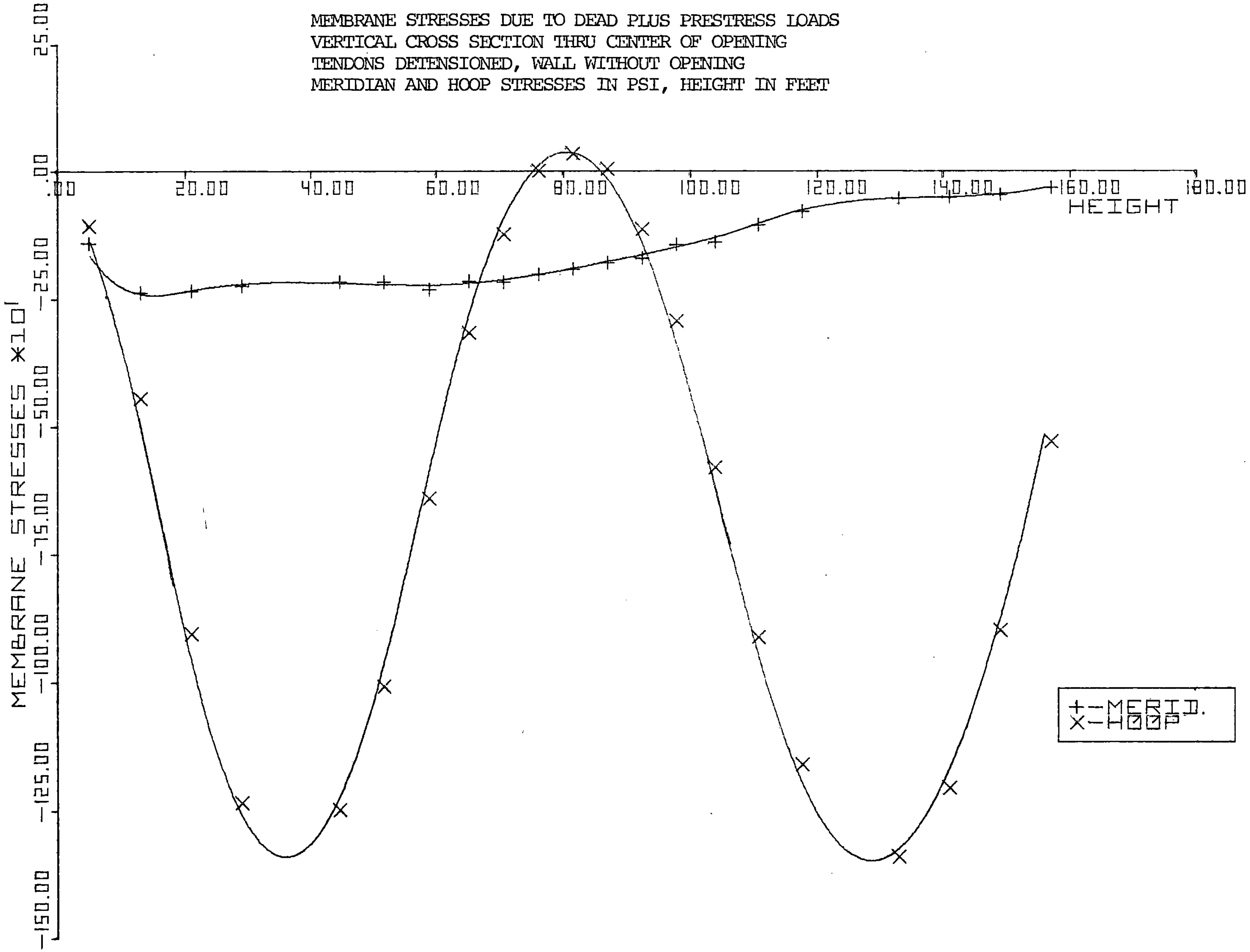
FINITE ELEMENT MODEL FOR CONTAINMENT SHELL ANALYSIS

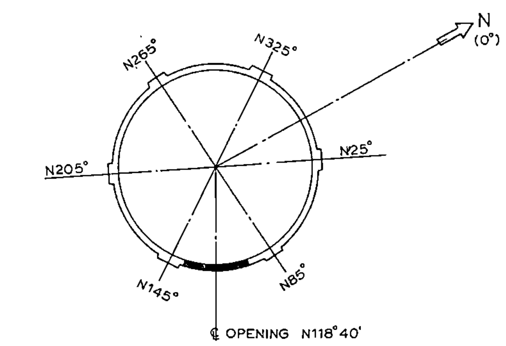
Figure 4.1-21



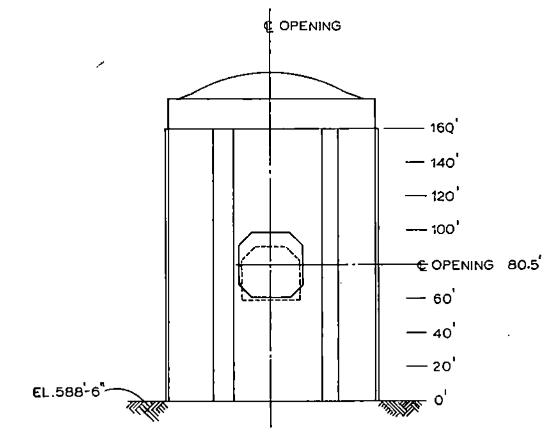
PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

THREE DIMENSIONAL PLOT
OF CONTAINMENT SHELL
(WITHOUT OPENING)
Figure 4.1-22



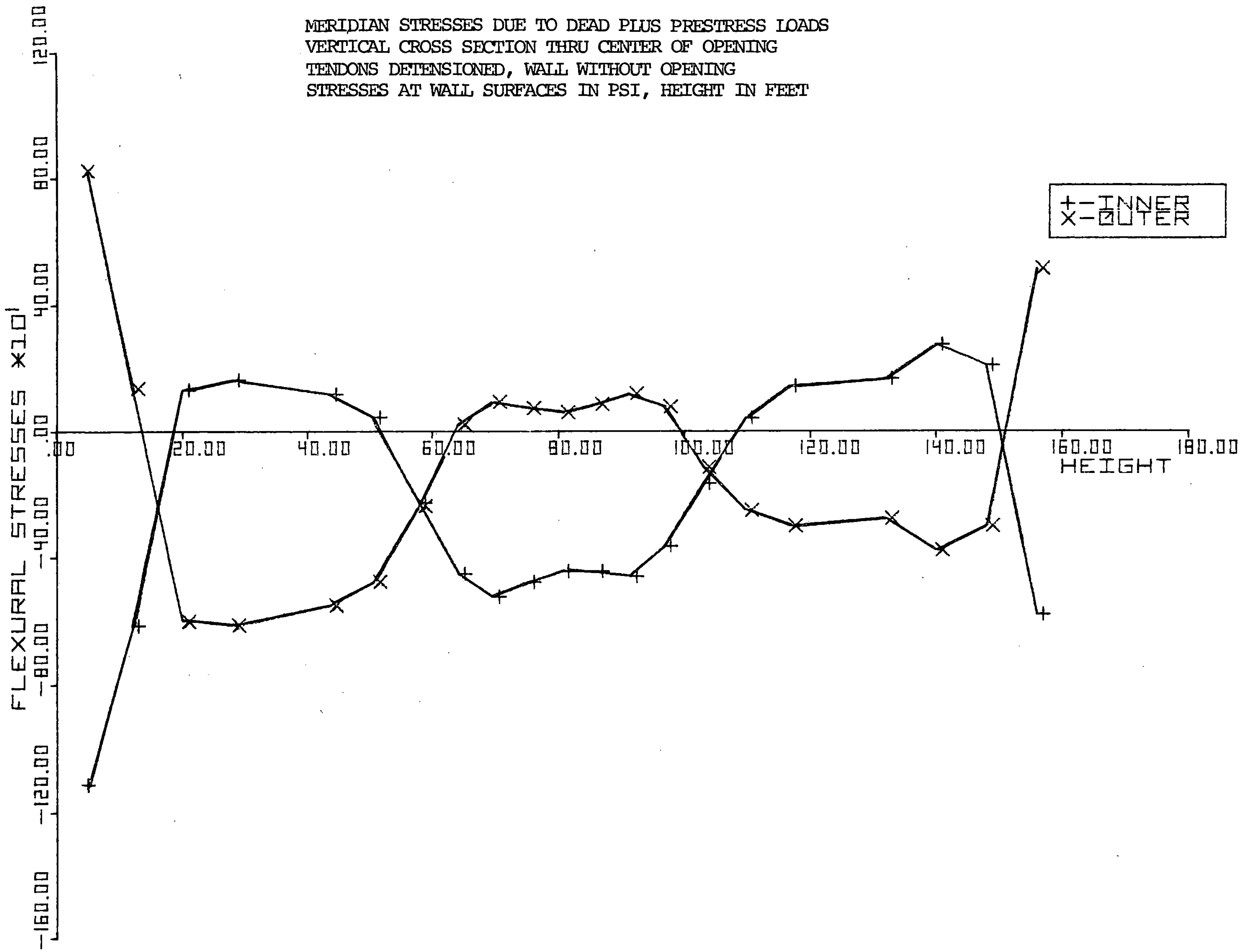


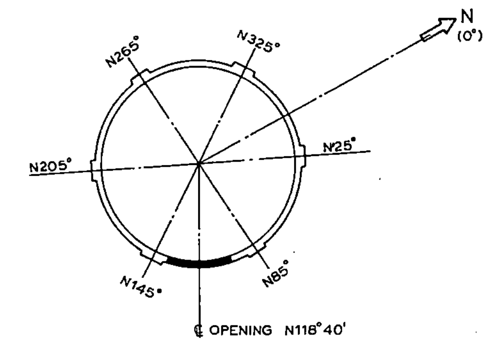
TYPICAL HORIZONTAL CROSS-SECTION



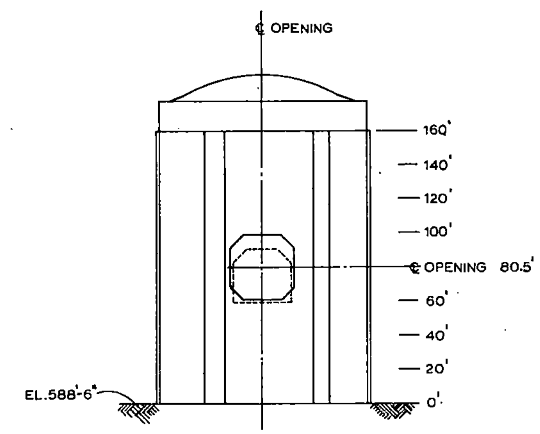
ELEVATION

PALISADES PLANT STEAM GENERATOR REPAIR REPORT
MEMBRANE STRESSES VERSUS HEIGHT (WITHOUT OPENING) Figure 4.1-23





TYPICAL HORIZONTAL CROSS-SECTION

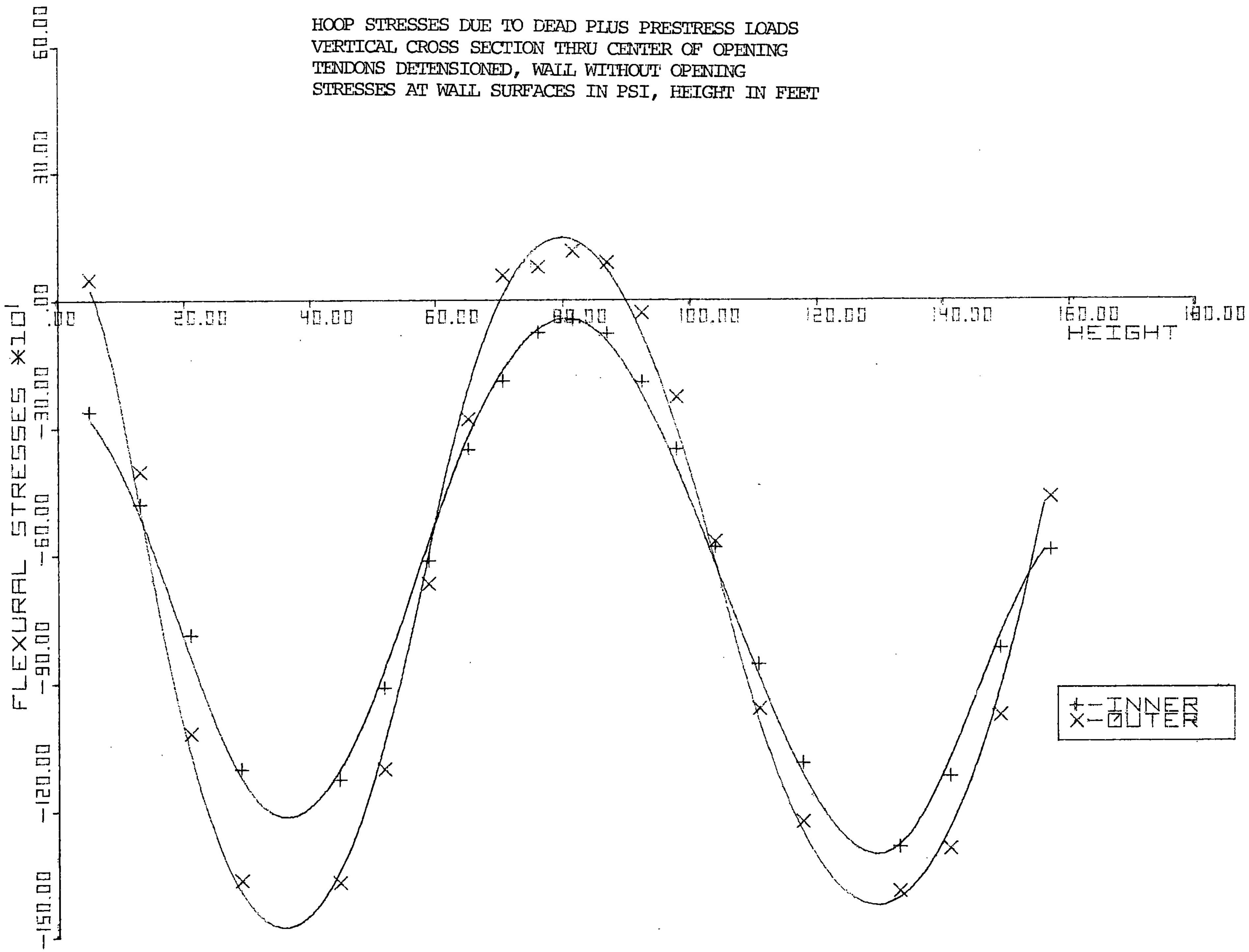


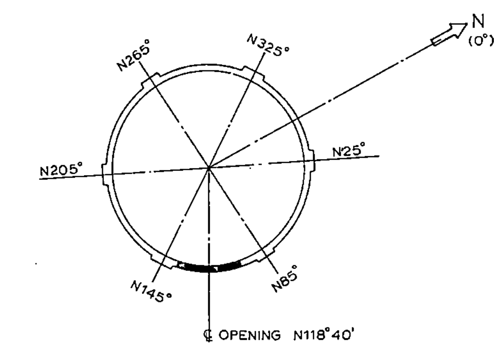
ELEVATION

PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

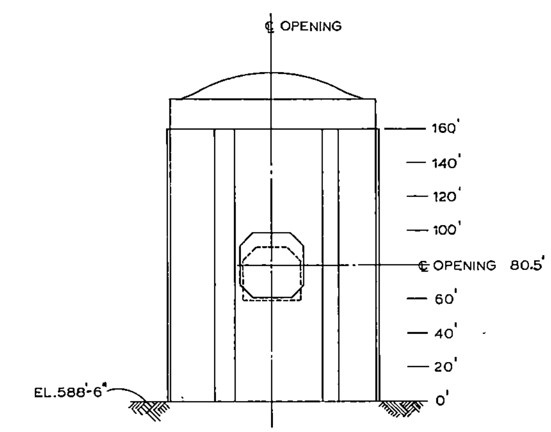
MERIDIAN STRESSES VERSUS HEIGHT (WITHOUT OPENING)

Figure 4.1-24





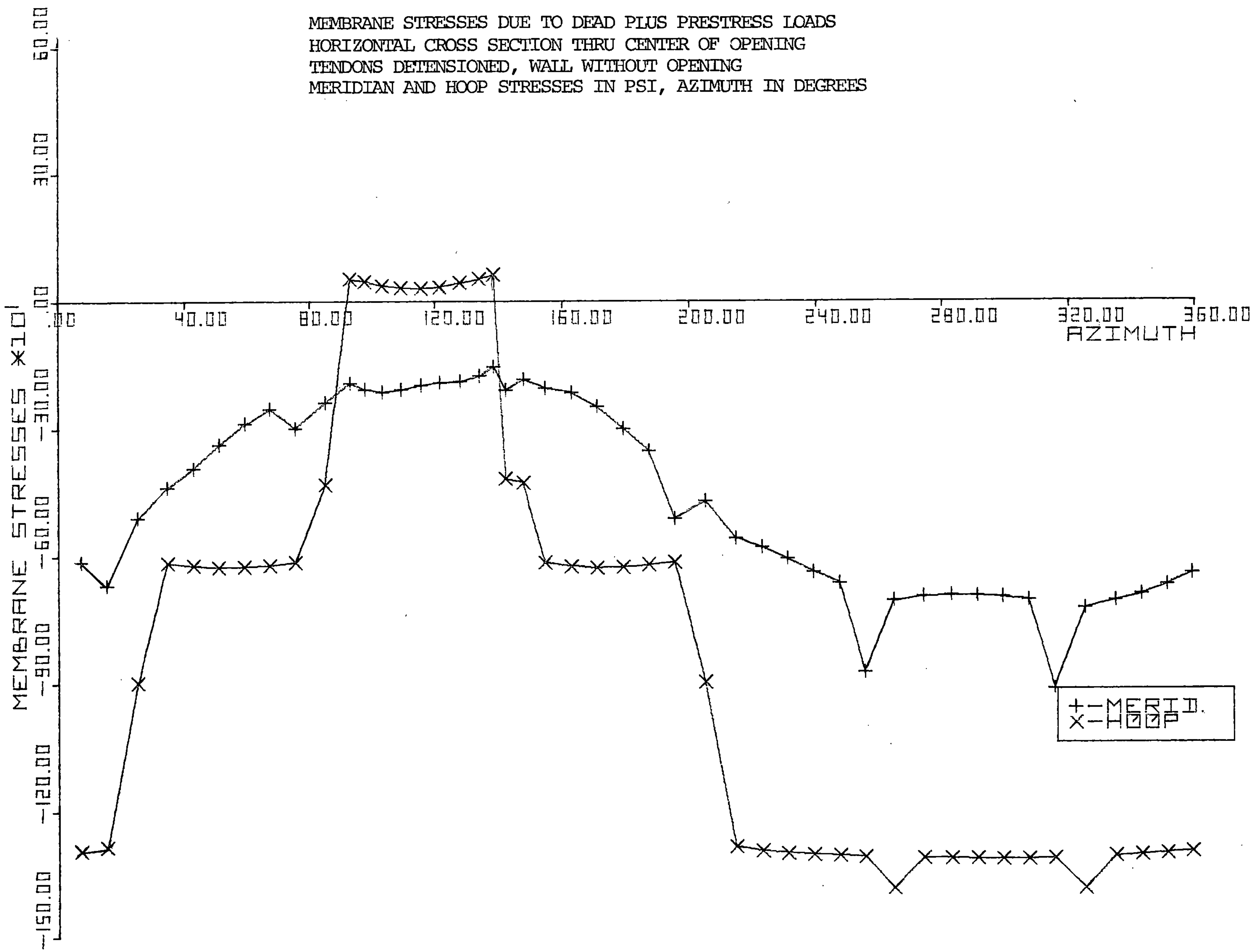
TYPICAL HORIZONTAL CROSS-SECTION

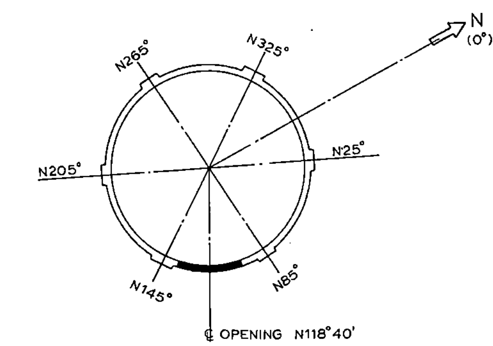


ELEVATION

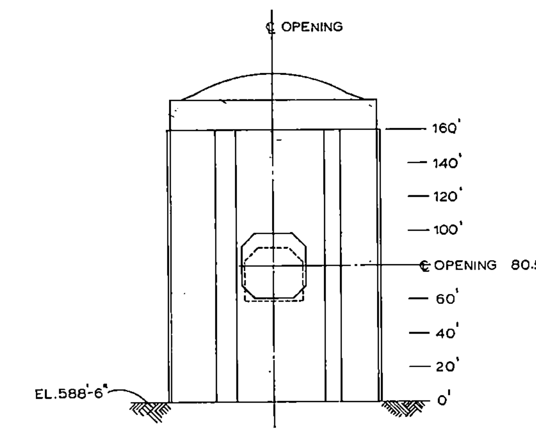
PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

HOOP STRESSES VERSUS HEIGHT
(WITHOUT OPENING)
Figure 4.1-25





TYPICAL HORIZONTAL CROSS-SECTION



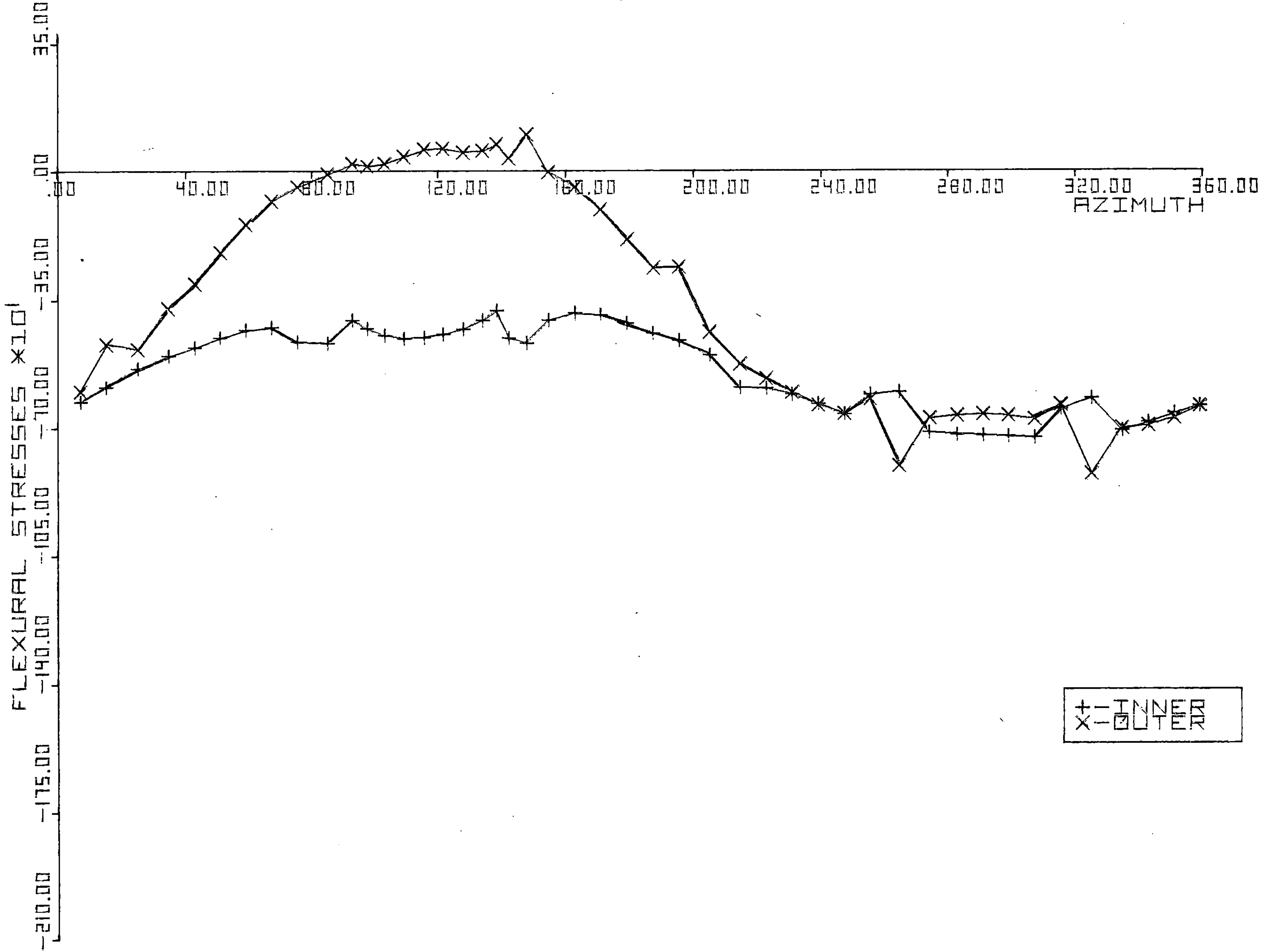
ELEVATION

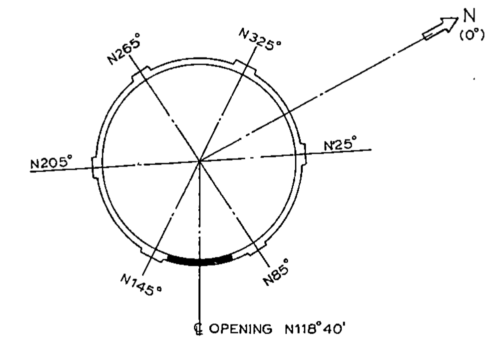
PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

MEMBRANE STRESSES VERSUS AZIMUTH (WITHOUT OPENING)

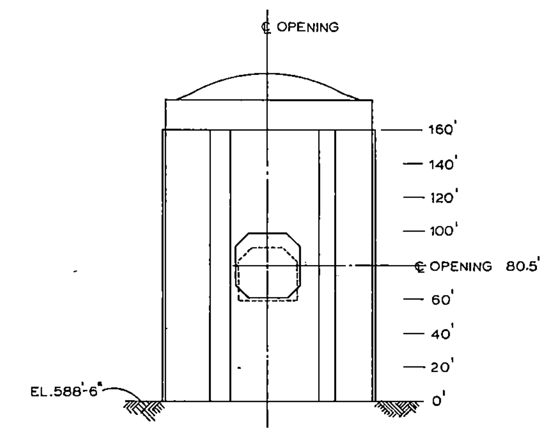
Figure 4.1-26

MERIDIAN STRESSES DUE TO DEAD PLUS PRESTRESS LOADS
HORIZONTAL CROSS SECTION THRU CENTER OF OPENING
TENDONS DETENSIONED, WALL WITHOUT OPENING
STRESSES AT WALL SURFACES IN PSI, AZIMUTH IN DEGREES





TYPICAL HORIZONTAL CROSS-SECTION

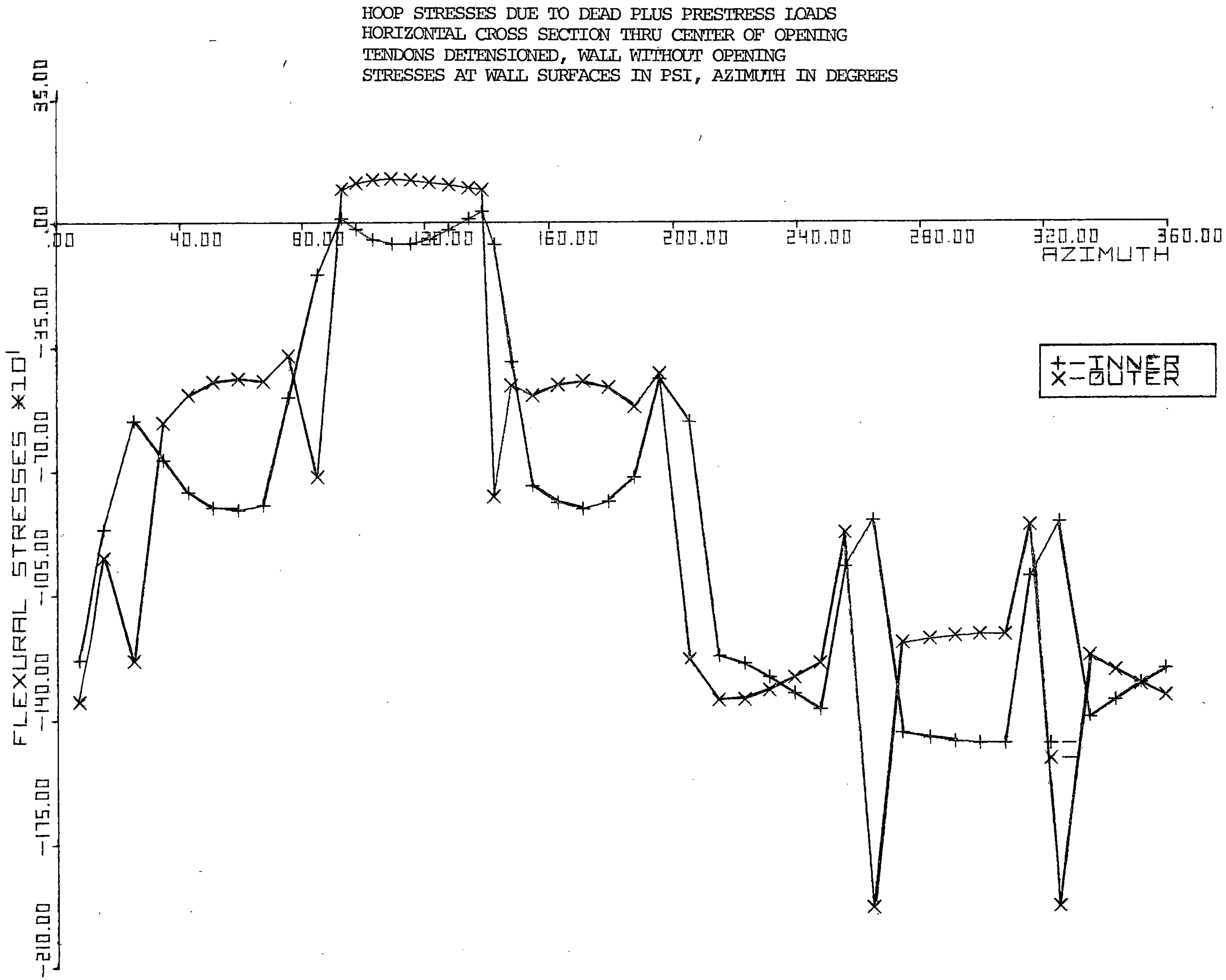


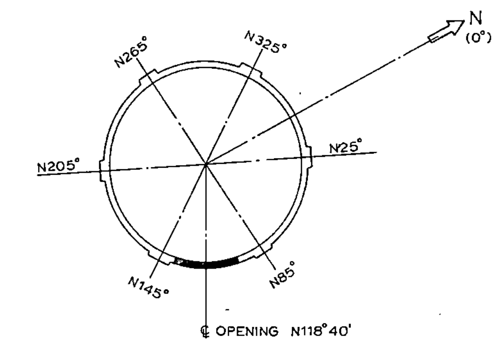
ELEVATION

PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

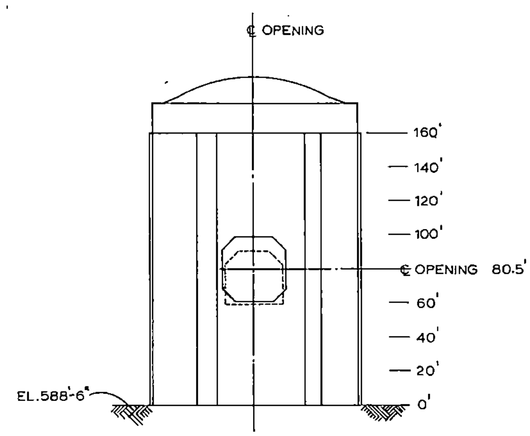
MERIDIAN STRESSES VERSUS
AZIMUTH (WITHOUT OPENING)

Figure 4.1-27





TYPICAL HORIZONTAL CROSS-SECTION

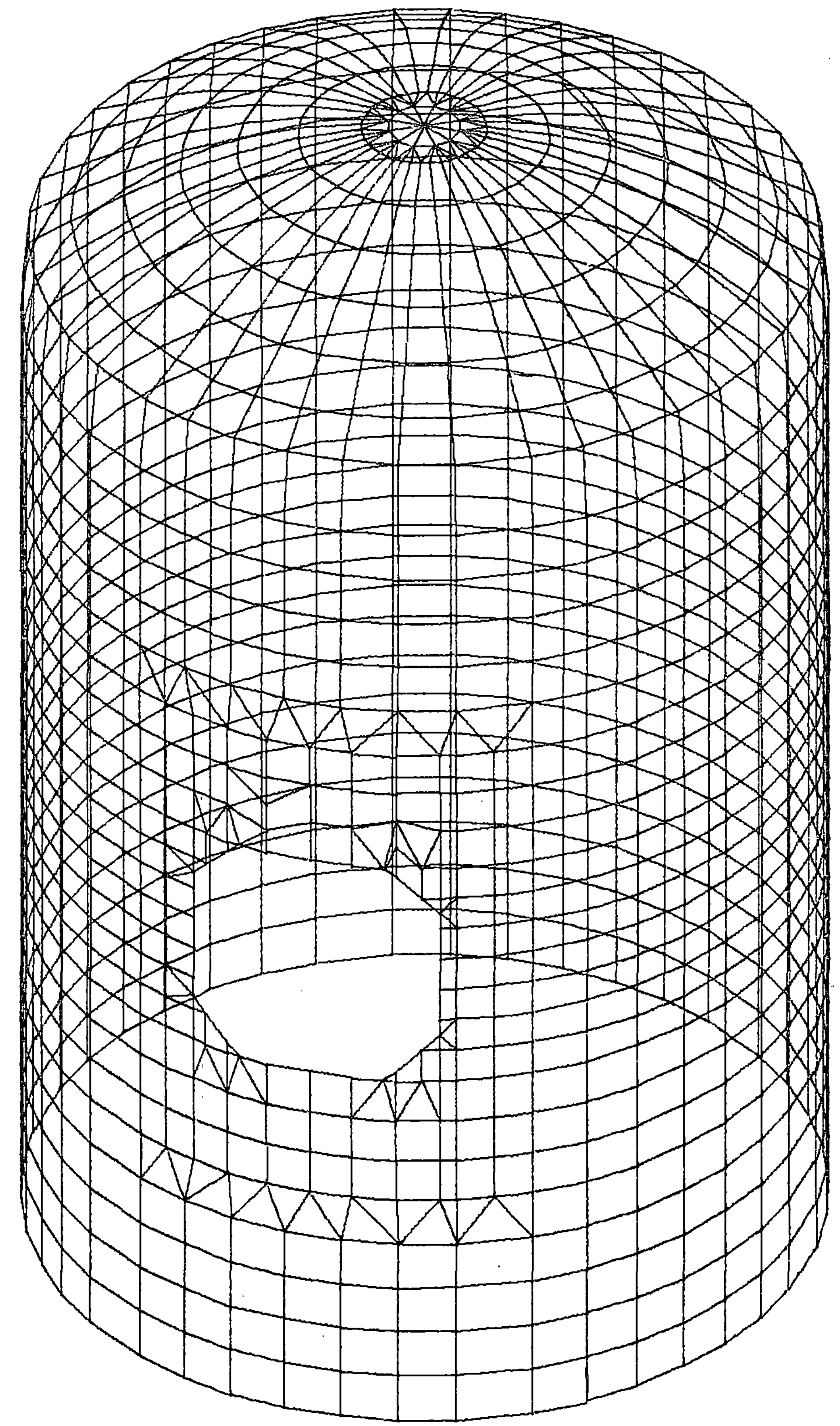


ELEVATION

PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

HOOP STRESSES VERSUS AZIMUTH
(WITHOUT OPENING)

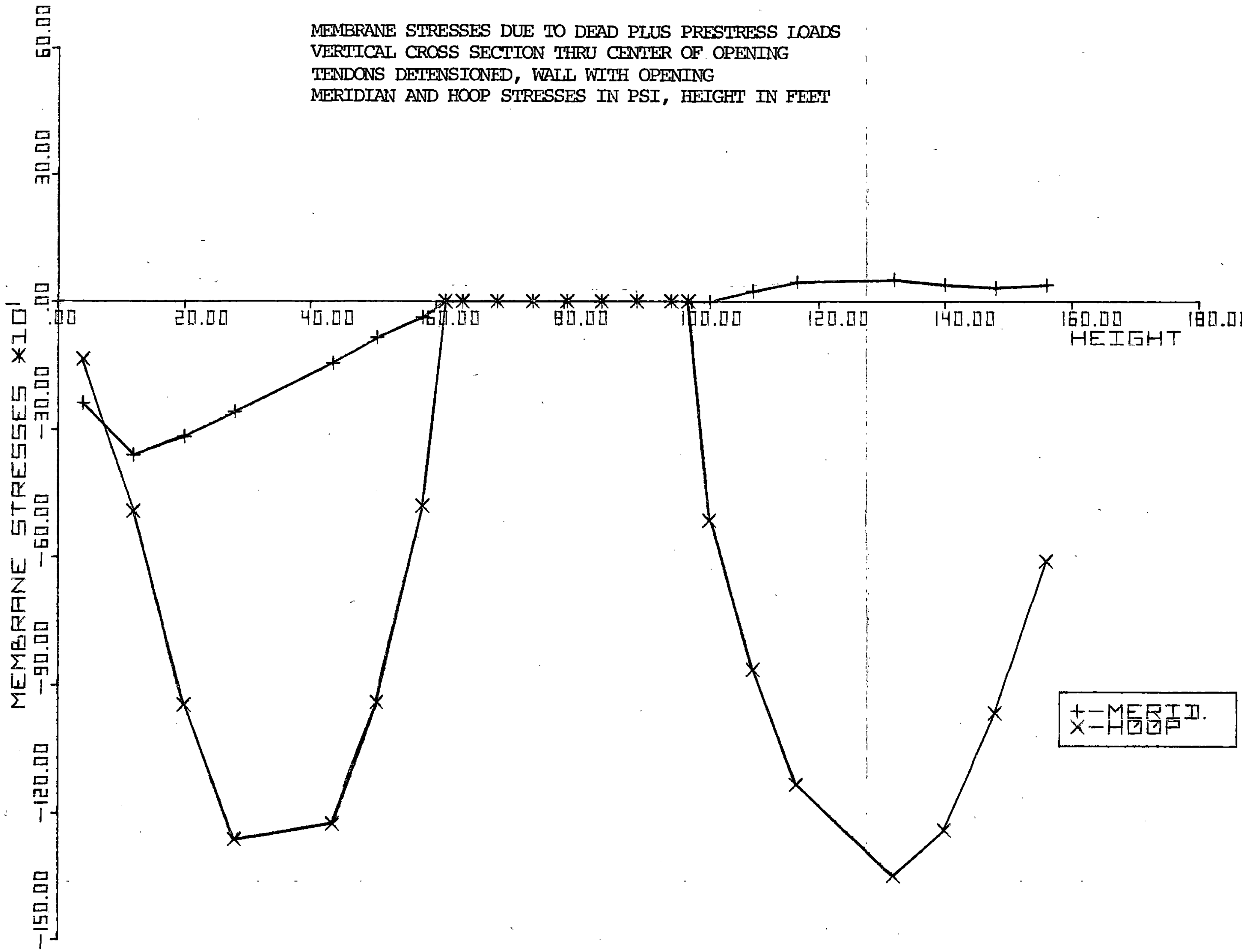
Figure 4.1-28

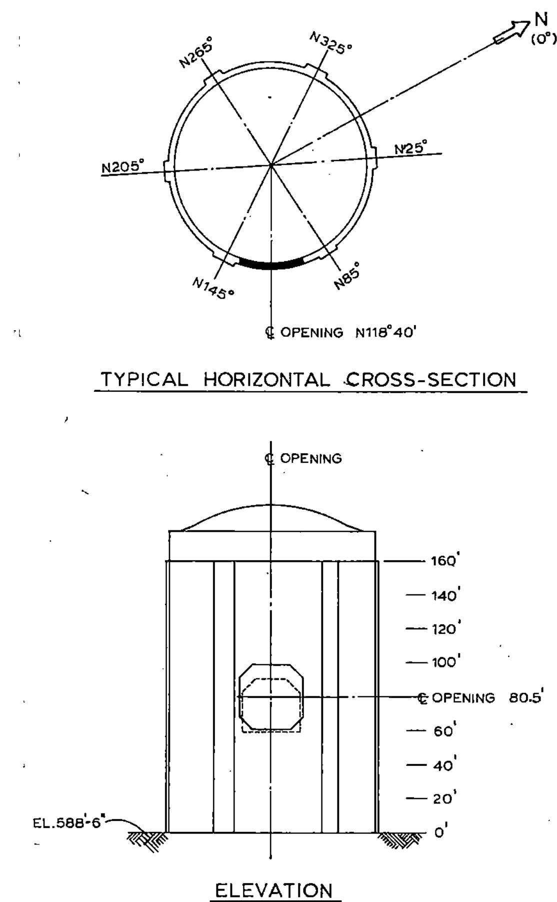


PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

THREE DIMENSIONAL PLOT OF CONTAINMENT (WITH OPENING)

Figure 4.1-29

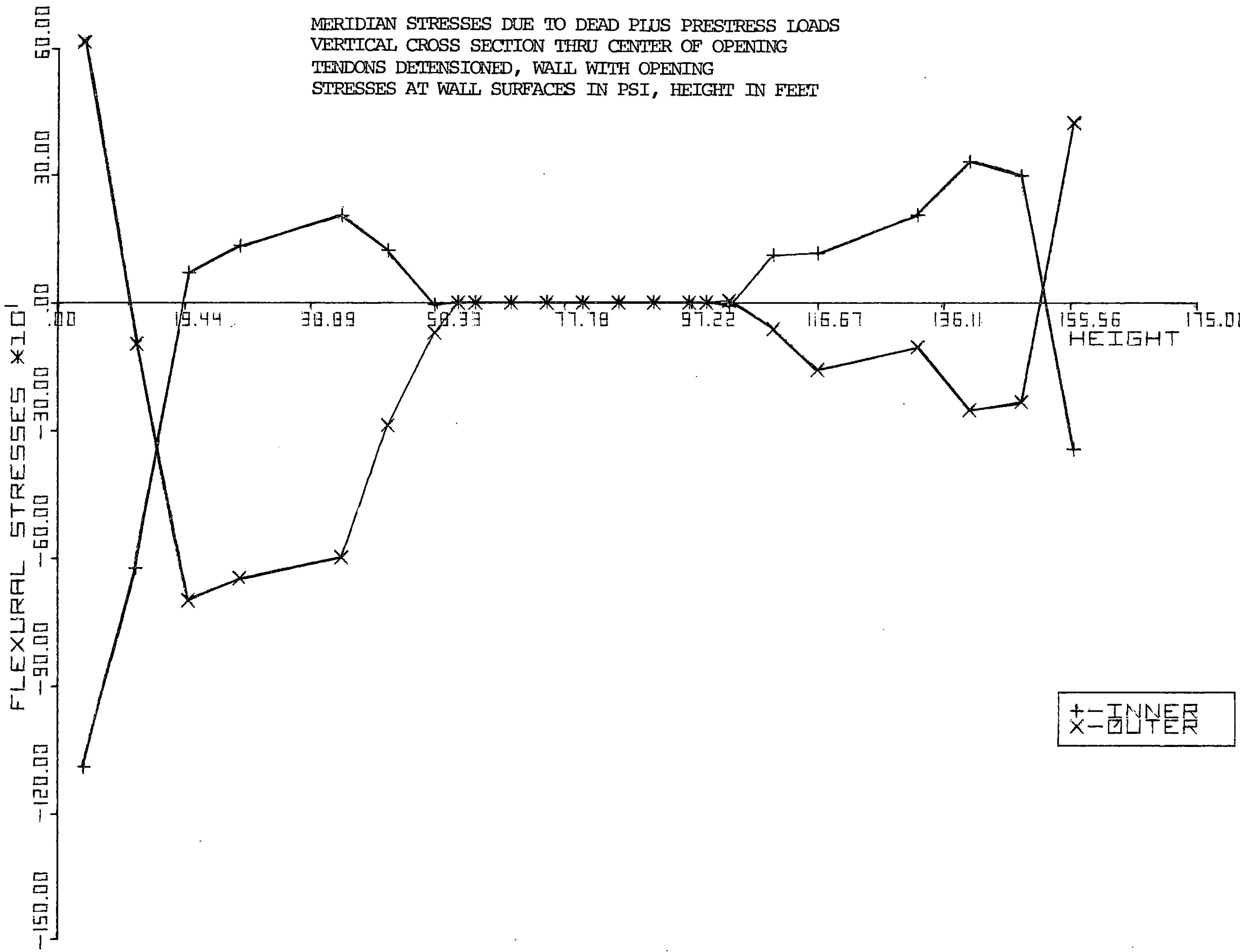


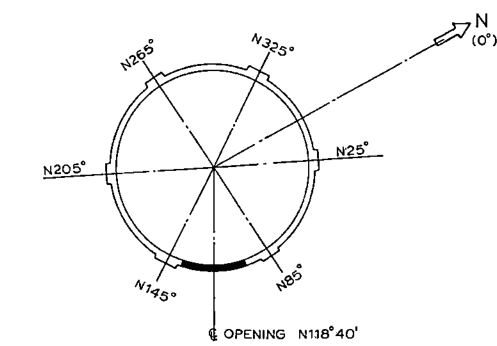


PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

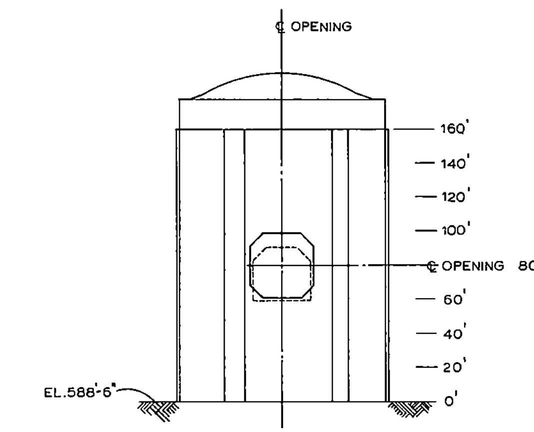
MEMBRANE STRESSES VERSUS
HEIGHT (WITH OPENING)

Figure 4.1-30





TYPICAL HORIZONTAL CROSS-SECTION

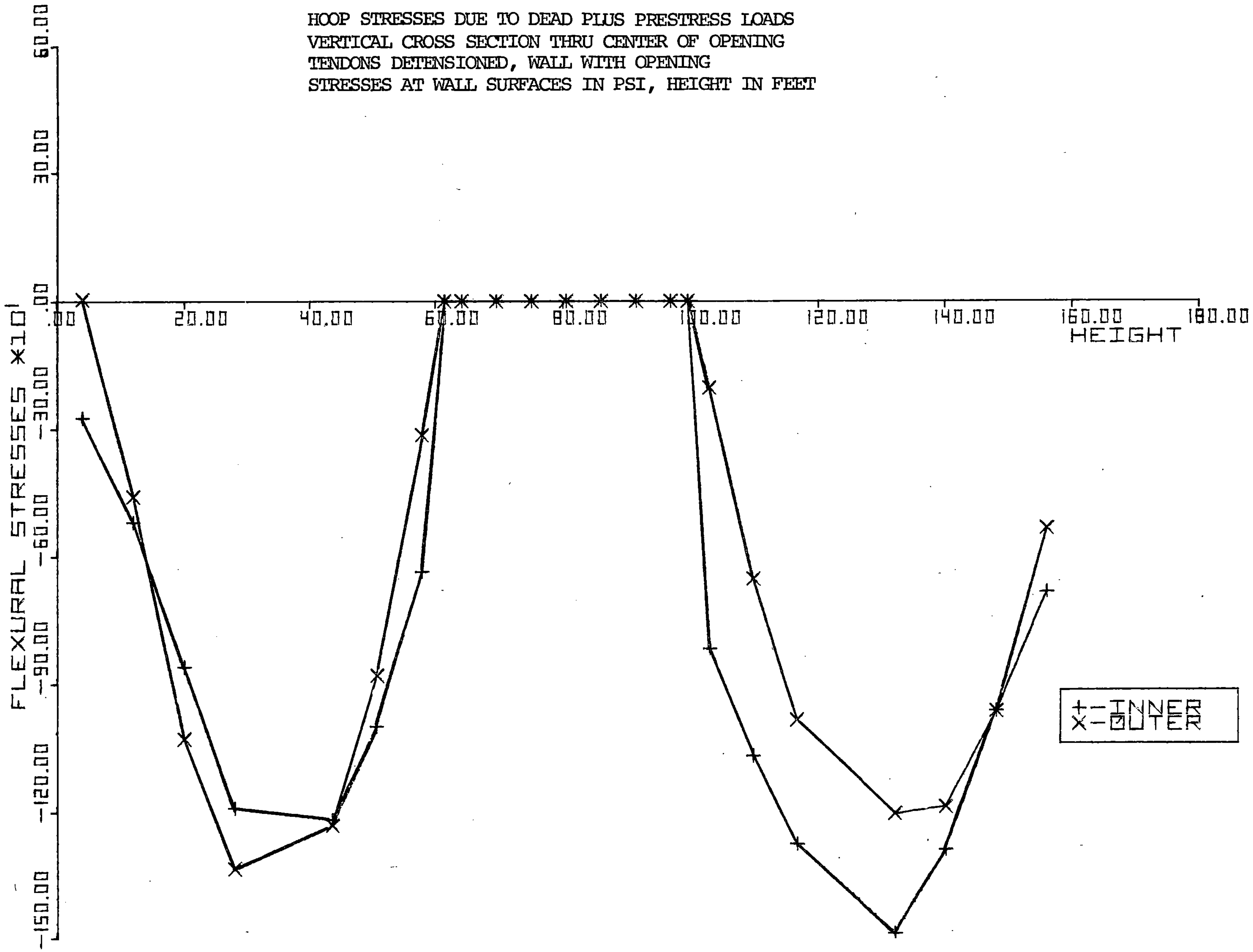


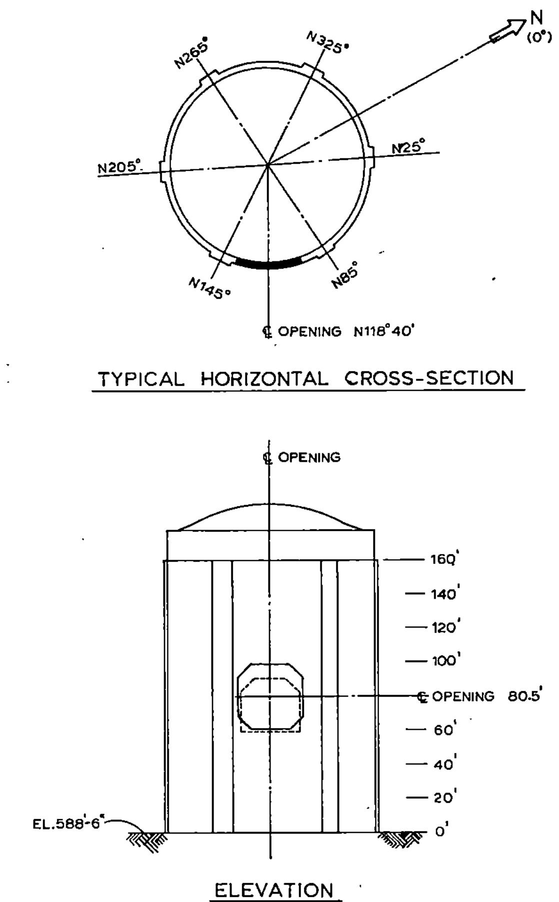
ELEVATION

PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

MERIDIAN STRESSES VERSUS HEIGHT (WITH OPENING)

Figure 4.1-31

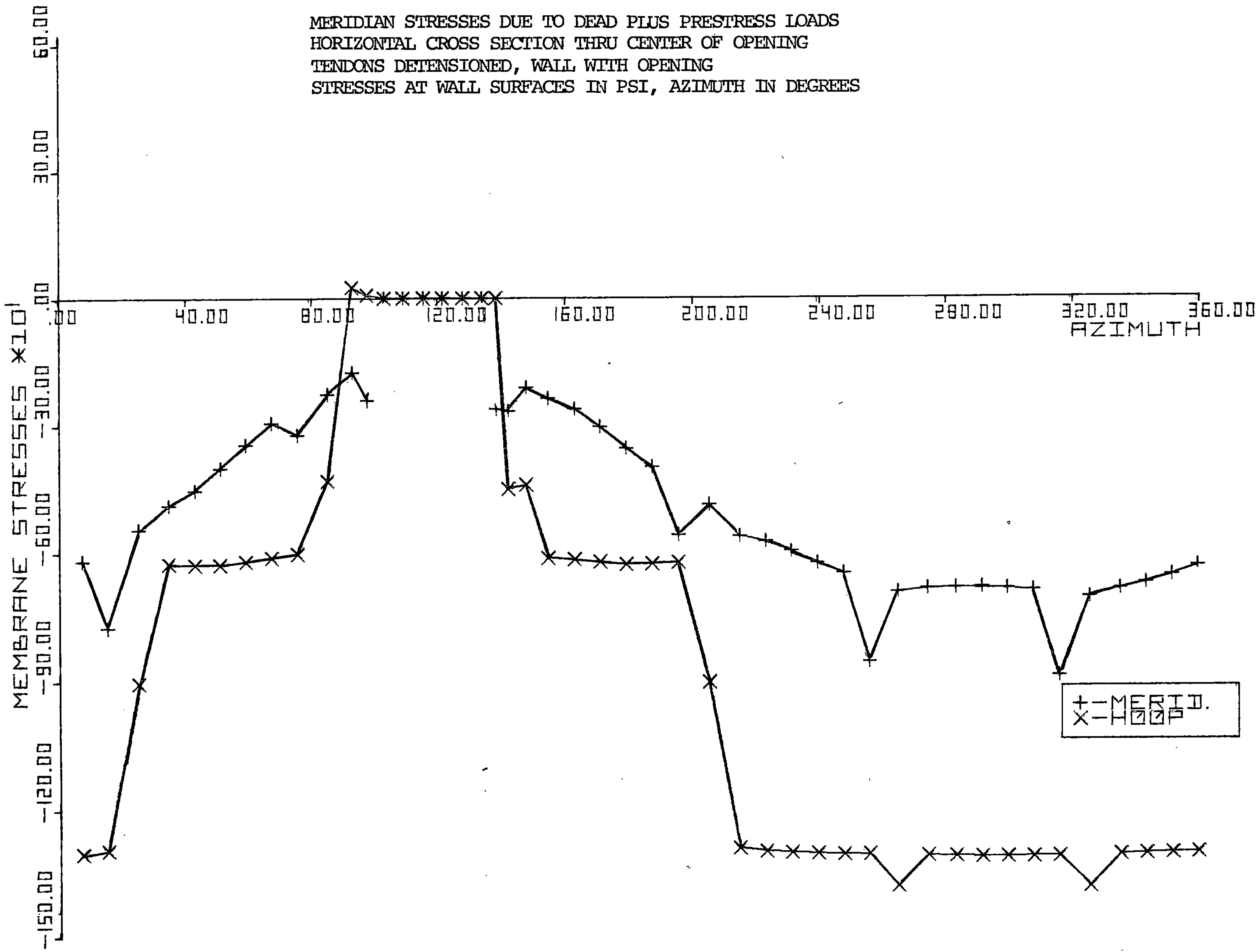


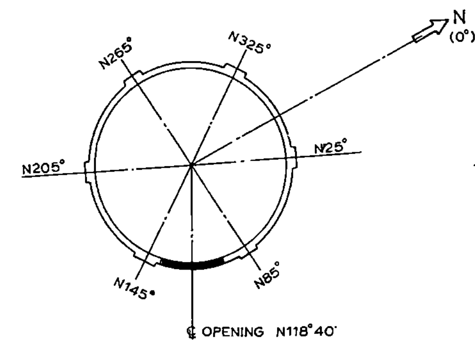


PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

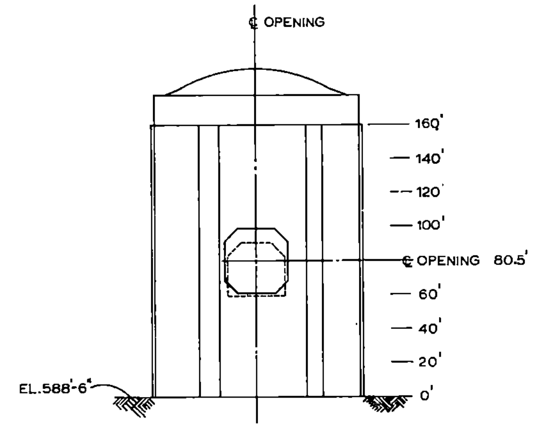
HOOP STRESSES VERSUS
HEIGHT (WITH OPENING)

Figure 4.1-32





TYPICAL HORIZONTAL CROSS-SECTION



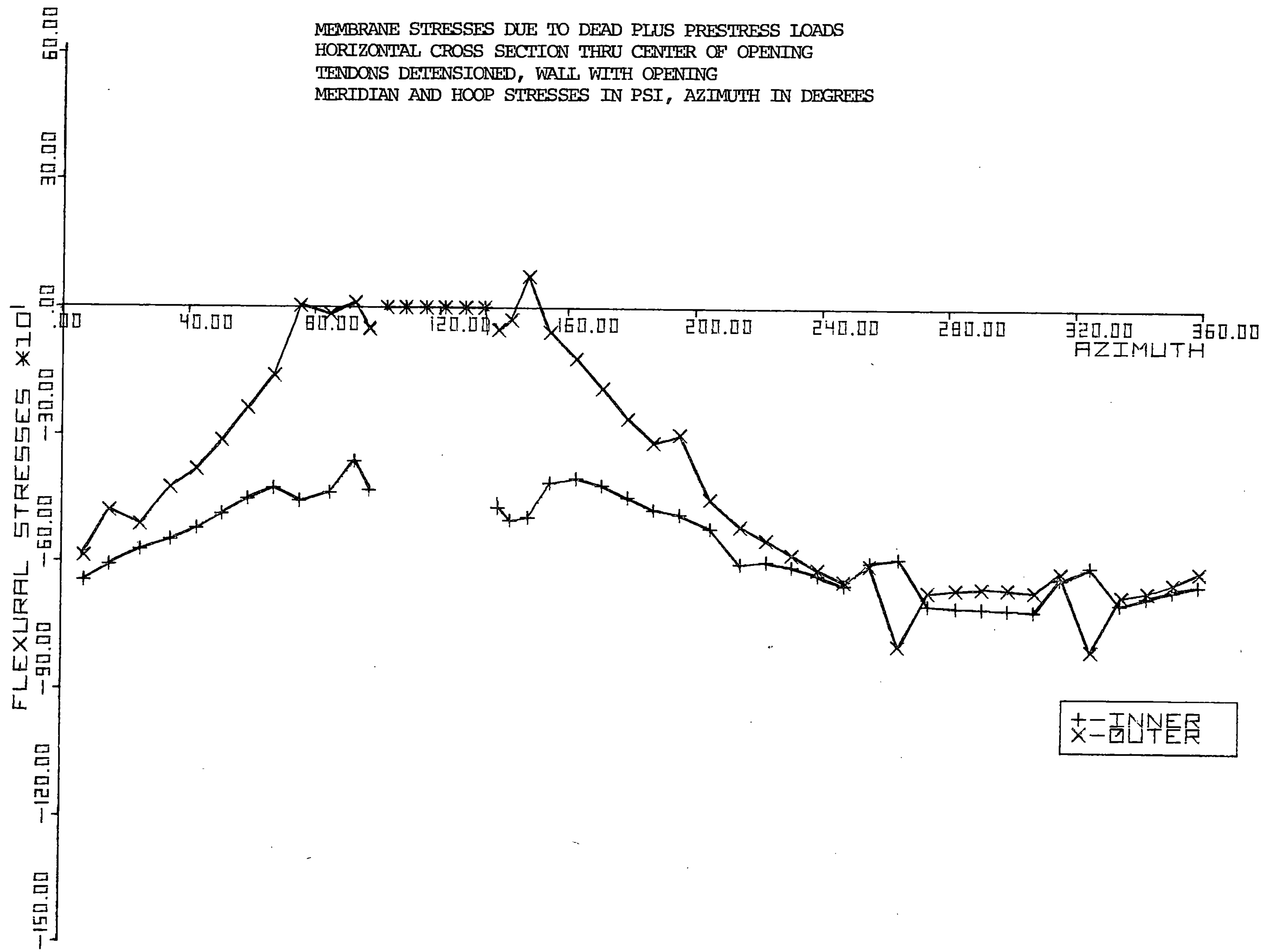
ELEVATION

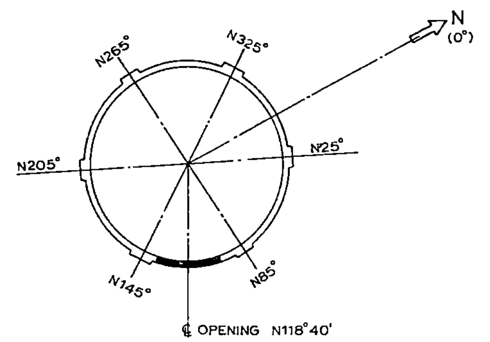
PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

MEMBRANE STRESSES VERSUS
AZIMUTH (WITH OPENING)

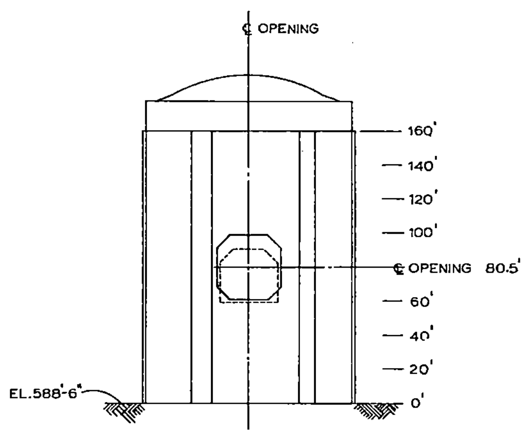
Figure 4.1-33

MEMBRANE STRESSES DUE TO DEAD PLUS PRESTRESS LOADS
HORIZONTAL CROSS SECTION THRU CENTER OF OPENING
TENDONS DETENSIONED, WALL WITH OPENING
MERIDIAN AND HOOP STRESSES IN PSI, AZIMUTH IN DEGREES





TYPICAL HORIZONTAL CROSS-SECTION



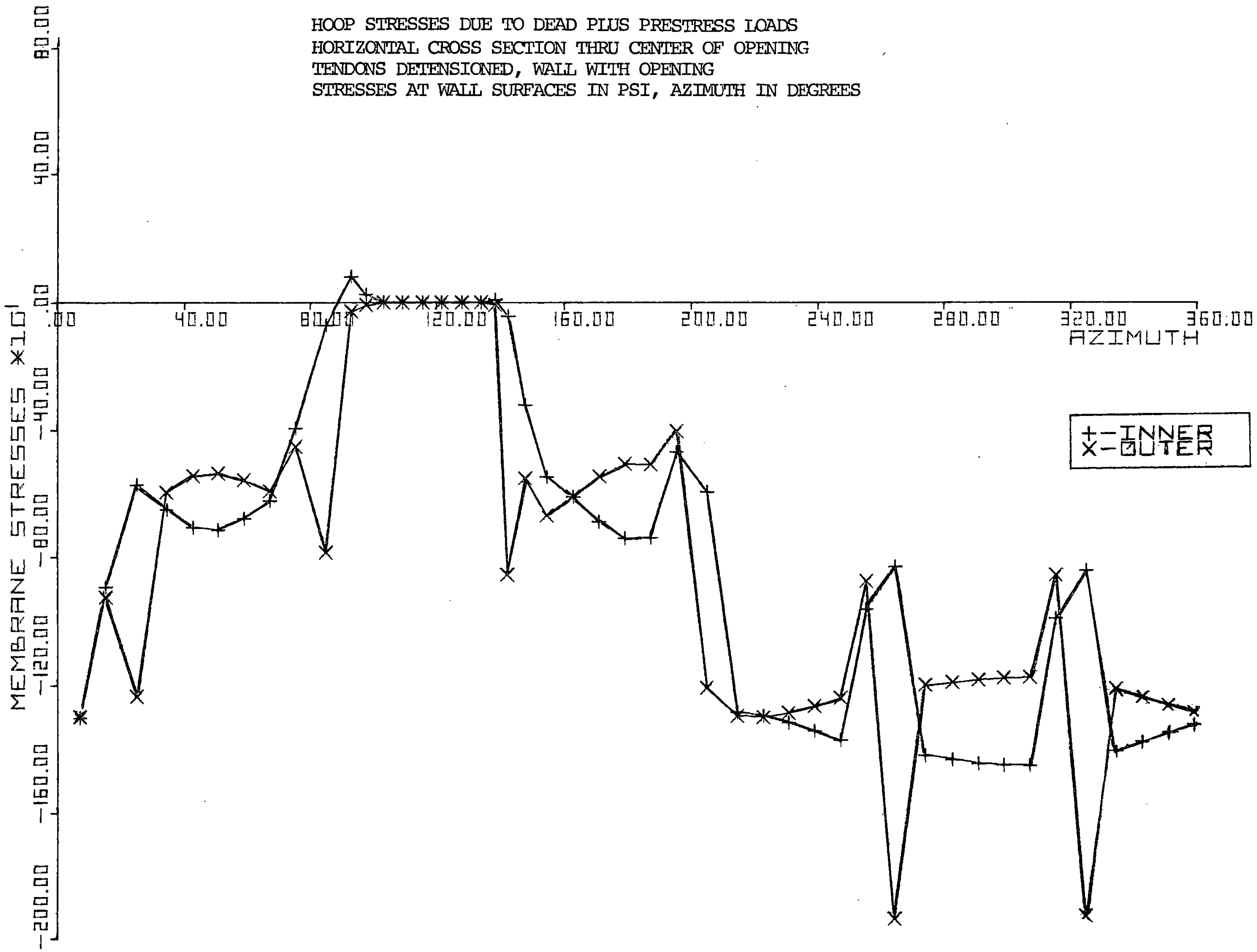
ELEVATION

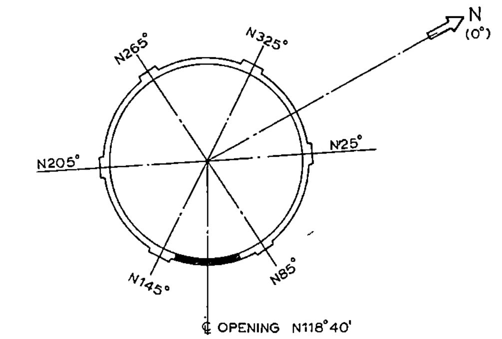
PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

MERIDIAN STRESSES VERSUS
AZIMUTH (WITH OPENING)

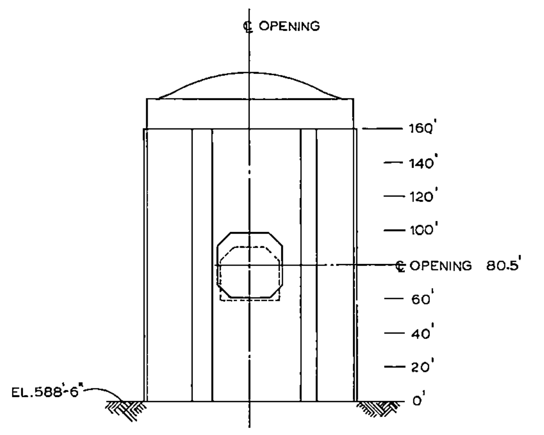
Figure 4.1-34

HOOP STRESSES DUE TO DEAD PLUS PRESTRESS LOADS
HORIZONTAL CROSS SECTION THRU CENTER OF OPENING
TENDONS DETENSIONED, WALL WITH OPENING
STRESSES AT WALL SURFACES IN PSI, AZIMUTH IN DEGREES





TYPICAL HORIZONTAL CROSS-SECTION



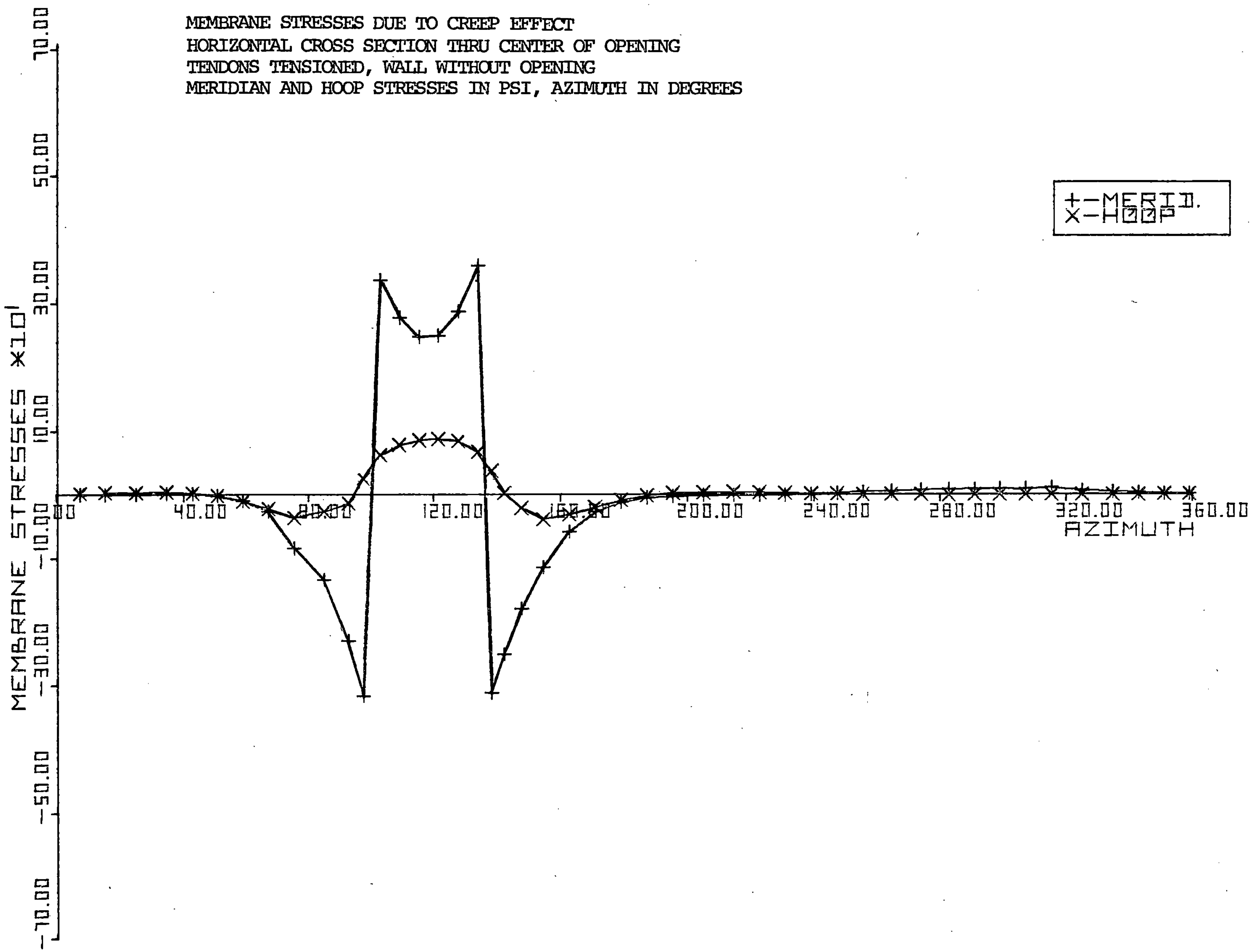
ELEVATION

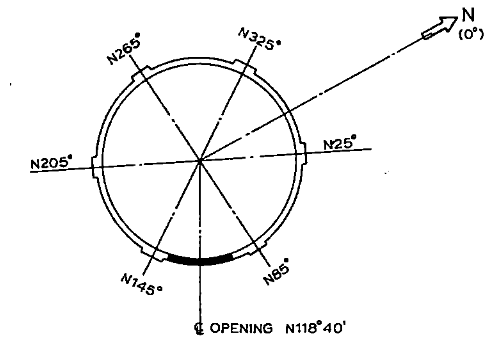
PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

HOOP STRESSES VERSUS
AZIMUTH (WITH OPENING)

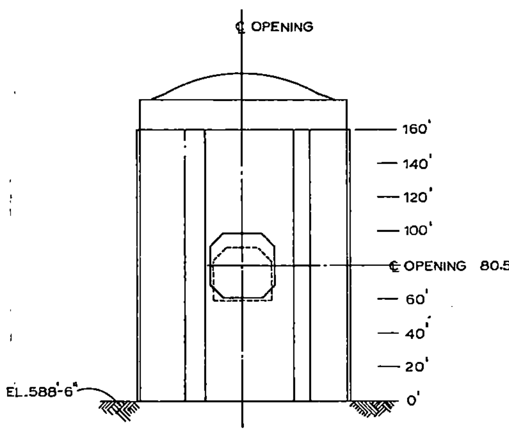
Figure 4.1-35

MEMBRANE STRESSES DUE TO CREEP EFFECT
HORIZONTAL CROSS SECTION THRU CENTER OF OPENING
TENDONS TENSIONED, WALL WITHOUT OPENING
MERIDIAN AND HOOP STRESSES IN PSI, AZIMUTH IN DEGREES





TYPICAL HORIZONTAL CROSS-SECTION

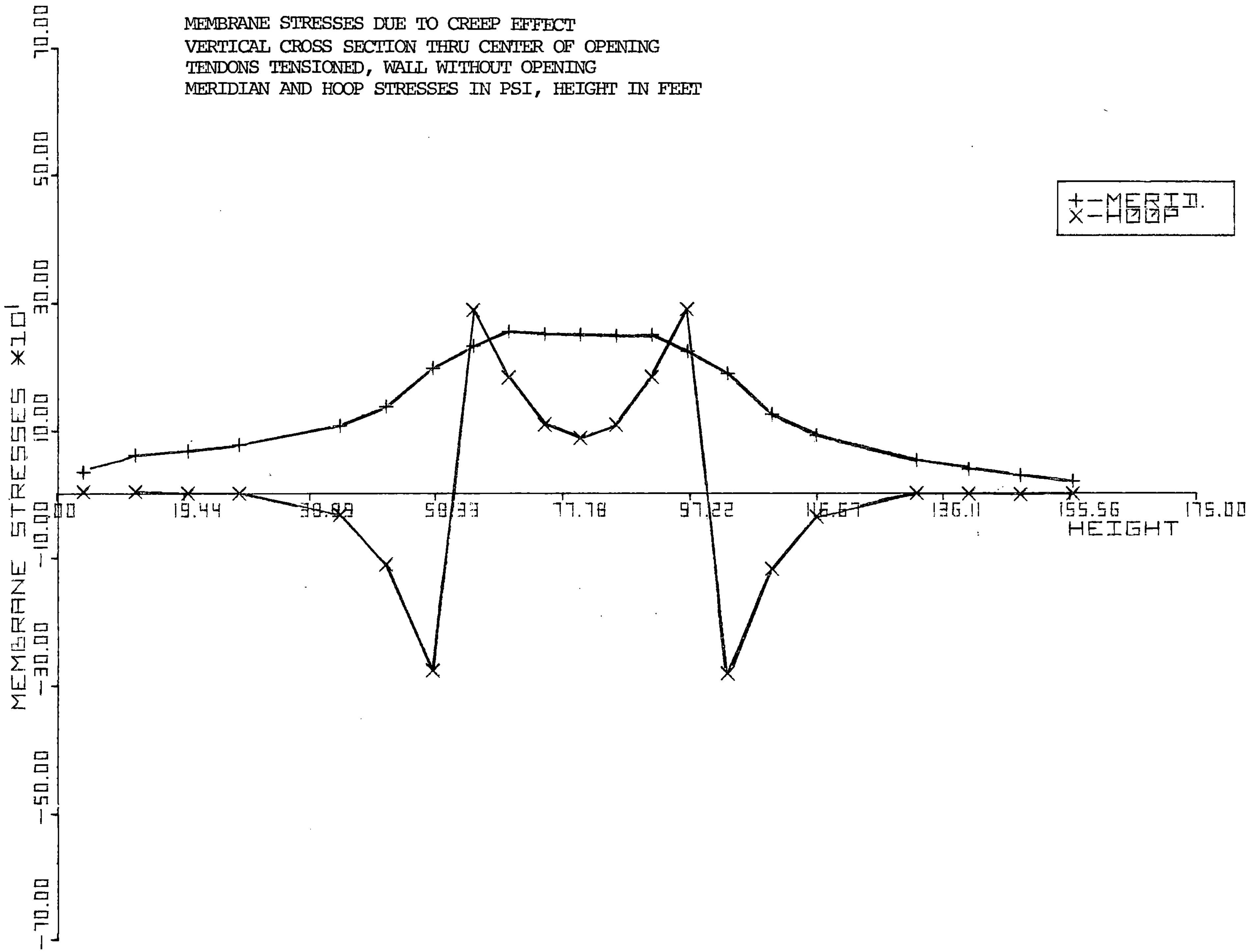


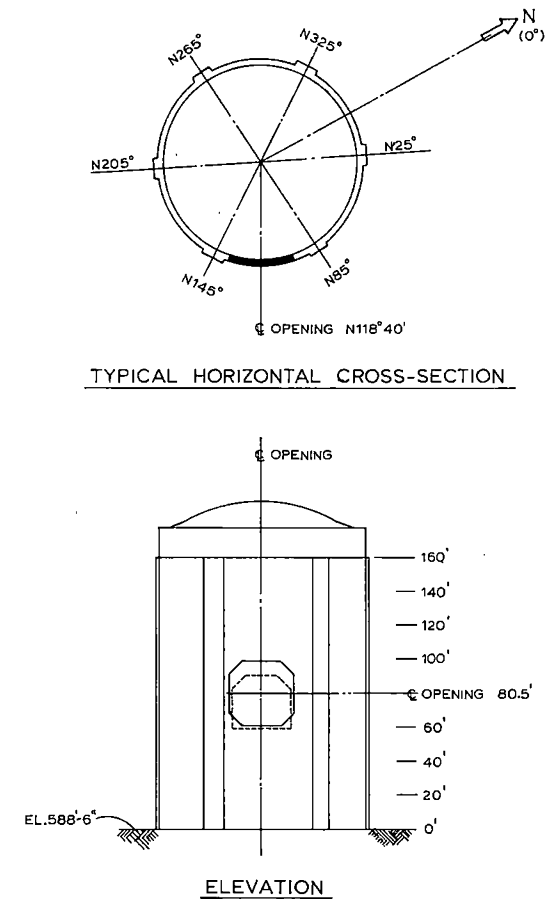
ELEVATION

PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

MEMBRANE STRESSES VERSUS HEIGHT (OPENING CLOSED)

Figure 4.1-36

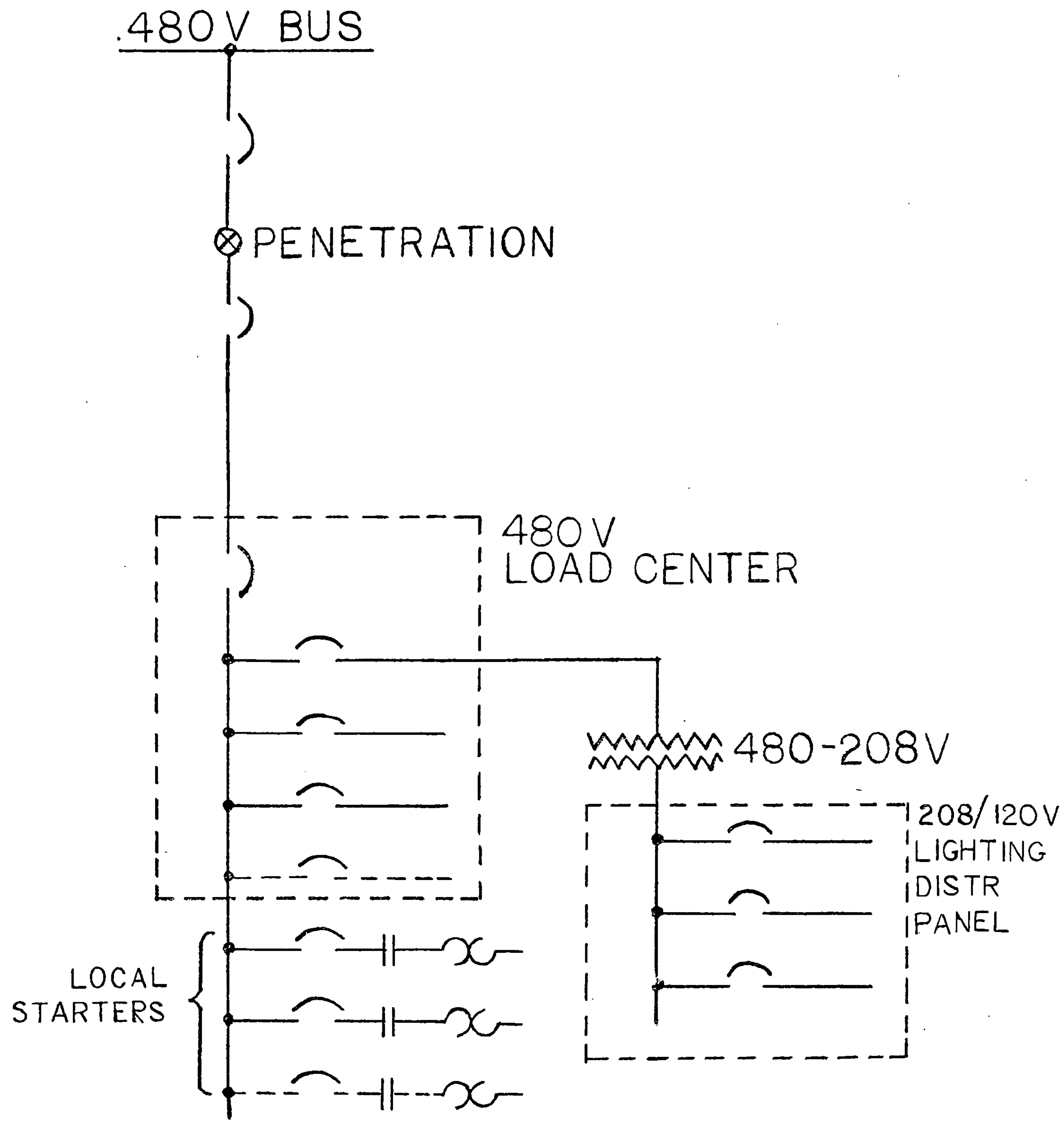




PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

MEMBRANE STRESSES VERSUS AZIMUTH (OPENING CLOSED)

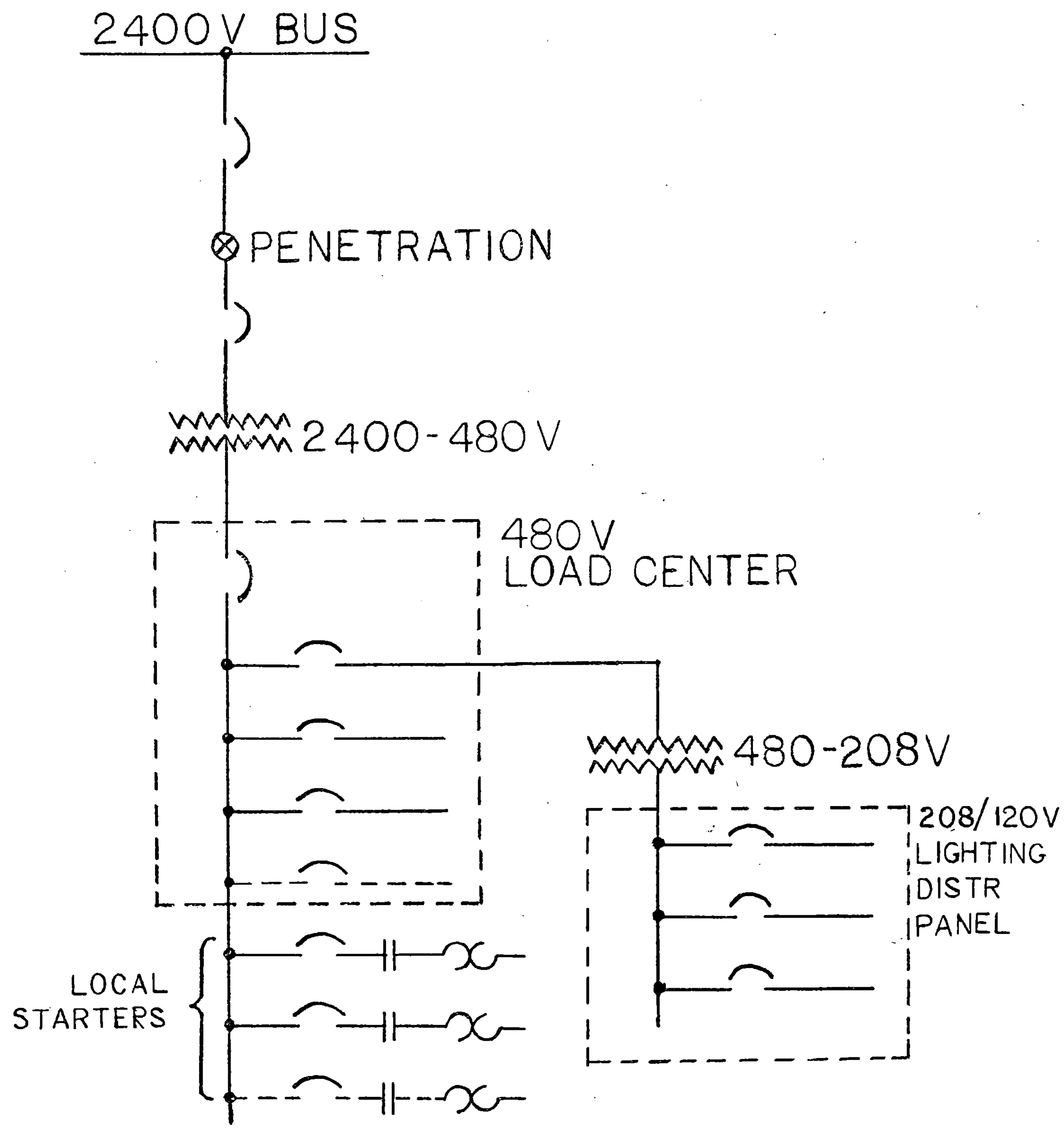
Figure 4.1-37



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

TEMPORARY ELECTRICAL POWER SUPPLIES - ALTERNATE-1

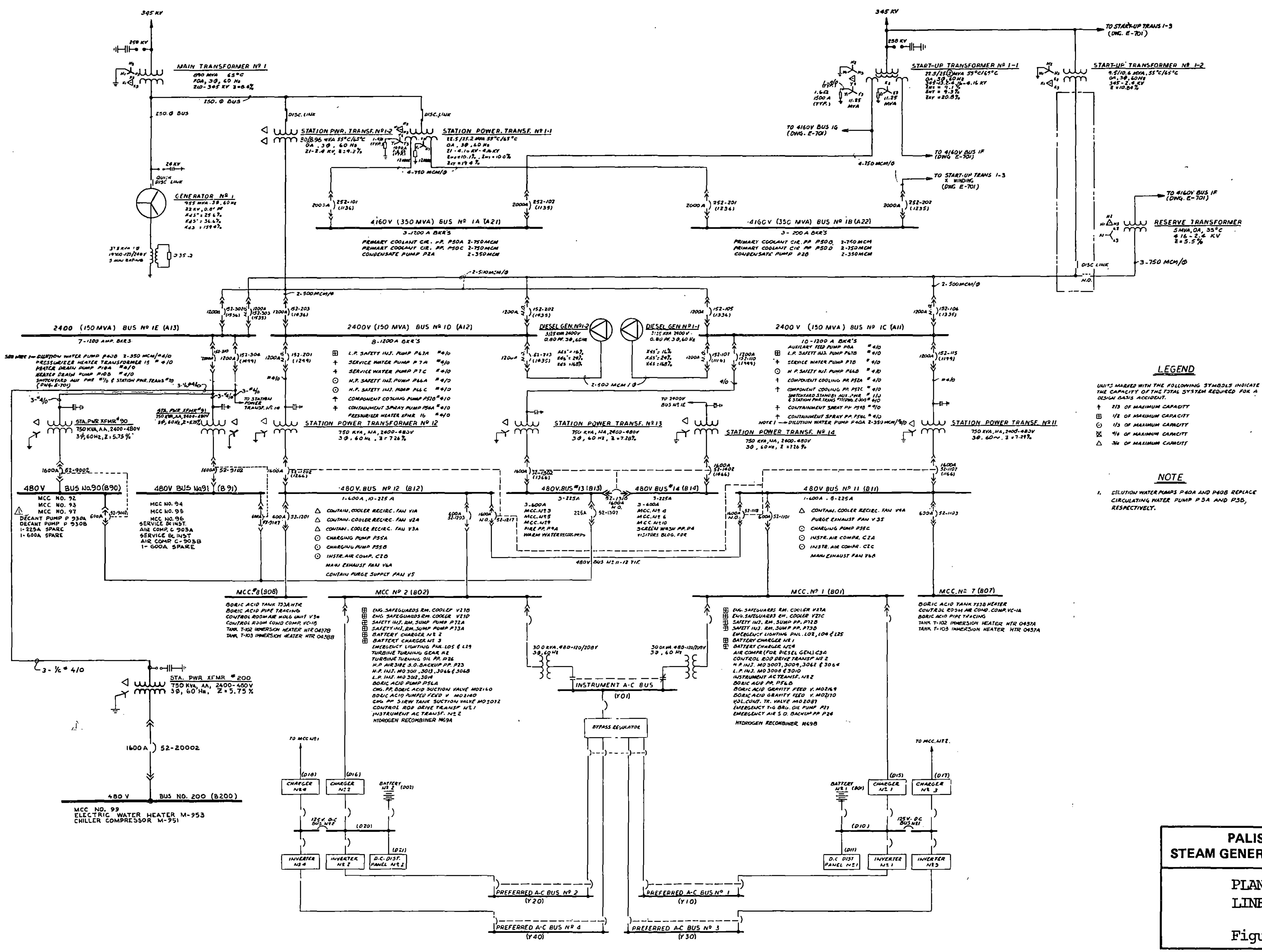
Figure 4.2-1



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

TEMPORARY ELECTRICAL POWER SUPPLIES - ALTERNATE-2

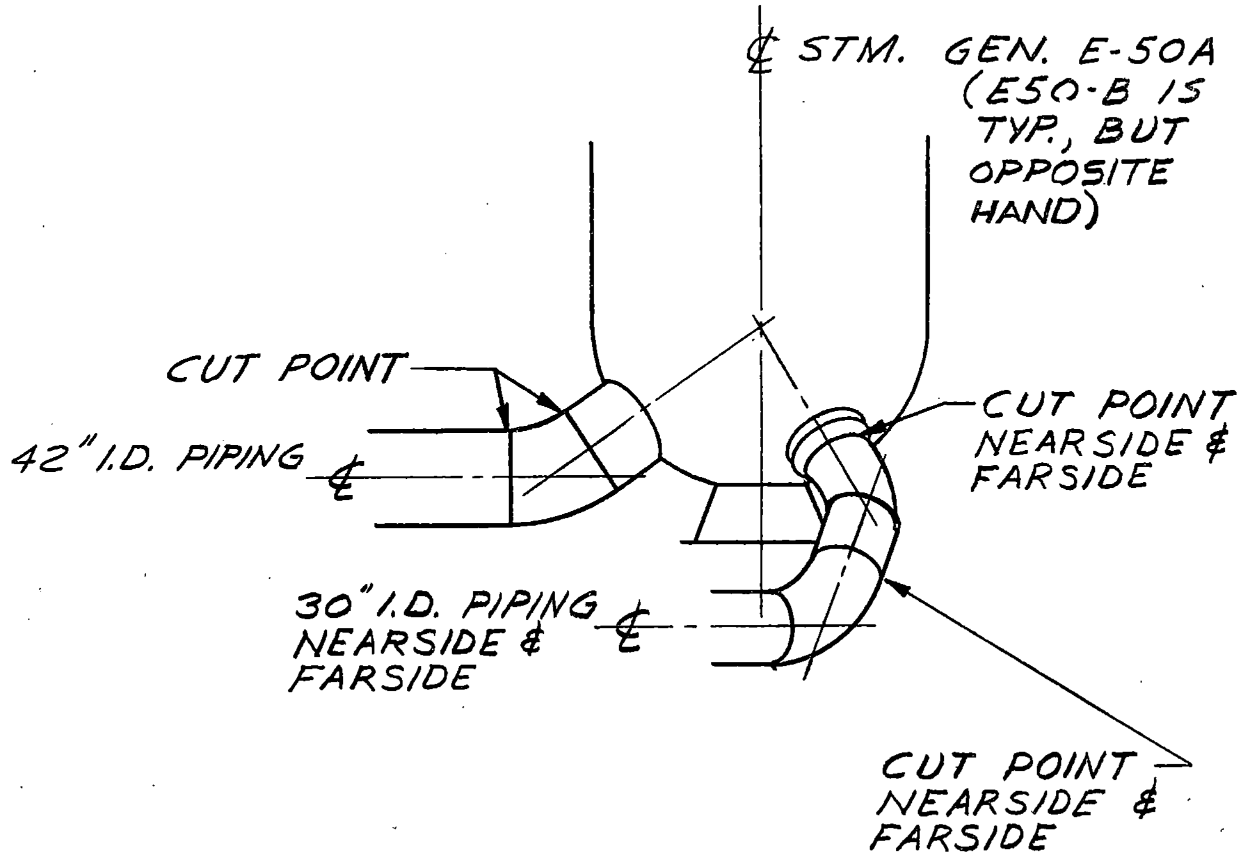
Figure 4.2-2



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

PLANT SINGLE LINE DIAGRAM

Figure 4.2-3

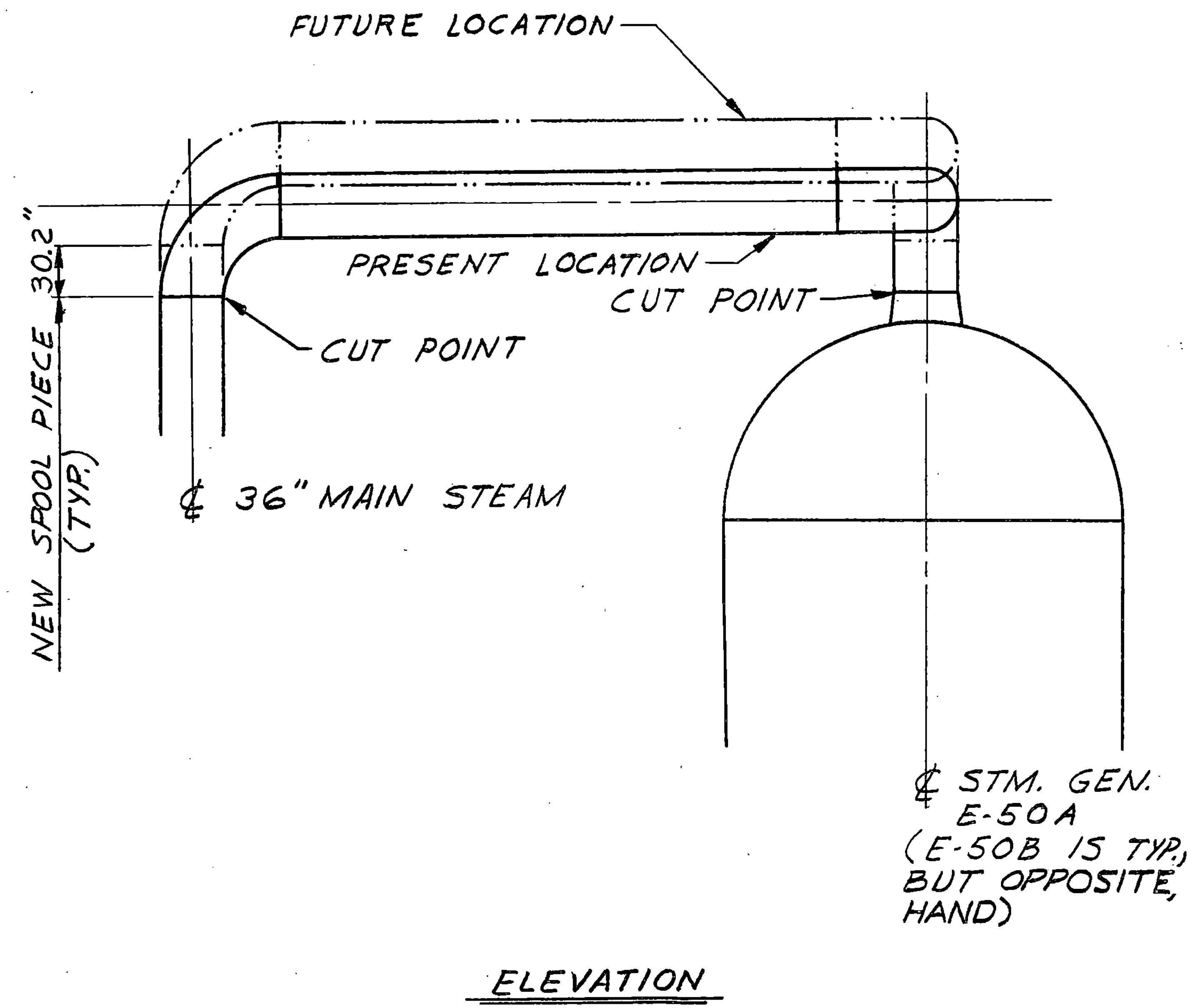


ELEVATION

PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

PRIMARY COOLANT
PIPING CUT POINTS

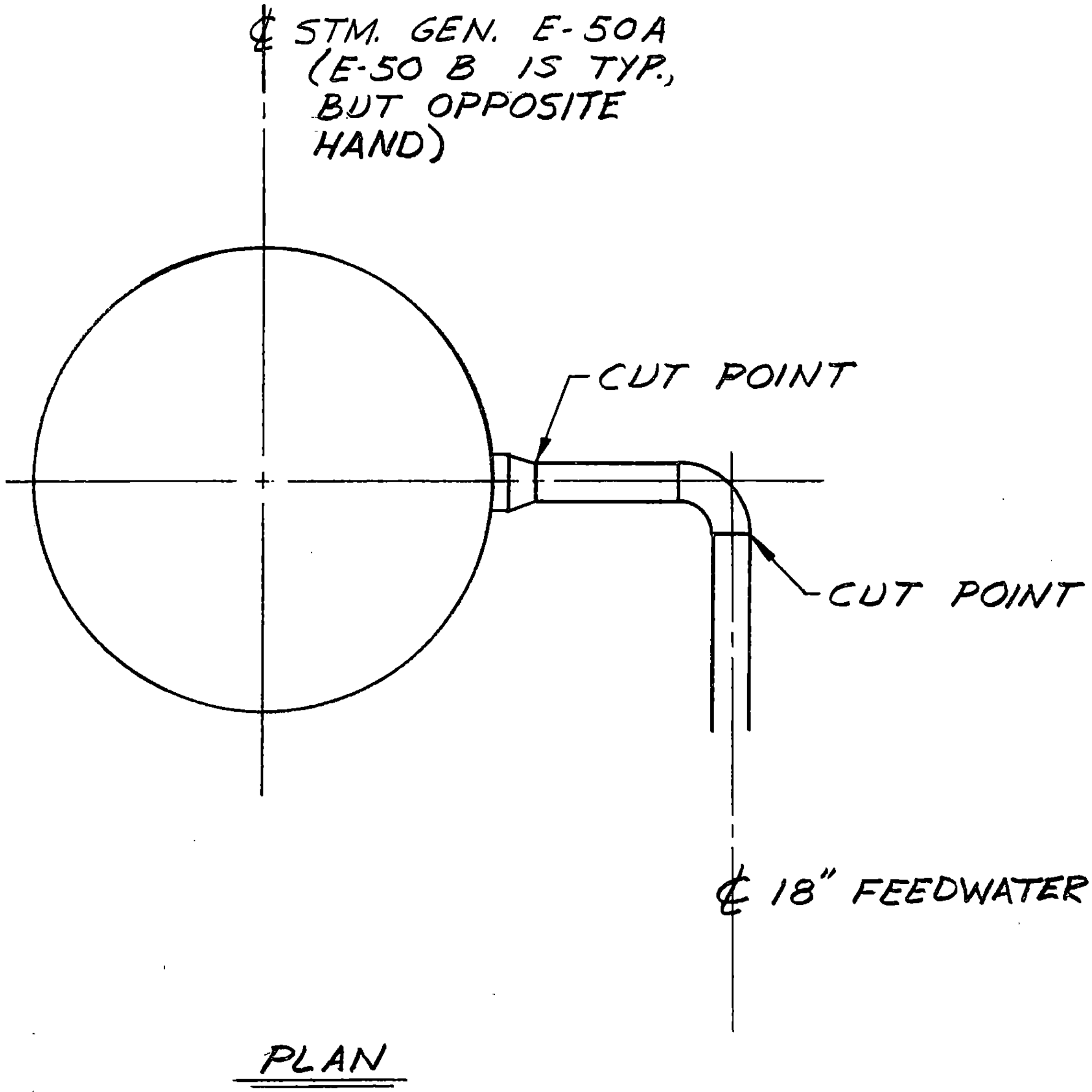
Figure 4.2-4



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

MAIN STEAM PIPING
CUT POINTS

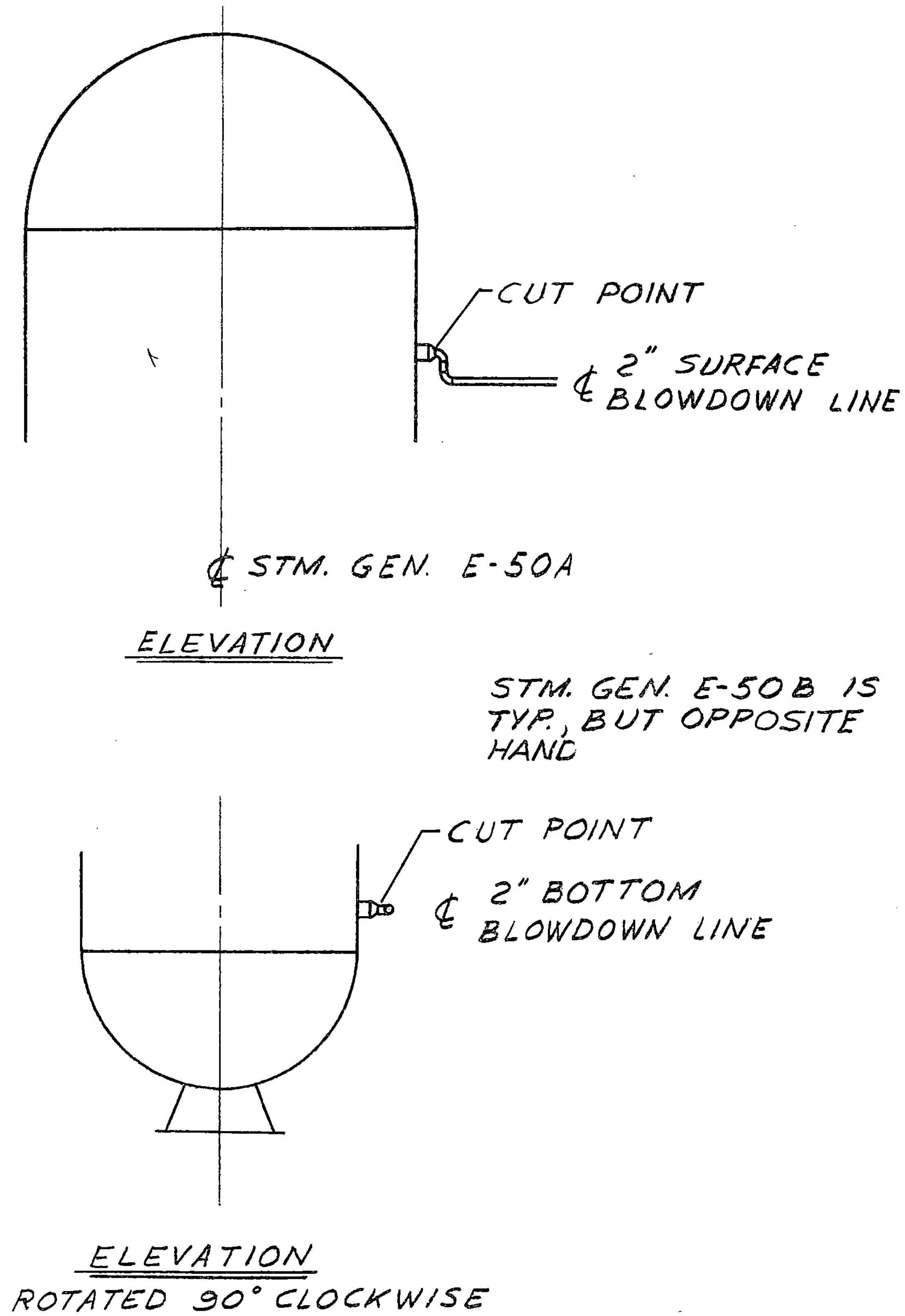
Figure 4.2-5



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

FEEDWATER PIPING
CUT POINTS

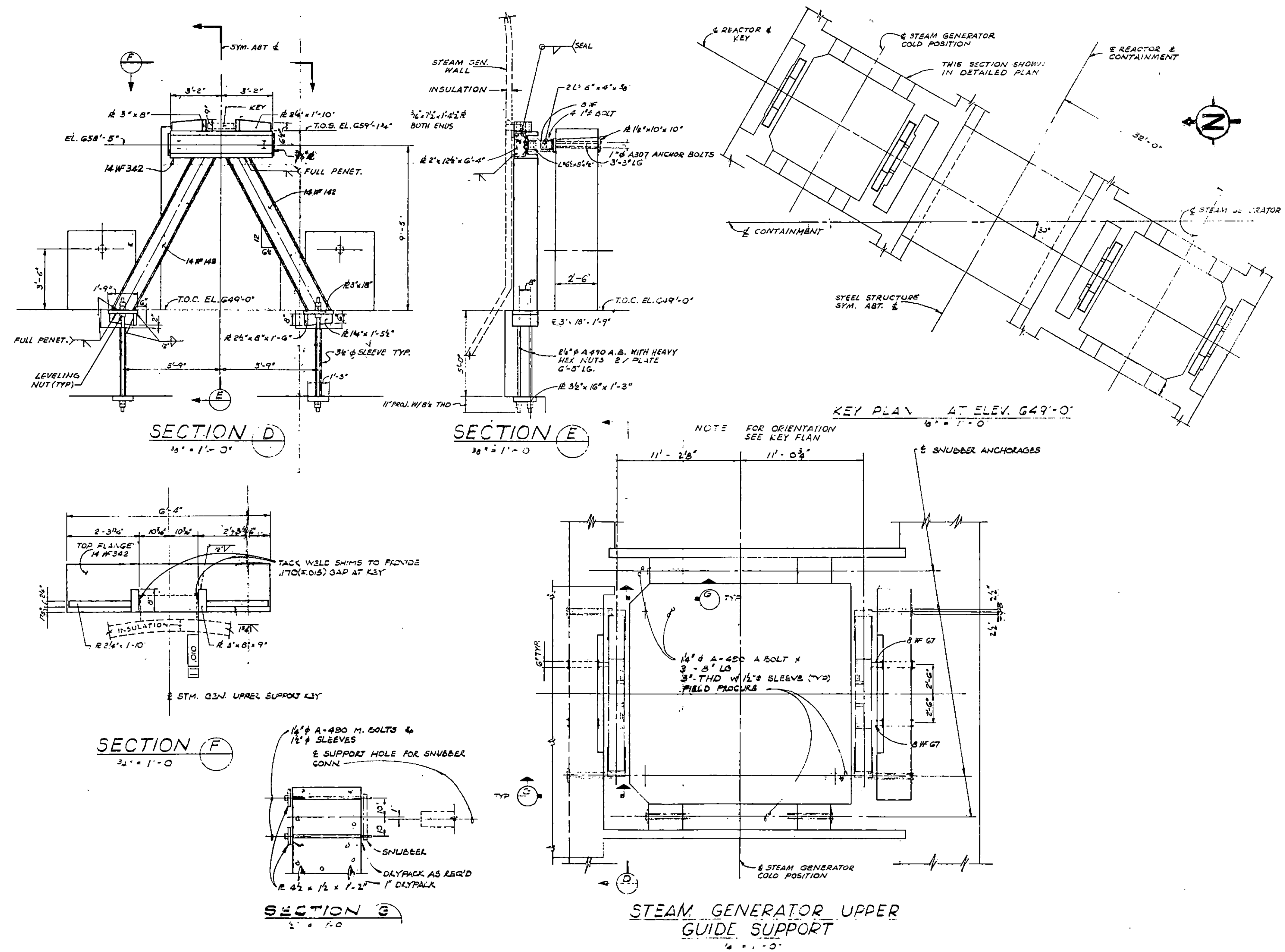
Figure 4.2-6



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

BLOWDOWN PIPING
CUT POINTS

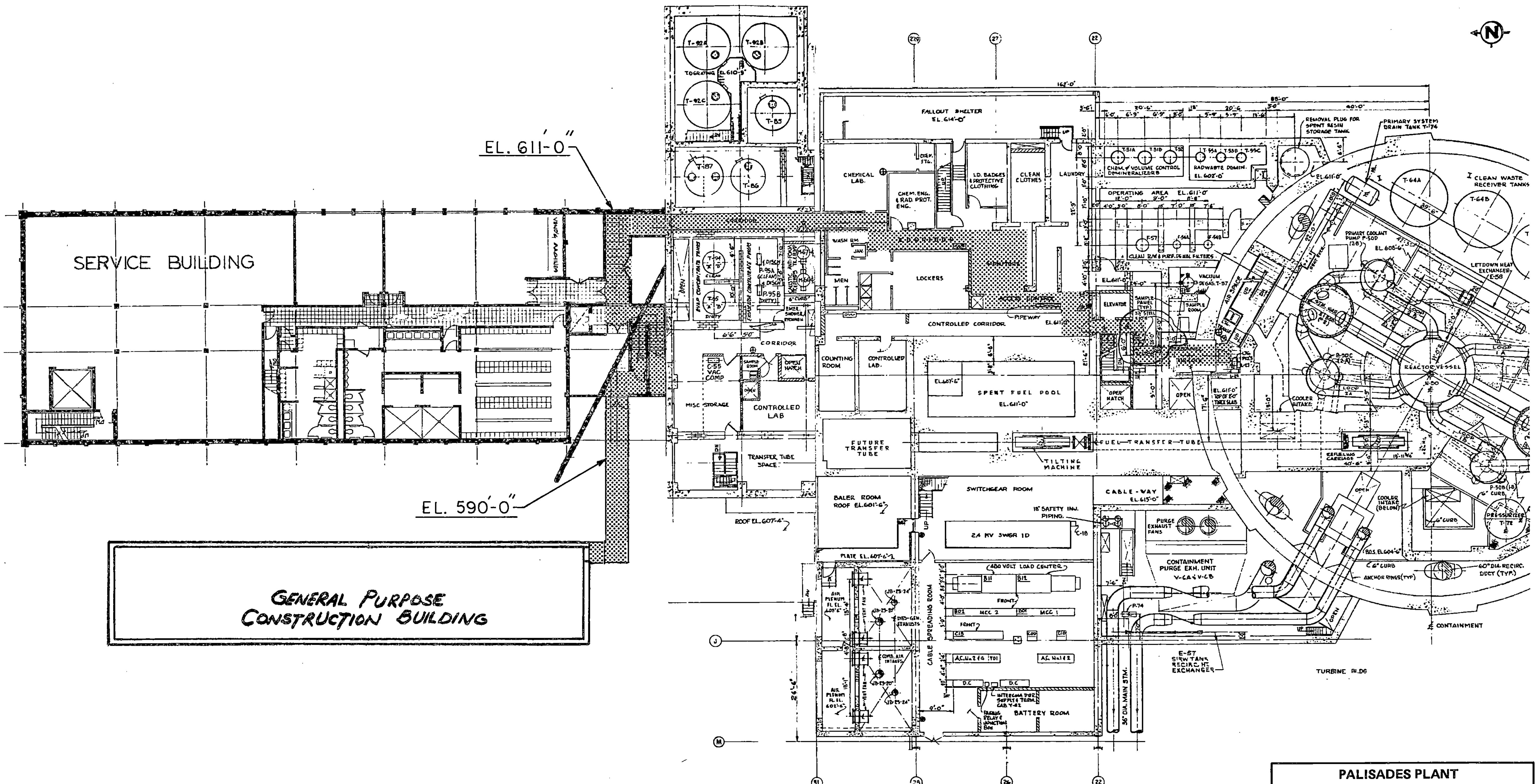
Figure 4.2-7



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

STEAM GENERATOR
UPPER SUPPORT DETAILS

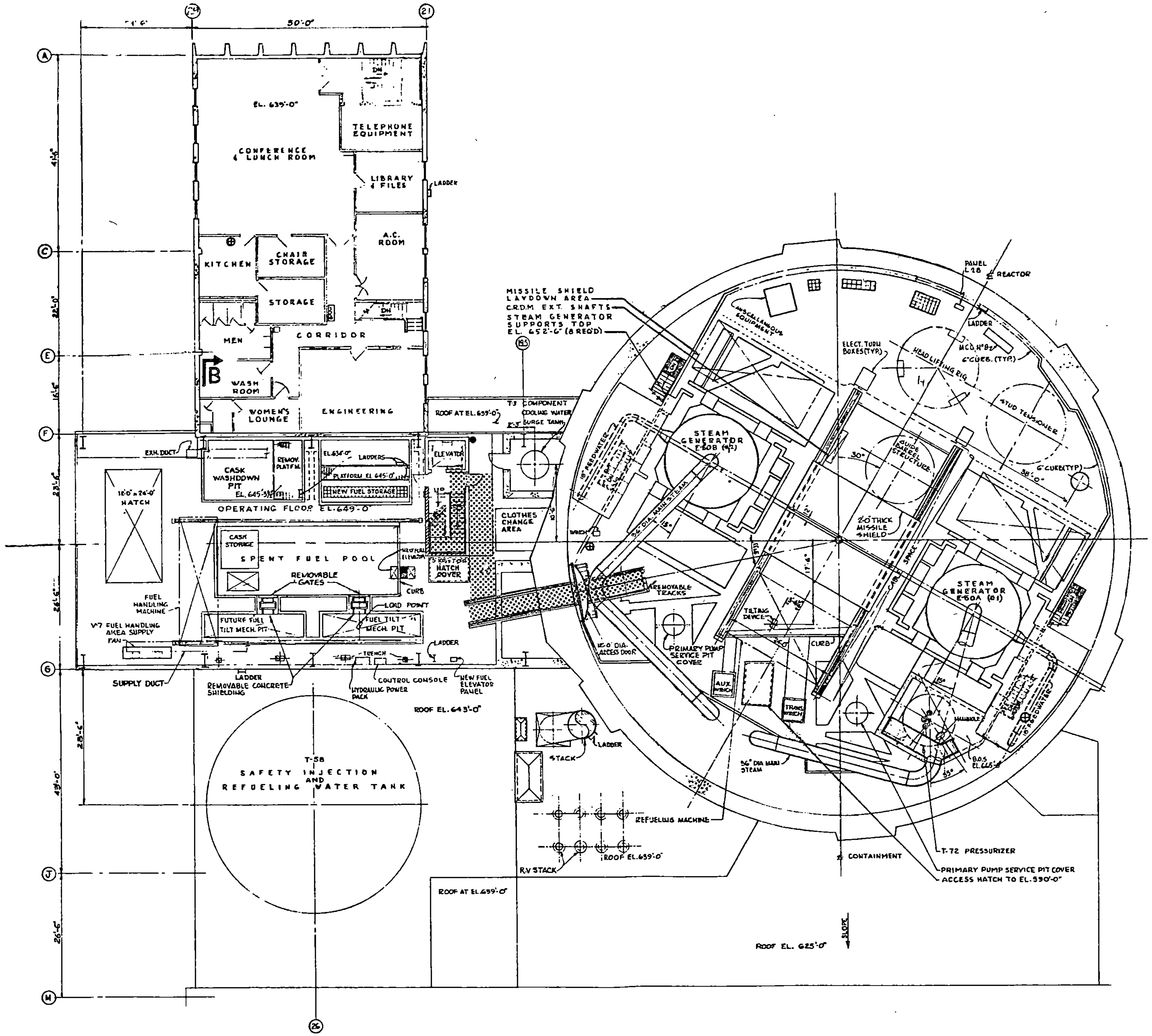
Figure 4.2-8



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

ACCESS CONTROL (EL 590'-0" AND EL 611'-0")

Figure 4.3-1



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

ACCESS CONTROL
(EL 649'-0")

Figure 4.3-2

(APPROXIMATELY 10 WEEKS AFTER SHUTDOWN DURING 1978 REFUELING OUTAGE)

DATE: 3/ 8/ 78 TIME 09:00

INSTRUMENT: Rad Gun SERIAL NO. 2250

6 HOURS ______ 24 HOURS ______ ____ DAYS ____

"A" STEAM GENERATOR:

1. Bottom of Hot Leg 35 mr/hr
2. Bottom of Cold Leg 35 mr/hr
3. Man Way of Cold Leg 1200 mr/hr
4. Man Way of Hot Leg 1000 mr/hr

"B" STEAM GENERATOR:

5. Bottom of Hot Leg 40 mr/hr
6. Bottom of Cold Leg 25 mr/hr
7. Man Way of Cold Leg 1000 mr/hr
8. Man Way of Hot Leg 1100 mr/hr

PRIMARY COOLANT PUMP SUCTION SIDE:

9. "A" Primary Coolant Pump 25 mr/hr
10. "B" Primary Coolant Pump 28 mr/hr
11. "C" Primary Coolant Pump 30 mr/hr
12. "D" Primary Coolant Pump 28 mr/hr

13. Regenerative Heat Exchanger Line 600 mr/hr
5 Feet Above Floor

14. Letdown Heat Exchanger Line 170 mr/hr
Bottom of Inlet Piping

PRIMARY COOLANT LOOP DRAIN LINE:

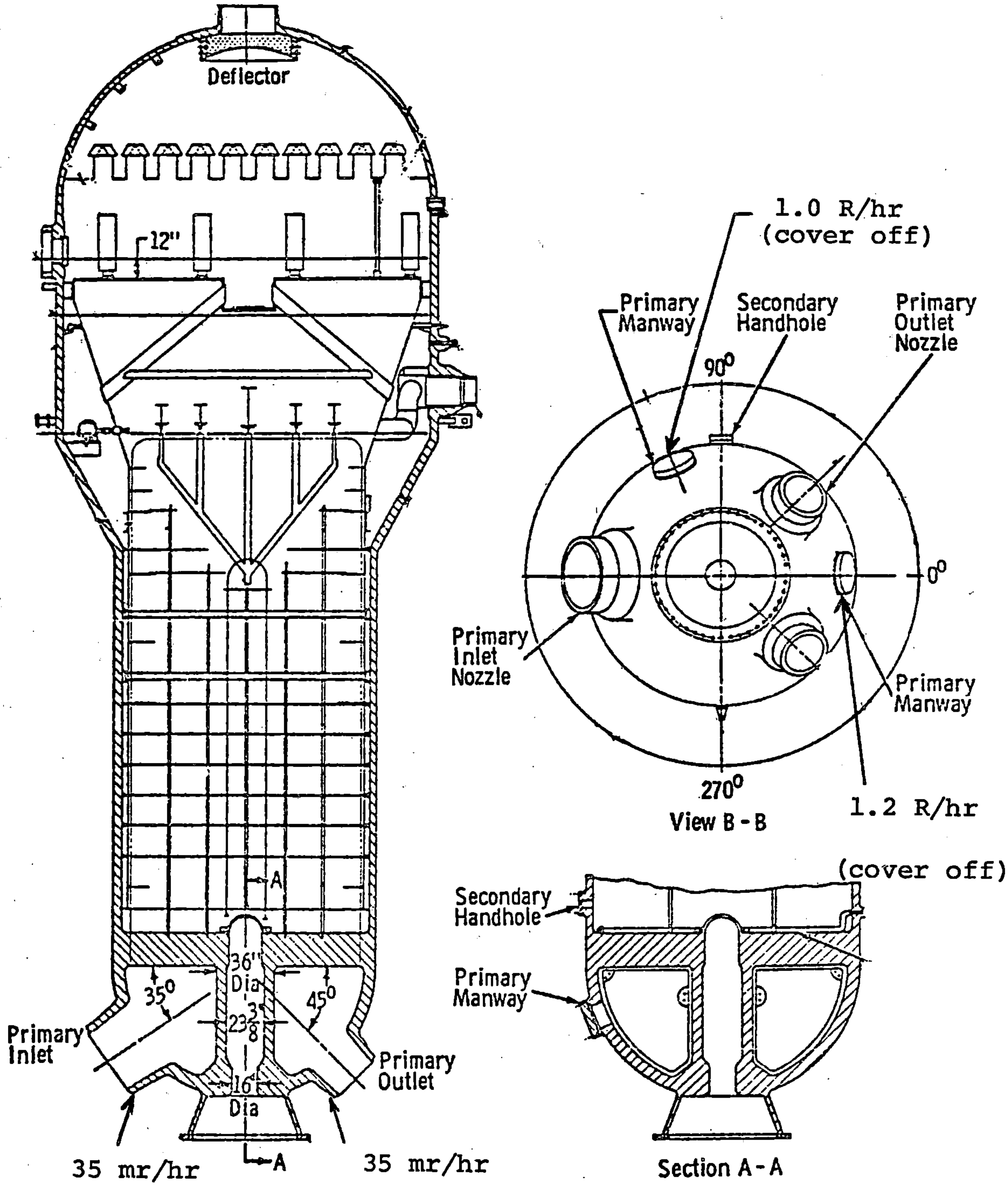
15. "A" Primary Coolant Pump Loop 1A Drain Line ______ mr/hr
16. "B" Primary Coolant Pump Loop 1B Drain Line ______ mr/hr
17. "C" Primary Coolant Pump Loop 2A Drain Line ______ mr/hr
18. "D" Primary Coolant Pump Loop 2B Drain Line ______ mr/hr

19. Primary Coolant Pump Seal Housing (A and D Pumps) ______ A mr/hr
______ D mr/hr
20. Pressurizer Spray Valves (Above Shut-Down Cooling Lines) 250 mr/hr
21. Pressurizer Surge Lines - Two Points #1 ______ mr/hr #2 ______ mr/hr
22. Control Rod Drive (4 points) Above Insulation Heat #1 ______ mr/hr
#2 ______ mr/hr #3 ______ mr/hr #4 ______ mr/hr

Contact Reading Collected by: Ned Campbell
Technician's Signature

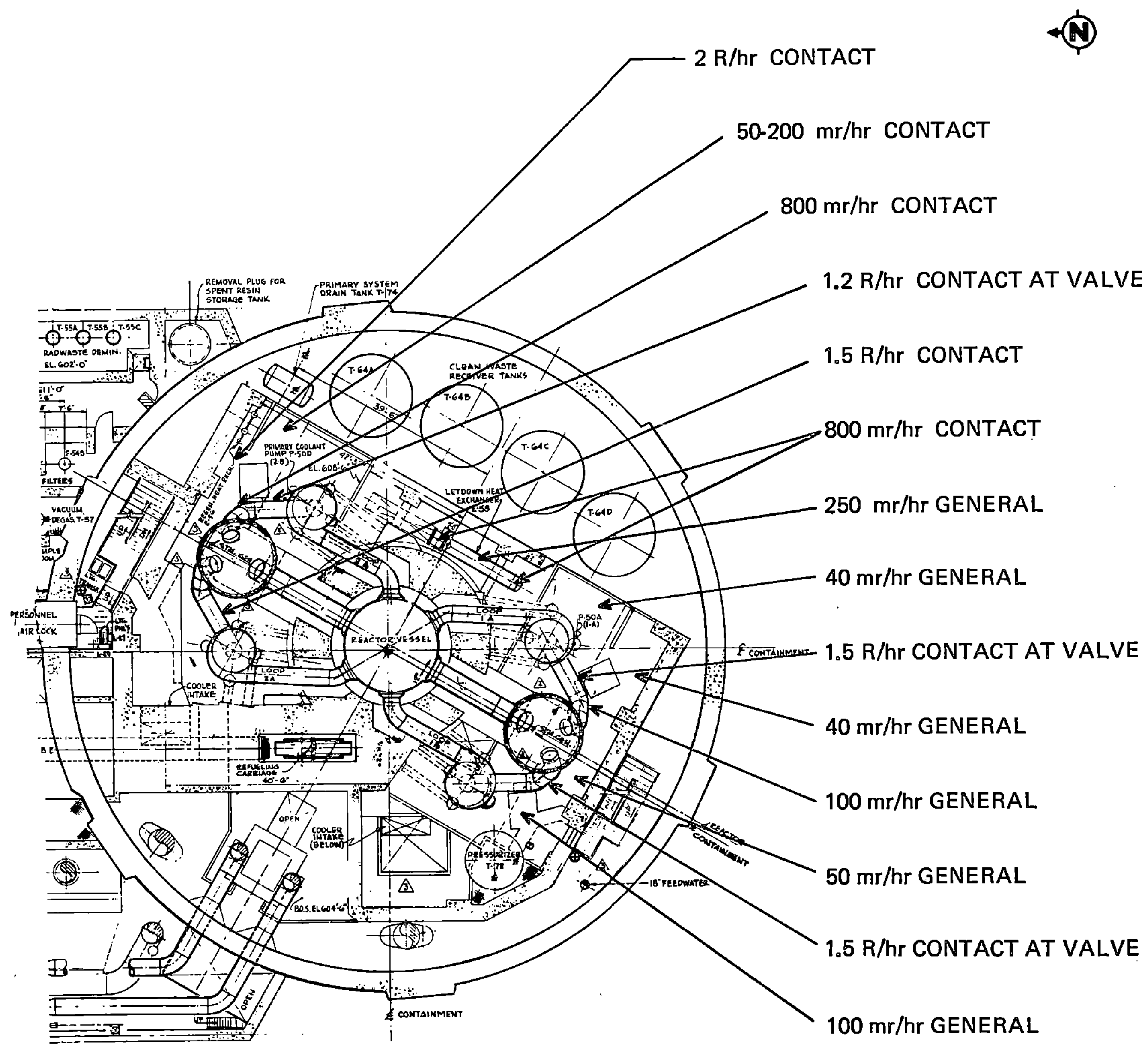
Reviewed by Radiation Protection Supervisor: JD Mills
Signature

PALISADES PLANT STEAM GENERATOR REPAIR REPORT
PRIMARY COOLANT PIPING CONTACT RADIATION SURVEY Figure 4.3-3



PALISADES PLANT STEAM GENERATOR REPAIR REPORT
AVERAGE RADIATION FIELDS 10 WEEKS AFTER SHUTDOWN Figure 4.3-4

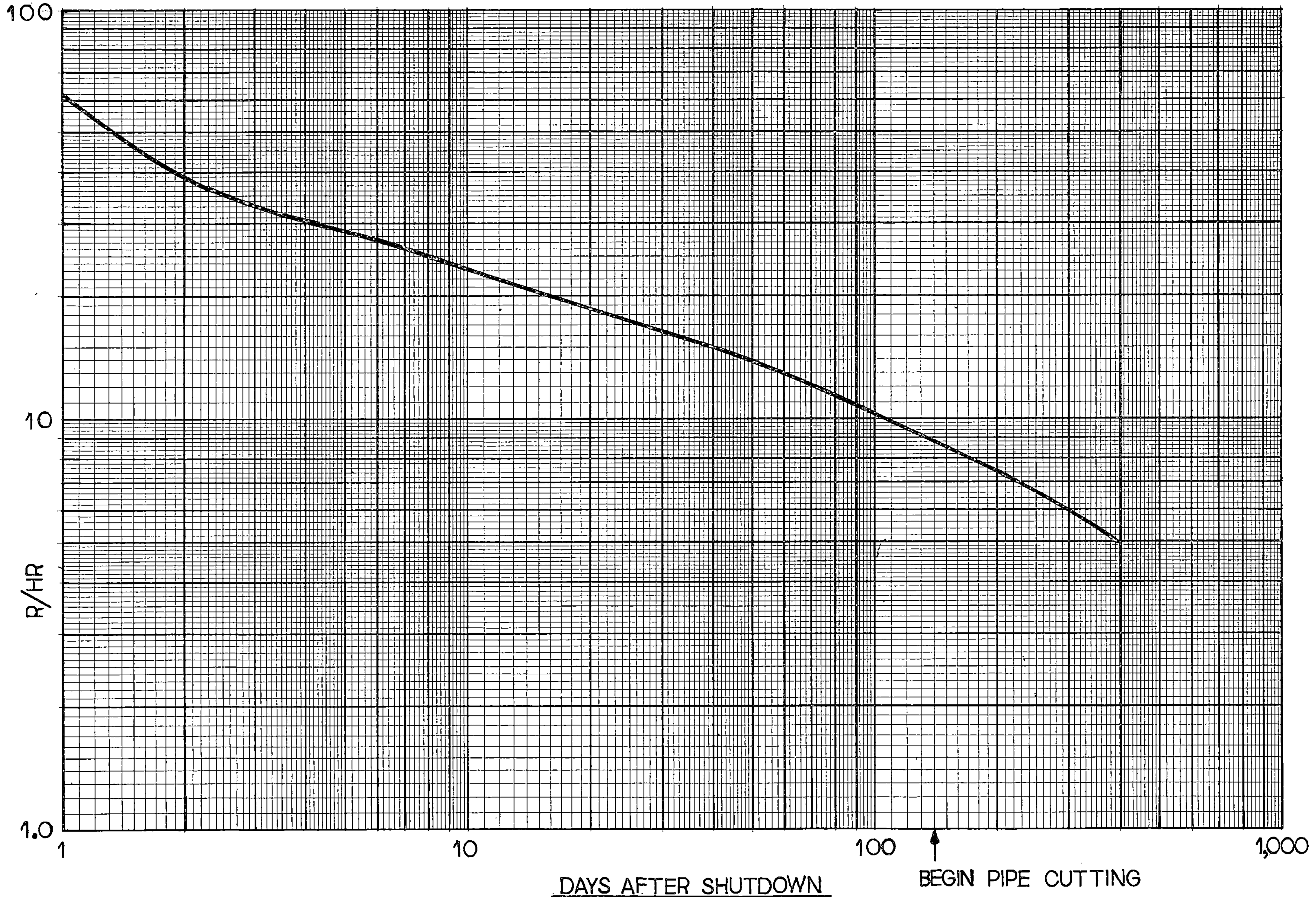
NOTE: READINGS REPRESENT RADIATION LEVELS APPROXIMATELY ONE DAY POST-SHUTDOWN AS-FOUND (NOT ADJUSTED FOR PROJECTED INCREASE AT THE TIME OF THE STEAM GENERATOR REPLACEMENT).



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

RADIATION SURVEY
(EL 607'-6")

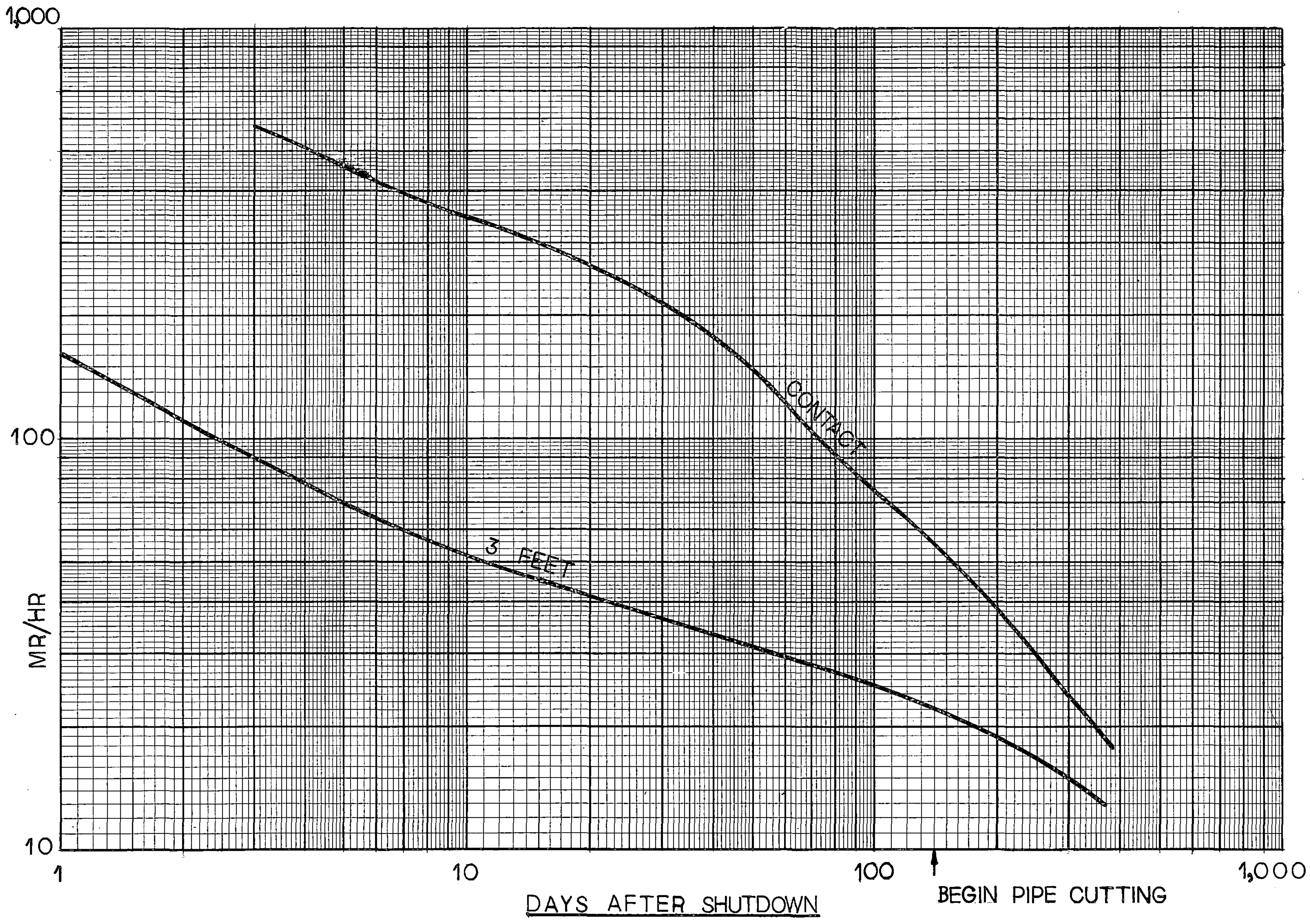
Figure 4.3-5



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

MAXIMUM DOSE RATE INSIDE
STEAM GENERATORS

Figure 4.3-6



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

GENERAL RADIATION FIELD
NEAR STEAM GENERATOR PIPING

Figure 4.3-7

5.0 RETURN TO SERVICE TESTING

As a part of the steam generator repair program, a preoperational, startup, and hot functional test program will be conducted to ensure that the plant is safely returned to full power operation. To meet this objective, the program will require testing of all newly installed equipment, as well as testing of those pieces of equipment that have been affected by the steam generator replacement field construction efforts. Additionally, this test program will include testing of the safety-related equipment in accordance with the Technical Specification's requirements and other testing that is routinely performed after normal fuel reloadings and before return to full power operation.

The test program would include the following:

a. Preoperational checks and inspections to ensure that all newly installed safety-related equipment and equipment that was affected is prepared for functional testing. This would include such things as flushing and cleaning, leak checks, electrical continuity checks, visual checks, instrument calibration checks, verification of valve lineups, and hand rotating pumps.

b. Functional checks of equipment that has been newly installed. This would include such things as steam generator water level instrument checks, steam generator blowdown performance testing, and steam generator performance testing.

c. Functional checks, during and after construction, of any equipment that has been affected by repair. On the basis of a system-by-system review, this would include starting, running and monitoring of pumps, valves, and ancillary systems.

d. Surveillance of equipment in accordance with the technical specifications current at the time of the steam generator replacement, such as valve operability and pump operability

e. Startup testing which would normally be performed between routine fuel loading and return to full power operations, such as rod drop tests and low power physics tests

f. Review of jumper log, etc, to ensure that any temporary jumpers, etc, have been properly dispositioned

g. Performance of an integrated leak rate test and any other testing that may be necessary to return the containment building to service

h. A final overall review of the plant and systems to ensure readiness for return to service and power operation

6.0 SAFETY EVALUATIONS

6.1 FSAR EVALUATIONS

6.1.1 INTRODUCTION

The purpose of this section is to evaluate the impact, if any, of the replacement steam generators on the accident analysis transients for the Palisades Plant. Under the guidelines specified in 10 CFR 50.59, such an evaluation is required to verify that no unreviewed safety concerns or changes to the Palisades Plant Technical Specifications occur. This section provides a qualitative discussion of the effect on the accident analyses of steam generator parameter changes resulting from steam generator repair. Conclusions are made in this section concerning the validity of the original FSAR to the repaired units. Consistent with the requirements of 10 CFR 50.59, licensing regulations and guidelines of the original licensing of the Palisades Plant are assumed to apply, and only changes in the safety analyses due to the equipment changes are considered.

The relevant plant operating parameters and steam generator design parameters have been compared for the original and replacement steam generators in Section 2.1. While incorporating design improvements that will improve the flow distribution and tube bundle accessibility and reduce secondary side corrosion, the replacement steam generators continue to match the design performance of the original steam generators. It may be noted from Section 2.1 that there is very little effect on plant operating parameters due to the replacement of the steam generators. It is, therefore, to be anticipated that the impact on the accident analyses will be insignificant. The results of the accident evaluation show that no unreviewed safety concerns exist because of operation with the replacement steam generators.

6.1.2 NON-LOCA ACCIDENTS

Analyses of the following non-LOCA design basis events were originally presented in the Palisades Plant FSAR. These events were evaluated to determine the effect, if any, of the replacement steam generators on the plant transient response.

a. Control rod withdrawal

b. Boron dilution

c. Full-length control rod drop

d. Malpositioning of the part-length control rod group

e. Loss of coolant flow

f. Idle loop startup

g. Excessive feedwater

h. Excessive load increase

i. Loss of load

j. Loss of feedwater flow

k. Steam line rupture inside containment

l. Steam generator tube rupture

m. Control rod ejection

The excessive feedwater, excessive load increase, loss of load, loss of feedwater flow, steam line rupture, and steam generator tube rupture events are discussed in the succeeding subsections. The remaining events are primarily core-related and are not significantly affected by the replacement of the steam generators.

6.1.2.1 Excessive Feedwater

An excessive feedwater transient may be caused by a decrease in feedwater temperature or by an increase in feedwater flow. These conditions primarily affect reactor coolant parameters due to the resulting excessive heat removal from the primary system. Since the feedwater system flowrates have not been altered, the replacement of the steam generators has no effect on the results for this transient and the consequences will be no more adverse than those determined in analyses for the FSAR or the power uprating submittal.

6.1.2.2 Excessive Load Increase

The excess load transient may be initiated by an inadvertent opening of the turbine control valves, atmospheric steam dump valves, and/or steam bypass valve. The ensuing transient causes a high power level trip to protect the reactor core. At hot standby conditions, there may be an excessive reduction in the steam generator water inventory. However, the time required to boil the steam generators dry is in excess of that time predicted to empty the steam generators during a loss of feedwater flow transient. Since there are no changes in the valves identified above as the potential initiating mechanisms, and since an excess load transient is much less severe than a steam line break, the consequences of this transient are no more adverse with the replacement steam generators than those results reported for previous analyses. In addition, the results of these analyses are bounded by those of the main steam line break and loss of feedwater flow events.

6.1.2.3 Loss of Load

A loss of load transient leads to a rapid (and large) reduction in the power demand while the reactor is operating at full power. There is a corresponding reduction in the rate of heat removal from the primary coolant system. This leads to elevated pressurizer and steam generator pressures that cause the pressurizer and steam generator safety valves to open to minimize the peak primary and secondary pressures. Additional protection is provided by the high pressurizer pressure trip. As a result of the actuation of the reactor trip and the opening of the valves, the peak primary and secondary system pressures are no more adverse with the replacement steam generators than they were when the analyses were performed using the characteristics of the original steam generators.

6.1.2.4 Loss of Feedwater Flow

A complete loss of feedwater flow may be initiated by a rupture of the feedwater crossover line downstream of the main feedwater pumps or a condensate pump failure, which would cause low suction pressure on both feedwater pumps. The primary consequence of this accident is the reduction in, and eventual loss of, the primary coolant system heat sink. An analysis of this transient assuming the installation of the replacement steam generators shows that the consequences of this accident are no more adverse than those reported for analyses using the original steam generator characteristics. In fact, the increase in secondary water inventory for the replacement steam generators increases the predicted time to empty the steam generators.

6.1.2.5 Steam Line Break

A rupture in a main steam line would increase the rate of heat extractions of the steam generators and cause a rapid temperature reduction in the primary coolant. The fastest blowdown and the most rapid reactivity addition are associated with a break located at a steam generator nozzle. The replacement steam generators were analyzed for the full power Main Steam Line Break (MSLB) event using the NSSS simulation code version which is consistent with the analytical methods use for the corresponding FSAR analysis. Although the replacement steam generators have the increased water inventory (see Tables 2.1-1 and 2.1-2), the effect of including the steam nozzle flow restrictors is to decrease maximum power during the full power MSLB for the replacement steam generators. Therefore, the results of the full power MSLB event with the replacement steam generators are no worse than the results reported in the FSAR except for a slight increase in the total blowdown flow. The effect of this slight increase of blowdown flow is discussed in Section 6.1.4.

6.1.2.6 Steam Generator Tube Rupture

A steam generator tube rupture is a penetration of the barrier between the primary coolant system and the main steam system. A double-ended (guillotine) rupture of a steam generator U-tube at the tubesheet is postulated. Integrity of the barrier between the primary coolant system and main steam system is radiologically significant, since a leaking steam generator tube allows transport of primary coolant into the main steam system. Radioactivity contained in the reactor coolant can then mix with shell-side water in the affected steam generator and be expelled to the atmosphere. During normal plant operations, some of this radioactivity is transported through the turbine to the condenser, where the noncondensable radioactive materials are released via the condenser air ejectors.

The steam generator tube rupture event has been analyzed because the tube inside diameter is larger for the replacement steam generators than for the original steam generators. For a double-ended tube rupture within the replacement steam generators, the larger break area could result in a higher primary-to-secondary leak rate. The re-analysis confirms that the fluid leak rate with the replacement steam generators is higher, during the initial stages of this transient. However, the escalated decrease in the primary coolant inventory leads to an earlier reactor trip on low-pressurizer pressure and an earlier emptying of the pressurizer for this transient. Because during this transient, the reactor remains at power for a considerable period of time, a noticeable reduction in the time to trip the reactor causes a reduction in the total primary coolant activity transferred to the secondary side of the steam generators. Thus, a reduction in time to trip the reactor also reduces the total curie content transferred from the secondary side of the steam generators to atmosphere via the atmospheric dump valves or steam generator safety valves. The overall impact is that the radiological releases from the steam generator tube rupture event are no worse than the values reported in the corresponding FSAR analysis.

6.1.3 LOSS-OF-COOLANT ACCIDENT EVALUATION

A major primary coolant system pipe break would result in a rapid depressurization of the primary coolant system and subsequently in reactor trip and safety injection system actuation on either low pressurizer pressure or high containment pressure. The reactor trip and safety injection systems serve to mitigate the consequences of the event in the following ways:

a. Reactor trip and borated water injection, in addition to void formation as a result of the depressurization, cause a rapid reduction in core power to the fission product decay heat level.

b. Water injected by the safety injection system provides for core cooling and prevents excessive fuel and clad temperatures.

Safety injection system water is supplemented by the injection of borated water from the safety injection bottles. The safety injection tanks passively actuate when the primary coolant system pressure drops below 200 psia (plus the elevation head in the injection lines and bottles). The safety injection tanks affect a rapid refilling of the reactor vessel due to large capacity and, hence, strictly limit the period of time during which the reactor core remains uncovered.

The emergency core cooling system (i.e., the safety injection system in combination with the safety injection tanks) is designed so that the reactor can be safely shut down and the essential heat transfer geometry of the core preserved following the LOCA. More specifically, when the emergency core cooling system (ECCS) is degraded by the most severe active single failure, it is designed to meet the ECCS acceptance criteria as stated in Reference 4.

An evaluation was performed to determine the effects of the replacement steam generators on ECCS performance and is summarized below. The most recent loss-of-coolant accident analysis submitted for the Palisades Plant was used as the reference analysis in evaluating these effects. The most recent analysis, as documented in Reference 5, used the currently approved Exxon Nuclear Company WREM-II PWR Evaluation Model.

For this evaluation, sensitivity studies were conducted to determine the effect of changes in significant primary system operating parameters and steam generator characteristics on peak cladding temperature (PCT) for the most limiting large break LOCA as determined by the reference analysis. The methods used for these studies are identical to those used in the reference analysis. The sensitivity of PCT to steam generator tube plugging, primary system pressure, and core inlet temperature were evaluated. The results of this study are summarized below.

Parameter	Change in Parameter	Change in PCT
Plugged tubes	+ 850 tubes	+25F
Core inlet temperature	+ 8.5F	-18F
Primary system pressure	+ 90 psi	+54F

Based on these sensitivity studies and noting that the reference analysis assumed a total of 4,175 plugged tubes, it can be deduced that replacement of the steam generators without changing plant operating conditions should result in a net reduction in PCT with respect to the reference case of about 125F. This improvement is primarily due to reduced steam generator flow resistance during the reflood phase of the transient and, hence, higher core reflooding rates. However, because of higher expected primary system flowrates and higher expected secondary steam pressures with the new steam generators as compared to the referenced analysis, it is expected that the Palisades Plant will be able to operate at a slightly higher core inlet temperature (+5F) and a slightly higher primary system pressure (+40 psi) as compared to the core inlet temperature and pressure at which the reference analysis was performed. The increase in core inlet temperature should result in a slight reduction in PCT (-10F), whereas the increase in pressure should result in a slight increase in PCT (+25F). Taking all of these changes into account, the replacement steam generators should result in a net improvement in PCT for the limiting large break LOCA of about 110F.

Parameter	Change in Parameter	Change in PCT
Plugged tubes	-4175 tubes	-125F
Core inlet temperature	+5F	-10F
Primary system	+40 psi	+25F
	Net	-110F

Hence, it can be concluded that the replacement steam generators will have a beneficial effect on ECCS performance and that the ECCS acceptance criteria (1) will be met with the new steam generators installed in the Palisades Plant.

6.1.4 CONTAINMENT PRESSURE ANALYSIS

The effects of the replacement steam generators upon the containment pressure response analysis have been evaluated by assessment of the mass/energy releases to containment during the main steam line break (MSLB) and the loss-of-coolant accident (LOCA). FSAR Section 14.18 and Answer 14.11 of Amendment 14 (FSAR) indicated that for the original steam generators the MSLB at full load would be more severe from a containment pressure point of view than either the LOCA or the MSLB at no load.

The LOCA mass/energy release for the replacement steam generators at full power (2530 Mwt) was compared to the LOCA mass/energy for the original steam generators. It was concluded based on this comparative evaluation that the peak containment pressure following a LOCA would be slightly less than that predicted in the FSAR (51.0 psig).

The full power MSLB mass/energy release for the replacement steam generators was analyzed using analytical methods comparable to those used in the preparation of the FSAR, except that credit was taken for the steam nozzle flow restrictors in the replacement steam generators. To obtain full benefit of the flow restrictors, the analysis included a Main Steam Isolation System (MSIS) actuation on high containment pressure (5.75 psig) as described in Section 3.6. The containment response to a full power MSLB was analyzed using the version of the COPATTA computer program which is described in FSAR Section 14.18.1. Containment initial conditions, engineered safeguard equipment actuation times and containment heat sink data used for this analysis were identical to those presented in FSAR Section 14.18.1. Although the mass/energy data were developed by conservatively assuming the availability of off-site power, the containment response analysis conservatively assumed the loss of off-site power. Consequently, the single active failure assumed for the containment response analysis was a diesel-generator failure. This postulated active failure minimizes the engineered safeguard equipment available during the accident and maximizes containment pressure.

A peak containment building pressure of 47.6 psig was calculated for the full load MSLB with the replacement steam generators; this value is less than that predicted in the FSAR (51.8 psig). Since the zero power inventory for the replacement steam generators is slightly less than that in the original steam generators, the peak containment pressure for the no-load MSLB would be less for the replacement steam generators than that predicted in the FSAR.

Based on the foregoing, it is concluded that the peak containment pressure following either a MSLB or a LOCA would be no more severe for the replacement steam generators than for the original steam generators.

6.1.5 FSAR EVALUATION CONCLUSIONS

The conclusions based on the safety evaluation of the design basis events for the replacement steam generators are as follows:

a. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased.

b. The possibility for an accident or malfunction of a different type than any of those evaluated previously in the safety analysis report is not created.

c. The margin of safety as defined in the basis for any present technical specification is not reduced.

d. The Palisades Plant, equipped with the replacement steam generators, may be safely operated without presenting any undue hazard to the health and safety of the public.

6.2 CONSTRUCTION-RELATED EVALUATIONS

6.2.1 HANDLING OF HEAVY OBJECTS

Most of the equipment and construction-related tools required for the steam generator repair program will be brought into the containment via the construction and supplemental access routes as shown in Figures 4.3-1 and 4.3-2. Some equipment or tools which cannot be practically or conveniently transported through the access route described above will be transported through the fuel building floor hatch at el 649'-0". This equipment will be transported over the tilt pit areas away from the spent fuel pool. Movement of equipment or tools in the spent fuel area will be governed by the Palisades Plant Technical Specifications in effect at the time the repair is performed.

Construction-related incidents concerned with the handling of the steam generators are discussed in Section 4.1.4.

6.2.2 OFFSITE RADIOACTIVE RELEASES AND DOSE ASSESSMENT

Radioactive airborne and liquid offsite releases have been evaluated for the repair effort using conservative bounding parameters and assumptions.

6.2.2.1 Airborne Releases

Radioactive airborne effluent releases to the environment resulting from the repair effort have been estimated using the following assumptions and parameters:

a. Airborne releases are assumed to occur during the cutting operations.

b. The reactor coolant pipes and the steam generators are expected to be contaminated primarily by deposited corrosion products. Typical corrosion product activities expected on the primary side surfaces of the steam generators are given in Table 6.2-1. These activities have been increased by a factor of 3 for approximately 5 effective full power years (Reference 3) of additional reactor operation.

c. It is conservatively assumed that all the activity present in the vicinity of each cut will become airborne and be available for release to the environment.

d. One hundred forty days of radioisotope decay were assumed before cutting operations, on the basis of the earliest reasonable time as dictated by the repair effort. No credit was taken for radioisotope decay during cutting operations.

e. All primary coolant piping cuts are assumed to be made in specially-designed contamination control enclosures which will provide high efficiency filtration. The enclosures are assumed to be 90% efficient for capturing particulates. An additional 99% efficiency is assumed for the stack filter (HEPA) through which all plant ventilation flows. Further reductions in airborne radioactivity will occur through use of the two internal recirculation filters which are discussed in Section 4.3.3; however, no additional environmental release credit is assumed for these filters since their primary purpose is to reduce occupational doses and minimize personal respiratory protection devices. Radioactive airborne effluent releases to the environment based on the above assumptions are approximately 5.9×10^{-5} Ci. Details of the airborne effluent release by isotopes are given in Table 6.2-1.

6.2.2.2 Comparison with Observed Gaseous Releases and Estimated Doses During Normal Operation

The estimated releases of radioactive airborne effluents during the repair effort are found to be much smaller than the observed effluent releases at the Palisades Plant during 1977. Observed airborne effluent releases during 1977 are compared with estimated releases during the repair effort in Table 6.2-2. The estimated critical organ dose for the repair program was found to be less than 1.0% of the calculated critical organ dose for 1977.

6.2.2.3 Liquid Effluent Releases

Liquid effluent releases resulting from the repair effort were estimated using analogous data from previous refueling and steam generator inspection outages. The total radioactive effluent release estimated for the repair activity is shown in Table 6.2-4. The total includes laundry waste effluents expected during repair activities and the small amount of liquid waste generated as a result of local decontamination. The estimated specific activities of laundry wastewater are shown in Table 6.2-3. A description of the laundry waste treatment system is included in Section 4.3.6.3. A comparison of the average release for the repair effort is shown on Table 6.2-5.

6.2.2.4 Comparison with Observed Liquid Releases during Normal Operation

Observed liquid effluent releases during 1977 are compared with the estimated releases for the repair effort in Table 6.2-5. The total body and significant organ doses for the repair effort were roughly equivalent to the dose from liquid effluents during the year 1977.

6.2.2.5 Conclusion

The combined effect to the offsite dose from gaseous and liquid releases is less than that expected for a year of normal operation. The estimated dose to an individual in an unrestricted area from all pathways of exposure is much less than the limits specified in 10 CFR Part 50 (Appendix I).

6.3 FIRE PROTECTION

A fire prevention and protection program is currently in effect at the Palisades Plant. A complete fire analysis has been conducted for the plant and a report has been submitted to the Nuclear Regulatory Commission.

Work associated with the steam generator repair activities will be performed in containment and the yard area east and southeast of containment.

6.3.1 EXISTING FIRE PROTECTION

All permanent fire protection systems and equipment that can be maintained in its present mode of operation and position without being disrupted, removed, or relocated by the construction will be maintained as is throughout the construction period.

The present fire emergency operations plan will continue to be in effect during construction with inclusions pertaining to fire protection around the actual construction site.

The Consumers' Property Protection Department will be notified before altering the fire protection system, equipment, or plan.

During the construction progress, inspections will be made by members of the Consumers' Property Protection Department and N.M.L. fire protection engineers used by Consumers to check on compliance with established permanent and temporary fire protection systems, equipment, and policies.

6.3.2 FIRE PROTECTION DURING THE REPAIR PROGRAM

This section contains policies and procedures pertaining to fire protection that will be in operation and enforced during the repair program.

6.3.2.1 Portable Fire Extinguishers

Portable fire extinguishers will be located around the work areas to support construction activities. The types of fire extinguishers to be used are pressurized water, carbon dioxide, and dry chemical.

Each fire extinguisher location would be determined before the construction process so that each fire extinguisher would be in the most accessible location and conspicuous to view for locating by workers.

The type and size of fire extinguisher selected for each location will be dependent on the class (or classes) of fire that could be expected to occur at that location or in that area.

Fire extinguishers will be inspected and maintained according to N.F.P.A. Pamphlet 10. Any fire extinguisher discharged or damaged will immediately be removed from service to be recharged or repaired. During its absence, a fully charged unit of identical size and type will be placed in the location.

6.3.2.2 Fire Hose

A 2-1/2-inch gate valve will be attached to the fire hydrant at Hose House 5. During construction, 100 feet of 2-1/2-inch hose will be kept ready for use at the hydrant at the area of construction near containment. Two 100-foot lengths of 1-1/2-inch hose will be available for attaching to a gated wye (two 1-1/2-inch male x one 2-1/2-inch female) attached to the 2-1/2-inch hose at the construction site. Each length of 100-foot 1-1/2-inch hose will have attached a 1-1/2-inch water fog nozzle. These 1-1/2-inch hoses will be available to be used on either side of the construction site. Fire hose is also located at Hose Houses 4, 6, and 7 near the area of construction.

Two internal 1-1/2-inch fire-fighting standpipe hoses are located near the containment lock if a hose is needed inside of containment. New hoses for fire-fighting and fire protection inside of containment on the existing house service water system will be installed in the future.

6.3.2.3 Fire Emergency Reporting

A fire emergency reporting phone will be located in the immediate vicinity of the working area on the construction site. This communications system will be tied into the plant system for immediate notification of the plant fire brigade. All construction workers on site will be instructed in fire response/reporting procedures.

Portable bullhorns will be available at or near the work site.

6.3.2.4 Combustible Materials

A minimum amount of combustible materials will be used at the work area. Combustible materials not in use will be stored away from the work areas. This separation will provide a natural fire break.

All construction areas will be kept as free and clear of rubbish and combustible materials as possible. Metal containers will be strategically located around the construction areas for disposal of materials.

6.3.2.5 Welding and Cutting

All combustible materials to the extent practical at welding and cutting locations will be relocated before welding and cutting starts. Combustible materials that cannot be moved will be covered with noncombustible tarps, if possible, before welding and cutting and will remain covered until all welding and cutting operations are completed. Frequent tours by supervisors to inspect areas where welding and cutting is being done will help minimize the number of welding and cutting fires that could occur.

6.3.2.6 Flammable and Combustible Liquids

All flammable and combustible liquids used on the construction site will be handled, transferred, and stored in accordance with established Consumers' fire protection practices.

Transfer and storage of flammable and combustible liquids will be in areas away from ignition sources.

6.3.2.7 Smoking

Smoking will only be permitted in designated areas.

6.3.2.8 Electrical Wiring

All wiring used to provide electricity to construction equipment or regular lighting at the construction site will be of UL approved 3-wire type designed to be used in outdoor locations and able to handle the electrical load placed on it. The actual wire size and type will be selected by the contractor doing the work. Wiring will be installed in a professional manner and located to prevent possible damage.

All wiring will be properly fused.

6.3.2.9 Work Site Enclosure

Any work site enclosure erected to provide a work area protected from the elements will be constructed of fire-retardant material whenever possible. Exit passages and/or doors will be provided for easy exit in case of emergency.

6.3.2.10 Emergency Lighting

Emergency lighting will be provided at the work site for use in case of emergency or accident.

6.3.2.11 Access Into All Work Site Areas

An access road will be maintained into and through the construction area up to containment for access by vehicles (fire apparatus, ambulance, etc) and personnel.

6.3.2.12 Fire Brigade

The plant fire brigade will be available at all times in case of emergency. Construction workers will also be available for fire brigade operations, if needed.

6.3.3 CONCLUSION

The fire protection measures presently in effect at the Palisades Plant, augmented by the special temporary measures discussed above, provide reasonable assurance that potential fires can be readily detected and extinguished if they occur without causing significant damage to the facility.

PALISADES PLANT SGRR

TABLE 6.2-1

ESTIMATES OF AIRBORNE RELEASES TO ENVIRONMENT DURING STEAM GENERATOR REPAIR EFFORT

Isotope	Activity of Corrosion Products[1] at 140 days ($\mu Ci/in^2$)	Total Release[2] (μCi)
Cr-51	0.589	0.378
Mn-54	0.730	0.468
Co-57	0.312	0.200
Co-58	85.5	54.8
Fe-59	0.535	0.343
Co-60	4.27	2.74
Nb-95	0.040	0.026
Zr-95	0.085	0.054
TOTAL	92.2	59.1

NOTES:

(1) These are the activities presented in Table 4.4-1 converted to $\mu Ci/in^2$. The activities were increased by a factor of 3 to account for the expected activity build-up.

(2) The following technique was used to estimate the activity from each isotope released during cutting operations:

Airborne Activity Near cut (μCi)	=	Area of Cut ($.5\ in^2$)	x	Activity of Corrosion ($\mu Ci/in^2$)	x	Number of Cuts (No.)	x	Enclosure and Stack Filter Penetration (.1) (.01)

The total number of cuts on primary coolant piping which were assumed to total 12 where:

4 cuts (42-inch ID pipe)
8 cuts (30-inch ID pipe)

PALISADES PLANT SGRR

TABLE 6.2-2

COMPARISON OF GASEOUS EFFLUENT RELEASES

Isotope	Average 1977 Release (Ci)	Estimated Release During SG Repair Effort (Ci)
Noble Gases	59.89	Negligible
Iodines	1.51×10^{-2}	Negligible
Particulates[1]	1.1×10^{3}	$.059 \times 10^{3}$
Tritium	2.21	Negligible

NOTES:

[1] Approximately 29.0 and 26% of the total particulate release during the year 1977 are Co-58 and Co-60, respectively.

TABLE 6.2-3

ESTIMATED SPECIFIC ACTIVITY OF LAUNDRY WASTEWATER

ISOTOPE	SPECIFIC ACTIVITY Ci/cc(1)
Co-57	8.96×10^{-7}
Cs-134	4.64×10^{-5}
Cs-137	1.03×10^{-4}
Co-58	3.85×10^{-4}
Mn-54	2.63×10^{-5}
Co-60	7.22×10^{-5}
Fe-59	3.27×10^{-6}
Zn-65	4.09×10^{-7}
Zr-95	4.15×10^{-6}
Nb-95	7.2×10^{-6}
Sr-90	3.66×10^{-5}
Ni-63	2.65×10^{-5}

NOTE:

(1) Time averaged specific activity during a period of 365 days.

PALISADES PLANT SGRR

TABLE 6.2-4

ESTIMATED RADIOACTIVE LIQUID EFFLUENT RELEASED DURING THE STEAM GENERATOR REPAIR

ISOTOPE	RELEASE Ci
Co-57	6.85×10^{-4}
Cs-134	3.54×10^{-2}
Cs-137	7.85×10^{-2}
Co-58	2.95×10^{-1}
Mn-54	2.01×10^{-2}
Co-60	5.5×10^{-2}
Fe-59	2.5×10^{-3}
Zn-65	3.12×10^{-4}
Zr-95	3.17×10^{-3}
Nb-95	5.5×10^{-3}
Sr-90	2.79×10^{-2}
Ni-63	2.02×10^{-2}
TOTAL	.544
H-3	1.91

PALISADES PLANT SGRR

TABLE 6.2-5

COMPARISON OF RADIOACTIVE LIQUID EFFLUENT RELEASES

ISOTOPE	AVERAGE 1977 RELEASE (Ci)	ESTIMATED RELEASE DURING SG REPAIR EFFORT (Ci)
Fission and activation products	.093	.544
Tritium	55.8	1.91
Total	55.9	2.45

7.0 ENVIRONMENTAL ASPECTS OF THE REPAIR PROGRAM

7.1 GENERAL

The following sections present information and the assessment of environmental impact of the proposed steam generator repair program. The estimated environmental impact of the repair activity and disposal of the removed steam generators is expected to be negligible and temporary. The proposed activity will cause little additional environmental impact over that of normal plant operation.

Construction activities, particularly the barge slip preparation, will be carried out in conformance with local, state, and federal regulations. When the facility is returned to service after the repair, water use, occupational exposures, and radiological releases are expected to be less than those associated with current facility operation.

It may be necessary to store the old steam generators onsite in an engineered storage facility that will provide shielding for the direct radiation. The steam generators will be sealed to contain any airborne radionuclides. The steam generators may be shipped offsite to a federally licensed storage facility, if available. If shipped offsite, all local, state, and federal regulations pertaining to the shipment of radiological materials will be followed.

7.2 RESOURCES COMMITTED

7.2.1 NONRECYCLABLE BUILDING MATERIALS

The steam generator repair program at the Palisades Plant will require the commitment of various irretrievable building materials. The quantitative estimates for the nonrecyclable building materials are as follows:

Concrete	1,000,000 pounds
Structural steel	50,000 pounds
Alloy or stainless steel	1,594,000 pounds
Tendons (53)	150,000 pounds
Cable (copper)	400 pounds
Inconel	280,000 pounds
Pipe	5,000 pounds
Wood	15,000 board feet

7.2.2 LAND RESOURCES

The steam generator repair program will have minimal impact on the existing site in terms of land use. The construction of facilities (see Section 4.1.1.2) to store the steam generators will require some excavation, leveling, and foundation work. If necessary, a building will be located near the temporary barge slip (see Figure 4.1-1). The extent of the disturbance will be temporary, negligible, and of minor impact. This area had been previously excavated during plant construction.

7.2.3 WATER RESOURCES

During the repair effort, construction water will be supplied from existing Palisades Plant water sources. No requirements for commitments of new water sources have been identified for the repair effort. Since water consumption during the extended shutdown is expected to be less than during plant operation, water consumption during the repair effort will result in a reduction in plant water usage.

A temporary barge slip will be constructed just north of the existing Palisades Plant (see Figure 4.1-2) to receive and offload two replacement steam generators. The temporary barge slip will be approximately 110 feet long, 50 feet wide, and 18 feet deep. The total dredged quantities are anticipated to be approximately 12,000 cubic yards based upon a lake elevation of 579'-0" (USGS). The dredged material will be disposed of according to Guidelines for the Pollution Classification of Great Lakes Harbor Sediments, U.S. EPA Region V. The dredging will be conducted according to specifications of the U.S. Corps of Engineers permit. Steel sheet piles or sunken ship hulls may be used to provide wave and scour protection for the barge. Depending on lake conditions, the barge slip may have to be periodically redredged to specifications if suspended materials (sand) are deposited there before delivery of the generators.

The dredging activity will temporarily increase turbidity in the site vicinity and remove a small number of benthic organisms with the spoils. The piling or sunken hulls, if used, will provide substrate suitable for attachment of periphyton and filamentous algae, but large growths are not expected. The impacts of the dredging will only have a temporary impact upon the aquatic biota in the immediate plant vicinity. Removal of the temporary barge slip facility will result only in some temporary increase in turbidity along the shore line.

7.3 WASTEWATER

7.3.1 SANITARY FACILITIES

Since the repair activities will take place in locations near which permanent sanitary facilities are not readily accessible (e.g., the containment and laydown area), portable units will be used. There will be no modification to existing sanitary facilities as a result of the repair activity.

7.3.2 LAUNDERING OPERATIONS

Laundry wastewater generated during the repair activities will be produced in the existing facility. A description of the laundry waste processing scheme is included in Section 4.3.6.3.

7.4 CONSTRUCTION

Construction activities at the time of the repair effort will satisfy applicable laws that are in force at that time. These activities will have a negligible effect on noise levels, dust, or smoke.

7.4.1 NOISE

Values and calculation methods described in Reference 6 have been used to examine the maximum sound pressure level (SPL) expected at the site boundary (0.4 miles) because of construction noise. The maximum source SPL is expected to be less than 94 dBA at 50 feet, which will attenuate to less than 62 dBA at the site boundary. The site boundary maximum expected SPL is within the acceptable limits for permissible outdoor noise levels for sleeping with open windows. Moreover, the site is located in a low population area. On the basis of these facts, it is concluded that the additional noise resulting from the repair program for the steam generators is expected to have negligible impact on the local area.

To protect personnel located on the site, Occupational Safety and Health Administration standards (state and federal) will be followed.

7.4.2 DUST

Dust, if any, will be abated by periodically spraying with water or other dust control measures. The frequency of spraying and the quantity of water sprayed will be determined by visual inspection of the areas and will vary with the weather conditions.

7.4.3 OPEN BURNING

Open burning is not anticipated during the steam generator repair effort. However, should the necessity arise, applicable county and state regulations for open burning will be followed.

7.5 RADIOLOGICAL MONITORING

Radioactive effluent release points during steam generator repair activities will be the same as during normal plant operations; therefore, the plant radioactive process monitors will not be affected. Since releases of radioactive effluents during the repair program will be similar to those from the operating plant, and their potential exposure pathway will be the same as for existing plant operations, these effluents will be monitored in accordance with the existing environmental monitoring program at Palisades Plant.

7.6 RETURN TO OPERATION

7.6.1 WATER USE

Water consumption during post repair plant outages is expected to be appreciably less than is currently required as a result of repairs to degraded tubing.

Periodic plant shutdown for steam generator inspection consumes large quantities of pure water. During shutdown, the steam generator water level is controlled on the low side (between 20 to 40% of operating band) to aid in chemical layup. Approximately 10,000 gallons of water are required to place the two steam generators into wet chemical layup.

If a steam generator requires draining (for tube plugging, tube sleeving, or tube removal) during the inspection, it would require an additional 24,000 gallons to refill. Other plant requirements for pure water include 70,000 to 85,000 gallons for heater train, condensate polishers, and hotwell. This is the amount necessary to refill the systems if maintenance had been performed that required draining the systems. Approximately 75,000 gallons would be required for primary system dilutions to return to power. Depending on various chemical parameters, as much as 50% of this water could be recovered through the plant recovery systems, such as clean radwaste system, boric acid recycle system, and steam generator blowdown recovery system.

Following replacement of the steam generators, it is expected that forced outages associated with steam generator tube plugging and/or tube sleeving will be essentially eliminated; however, it is not anticipated that the water consumption associated with the current inspection program will be significantly reduced because of the continuing requirement to inspect (eddy current test) the steam generator tubing at regular intervals.

7.6.2 OPERATIONAL EXPOSURES

Section 4.3.7 discusses the future reduction in man-rem exposure as a consequence of the repair program. A potential savings of 250 man-rem/yr may be realized because of the expected elimination of the necessity to plug tubes in the repaired steam generators and the decrease in the number of inspections required (Regulatory Guide 1.83).

7.6.3 RADIOLOGICAL RELEASES

Although the Palisades Plant has experienced only one primary to secondary leak from tube failure (1974), the repair of the steam generators should reduce the probability of future secondary releases as a consequence of the same tube failure mechanism.

8.0 EVALUATION OF ALTERNATIVES

8.1 INTRODUCTION

In view of the uncertainty which existed with regard to the continued full-power operation of the Palisades Plant as a consequence of corrosion-related steam generator tube degradation, Consumers has investigated a number of potential repair concepts. Those alternatives that have been found feasible, as well as practical, have been classified as "minor" or "major." Minor repair alternatives allow retention of the existing generators, whereas major repair alternatives would involve the replacement of substantial portions of the original units.

In the former category, sleeving appears to offer the most promise, but the "denting" phenomenon has the potential for severely limiting its application. Chemical cleaning has also been extensively investigated as a method to restore the existing units, but its effectiveness and the absence of adverse side effects has not been satisfactorily demonstrated. Continued tube plugging is obviously the least desirable alternative because of the eventual impact the plugging will have on plant output.

In the latter category, the field of practical repair alternatives available to restore the original steam generating capability of the Palisades Plant appears limited to the following:

a. Retubing the original units within the containment

b. Repair, limited to replacement of the evaporator sections only

c. Repair utilizing complete replacement units

The following discussions of these alternatives are based on feasibility studies which have been performed for Consumers during the period 1974 through 1977. The repair options have been evaluated and compared on the basis of the following factors:

a. Plant outage requirements

b. Direct cost of repair

c. Radiological aspects

d. Equipment procurement lead times

e. Special tooling development requirements

f. Plant structural consideration

g. Equipment design improvement possibilities

h. Site preparation requirements

i. Radwaste disposal considerations

On the basis of comparisons of the three alternatives, complete replacement of the existing steam generators with compatible spares appears to offer the optimum repair solution if a program of major repair becomes necessary at the Palisades Plant.

8.2 CONTINUED TUBE PLUGGING AND PLANT DERATE

The effect of future tube plugging on plant power output with the current steam generators has been evaluated. No derating is projected before 500 additional tubes (a total of 4,175) are plugged. In the absence of mitigating measures, it is estimated that the plant would be derated from its currently rated level of 2530 MWt by approximately 14% if a total of 7,175 tubes were plugged, and by 30% if a total of 10,000 tubes were plugged. These estimates are based upon the most recent safety analyses for 2530 MWt generation and on an empirical relationship between core flow and number of tubes plugged, which was derived on the basis of actual plant measurements.

8.3 IN-PLACE TUBE SLEEVING

Sleeving is the insertion of a thin-walled tube insert that is positioned in the vertical section of the tube, spanning a degraded area, that is hydraulically expanded in place. The feasibility of local repair, via sleeving, of steam generator tubes that have suffered external wall thinning has been previously demonstrated at the Palisades Plant. Such technique will restore the structural integrity of the tube and yet avoid reducing heat transfer surface. The use of sleeving techniques for complete repair of the Palisades Plant steam generators is not, however, considered feasible at the present time. Sleeve insertion and inspection is a time-consuming process requiring extensive baseline and inservice inspections. Defects at certain positions cannot be repaired by sleeving because of the inability to install sleeves at those locations (bends, etc). The sleeving process, as presently applied, could not be used to repair areas which have experienced significant denting. Although those conditions do not currently exist in the Palisades Plant steam generators, and their occurrence in the future is not expected, these restrictions create a degree of uncertainty with respect to future application of sleeving.

8.4 IN-PLACE TUBE REPLACEMENT

Because of the limited size of the Palisades containment building equipment hatch, retubing is the only major option that would not entail the construction of a temporary opening in the containment building in order to accomplish a major repair. The hatch diameter is 12 feet, whereas the steam generator, even at its narrowest, is more than 14 feet in width.

The general concept for in-place retubing would be to remove the secondary head, abrasively cut and remove the old tubes and structural material, remove the tube stubs from the tubesheet, install new tubes and support structures, replace steam drying and separating equipment, and replace the secondary head. All equipment and material could be transferred to and from the containment through the existing equipment hatch. A new tube bundle shroud and new eggcrate supports would be shop-fabricated in sections and aligned and assembled within the containment before installation in the steam generator shells.

The conceptual schedule for the retubing alternative is estimated at approximately 3 years, which does not allow for any decontamination beyond local measures. Activity levels within the containment and the primary coolant system will dictate the extent to which decontamination is required. Consumers has explored aspects of decontaminating the primary system, and the results of those studies suggest that full decontamination could add approximately 1 year to the retubing program, although for purposes of comparison with other alternatives, decontamination has been limited to localized efforts only.

The success of any program of retubing is crucially dependent upon the prior development of special tooling that would be necessary to remove the tube stubs from the tube sheet and prepare the holes for the welding of the new tubes. Despite prior local decontamination and the use of special tooling, it is, nevertheless, expected that an appreciable number of manhours will be expended in the primary head, near the tubesheet, contributing significantly to the man-rem dose estimated for retubing.

Aside from the lengthy outage required for such a program, an additional negative aspect of retubing is the limited ability to modify the existing steam generators in order to improve performance and maintainability.

The estimated direct cost for a retubing program, assuming minimal decontamination is required, is approximately $75 million (see Table 8.4-1).

8.5 REPLACEMENT WITH COMPLETE UNITS

Because of the negative aspects associated with a program of retubing, the feasibilities of various steam generator replacement schemes were also investigated. These studies concluded that complete replacement could be accomplished in approximately two-thirds of the time required for retubing, and that even though a temporary construction opening in the containment was necessary, the removal and replacement of the steam generators could be performed without disturbing any important structural elements within the containment.

Three alternative schemes involving construction openings were evaluated. Two programs were based upon construction openings in the side of the containment but at different locations. The third scheme was based upon an opening in the containment dome, through which the units could be removed and replaced.

The preferred repair scheme, and the one for which NRC approval is being sought, is based upon removal and replacement through a temporary side opening located above the 649-foot level and centered horizontally about the containment radius at approximately 118 degrees (north is 0 degrees). The repair program based on that alternative is discussed at length in Section 4.0. Briefly, it is planned that the new units would be shipped by barge and unloaded at a new temporary barge slip at the plant site. The containment would be partially detensioned and the construction opening cut. The old units would be cut free, rigged vertically from their cavities, lowered to a horizontal position, and removed from the containment. The installation would then follow in reverse order. Decontamination would be limited to local efforts in the area of major pipe cuts, supplemented with the wide use of temporary shielding. The estimated man-rem dose for a repair program based on such a replacement scheme would be in the range of 1,200 to 5,000 man-rems.

The length of the repair outage associated with complete replacement is estimated at approximately 2 years, based upon the currently anticipated scope of work. The related direct cost in 1983 dollars is estimated at approximately $75 million (see Table 8.4-1).

Some important direct benefits of a repair program based upon replacement relate to those steam generator design improvements that could be readily accommodated in new units but which would prove to be extremely difficult to incorporate into the existing generators under a program of retubing. Included in that category are:

a. High capacity blowdown capability

b. Recirculation/chemical cleaning capabilities

c. Visual inspection provisions for critical areas

d. Flow limiting nozzle design

e. Enlarged manways

f. Primary head drains

All of these features have been included in the design of the replacement units for the Palisades Plant and are discussed more fully in Section 2.0.

Since the procurement lead time for replacement steam generators is approximately 4 years, the assessment of the future operating performance of existing units is of crucial importance in the decision between alternatives and the decision as to when to proceed with planning for a repair program based on replacement. A parametric analysis of the various steam generator failure mechanisms was conducted, utilizing stochastic modeling techniques, to forecast future unit performance. As a result of that study, it was concluded that there is a significant probability that the existing steam generators will operate, without derating, until the time that replacement units are available.

8.6 REPLACEMENT OF STEAM GENERATOR EVAPORATOR SECTIONS

The third major repair alternative considered for the Palisades Plant is a variation of the replacement concept discussed previously in Section 8.5. It is different in that the upper shell and head portion of each old unit would be reused. That variation would require a circumferential cut at some appropriate location on the shell in order to separate the evaporator section from the steam drum. The removal and installation procedure for the new tube bundle assemblies would closely follow that for the complete assemblies with the additional requirement of the completion weld on the transition cone.

Other than the reduced scope of rigging related to the lower weight of the evaporator bundle, there is nothing to recommend this option over the complete replacement concept: the temporary containment opening is still required, the barge slip is still necessary, and the reduced cost of the hardware is more than offset by the additional time required to accomplish the completion weld on the cone. Furthermore, because of the interferences associated with the probable location of the completion weld, it appears that it might be necessary to perform all cutting, welding, and stress relieving on the transition cone in a separate facility outside containment. On that basis, an additional outage period of 8 months would be required.

8.7 MAN-REM CONSIDERATIONS

The detailed development of the man-rem dose estimate for steam generator repair utilizing complete units is presented in Section 4.3.7. The results of that assessment would suggest that a dose in the range of 1,200 to 5,000 man-rem could be expected as a consequence of replacing the steam generators. Although a comparable dose assessment of the evaporator replacement option was not performed, the projected dose is expected to be slightly greater because of the extended outage period.

The man-rem dose associated with a retubing program was developed in a manner similar to that for replacement. In the absence of preliminary decontamination, and using worker residence times in contaminated areas comparable to those used in the man-rem assessment for replacement, a dose of 40,250 man-rems was estimated for retubing in-place. The major contributor to that estimate is the dose incurred while performing tube stub removal and welding activities at the tubesheet. Various combinations of decontamination and mechanized welding programs could conceivably reduce that estimate by a significant amount, but considerable development is necessary in either case before they could be considered reliable options.

Since the need for extensive steam generator inspection and tube plugging operations should be eliminated by the repair, yearly exposures presently incurred for these operations should be significantly reduced. It is estimated that at least 250 man-rem would be saved per year, or a total of 7,500 man-rem over the plant's remaining lifetime of approximately 30 years. The savings above the replacement dose of 1,200 to 5,000 man-rem equals 2,500 to 6,300 man-rem.

Based on the biological savings of $1,000/man-rem (10 CFR 50, Appendix I), this results in a savings to society of $2,500,000 to $6,300,000. In-place decontamination before removal could possibly result in lowering the total man-rem incurred, but has been found impractical on a cost-benefit basis (see Section 9.2).

8.8 CONCLUSIONS

As a result of Consumers' investigations and comparisons of alternatives, it is evident, from both an economic and a technical viewpoint, that complete replacement represents the optimum method of major steam generator repair for Palisades Plant if repair becomes necessary.

The key factor in the economic comparison of alternatives is the cost of replacement power associated with the plant outage required to perform the repair. In that category, replacement has the clear advantage. Similarly, from the standpoint of ALARA consideration, replacement is the preferred method among alternatives.

On a technical basis, the installation of replacement steam generators would permit the inclusion of important steam generator design features that, in some instances, would be impossible to incorporate in the existing units in conjunction with a program of retubing. In addition, the success of a retubing program is dependent upon the development of special tooling, while the replacement option is not similarly constrained.

The decision to replace, and the timing of the repair, will be based upon the conditions of the old units with respect to derating, the adequacy of system reserve margins, the future inspection requirements of the degraded generators, and other relevant factors.

Consumers will continue to investigate sleeving and other qualified means of steam generator repair in order to extend the useful lives of the original units for as long as practical.

A tabular comparison of the major repair alternatives is presented in Table 8.8-1.

TABLE 8.4-1

STEAM GENERATOR REPAIR ALTERNATIVE COSTS

(Costs in Millions of Dollars)

Item	Replacement Complete Units	Replacement Evaporator Sections	Retube In-Place
Engineering and material	26.5	21.5	7.7
Construction	9.5	12.0	21.8
Licensing	5.0	5.0	2.5
Disposal	2.0	2.0	0.2
Consumers directs	1.0	1.0	0.8
Escalation	10.5*	12.8*	18.2*
Subtotal - directs	54.5	54.3	51.2
Administrative and general	1.5	1.5	1.0
Insurance and taxes	1.0	1.0	1.0
Allowance for funds used during construction (AFUDC)	11.5**	12.0**	9.7**
Subtotal - overheads	14.0	14.5	11.7
Consumers' contingency and rounding	6.5	11.2	12.1
Total estimated project cost	75.0***	80.0***	75.0***

Notes:

*Includes escalation for material and steam generator costs, as applicable.

**Includes AFUDC for material and steam generator payments, as applicable.

***Total estimated cost is for a commercial operation date of April 1, 1983.

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TABLE 8.8-1

COMPARISON OF MAJOR REPAIR ALTERNATIVES

		Replacement Complete Units	Replacement Evaporator Sections	Retube In-Place
1.	Length of repair outage (months) (Local decontamination only)	24	30	36
2.	Cost of replacement power ($\$x10^6$)	200	250	300
3.	Direct cost of repair ($\$x10^6$) (1983)	75	80	75
4.	Dose to workers (man-rem)	1,200-5,000	1,200-5,000	40,000
5.	Equipment procurement lead times (yr)	4	3	Not critical
6.	Special tooling requirements	None	None	a) Tube stub removal b) Tube sheet weld prep c) Tube/tubesheet weld d) Tube expansion
7.	Plant structural considerations	Temporary containment opening required	Temporary containment opening required	None
8.	Equipment design improvements which can be incorporated	a) SS egg crate tube supports b) Flow distribution baffle c) Increased capacity blowdown d) Inspection ports e) Primary head drains f) Larger primary manways g) Larger secondary manways h) Recirculation/chemical cleaning provisions i) Steam nozzle flow restrictor	a) Same b) Same c) Same d) Same e) Same f) Same	a) Same b) Same
9.	Site preparation requirements	Temporary barge slip	Same	None

9.0 COST BENEFIT ANALYSIS FOR THE DECONTAMINATION, STORAGE, AND DISPOSAL OF THE OLD STEAM GENERATORS CONSIDERING ALARA

9.1 INTRODUCTION

The following are evaluated on a cost-benefit basis:

a. In-place decontamination

b. Steam generator storage and disposal methods

The cost of each method compared with benefits gained in the total man-rem reduction to workers is in accordance with the philosophy of reducing worker dose to levels which are ALARA.

9.2 STEAM GENERATOR IN-PLACE DECONTAMINATION

A study has been performed by United Nuclear Industries (UNI) to evaluate the best method of decontamination of the Palisades steam generators. Its report (Reference 2) indicated that a combination acid/base flush with several rinses will lower fields inside the steam generators at the time of replacement to approximately 260 mrem/hr. Total costs of decontamination and waste disposal, including equipment, chemicals, and processing of solutions, is estimated at approximately $5 million. Decontamination time for one steam generator is optimistically estimated to be 17 weeks, with a total of 30 weeks required for both steam generators.

A man-rem assessment was made for the repair effort considering in-place decontamination of the steam generators. The dose assessment was based on the radiation field information and tasks required for removal presented in Section 4.3.7.2. A dose rate reduction factor of 25 was obtained from analysis of the decontamination study as applied to fields in various areas within 6 feet of the steam generators.

Assuming in-place decontamination before the initiation of the repair effort, it is estimated that the repair effort would require 200-300 man-rem, a savings of approximately 1,000-4,800 man-rem (see Table 4.3-2). The estimate does not consider man-rem incurred during the decontamination effort. Although the exposures from decontamination operations are difficult to quantify, it is known that approximately 60 curies of radioactivity would be removed and require handling. Since a considerable volume of the solutions are not compatible with the installed radwaste processing system, much manual contact would be required for solidification and shipping of this waste. In addition, considerable work is required in high radiation fields to connect piping and modify recirculation paths to avoid material incompatibilities. Thus, the 1,000-4,800 man-rem saving in replacement exposures is negated by exposures associated with decontamination efforts.

Considering that the estimated biological cost of a man-rem is $1,000 (10 CFR 50, Appendix I), the combined considerations of dose reduction, schedule penalty, and large capital cost associated with decontamination do not indicate a benefit for full-scale steam generator decontamination. Therefore, it is concluded that in-place decontamination of the steam generators is not practicable.

9.3 STEAM GENERATOR STORAGE AND DISPOSAL

The old steam generators may be stored onsite, at least temporarily, before eventual disposal. The alternatives associated with the storage and disposal are addressed below.

9.3.1 LONG-TERM STEAM GENERATOR STORAGE ONSITE

As discussed in Section 4.4.6, the steam generators would be sealed before storage to ensure complete encapsulation of residual contamination and placed in a storage facility. The storage facility would provide adequate shielding around the steam generators. Access control and monitoring measures would be implemented during the storage period. At the end of the plant lifetime, disposition of the steam generators will be accomplished in conjunction with plant decommissioning.

9.3.2 IMMEDIATE SHIPMENT BY BARGE

The immediate shipment of the steam generators in one piece by barge as discussed in Section 4.4.3.3 is currently considered to be a viable method of disposal. Immediately upon removal from the containment, the steam generators would be loaded on barges and shipped to a licensed depository.

9.3.3 SHORT-TERM STORAGE ONSITE WHILE UNITS ARE CUT UP FOR SHIPMENT (WITHOUT DECONTAMINATION)

If the steam generators are to be cut up for shipment, additional contamination control measures, as well as shielding, would have to be employed as discussed in Subsection 4.4.3.2. For this option, an enclosure with appropriate controls for airborne and liquid effluents will be required in addition to the requirements set forth in Section 4.4.2. Techniques for cutting and packaging are not well established, making cost and dose calculations uncertain.

9.3.4 CONCLUSIONS

The present-day conceptual cost and man-rem estimate for each method of steam generator storage and disposal are summarized below.

		Approximate Cost	Man-Rem
a.	Immediate shipment by barge	$353,000	1-5
b.	Long-term storage with disposition during decommissioning	$2,560,000	5-10
c.	Cut up and disposal near term with no decontamination	$1,756,000	575-750

It is apparent that immediate shipment by barge would be beneficial from the standpoint of both man-rem exposure and cost.

10.0 REFERENCES

1. BSAP - Bechtel Structural Analysis Program, Bechtel Power Corporation, June 1978

2. United Nuclear Industries, Inc. Study, Decontamination of Palisades Steam Generators, November 1, 1975

3. Electrical Power Research Institute (EPRI 404-2) Primary System Shutdown Radiation Levels at Nuclear Power Generating Station, p.56, December 1975

4. Acceptance Criteria for Emergency Core Cooling Systems for Light Water-Cooled Nuclear Power Reactors, 10 CFR 50.46 and Appendix K of 10 CFR 50, Federal Register, Volume 39, Number 3, January 4, 1974

5. LOCA Analysis for Palisades at 2530 MWt Using the ENC WREM-II PWR ECCS Evaluation Model, Exxon Nuclear Co. XN-NF-77-24, July 1977

6. L.L. Beranek, Noise and Vibration Control, McGraw-Hill Book Company, Chapters 7 and 18, 1971

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A-1(a) Evaluation of Alternatives

Table 8.8-1, Comparison of Major Repair Alternatives, estimates the cost of replacement power during the 24-month period required for replacement of the steam generators at a total cost of $200 million. The report stated that a 4-year procurement lead time is required for delivery of the new steam generators. Give a detailed cost breakdown of the type and cost of replacement power during this 24-month period as well as for the time periods for alternate repairs. The replacement power cost estimates should be in mills/kWh and thousands of dollars per day. Based on required procurement times, the earliest time of replacement would occur in 1983. Include the reference year when replacement power will be required for each type of steam generator repair.

RESPONSE:

The time estimates required to repair the Palisades steam generators by means of replacement with complete units, replacement with evaporator sections, or in-place tube replacement are 24, 30, and 36 months, respectively. The repair outage in each case is assumed to start in 1981 and will continue for the above stated time periods. The average cost of net replacement power required during these outages is estimated to be 27.0 mills/kWh, which is equivalent to $275,000 per day. This replacement power cost is based on a net plant output of 675 MW, a capacity factor of 0.75, and a scheduled outage of 3 months for every 18 months of normal operation. The source of replacement power will be a combination of CPCo system generation and purchased power from other sources, depending on cost, availability, and time of day.

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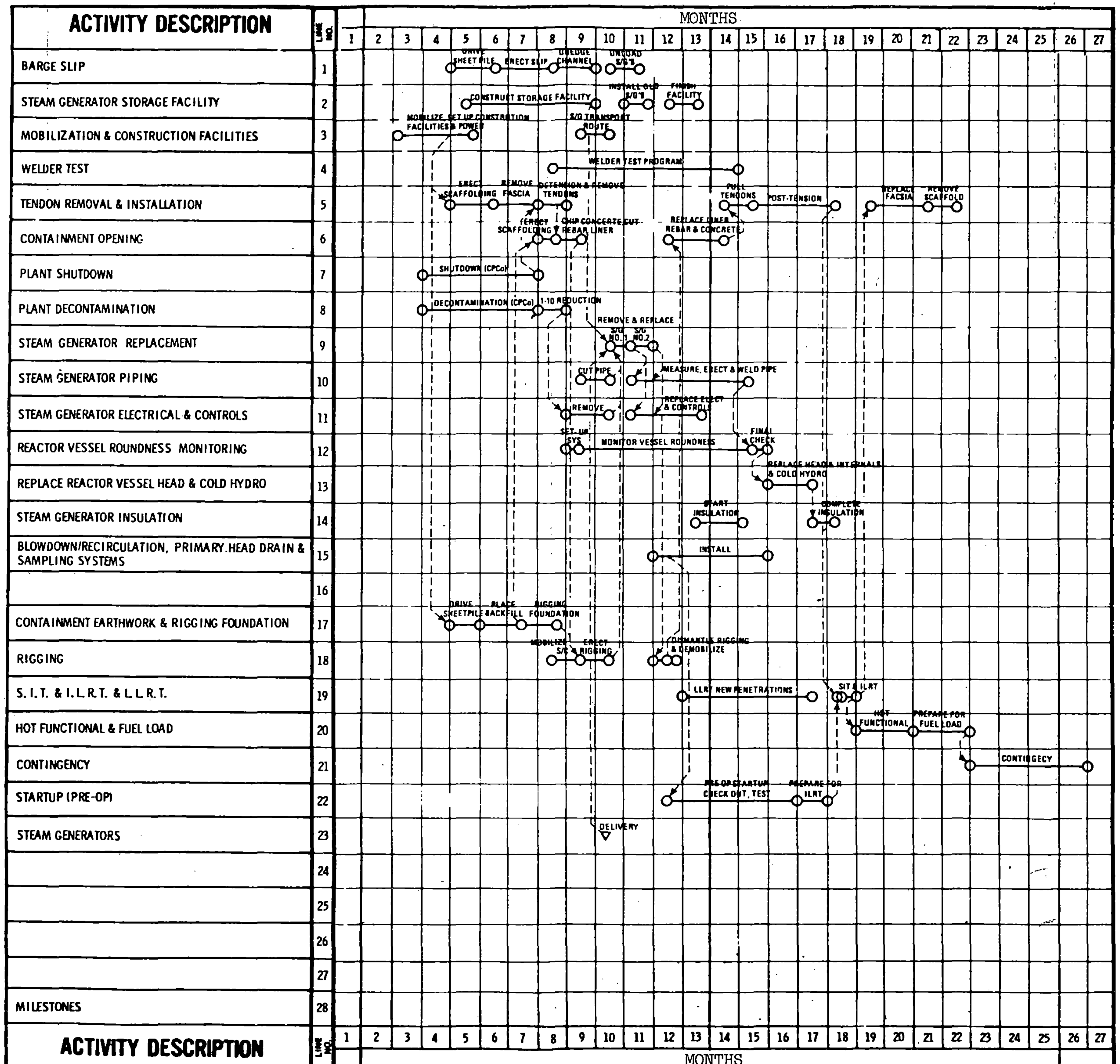
A-1(b) Length of Repair Outages

Sections 8.4, 8.5, and 8.6 give the estimated times of repairs as 36 months for retubing the original units, 24 months for complete replacement of the units, and 30 months for repair of the evaporator sections only. Table 8.4-1, Steam Generator Alternative Costs, gives the total estimated project costs of all three alternatives. Complete replacement, $75.0 million; evaporator repair, $80.0 million; and retubing, $75.0 million. The project costs are fairly close, however, the big difference is the cost of replacement power associated with plant shutdown. Because of the extensive period of plant outage and high cost of replacement power, include a detailed explanation for these required times. Are the 24, 30, and 36-month periods based on single shifts or have double shifts been considered in an attempt to reduce the high cost of replacement power.

RESPONSE:

The attached Figures A-1(b)-1 and A-1(b)-2 provide the bases for the nominal outage durations of 24, 30, and 36 months to repair the Palisades steam generators by means of replacement with complete units, replacement with evaporator sections, or in-place tube replacement. Minor adjustments to these schedules in the areas of post-construction testing or contingency requirements are responsible for any differences which exist between the outage duration shown on these figures and the nominal times used in the report. In each case, the schedule is based on conducting the repair on a 6-day workweek, consisting of two 10-hour shifts per day for critical path activities. As noted in Section 8.6 of the report, the evaporator replacement alternative is a variation of the repair concept based on replacement with complete units, with the additional work of cutting, welding, and stress relieving on the transition cone being responsible for the schedule extension of from 6 to 8 months.



LEGEND:

ACTIVITY O———O

RESTRAINT O- - - - -O

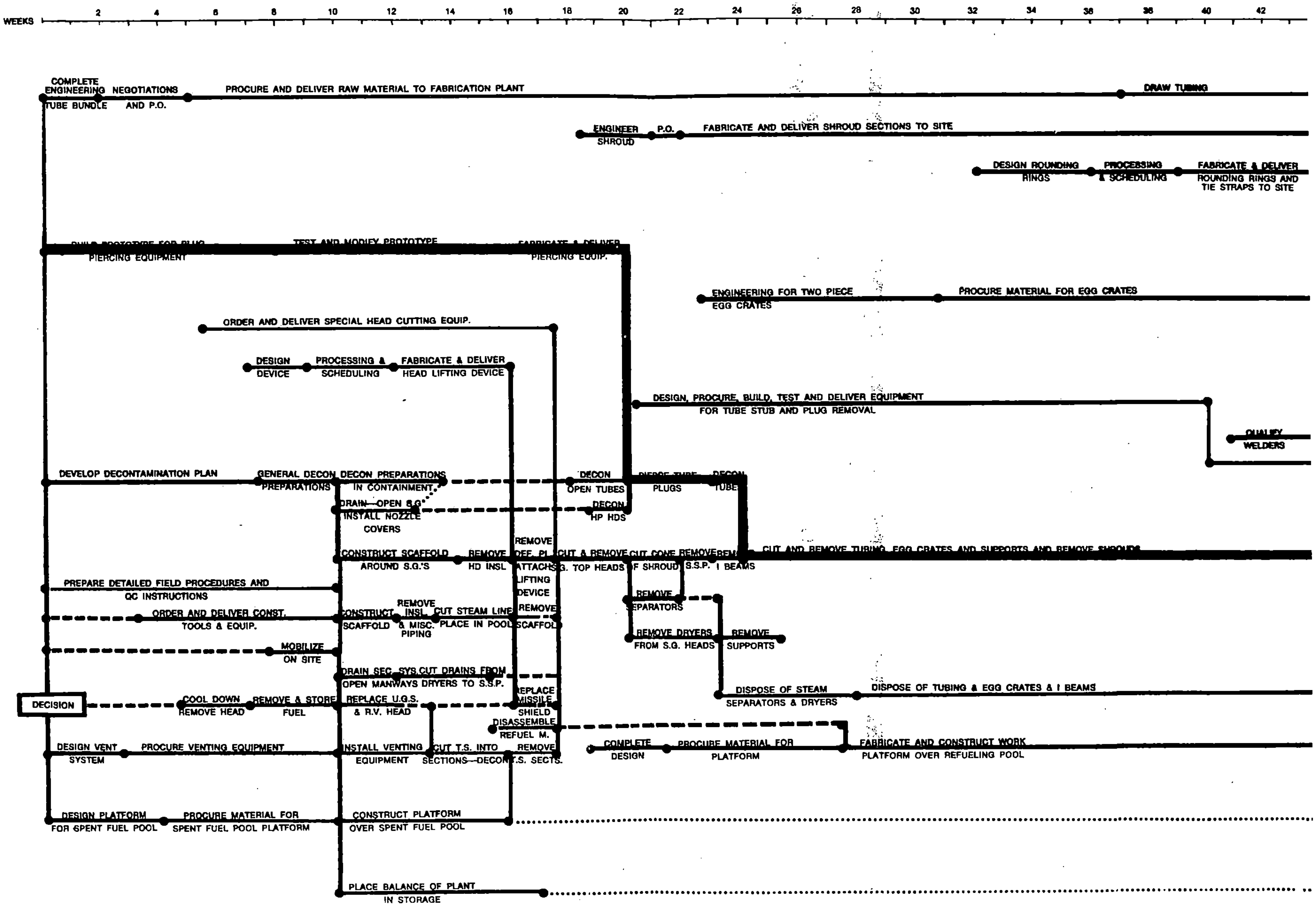
NOTES:

1. THE FOLLOWING DURATIONS ARE BASED ON 2 SHIFTS OF 10 HOURS PER DAY 6 DAYS PER WEEK: LINE 10 MEASURE, ERECT & WELD S/G PIPING: LINE 14 INSULATION: LINE 5 DETENSION & REMOVE TENDONS AND PULL TENDONS: LINE 19 ILRT TEST: LINE 6 CHIP CONCRETE & CUT REBAR & LINER'
2. RIGGING ACTIVITY DURATIONS ARE BASED ON A 60 HR(DAYLIGHT) WORK WEEK
3. THE REMAINING DURATIONS ARE BASED ON A 40 HOUR WORK WEEK

PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

STEAM GENERATOR REPLACEMENT SCHEDULE

Figure A-1(b)-1



WEEKS
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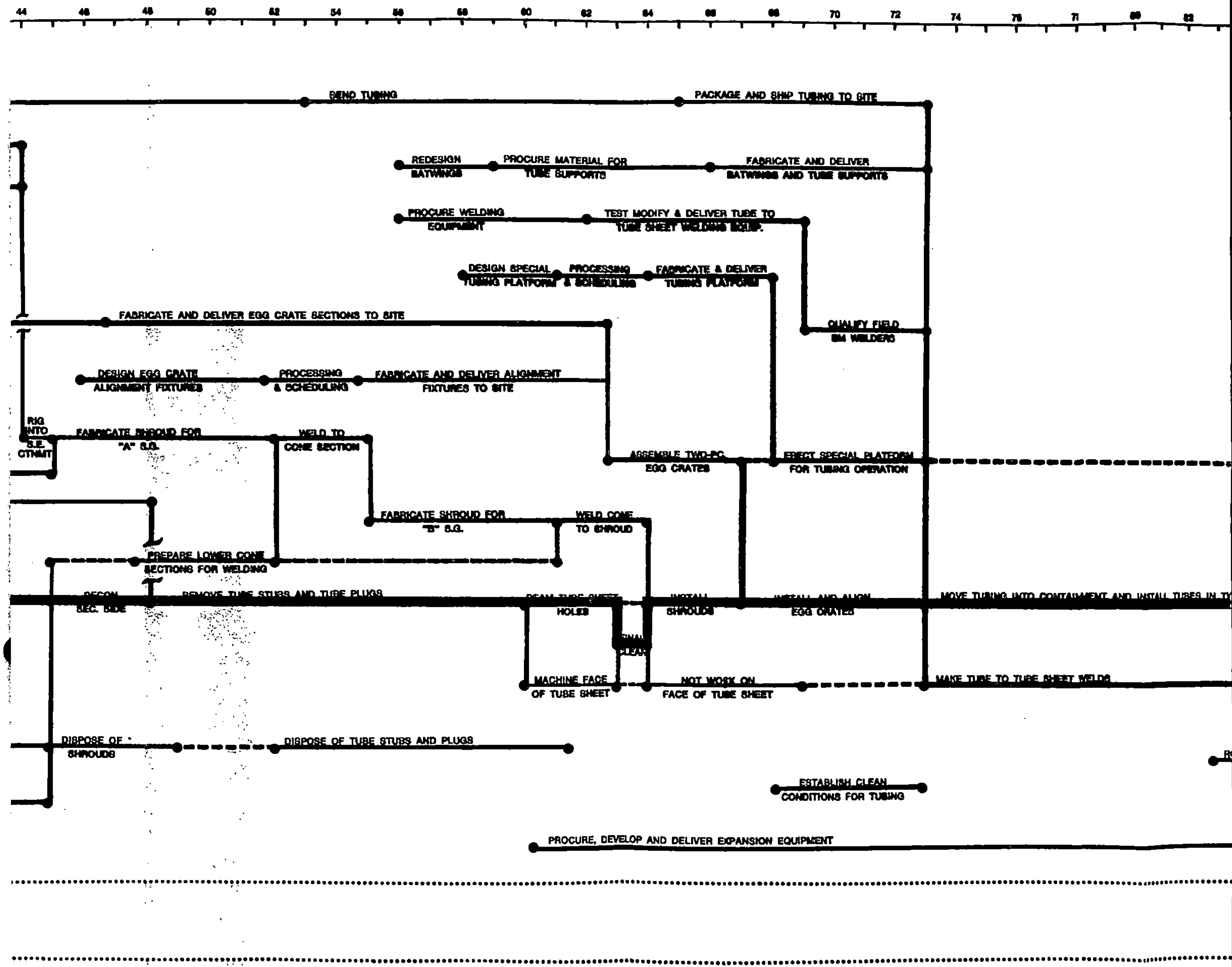
COMPLETE ENGINEERING TUBE BUNDLE
NEGOTIATIONS AND P.O.
PROCURE AND DELIVER RAW MATERIAL TO FABRICATION PLANT
DRAW TUBING
ENGINEER SHROUD
P.O.
FABRICATE AND DELIVER SHROUD SECTIONS TO SITE
DESIGN ROUNDING RINGS
PROCESSING & SCHEDULING
FABRICATE & DELIVER ROUNDING RINGS AND TIE STRAPS TO SITE
PIERCING EQUIPMENT
TEST AND MODIFY PROTOTYPE
PIERCING EQUIP.
ENGINEERING FOR TWO PIECE EGG CRATES
PROCURE MATERIAL FOR EGG CRATES
ORDER AND DELIVER SPECIAL HEAD CUTTING EQUIP.
DESIGN DEVICE
PROCESSING & SCHEDULING
FABRICATE & DELIVER HEAD LIFTING DEVICE
DESIGN, PROCURE, BUILD, TEST AND DELIVER EQUIPMENT FOR TUBE STUB AND PLUG REMOVAL
QUALIFY WELDERS
DEVELOP DECONTAMINATION PLAN
GENERAL DECON PREPARATIONS
DECON PREPARATIONS IN CONTAINMENT
DECON OPEN TUBES
PLUGS
DECON TUBES
DRAIN—OPEN S.G. INSTALL NOZZLE COVERS
DECON HP HDS
CONSTRUCT SCAFFOLD AROUND S.G.'S
REMOVE HD INSL
REMOVE DEF. PL ATTACHS LIFTING DEVICE
CUT & REMOVE S.G. TOP HEADS
CUT CONE OF SHROUD
REMOVE S.S.P.
REMOVE I BEAMS
CUT AND REMOVE TUBING, EGG CRATES AND SUPPORTS AND REMOVE SHROUDS
PREPARE DETAILED FIELD PROCEDURES AND QC INSTRUCTIONS
REMOVE SEPARATORS
ORDER AND DELIVER CONST. TOOLS & EQUIP.
CONSTRUCT SCAFFOLD
REMOVE INSL. & MISC. PIPING
CUT STEAM LINE PLACE IN POOL
REMOVE SCAFFOLD
REMOVE DRYERS FROM S.G. HEADS
REMOVE SUPPORTS
MOBILIZE ON SITE
DRAIN SEC. SYS. OPEN MANWAYS
CUT DRAINS FROM DRYERS TO S.S.P.
DECISION
COOL DOWN REMOVE HEAD
REMOVE & STORE FUEL
REPLACE U.G.S. & R.V. HEAD
REPLACE MISSILE SHIELD
DISPOSE OF STEAM SEPARATORS & DRYERS
DISPOSE OF TUBING & EGG CRATES & I BEAMS
DISASSEMBLE REFUEL M.
DESIGN VENT SYSTEM
PROCURE VENTING EQUIPMENT
INSTALL VENTING EQUIPMENT
CUT T.S. INTO SECTIONS—DECON
REMOVE T.S. SECTS.
COMPLETE DESIGN
PROCURE MATERIAL FOR PLATFORM
FABRICATE AND CONSTRUCT WORK PLATFORM OVER REFUELING POOL
DESIGN PLATFORM FOR SPENT FUEL POOL
PROCURE MATERIAL FOR SPENT FUEL POOL PLATFORM
CONSTRUCT PLATFORM OVER SPENT FUEL POOL
PLACE BALANCE OF PLANT IN STORAGE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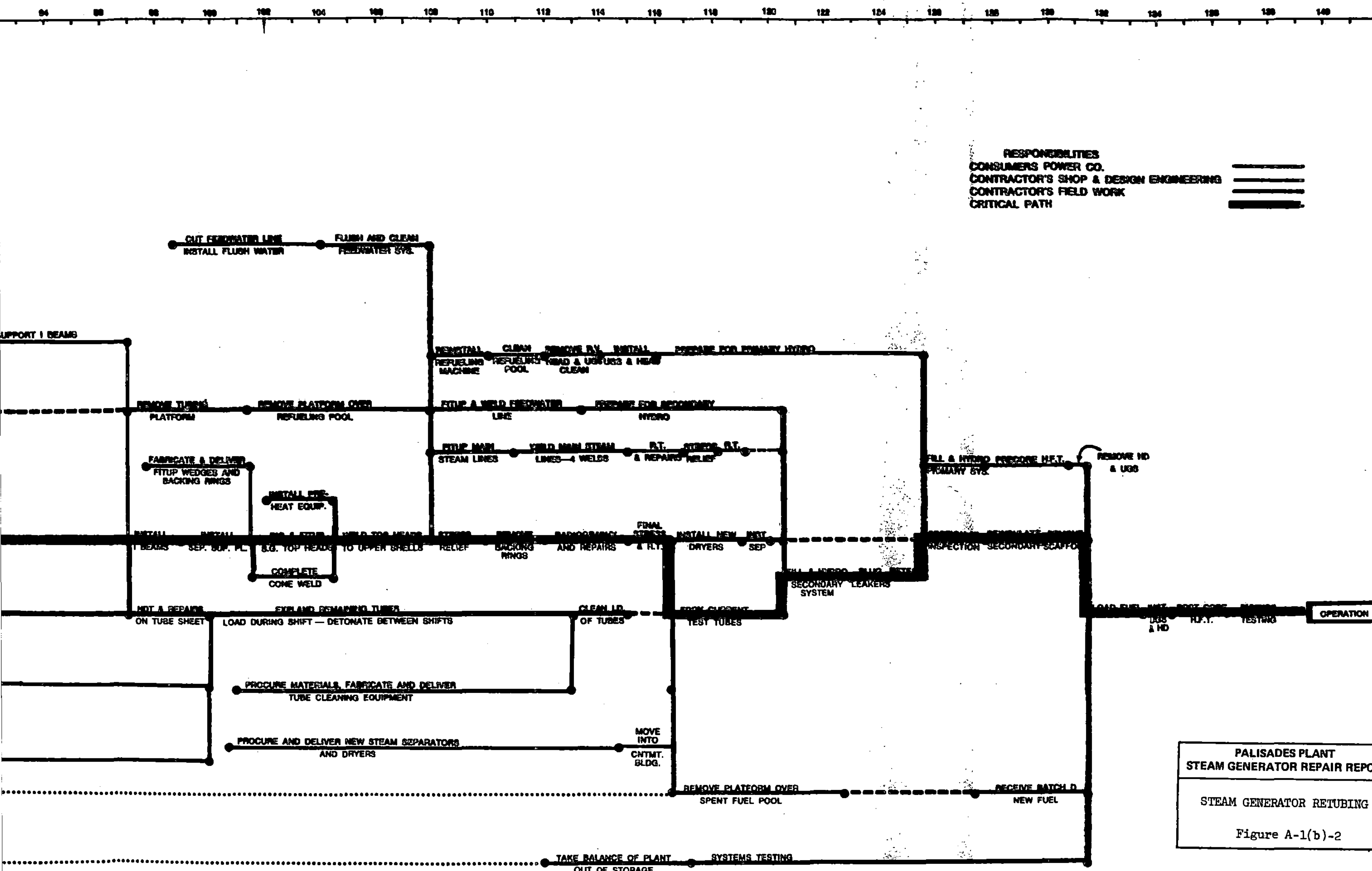
SEND TUBING
PACKAGE AND SHIP TUBING TO SITE
REDESIGN BATWINGS
PROCURE MATERIAL FOR TUBE SUPPORTS
FABRICATE AND DELIVER BATWINGS AND TUBE SUPPORTS
PROCURE WELDING EQUIPMENT
TEST MODIFY & DELIVER TUBE TO TUBE SHEET WELDING EQUIP.
DESIGN SPECIAL TUBING PLATFORM
PROCESSING & SCHEDULING
FABRICATE & DELIVER TUBING PLATFORM
QUALIFY FIELD WELDERS
FABRICATE AND DELIVER EGG CRATE SECTIONS TO SITE
DESIGN EGG CRATE ALIGNMENT FIXTURES
PROCESSING & SCHEDULING
FABRICATE AND DELIVER ALIGNMENT FIXTURES TO SITE
RIG INTO S.E. CTNMT
FABRICATE SHROUD FOR "A" S.G.
WELD TO CONE SECTION
ASSEMBLE TWO-PC. EGG CRATES
ERECT SPECIAL PLATFORM FOR TUBING OPERATION
FABRICATE SHROUD FOR "B" S.G.
WELD CONE TO SHROUD
PREPARE LOWER CONE SECTIONS FOR WELDING
DECON SEC. SIDE
REMOVE TUBE STUBS AND TUBE PLUGS
REAM TUBE SHEET HOLES
FINAL CLEAN
INSTALL SHROUDS
INSTALL AND ALIGN EGG CRATES
MOVE TUBING INTO CONTAINMENT AND INSTALL TUBES IN TY
MACHINE FACE OF TUBE SHEET
NOT WORK ON FACE OF TUBE SHEET
MAKE TUBE TO TUBE SHEET WELDS
DISPOSE OF SHROUDS
DISPOSE OF TUBE STUBS AND PLUGS
ESTABLISH CLEAN CONDITIONS FOR TUBING
PROCURE, DEVELOP AND DELIVER EXPANSION EQUIPMENT

FIG A-1(b)-2



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

STEAM GENERATOR RETUBING

Figure A-1(b)-2

FIG A-1(b)-2

PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

STEAM GENERATOR RETUBING

Figure A-1(b)-2

PALISADES PLANT SGRR

A-2 8.7 Man-Rem Consideration

The three alternate repair methods have comparable direct costs. The Palisades Plant Steam Generator Repair Report, Docket Number 50-255, also shows the occupational exposure of 1,200-5,000 man-rem for complete replacement of steam generators and replacement of evaporator sections. Retubing the steam generators, in place, has an estimated 40,000 man-rem exposure, Table 8.8-1, Comparison of Repair Alternatives. Based on biological savings of $1,000/man-rem cost is $35,000,000, if Alternate 3 is selected over the other alternatives. Give a detailed breakdown of how the man-rem exposure was estimated for retubing the steam generators. Also, include a breakdown of man-rem exposures for the other alternatives. The man-rem estimate should be of same detail as Table 17, Comparison of Exposure Estimates in NUREG/CR-0199, "Radiological Assessment of Steam Generator Removal and Replacement."

RESPONSE:

The three alternate repair methods described in Table 8.8-1 were replacement (complete units), replacement (evaporator sections), and retube in place.

A detailed breakdown of how the man-rem exposure was estimated for replacement of the steam generators as complete units has been shown in the repair report as Section 4.3.7 and Table 4.3.2.

The manhours and corresponding man-rem estimates associated with the replacement of the steam generator evaporator sections closely follow the manhour estimates made in Table 4.3.2. The difference would be a man-rem increase in Work Areas 10 and 11 which would reflect the additional time involved for a circumferential cut and weld on the shell of the steam generators.

Since the significant influence on the total man-rem for either replacement alternative is the welding and cutting technique used on the primary coolant piping (see footnote, Table 4.3.2), it is felt that the 1,200-5,000 man-rem range presented in Tables 8.8-1 and 4.3.2 is an adequate description of either replacement as a complete unit or replacement of evaporator sections only.

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The remaining repair technique described in Table 8.8-1 is to retube the steam generators in place. The following is a breakdown of how the man-rem exposure was estimated for the rerubing alternative.

Items (A-F) were used as the basis for the man-rem estimate.

A. TASK

TASK	TIME, WEEKS
1. Tube removal	20
2. Tube removal, 2 feet from tube sheet	2
3. Tube stub removal	16
4. Tube replacement, rolling, welding, etc	27
Total	65

B. It has been estimated that approximately 300 boilermakers would be required to complete the above tasks. It is assumed that all 300 craftworkers can be used interchangeably.

C. Tube removal would begin at Week 46 and tube welding and rolling would begin at Week 73. The radiation dose rate at the tubesheet is estimated as 0.8 r/hr at Week 73. The radiation dose rate at the tube bend area is 5 mr/hr at Week 46. The radiation dose rate remains constant with respect to height in the steam generators from the U-bend area to within 2 feet of the tubesheet at which point it increases to the dose rate specified at the tubesheet.

D. It has been estimated that only 35% of the workers' time will be spent in the specific radiation field (see Repair Report, Section 4.3.7). The remaining time is spent in security, access control, and time spent donning and removing protective clothing in nonradiation areas.

E. Craftworkers are available at 150/shift, working 10 hours per shift, 2 shifts per day for a 6-day week.

F. Reduction factors used for local decontamination and shielding during tube stub removal and tube replacement are 0.2 and 0.1, respectively.

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Using the above information, the integrated dose for the retubing alternative is as follows:

0.005 r/hr x 150 men x 20 hrs/day x 20 wks x 6 days/wk x .35 = 630 man-rem

\+ 0.8 r/hr x 150 men x 20 hrs/day x 2 wks x 6 day/wk x .35 = 10,080 man-rem

\+ 0.8 r/hr x 150 men x 20 hrs/day x 16 wks x 6 days/wk x .35 x 0.2 = 16,128 man-rem

\+ 0.8 r/hr x 150 men x 20 hrs/day x 27 wks x 6 days/wk x .35 x 0.1 = 13,608 man-rem

Total = 630 + 10,080 + 16,128 + 13,608 = 40,446 man-rem

PALISADES PLANT SGRR

B-1(a) The original Palisades steam generators appear to be unusually prone to promoting hideout of impurities from the coolant. This may be a result of the extremely dense forest of tubes and relatively small tube-to-tube spacing. Therefore, indicate:

(a) The number of spacing of tubes in the replacement generators and how they compare with the existing generators

RESPONSE:

The replacement steam generators will have 8,182 heat transfer tubes of 0.750-inch OD and .042-inch average wall. The original steam generators have 8,519 heat transfer tubes of 0.750-inch OD and .048-inch average wall. The number of heat transfer tubes for the replacement steam generator was established, using an equivalent UA basis, i.e., (UA) replacement steam generators = (UA) original steam generator. This equivalent UA basis will provide the replacement steam generators with essentially the same thermal performance capability as the original steam generator. The tube bundle geometry is arranged using a 1-inch triangular pitch pattern for the tube holes and is identical to the spacing pattern utilized on the original generators.

PALISADES PLANT SGRR

B-1(b) Details of design changes made to improve the flow characteristics to eliminate or reduce the potential for sludge accumulation, and . . .

RESPONSE:

Design features which enhance the flow distribution and circulation which have been included in the design of the replacement steam generators are:

1. Eggcrate type tube supports
2. Tube lane divider plate
3. Increased blowdown capability

The eggcrate type tube support system used on the replacement steam generator offers several advantages over the drilled tube support plates used on the original generators from a hydraulic standpoint. Since an eggcrate support is of a relatively opentype construction, localized crevices adjacent to tube surfaces are minimized. The eggcrate tube support system in the steam generator provides a minimum number of potential localized steam blanketed areas which might be present in the annular gaps between tubes and drilled support plate holes. The open flow area of each eggcrate support is calculated to be 69% of the total flow area. From a different point of view, the eggcrate support obstructs only 31% of the available secondary flow area. The large open flow areas available in the eggcrate support system avoids the accumulation of boiler water deposits by minimizing local flow eddies and flat surfaces which are present in other commonly used tube bundle support systems. The tube support design on the replacement steam generator has also been designed to reduce the number of supports, both horizontal and vertical grids in the tube bend region. In addition, the vertical tube pitch in the bend region has been increased from 1.0 inches to 1.75 inches which significantly reduces flow resistance in the bend region. The resultant effect is that the tube support system selected for the replacement steam generator is a design which reduces the potential for localized corrosion and increases internal recirculation.

On the replacement steam generator, a divider plate is mounted in the tube lane between the hot and cold leg sides of the tube bundle. This divider plate prevents the preferential bypass of the tube bundle by recirculating water exiting from the downcomer along the tube lane, thereby forcing the recirculation fluid from the downcomer to be directed across the tube bundle, ensuring relatively high radial fluid velocities throughout the region above the tubesheet.

PALISADES PLANT SGRR

The blowdown capabilities for the replacement steam generators compliment the improved hydraulics associated with the replacement generators for solids control within the steam generator. With the tube lane divider plate preventing flow bypass, the lowest radial recirculation fluid flow velocities exist in or near the open center region of the tube bundle where dropout of the solid particles occur. It is in this region that the blowdown ducts are located to take suction in a circular pattern.

PALISADES PLANT SGRR

B-1(c) Indicate what water chemistry controls you plan to use to minimize degradation of the replacement generators.

RESPONSE:

Primary Coolant System

We plan to maintain our current program to control water chemistry in the primary coolant system. Primary coolant system operating parameters are to be maintained as follows:

pH	4.5 to 10.2
Lithium	0.2 ppm to 1.0 ppm
Cl	<0.12 ppm
Dissolved O_2	<0.10 ppm
F	<0.10 ppm

pH is controlled by a balance between boric acid concentration and lithium concentration maintained by purification flowrate via the letdown system.

F and Cl will be controlled by purification flowrate via the letdown system and/or the makeup source (makeup demineralizers).

Dissolved O_2 will be maintained as low as possible using a continued excess of hydrogen during operation and will be controlled using hydrazine, as necessary, at system startup or shutdown.

Other corrosion products may be controlled by purification flowrate via the letdown system.

Secondary Systems

We plan to maintain our current "all volatile chemistry" program to control water chemistry in the shell side of the steam generators and in the secondary system. Steam generator operation parameters are to be maintained as follows:

pH	8.2 to 9.2
Cl	<0.1 ppm
Dissolved O_2	<0.01 ppm
Specific conductivity	<7 mmho/cm

The "all volatile chemistry" program (in effect for over 3 years at Palisades) consists primarily of adding hydrazine and morpholine for control of dissolved O_2 and pH, respectively. Dissolved O_2 will be maintained as low as practical.

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Cl will be controlled by flowrate through the full flow condensate polishing demineralizers and/or the makeup source (makeup demineralizers).

During wet layup, the shell side of the steam generators will be recycled to facilitate mixing of chemicals and improve the turnover time of the water inventory in the steam generators (reference Page 3-1 of the SGRR). Additional hydrazine and morpholine will be used for control of dissolved O_2 and pH, respectively.

Since our earlier problems with steam generator degradation, modifications have been completed for both the condensate and feedwater systems to improve secondary chemistry as follows:

1. To improve water purity during startup, shutdown, and abnormal operations, full flow condensate polishing demineralizers have already been installed with feedwater recirculation capability.

2. To reduce impurities and to maintain integrity of the secondary feedwater and condensate system(s), stainless steel tubing has been installed in the Numbers 1 through 4 feedwater heaters and the main condenser has been retubed using primarily 90-10 Cu Ni tubing with small quantities of stainless steel tubing in the periphery tubes and in the air removal section.

3. To improve steam generator chemistry, a new makeup demineralizer and water storage facility has been installed, which produces a better quality of water and provides for a large reservoir of water for blowdown considerations.

PALISADES PLANT SGRR

B-2 The eggcrate tube supports in the replacement steam generators will be fabricated of Type 409 stainless steel. Since Type 409 is only marginally a stainless steel, provide documentation of technical data showing its behavior under off normal water chemistry conditions.

RESPONSE:

Combustion Engineering, in conjunction with the Electric Power Research Institute, has performed model steam generator testing in various faulted environments where ferritic stainless steels were included side by side with carbon steel. The test results demonstrate conclusively that 409 ferritic stainless steels perform better than carbon steel in all environments tested. General corrosion of 409 is less than that of carbon steel by at least a factor of four and localized attack in heat transfer areas is orders of magnitude less for 409. The following table (B-2-1) provides general corrosion rates.

Localized corrosion comparisons of carbon steel with 409 at heat transfer/tube support locations were conducted in a model boiler which operated for 282 days, utilizing intermittent seawater injections at concentrations of from less than 50 ppb to 30 ppm chloride. The average daily chloride concentration was approximately 4.8 ppm (conductivity 5 to 200 μmhos/cm). Test results are shown in the following table (B-2-2).

PALISADES PLANT SGRR

TABLE B-2-1

GENERAL CORROSION RATES IN FAULTED
VOLATILE CHEMISTRY MODEL BOILER ENVIRONMENTS

		Average Surface Corrosion Rate (Mils per year)	
Faulted Chemistry Condition	Days Steaming	Carbon Steel	409 Stainless
Fresh water condenser leakage (12-15 μmhos/cm)	287	0.08	0.01
Ammonia cycle condensate polisher effluent (12-15 μmhos/cm)	179	0.16	0.04
Intermittent sea-water condenser leakage (5-200 μmhos/cm)	179	0.11	0.03
Intermittent sea-water condenser leakage (10-20 μmhos/cm)	179	0.24	0.04
Condensate polisher resin fines addition (20-60 μmhos/cm)	273	1.17	0.07

NOTE: Normal steam generator conductivity is maintained at approximately 3.0 μmhos/cm with an alarm setpoint at 4.0 μmhos/cm

PALISADES PLANT SGRR

TABLE B-2-2

TUBE SUPPORT MATERIALS COMPARISON CORROSION TESTING AT HEAT TRANSFER/SUPPORT LOCATIONS (MODEL BOILER TESTING)

Parameter	Carbon Steel Support Plate	409 Eggcrate
Denting	8 mils radially	None
Support plate cracking	Extensive	None
Corrosion of support material	Extensive	6.4 mils maximum pitting

PALISADES PLANT SGRR

B-3 Indicate what materials will be used for the batwing strips in the replacement generators.

RESPONSE:

The batwing strips on the replacement steam generator will be fabricated from 409 ferritic stainless steel strips. All tube support structures, including eggcrate supports on the replacement steam generators, will be fabricated using 409 stainless steel material.

PALISADES PLANT SGRR

B-4 Discuss what effects the thinner tube walls will have on the existing plugging criteria.

RESPONSE:

The tube plugging criteria established for the original Palisades steam generator was based upon a minimum tube wall thickness of .017 inches for the steam generator tubes and tube supports to retain structural adequacy during a hypothetical large pipe break. Utilizing this minimum tube wall thickness, it was determined that tubes having a local uniform degradation up to 64% of the nominal tube wall could withstand the postulated accident conditions and still meet the requisite code and regulatory guide criteria for faulted conditions.

For the replacement steam generator, it is anticipated that the results of similar analysis will demonstrate the structural adequacy of the tubes and tube supports, for the same accident conditions at 2,530 MW t, while retaining the same minimum tube wall thickness requirements as the original steam generator. Since the replacement steam generator will utilize tubes having a nominal wall thickness of .042 inches, the allowable degradation value, when correlated on a percentage basis, is anticipated to be 59% of the nominal tube wall.

PALISADES PLANT SGRR

C-1 Provide the following additional information regarding ALARA considerations (Sections 1.1.5 and 4.3.5):

(1) Duration of exposure associated with anticipated replacement/repair tasks

(2) Repetition rate of the tasks

(3) Numbers of work force exposed during each task

(4) Occupational exposures associated with anticipated replacement/repair activities

RESPONSE:

Tables (C-1-1 through C-1-5) provide the requested additional information. The manhour and corresponding man-rem estimates have changed from the original presented in Table 4.3.2. The changes are based on modifications to the welding techniques, described as Alternative (A_1) and Alternative (A_2), and further development of work packages.

The new exposure estimates are as follows: 1,547 to 2,808 man-rem (A_1), based on manual welding the reactor coolant pipe carbon steel portion and machine welding the cladding (with remote viewing), and 1,537 to 2,663 man-rem (A_2) based on machine welding (with remote viewing) both the carbon steel and cladding. The man-rem range reflects two analyses, for 140 and 42 days after shut down for the commencement of primary system pipe cutting. These estimates do not include a contingency.

It should be noted that Work Area 8, which represents manhours spent inside the reactor coolant pipe, has been expanded to appropriately differentiate radiation field levels before and after local decontamination.

TABLE C-1-1
(Sheet 1)

MANUAL WELDING OF RC PIPE WITH MACHINE CLADDING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
1	Scaffold, cut, and remove construction opening liner plate.	180	6	1	1	1,080	
		83	6	2	1	498	
2	Install new liner plate for construction opening, including fitup, scaffolding welding, etc.	233	6	3	1	1,398	
		100	6	2	1	600	
3	Cover and uncover spent fuel pool (protective cover).	270	7	16	0	1,890	
		116	7	2	0	812	
		240	7	1	0	1,680	
4	Cut reactor coolant pipe.	15	2	4	12	360	Inside radiation control envelope
		15	2	6	12	360	
		17	2	2	12	408	
		12	2	9	12	288	
5	Machine weld preparation on ends of reactor coolant pipes.	58	2	7	18	2,088	12 weld preps would be on decontaminated pipe and 6 on pipe attached to reactor.
		35	2	2	18	1,260	
		23	2	9	18	828	
6	Rig, fitup, line up, and tack in place reactor coolant pipe closure spools.	28	5	7	12	1,680	
		17	5	2	12	1,020	
		11	5	9	12	660	

*Refer to Table C-1-5

TABLE C-1-1
(Sheet 2)

MANUAL WELDING OF RC PIPE WITH MACHINE CLADDING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
7	Weld hot leg reactor coolant pipe (carbon steel).	131	4	5	4	2,096	Manual welding 8C level
		2	1	8C	4	8	
		131	4	9	4	2,104	
		113	4	2	4	1,800	
8	Clad hot leg reactor coolant pipe (stainless steel)	2	1	8C	4	8	Utilizing machine welding with remote welding
		50	2	9	4	400	
		30	2	2	4	240	
9	Weld cold leg reactor coolant pipe (carbon steel).	89	4	5	8	2,864	Manual welding
		2	1	8C	8	16	Manual welding
		91	4	9	8	2,896	
		77	4	2	8	2,448	
10	Clad cold leg reactor coolant pipe (stainless steel)	2	1	8C	8	16	Machine welding with remote viewing
		45	2	9	8	720	
		28	2	2	8	448	
11	Stress relieve reactor coolant pipe.	22	4	5	12	1,056	
		2	1	8C	12	24	
		22	4	9	12	1,056	
		20	4	2	12	960	

*Refer to Table C-1-5

TABLE C-1-1
(Sheet 3)

MANUAL WELDING OF RC PIPE WITH MACHINE CLADDING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
12	X-Ray and NDT reactor coolant pipe.	1/2	1	8	48	24	Leave pill guide in place 8C when possible in order to reduce exposure.
		4	2	5	120	1,960	
		3	2	2	120	720	
		1	2	9	120	240	
13	Install cleanliness plugs in reactor coolant pipe prior to cutting pipe.	1/2	1	8A	6	3	
		1	2	4	6	12	
14	Clean inside of reactor coolant pipe after welding.	1	1	8C	12	8	
15	Cover steam generator reactor coolant nozzles	1	4	8B	6	24	
		1	4	1	6	24	
		1	4	9	6	24	
		4	2	7	6	48	(Seal weld only)
16	Cover ends of reactor coolant pipe spools (temporary).	1	4	8B	18	72	
		2	4	7	18	144	
		1/2	4	1	18	36	
		1	4	2	18	72	
17	Rig reactor coolant pipes to decontamination area.	3	4	5	6	72	
		1	4	2	6	24	

*Refer to Table C-1-5

TABLE C-1-1
(Sheet 4)

MANUAL WELDING OF RC PIPE WITH MACHINE CLADDING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
18	Measure reactor coolant pipe closure spools.	4	3	5	12	144	
		1	3	2	12	36	
19	Remove insulation from steam generators.	38	4	2	2	304	
		32	4	4	2	256	
		25	4	9	2	200	
		32	4	10	2	256	
20	Reinsulate new steam generator.	180	4	2	2	1,440	
		120	4	9	2	960	
		150	4	11	2	1,200	
		150	4	17.	2	1,200	
21	Remove and replace reactor coolant pipe insulation.	5	2	2	12	120	
		3	2	4	12	72	
		5	2	5	12	120	
		3	2	9	12	72	
22	Cut, remove, bevel, erect, weld, stress relieve, and insulate main steam lines at top of steam generator.	260	3	2	2	1,560	
		277	3	9	2	1,662	
		40	3	10	2	240	
		148	3	11	2	888	
		148	3	12	2	888	
23	X-Ray and NDT main steam line.	40	2	2	2	160	(6 weld)
		25	2	9	2	100	
		65	2	11	2	260	

*Refer to Table C-1-5

TABLE C-1-1
(Sheet 5)

MANUAL WELDING OF RC PIPE WITH MACHINE CLADDING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
24	Cut, bevel, erect, weld, stress reileve, and insulate feedwater line.	162	2	2	2	648	
		107	2	9	2	428	
		25	2	10	2	100	
		245	2	11	2	980	
25	X-Ray and NDT feedwater line.	20	2	2	2	80	(4 welds)
		45	2	11	2	180	
26	Remove and replace miscellaneous small pipe near steam generators.	60	4	2	1	240	
		50	4	4	1	200	
		40	4	9	1	160	
		50	4	17	1	200	
27	Move component cooling water tank and clean resin tank out of way and reinstall.	75	4	2	1	300	
		50	4	9	1	200	
		125	4	14	1	500	
28	Install blowdown, tank, pump, and insulation.	40	3	2	1	120	
		20	3	9	1	60	
		67	3	15	1	201	
29	Install new blowdown piping inside containment.	122	6	2	1	732	
		13	6	4	1	78	
		92	6	5	1	552	
		177	6	9	1	1,062	

*Refer to Table C-1-5

TABLE C-1-1
(Sheet 6)

MANUAL WELDING OF RC PIPE WITH MACHINE CLADDING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
30	Install new blowdown piping outside containment.	193	6	2	1	1,158	
		128	6	9	1	768	
		322	6	15	1	1,932	
31	Remove electrical inside containment so steam generator can be removed.	37	4	2	1	148	
		63	4	4	1	252	
		25	4	9	1	150	
32	Reinstall electrical inside containment.	90	8	2	1	720	
		60	8	9	1	480	
		75	8	11	1	600	
		75	8	17	1	600	
33	Install electrical for new blowdown system.	92	4	2	1	368	
		12	4	9	1	48	
		173	4	15	1	692	
		32	4	17	1	128	
34	Install and remove equipment required to monitor position of reactor and steam generators during weldup of reactor coolant pipe.	30	2	2	1	60	
		-	-	10	-	-	
		130	1	9	1	130	
		20	2	17	1	40	
		20	2	18	1	40	

*Refer to Table C-1-5

TABLE C-1-1
(Sheet 7)

MANUAL WELDING OF RC PIPE WITH MACHINE CLADDING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
35	Mobilize, install cages. Remove dome and buttress facia, relax tendons, remove tendons, chip concrete, cut rebar and tendon sheathing for containment construction opening.	1,075	20	1	1	21,500	
36	Replace opening on containment, including the following: Replace tendon sheathing, rebar, concrete and tendons, stress tendons and replace dome and buttress facia, demobilize.	1,275	15	1	1	19,125	
37	Construct and remove barge slip.	829	14	1	1	11,606	
38	Foundations for rigging equipment, including sheetpiling, earthwork, concrete foundations and removal of foundations (at containment building).	692	12	1	1	8,304	
39	Mobile heavy lift rigger.	100	15	1	1	1,500	
40	Assemble 4 crawlers.	150	10	1	1	1,500	

*Refer to Table C-1-5

TABLE C-1-1
(Sheet 8)

MANUAL WELDING OF RC PIPE WITH MACHINE CLADDING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
41	Assemble jacking frame at barge.	62	12	1	1	744	
42	Assemble jacking frame at containment.	78	12	1	1	936	
43	Preassemble equipment for inside containment.	78	12	1	1	936	
44	Install lifting equipment inside containment.	112	15	2	1	1,680	
		158	15	12	1	2,370	
		105	15	13	1	1,575	
45	Remove existing steam generators from containment (rigging).	23	15	1	2	690	
		26	15	2	2	780	
		44	15	4	2	1,320	
		19	15	19	2	570	
46	Transport and store existing steam generators.	31	10	19	2	620	
47	Receive and ballast barge.	20	5	1	1	100	
48	Offload, store, load, and transport new steam generators.	158	10	1	2	3,160	
49	Rerig as required to install.	16	15	1	2	480	

*Refer to Table C-1-5

TABLE C-1-1
(Sheet 9)

MANUAL WELDING OF RC PIPE WITH MACHINE CLADDING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
50	Install new steam generators.	19	15	1	2	570	
		21	15	2	2	630	
		37	15	12	2	1,110	
		12	15	17	2	360	
51	Remove all external rigging equipment from site.	225	15	1	1	3,375	
52	Remove all rigging equipment from containment.	75	15	2	1	1,125	
		105	15	12	1	1,575	
		70	15	13	1	1,050	
53	Remove internal rigging equipment from site.	63	15	1	1	945	
54	Miscellaneous rigging	150	5	1	1	750	
55	Steam generator storage building.	406	10	1	1	4,060	
56	Cut and remove top support of steam generator.	8	5	10	1	40	
		5	5	2	1	25	
		3	5	9	1	15	
57	Remove shims other steam generator top support.	6	4	10	1	24	
		4	4	2	1	16	
		2	4	9	1	8	

TABLE C-1-1
(Sheet 10)

MANUAL WELDING OF RC PIPE WITH MACHINE CLADDING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
58	Remove hydraulic snubbers.	4	5	10	8	160	
		3	5	2	8	120	
		1	5	9	8	40	
59	Unbolt existing steam generator.	4	3	4	2	24	
		3	3	2	2	18	
		1	3	9	2	6	
60	Remove shims at steam generator base.	15	3	4	2	90	
		9	3	2	2	54	
		6	3	9	2	36	
61	Bolt down steam generators.	10	3	5	2	60	
		6	3	2	2	36	
		4	3	9	2	24	
62	Reshim bottom steam generator (sliding base).	50	3	5	2	300	
		30	3	2	2	180	
		20	3	9	2	120	
63	Replace top steam generator support.	120	3	11	2	720	
		72	3	2	2	432	
		48	3	9	2	288	
64	Reshim top steam generator supports.	40	3	11	2	240	
		24	3	2	2	144	
		16	3	9	2	96	

*Refer to Table C-1-5

TABLE C-1-1
(Sheet 11)

MANUAL WELDING OF RC PIPE WITH MACHINE CLADDING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
65	Reinstall steam generator hydraulic snubbers.	30	4	11	8	960	
		18	4	2	8	576	
		12	4	9	8	384	
66	Miscellaneous pipe operations (welders tests, material hangling, scaffolding, training, hangers and supports, line testing, cleanup, tents).			1		25,088	Welders tests, training, material handling and fabrication of tents are in Location 1. Remainder of manhours were allocated on the basis of piping manhours in each location.
				2		7,030	
				4		952	
				5		3,699	
				8		20	
				9		6,982	
				10		552	
				11		1,088	
				12		413	
				15		914	
				17		87	
67	Distributables (startup, cleanup, scaffolding, welders tests other than pipe fitters, miscellaneous).			1		11,436	Welders tests and miscellaneous in Area 1. Remainder of manhours allocated on the basis of direct manhours excluding Area 1.
				2		13,298	
				3		1,003	
				4		1,526	
				5		4,883	
				6		131	
				7		1,613	
				8		51	
				9		9,766	
				10		567	
				11		2,921	

*Refer to Table C-1-5

TABLE C-1-1
(Sheet 12)

MANUAL WELDING OF RC PIPE WITH MACHINE CLADDING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
67	(Continued)			12		2,616	
				13		1,090	
				14		218	
				15		1,526	
				16		785	
				17		1,090	
				19		480	
68	Nonmanual			1		99,160	Office personnel and
				2		6,511	50% of engineers and
				3		53	supervision's man-
				4		1,110	hours are in Location
				5		2,872	#1. Remainder of man-
				6		16	hours by discipline
				7		85	were allocated to the
				8		14	proper location based
				9		5,770	on direct manhours
				10		426	expended in that loca-
				11		1,176	tion by discipline.
				12		895	
				13		149	For example: The man-
				14		11	hours for electrical
				15		1,143	engineers and supts
				16		141	were allocated based
				17		401	on the electrical
				19		67	direct hours expended
							on each task at
							each location.

*Refer to Table C-1-5

TABLE C-1-2
(Sheet 1)

SUMMARY OF MANHOURS FOR ALL TASKS BY LOCATION

MANUAL WELDING OF REACTOR COOLANT C.S. PIPE WITH
MACHINE CLADDING AND REMOTE VIEWING

TASK NO.	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
1		498	1,080																
2		600	1,398																
3	1,680	812														1,890			
4		408		360		360			288										
5		1,260					2,088		828										
6		1,020					1,680		660										
7		1,800			2,096			8	2,104										
8		240						8	400										
9		2,448			2,864			16	2,896										
10		448						16	720										
11		960			1,056			24	1,056										
12		720			960			24	240										
13				12				3											
14								8											
15	24						48	24	24										
16	36	72					144	72											
17		24			72														
18		36			144														
19		304		256					200	256									
20		1,440							960		1,200						1,200		
21		120		72	120				72										
22		1,560							1,662	240	888	888							
23		160							100		260								
24		648							428	100	980								
25		80									180								
26		240		200					160								200		
27		300							200					500					

TABLE C-1-2
(Sheet 2)

SUMMARY OF MANHOURS FOR ALL TASKS BY LOCATION

MANUAL WELDING OF REACTOR COOLANT C.S. PIPE WITH
MACHINE CLADDING AND REMOTE VIEWING

TASK NO.	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
28		120							60						201				
29		732		78	552				1,062										
30		1,158							768						1,932				
31		148		252					150										
32		720							480		680						600		
33		368							48						692		128		
34		60							130								40	40	
35	21,500																		
36	19,125																		
37	11,606																		
38	8,304																		
39	1,500																		
40	1,500																		
41	744																		
42	936																		
43	936																		
44		1,680																	
45	690	780		1,320								2,370	1,575						
46																			570
47	100																		620
48	3,160																		
49	480																		
50	570	630										1,110					360		
51	3,375																		
52		1,125										1,575	1,050						
53	945																		

TABLE C-1-2
(Sheet 3)

SUMMARY OF MANHOURS FOR ALL TASKS BY LOCATION

MANUAL WELDING OF REACTOR COOLANT C.S. PIPE WITH
MACHINE CLADDING AND REMOTE VIEWING

TASK NO.	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
54	750																		
55	4,060																		
56		25							15	40									
57									8	24						16			
58		120							40	160									
59		18		24					6										
60		54		90					36										
61		36			60				24										
62		180			300				120										
63		432							288		720								
64		144							96		240								
65		576							384		960								
66	25,088	7,030		952	3,699			20	6,982	552	1,088	413			914		87		
Total Directs	107,061	32,334	2,478	3,640	11,923	360	3,960	223	23,662	1,372	7,116	6,356	2,625	500	3,739	1,906	2,615	40	1,190
67	11,436	13,298	1,003	1,526	4,883	131	1,613	51	9,766	567	2,921	2,616	1,090	218	1,526	785	1,090	-	480
68	99,160	6,511	53	1,110	2,872	16	85	14	5,770	426	1,176	895	149	11	1,143	141	401	-	67
Total	217,657	52,143	3,534	6,276	19,678	507	5,658	288	39,198	2,365	11,213	9,867	3,864	729	6,408	2,832	4,106	40	1,737

TABLE C-1-3
(Sheet 1)

MACHINE WELDING OF R.C. PIPE WITH REMOTE VIEWING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
1	Scaffold, cut, and remove construction opening liner plate.	180	6	3	3	1,080	
		83	6	2	1	498	
2	Install new liner plate for construction opening, including fitup, scaffolding, welding, etc.	233	6	3	1	1,398	
		100	6	2	1	600	
3	Cover and uncover spent fuel pool (protective cover).	270	7	16	0	1,890	
		116	7	2	0	812	
		240	7	1	0	1,680	
4	Cut reactor coolant pipe.	15	2	4	12	360	Inside radiation control envelope
		15	2	6	12	360	
		17	2	2	12	408	
		12	2	9	12	288	
5	Machine weld prep on ends of reactor coolant pipes.	58	2	7	18	2,088	12 weld preps would be on decontaminated pipe and 6 on pipe attached to reactor.
		35	2	2	18	1,260	
		23	2	9	18	828	
6	Rig, fitup, lineup and tack in place reactor coolant pipe closure spools.	28	5	7	12	1,680	
		17	5	2	12	1,020	
		11	5	9	12	660	

*Refer to Table C-1-5

TABLE C-1-3
(Sheet 2)

MACHINE WELDING OF R.C. PIPE WITH REMOTE VIEWING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
7	Weld hot leg reactor coolant pipe (carbon steel).	4	2	7	4	32	Utilizing machine welding with remote viewing.
		3	1	8C	4	12	
		234	1	5	4	935	
		146	3	2	4	1,753	
		273	3	9	4	3,276	
8	Clad hot leg reactor coolant pipe (stainless steel)	2	1	8C	4	8	Utilizing machine welding with remote viewing.
		50	2	9	4	400	
		30	2	2	4	240	
9	Weld cold leg reactor coolant pipe (carbon steel).	4	2	7	8	64	
		2	1	8	8	16	
		160	1	5	8	1,280	
		100	3	2	8	2,400	
		186	3	9	8	4,464	
10	Clad cold leg reactor coolant pipe (stainless steel).	2	1	8C	8	16	Machine welding with remote viewing.
		45	2	9	8	720	
		28	2	2	8	448	
11	Stress relieve reactor coolant pipe.	22	4	5	12	1,056	
		2	1	8C	12	24	
		22	4	9	12	1,056	
		20	4	2	12	960	

*Refer to Table C-1-5

TABLE C-1-3
(Sheet 3)

MACHINE WELDING OF R.C. PIPE WITH REMOTE VIEWING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
12	X-Ray and NDT reactor coolant pipe.	1/2	1	8C	48	24	Leave pill guide in place 8C when possible in order to reduce exposure.
		4	2	5	120	1,960	
		3	2	2	120	720	
		1	2	9	120	240	
13	Install cleanliness plugs in reactor coolant pipe prior to cutting pipe.	1/2	1	8A	6	3	
		1	2	4	6	12	
14	Clean inside of reactor coolant pipe after welding.	1	1	8C	12	8	
15	Cover steam generator reactor coolant nozzles.	1	4	8B	6	24	
		1	4	1	6	24	
		1	4	9	6	24	
		4	2	7	6	48	(Seal weld only)
16	Cover ends of reactor coolant pipe spools (temporary).	1	4	8B	18	72	
		1	4	7	18	144	
		1/2	4	1	18	36	
		1	4	2	18	72	
17	Rig reactor coolant pipes to decontamination area.	3	4	5	6	72	
		1	4	2	6	24	

*Refer to Table C-1-5

TABLE C-1-3
(Sheet 4)

MACHINE WELDING OF R.C. PIPE WITH REMOTE VIEWING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
18	Measure reactor coolant pipe closure spools.	4	3	5	12	144	
		1	3	2	12	36	
19	Remove insulation from steam generators.	38	4	2	2	304	
		32	4	4	2	256	
		25	4	9	2	200	
		32	4	10	2	256	
20	Reinsulate new steam generator.	180	4	2	2	1,440	
		120	4	9	2	960	
		150	4	11	2	1,200	
		150	4	17	2	1,200	
21	Remove and replace reactor coolant pipe insulation.	5	2	2	12	120	
		3	2	4	12	72	
		5	2	5	12	120	
		3	2	9	12	72	
22	Cut, remove, bevel, erect, weld, stress relieve, and insulate main steam lines at top of steam generator.	260	3	2	2	1,560	
		277	3	9	2	1,662	
		40	3	10	2	240	
		148	3	11	2	888	
		148	3	12	2	888	

*Refer to Table C-1-5

PALISADES PLANT SGRR

TABLE C-1-3
(Sheet 5)

MACHINE WELDING OF R.C. PIPE WITH REMOTE VIEWING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
23	X-Ray and NDT main steam line.	40	2	2	2	160	(6 welds)
		25	2	9	2	100	
		65	2	11	2	260	
24	Cut, bevel, erect, weld, stress relieve, and insulate feedwater line.	162	2	2	2	648	
		107	2	9	2	428	
		25	2	10	2	100	
		245	2	11	2	980	
25	X-Ray and NDT feedwater line.	20	2	2	2	80	(4 welds)
		45	2	11	2	180	
26	Remove and replace miscellaneous small pipe near steam generators.	60	4	2	1	240	
		50	4	4	1	200	
		40	4	9	1	160	
		50	4	17	1	200	
27	Move component cooling water tank and clean resin tank out of way and reinstall.	75	4	2	1	300	
		50	4	9	1	200	
		125	4	14	1	500	

*Refer to Table C-1-5

TABLE C-1-3
(Sheet 6)

MACHINE WELDING OF R.C. PIPE WITH REMOTE VIEWING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
28	Install blowdown, tank, pump, and insulation.	40	3	2	1	120	
		20	3	9	1	60	
		67	3	15	1	201	
29	Install new blowdown piping inside containment.	122	6	2	1	732	
		13	6	4	1	78	
		92	6	5	1	552	
		177	6	9	1	1,062	
30	Install new blowdown piping outside containment.	193	6	2	1	1,158	
		128	6	9	1	768	
		322	6	15	1	1,932	
31	Remove electrical inside containment so steam generator can be removed.	37	4	2	1	148	
		63	4	4	1	252	
		25	4	9	1	150	
32	Reinstall electrical inside containment.	90	8	2	1	720	
		60	8	9	1	480	
		75	8	11	1	600	
		75	8	17	1	600	
33	Install electrical for new blowdown system.	92	4	2	1	368	
		12	4	9	1	48	
		173	4	15	1	692	
		32	4	17	1	128	

*Refer to Table C-1-5

TABLE C-1-3
(Sheet 7)

MACHINE WELDING OF R.C. PIPE WITH REMOTE VIEWING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
34	Install and remove equipment required to monitor position of reactor and steam generators during weldup of reactor coolant pipe.	30	2	2	1	60	
		–	–	10	–	–	
		130	1	9	1	130	
		20	2	17	1	40	
		20	2	18	1	40	
35	Mobilize, install cages. Remove dome and buttress facia, relax tendons, remove tendons, chip concrete, cut rebar and tendon sheathing for containment construction opening.	1,075	20	1	1	21,500	
36	Replace opening on containment, including the following: Replace tendon sheathing, rebar, concrete and tendons, stress tendons and replace dome and buttress facia, demobilize.	1,275	15	1	1	19,125	
37	Construct and remove barge slip.	829	14	1	1	11,606	

*Refer to Table C-1-5

TABLE C-1-3
(Sheet 8)

MACHINE WELDING OF R.C. PIPE WITH REMOTE VIEWING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
38	Foundations for rigging equipment, including sheet-piling, earthwork, concrete foundations, and removal of foundations (at containment building).	692	12	1	1	8,304	
39	Mobile heavy lift rigger	100	15	1	1	1,500	
40	Assemble 4 crawlers.	150	10	1	1	1,500	
41	Assemble jacking frame at barge.	62	12	1	1	744	
42	Assemble jacking frame at containment.	78	12	1	1	936	
43	Preassemble equipment for inside containment.	78	12	1	1	936	
44	Install lifting equipment inside containment.	112 158 105	15 15 15	2 12 13	1 1 1	1,680 2,370 1,575	

*Refer to Table C-1-5

PALISADES PLANT SGRR

TABLE C-1-3
(Sheet 9)

MACHINE WELDING OF R.C. PIPE WITH REMOTE VIEWING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
45	Remove existnig steam generators from containment (rigging).	23	15	1	2	690	
		26	15	2	2	780	
		44	15	4	2	1,320	
		19	15	19	2	570	
46	Transport and store existing steam generators.	31	10	19	2	620	
47	Receive and ballast barge.	20	5	1	1	100	
48	Offload, store, load, and transport new steam generators.	158	10	1	2	3,160	
49	Rerig as required to install.	16	15	1	2	480	
50	Install new steam generators.	19	15	1	2	570	
		21	15	2	2	630	
		37	15	12	2	1,110	
		12	15	17	2	360	
51	Remove all external rigging equipment from site.	225.	15	1	1	3,375	

*Refer to Table C-1-5

TABLE C-1-3
(Sheet 10)

MACHINE WELDING OF R.C. PIPE WITH REMOTE VIEWING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
52	Remove all rigging equipment from containment.	75	15	2	1	1,125	
		105	15	12	1	1,575	
		70	15	13	1	1,050	
53	Remove internal rigging equipment from site.	63	15	1	1	945	
54	Miscellaneous rigging.	150	5	1	1	750	
55	Steam generator storage building.	406	10	1	1	4,060	
56	Cut and remove top support of steam generator.	8	5	10	1	40	
		5	5	2	1	25	
		3	5	9	1	15	
57	Remove shims other steam generator top support.	6	4	10	1	24	
		4	4	2	1	16	
		2	4	9	1	8	
58	Remove hydraulic snubbers.	4	5	10	8	160	
		3	5	2	8	120	
		1	5	9	8	40	
59	Unbolt existing steam generator.	4	3	4	2	24	
		3	3	2	2	18	
		1	3	9	2	6	

*Refer to Table C-1-5

TABLE C-1-3
(Sheet 11)

MACHINE WELDING OF R.C. PIPE WITH REMOTE VIEWING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
60	Remove shims at steam generator base.	15	3	4	2	90	
		9	3	2	2	54	
		6	3	9	2	36	
61	Bolt down steam generators	10	3	5	2	60	
		6	3	2	2	36	
		4	3	9	2	24	
62	Reshim bottom steam generator (sliding base).	50	3	5	2	300	
		30	3	2	2	180	
		20	3	9	2	120	
63	Replace top steam generator support.	120	3	11	2	720	
		72	3	2	2	432	
		48	3	9	2	288	
64	Reshim top steam generator supports.	40	3	11	2	240	
		24	3	2	2	144	
		16	3	9	2	96	
65	Reinstall steam generator hydraulic snubbers.	30	4	11	8	960	
		18	4	2	8	576	
		12	4	9	8	384	

*Refer to Table C-1-5

TABLE C-1-3
(Sheet 12)

MACHINE WELDING OF R.C. PIPE WITH REMOTE VIEWING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
66	Miscellaneous pipe operations (welders tests, material handling, scaffolding, training, hangers and supports, line testing, cleanup, tests).			1		25,058	Welders tests, training, material handling and fabrication of tents are in Location 1. Remainder of manhours were allocated on the basis of piping manhours in each location.
				2		6,993	
				4		1,064	
				5		1,954	
				7		217	
				8		20	
				9		8,358	
				10		543	
				11		1,107	
				12		434	
				15		933	
				17		87	
67	Distributables (startup, cleanup, scaffolding, welders tests other than pipe fitters, miscellaneous).			1		11,400	
				2		13,385	
				3		1,003	
				4		1,526	
				5		3,052	
				6		131	
				7		1,744	
				8		---	

*Refer to Table C-1-5

TABLE C-1-3
(Sheet 13)

MACHINE WELDING OF R.C. PIPE WITH REMOTE VIEWING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
67	(Continued)				9	11,466	
					10	567	
					11	2,921	
					12	2,616	
					13	1,090	
					14	218	
					15	1,526	
					16	785	
					17	1,090	
					19	480	
68	Nonmanual				1	99,160	Office personnel +50%
					2	6,431	of engineers and
					3	53	supervision manhours
					4	1,191	are in Location 1.
					5	1,648	Remainder of manhours
					6	16	by discipline were
					7	238	allocated to the
					8	61	proper location based
					9	6,753	on direct manhours
					10	420	expended in that
					11	1,191	location by discipline.
					12	911	

*Refer to Table C-1-5

TABLE C-1-3
(Sheet 14)

MACHINE WELDING OF R.C. PIPE WITH REMOTE VIEWING

TASK NUMBER	DESCRIPTION	DURATION (HOURS)	AVERAGE NUMBER OF PERSONNEL	*LOCATION	NUMBER OF TIMES TASK PERFORMED	MANHOURS	COMMENTS
68	(Continued)				13	149	For example: The manhours for electrical engineers and supts were allocated based on the electrical direct hours expended on each task at each location.
					14	11	
					15	1,158	
					16	141	
					17	401	
					18	---	
					19	67	

*Refer to Table C-1-5

TABLE C-1-4
(Sheet 1)

SUMMARY OF MANHOURS FOR ALL TASKS BY LOCATION

WELDING OF REACTOR COOLANT PIPE BY
MACHINE WITH REMOTE VIEWING

TASK NO.	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
1		498	1,080																
2		600	1,398																
3	1,680	812														1,890			
4		408		360		360			288										
5		1,260					2,088		828										
6		1,020					1,680		660										
7		1,753			935		32	12	3,276										
8		240						8	400										
9		2,400			1,280		64	16	4,464										
10		448						16	720										
11		960			1,056			24	1,056										
12		720			960			24	240										
13				12				3											
14								8											
15	24						48	24	24										
16	36	72					144	72											
17		24			72														
18		36			144														
19		304		256					200	256									
20		1,440							960		1,200						1,200		
21		120		72	120				72										
22		1,560							1,662	240	888	888							
23		160							100		260								
24		648							428	100	980								
25		80									180								

TABLE C-1-4
(Sheet 2)

SUMMARY OF MANHOURS FOR ALL TASKS BY LOCATION

WELDING OF REACTOR COOLANT PIPE BY
MACHINE WITH REMOTE VIEWING

TASK NO.	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
26		240		200					160								200		
27		300							200					500					
28		120							60						201				
29		732		78	552				1,062										
30		1,158							768						1,932				
31		148		252					150										
32		720							480		600						600		
33		368							48						692		128		
34		60							130								40	40	
35	21,500																		
36	19,125																		
37	11,606																		
38	8,304																		
39	1,500																		
40	1,500																		
41	744																		
42	936																		
43	936																		
44		1,680																	
45	690	780		1,320								2,370	1,575						570
46																			620
47	100																		
48	3,160																		
49	480																		
50	570	630										1,110					360		
51	3,375																		
52		1,125										1,575	1,050						
53	945																		

TABLE C-1-4
(Sheet 3)

SUMMARY OF MANHOURS FOR ALL TASKS BY LOCATION

WELDING OF REACTOR COOLANT PIPE BY
MACHINE WITH REMOTE VIEWING

TASK NO.	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
54	750																		
55	4,060																		
56		25							15	40									
57									8	24						16			
58		120							40	160									
59		18		24					6										
60		54		90					36										
61		36			60				24										
62		180			300				120										
63		432							288		720								
64		144							96		240								
65		576							384		960								
Subtotal	82,021	25,209	2,478	2,664	5,479	360	4,056	207	19,453	820	6,028	5,943	2,625	500	2,825	1,906	2,528	40	1,190
66	25,058	6,993	-	1,064	1,954	-	217	20	8,358	543	1,107	434			933		87		
Total Directs	107,079	32,202	2,478	3,728	7,433	360	4,273	227	27,811	1,363	7,135	6,377	2,625	500	3,758	1,906	2,615	40	1,190
67	11,400	13,385	1,003	1,526	3,052	131	1,744	-	11,466	567	2,921	2,616	1,090	218	1,526	785	1,090	-	480
68	99,160	6,431	53	1,191	1,648	16	238	61	6,753	420	1,191	911	149	11	1,158	141	401	-	67
Total	217,639	52,018	3,534	6,445	12,133	507	6,255	288	46,030	2,350	11,247	9,904	3,864	729	6,422	2,832	4,106	40	1,737

PALISADES PLANT SGRR
TABLE C-1-5
(SHEET 1)
MAN-REM ESTIMATE

Work Area (Location)	(1) Average Radiation Field (rem/hr)	Estimated Manhours in Radiation Field (Manual Welding) A_1	(Machine Welding) A_2	(2) Area Man-Rem Dose (Man-Rem) (Manual Welding)	(Machine Welding)
1. Outside of power plant building but within security fence.	$.5 \times 10^{-5}$	217659	217639	1.1	1.1
2. Checking in and out through security and health physics as well as time spent suiting up, cleaning up and moving to and from work areas for personnel working in radioactive areas.	.0025	52143	52018	130.3	130
3. Inside containment near new construction opening.	.001	3534	3534	3.5	3.5
4. Within 6' of outside of reactor coolant pipe or bottom of steam generator prior to removal of steam generators.	.030 - .050	6276	6445	188.3 - 313.8	193.3 - 322.3
5. Within 6' outside of reactor coolant pipe after steam generator's removal.	.010 - .030	19678	12133	196.8 - 590.3	121.3 - 364.0
6. Within 6' of outside of reactor coolant pipe or bottom of steam generators with partial exposure to inside of reactor coolant pipe prior to steam generator's removal.	.050 - 0.100	507	507	25.4 - 50.7	25.4 - 50.7
7. Within 6' of outside of reactor coolant pipe with partial exposure to inside of reactor coolant pipe after steam generator's removal.	.050 - .100	5658	6255	283 - 454.8	312.8 - 625.5

PALISADES PLANT SGRR
TABLE C-1-5
(SHEET 2)
MAN-REM ESTIMATE

Work Area (Location)	(1) Average Radiation Field (rem/hr)	Estimated Manhours in Radiation Field (Manual Welding) A_1	(Machine Welding) A_2	(2) Area Man-Rem Dose (Man-Rem) (Manual Welding)	(Machine Welding)
8. a) Inside of reactor coolant pipe before decontamination.	9.0 - 12.0	4.3	4.2	38.7 - 51.6	37.8 - 50.4
b) Outside of pipe w/partial exposure inside before decontamination.	1.0 - 2.0	136	134	136 - 272.0	134 - 268.0
c) Inside of reactor coolant pipe after decontamination.	.035	148	150	5.2	5.3
9. Low radiation area within containment	.005	39198	46030	196	230.2
10. Within 6' of top half of original steam generators (installed in place).	.005	2365	2350	11.8	11.8
11. Within 6' of top half of new steam generators (installed in place).	.001	11213	11247	11.2	11.3
12. Operating floor or containment.	.005 - .010	9867	9904	49.3 - 98.7	49.5 - 99.0
13. Inside containment, at polar crane elevation.	.001	3867	3864	3.9	3.9
14. Auxiliary building near clean resin tank and cooling water tank.	.001	729	729	0.7	0.7
15. Auxiliary building near blowdown tank.	.001	6408	6442	6.4	6.4
16. Spent fuel pool floor.	.005	2832	2832	14.2	14.2
17. Within 6' of the bottom half of new steam generators (in place).	.010 - .030	4106	4106	41.1 - 123.2	41.1 - 123.2

PALISADES PLANT SGRR
TABLE C-1-5
(SHEET 3)
MAN-REM ESTIMATE

Work Area (Location)	(1) Average Radiation Field (rem/hr)	Estimated Manhours in Radiation Field (Manual Welding) A_1	Estimated Manhours in Radiation Field (Machine Welding) A_2	(2) Area Man-Rem Dose (Man-Rem) (Manual Welding)	(2) Area Man-Rem Dose (Man-Rem) (Machine Welding)
18. Within 6' of the outside of the reactor vessel.	.100	40	40	4.0	4.0
19. Next to the existing steam generators outside of the containment	.020 - .030	1737	1737	34.7 - 52.1	34.7 - 52.1
20. Installation of shielding and local decontamination.	(3)	(3)	(3)	165.7 - 300.8	164.7 - 285.3
				1547 - 2807.6	1537 - 2662.9

NOTE:

(1) Reduction factors attributed to shielding and/or decontamination have been incorporated into these field estimates and will not be presented here as a separate column.

(2) Further reduction in area man-rem dose could occur as work packages and ALARA studies continue. The estimates are based on conservative assumptions.

(3) The manhour estimates for placement of shielding and local decontamination are tentative due to the continuation of ALARA analysis and work package development. The numbers presented here are an estimate and represent a percentage of the total man-rem.

PALISADES PLANT SGRR

C-2 Describe the designated contamination control envelopes and your plan to maintain occupational exposure within these envelopes "ALARA." Include also dose rates, exposure times, and numbers of workers involved in the tasks (Section 4.3.3).

RESPONSE:

The contamination control envelopes will be used for the cutting of reactor coolant piping (Task 4). Although the design of the envelopes has not been finalized, the envelopes will include a high efficiency filtration system. The flow of air within the envelopes will preclude the escape of contaminants through the tent openings used for entering and exiting the area.

Each of the 12 cuts on primary coolant piping will require two workers approximately 15 manhours. It is estimated that 720 man-hours will be spent within the control envelopes, with an average radiation field of 30-50 mr/hr. This results in an estimated 21-36 man-rem of occupational exposure.

As described in the Repair Report, Section 4.3-1, personnel involved in work within areas with a high level of contamination will wear two sets of protective clothing. Respiratory protection will be required in accordance with Palisades health physics procedures. Sheet lead, lead wool blanket, or other shielding will be used where possible in accordance with ALARA guidelines.

PALISADES PLANT SGRR

C-3 Provide a diagram showing the radiation surveys around the steam generator replacement/repair activities. Include similar radiation surveys for Figures 4.2-4 through 4.2-7. Include a table showing the whole body dose received during the inspection and plugging of the degraded steam generator tubes for 1976, 1977, and 1978.

RESPONSE:

1. Figures 4.3-3, 4.3-4, and 4.3-5, which are included in the Steam Generator Repair Report, show radiation surveys around the steam generators at various times after shutdown. The fields specified are representative of those expected for the replacement repair activities. It should be noted that the fields specified in Figure 4.3-5 are expected to decrease significantly at the time that pipe cutting begins, considered to be 42-140 days post-shutdown for study purposes. The decrease will follow the general radiation field near the steam generator piping shown in Figure 4.3-7. Figures 4.2-4 through 4.2-7 have been modified to include general field information in areas where replacement/repair activities will occur and are now designated as Figures C-3-1 through C-3-4.

2. As requested, Tables C-3-1 and C-3-2 show the whole body dose received for inspection and plugging steam generator tubes during 1976 and 1978, respectively.

TABLE C-3-1
PALISADES PLANT - RADIATION DOSE SUMMARY
STEAM GENERATOR WORK 1976

Activity	1976 Exposure	
1. ECT Personnel		
Inside steam generator (without shielding)	16.5R	
Outside steam generator	19.3R	
Total (received over a period of 21 days)		35.8R
2. Insert and remove shielding		
Inside steam generator	17.8R	
Outside steam generator	2.0R	
Total		19.8R
3. Insert templates		
Inside steam generator	25.3R	
Outside steam generator	2.8R	
total		28.1R
4. Brushing and rolling		
Inside steam generator	37.6R	
Outside steam generator	4.3R	
Total		41.9R

PALISADES PLANT SGRR

Activity	1976 Exposure	
5. Insert plugs		
Inside steam generator	28.8R	
Outside steam generator	2.9R	
Total		31.7R
6. Weld plugs		
Inside steam generator	54.4R	
Outside steam generator	15.3R	
Total		69.7R
7. QC inspection of above operation		
Inside steam generator	23.6R	
Outside steam generator	1.6R	
Total		25.2R
8. Engineers support of above operations		10.7R
Exposure accumulated inside steam generator	204.0R	
Exposure accumulated outside steam generator	48.2R	
TOTAL ACCUMULATED EXPOSURE		262.9R

NOTE: The exposure designated inside and outside steam generator are only estimates (although the total for each operation is accurate), since inside steam generator data is extracted from high radiation dose summary sheets which also include some outside steam generator work. The net result is that the inside steam generator data reads slightly higher and the outside steam generator data reads slightly lower than actually occurred. Radiation exposure to engineers is not included in this breakdown.

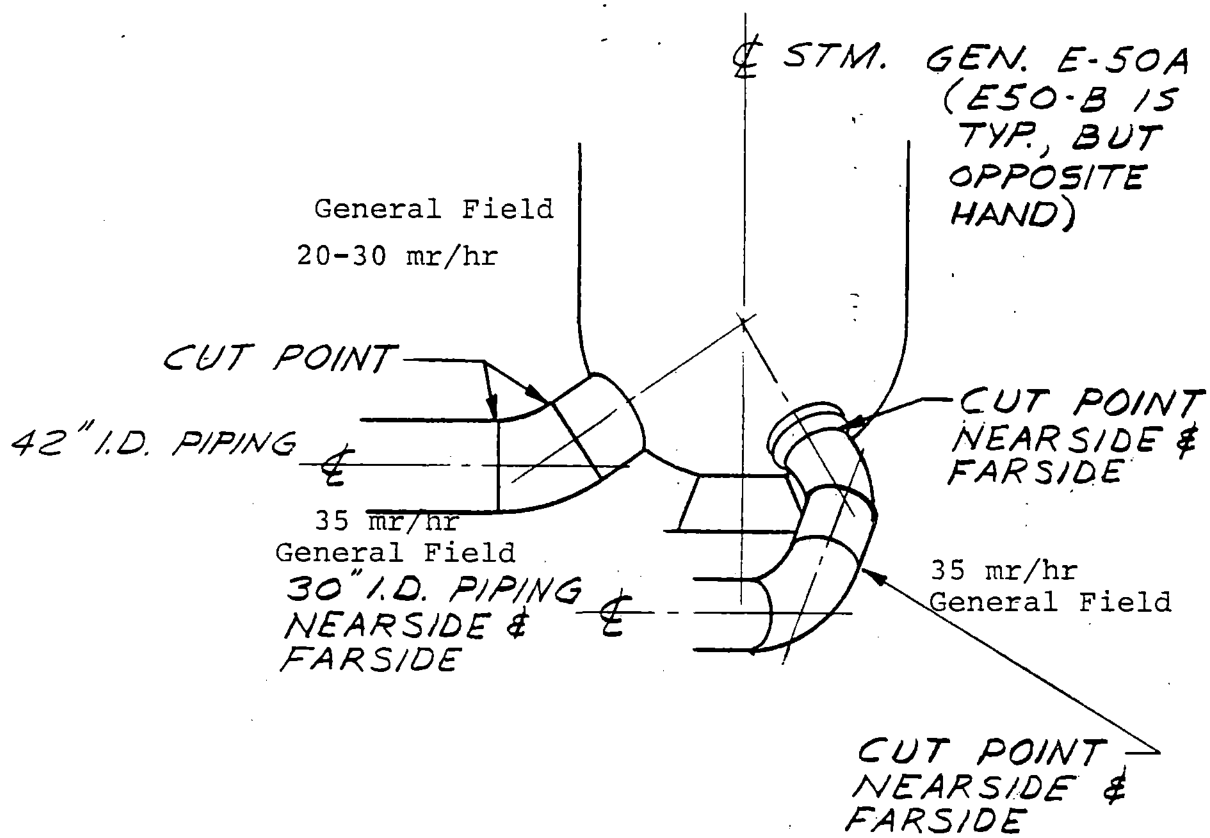
TABLE C-3-2
PALISADES PLANT - RADIATION DOSE SUMMARY
STEAM GENERATOR WORK 1978

Organization/Activity	*1978 Exposure (Level 9 R/hr 3.5 R/hr)
1. Consumers Power Company Repairmen	
a. Manway cover removal/vacuuming	5.6
b. Dam installation	45.0
c. Surge line shielding + miscellaneous	5.5
d. Flooring/RSS hot legs	8.5
e. Flooring, struts, tracks, bridge	
A cold leg	14.3
B cold leg	22.5

PALISADES PLANT SGRR

Activity			*1978 Exposure
	f. Tracks, bridge		
		A hot leg	6.1
		B hot leg	6.3
	g. General maintenance		1.9
	h. Tube plugging		
		A cold leg	3.1
		A hot leg	1.6
		B cold leg	3.9
		B hot leg	1.8
	i. Equipment removal/leg cleanup		6.8
	j. Dam removal		4.6
	k. Manway cover replacement		4.9
	l. Cont. cleanup		2.6
			145.0
2.	HP coverage		6.0
3.	NDT lab/contractors		44.3
4.	CE - Setup		2.4
	Deplugging	A cold leg	7.4
		A hot leg	4.3
	Sleeving		14.9
			29.0
		Total exposure	*224.3 Man-rem

NOTE: *These exposures are based on dosimeters and TLDs. Source is containment entry logs. These numbers should only be considered close approximations.



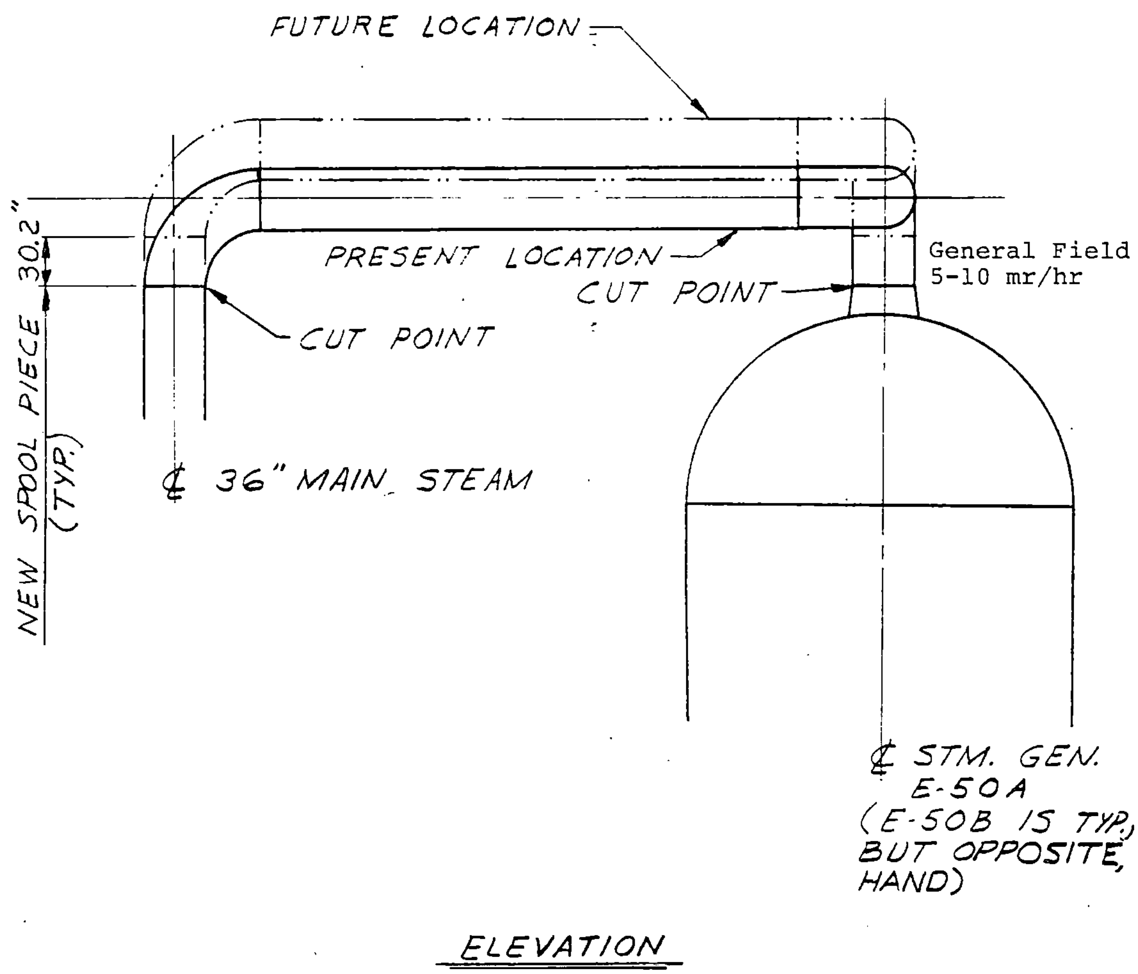
ELEVATION

PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

PRIMARY COOLANT PIPING CUT POINTS

Figure C-3-1

Revision 3
July 1979

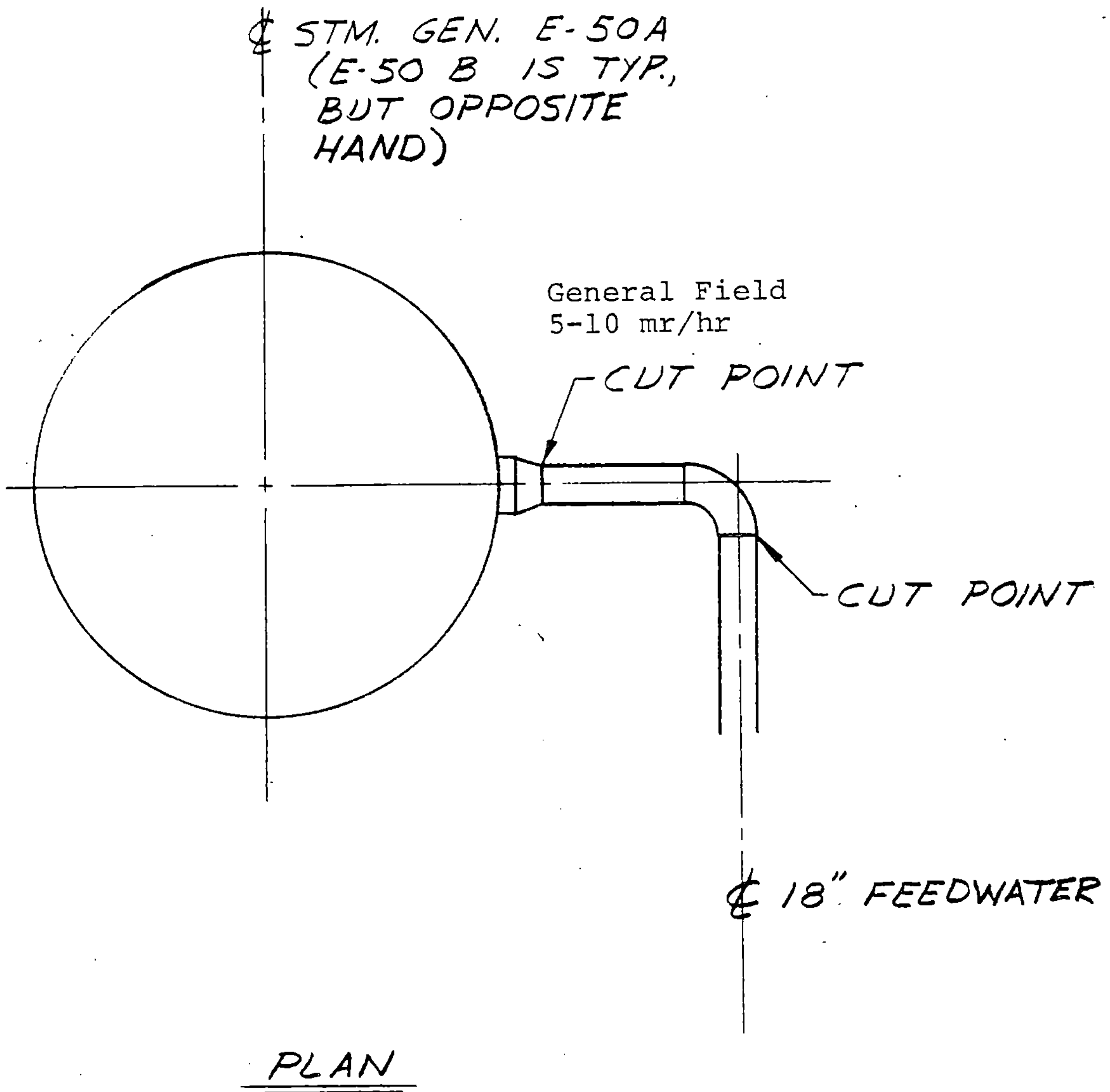


PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

MAIN STEAM PIPING
CUT POINTS

Figure C-3-2

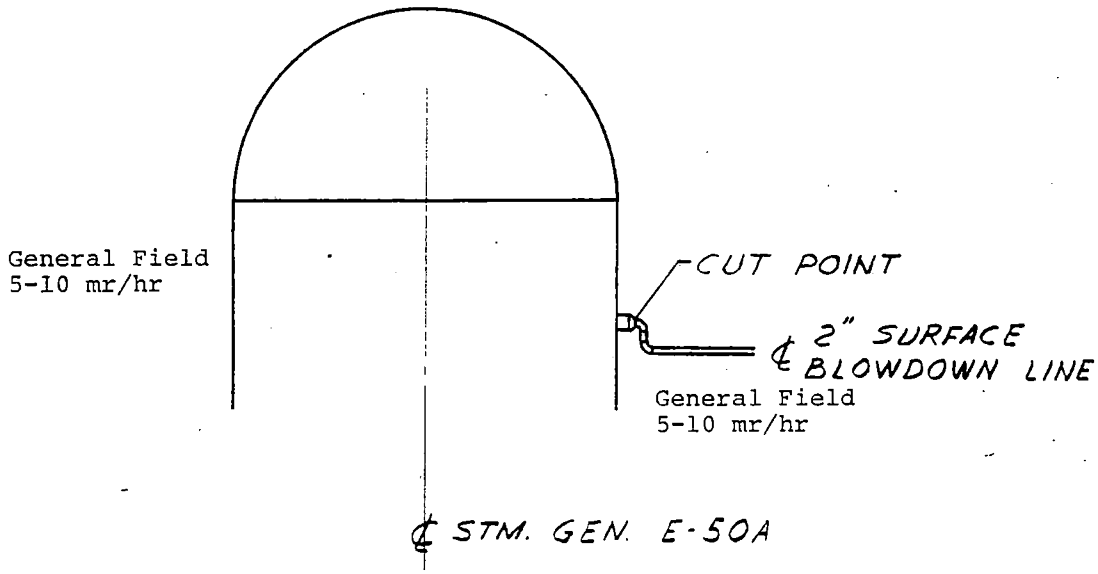
Revision 3
July 1979



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

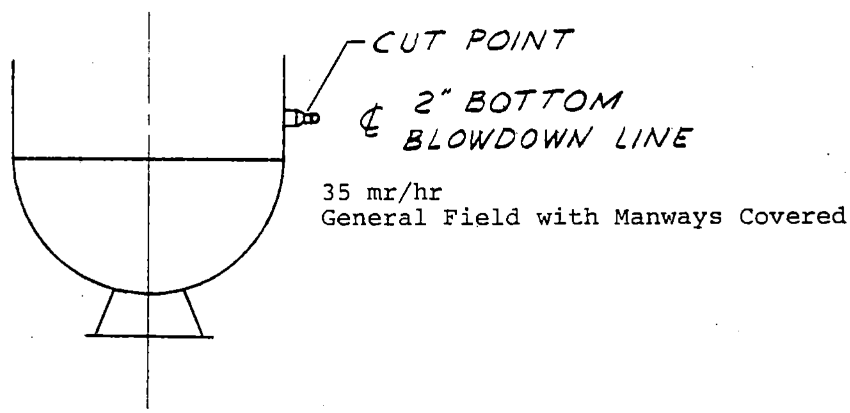
FEEDWATER PIPING
CUT POINTS
Figure C-3-3

Revision 3
July 1979



ELEVATION

STM. GEN. E-50B IS TYP., BUT OPPOSITE HAND



ELEVATION
ROTATED 90° CLOCKWISE

PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

BLOWDOWN PIPING
CUT POINTS

Figure C-3-4

Revision 3
July 1979

PALISADES PLANT SGRR

C-4 Discuss briefly how you would avoid imbalance of the permanent ventilation systems due to the additional construction -related ventilation equipment (portable fans, hoods and filters, etc).

RESPONSE:

Additional construction - related ventilation equipment will be used to supplement the permanent ventilation system and will remove fumes associated with welding and cutting operation as well as ventilating temporary enclosures. Significant imbalance of the permanent ventilation system is not expected since the construction related ventilation equipment will exhaust inside the containment after filtration, and/or have relatively low flowrate. However, should imbalance of the permanent ventilation system occur, balance can be restored by controlling the dampers on the permanent ventilation system.

PALISADES PLANT SGRR

C-5 Your description of compliance with Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Exposure at Nuclear Power Stations Will Be As Low As Reasonably Achievable", Revision 3, June 1978, states that the following considerations were not implemented:

1. Radiation zones in the containment work areas, identifying the exposure levels in each work zone. C.2.(a)

2. Streaming or scattering of radiation from installed shielding, such as plugs in open ended pipe lines following cutting. C.2.b.(4)

3. Outleakage of airborne contamination from the containment due to steam generator replacement/repair activities when the equipment hatch is open.

4. Operating experiences should be recorded, evaluated, and reflected in the selection of replacement instrumentation. C.2.(c)(3)

5. Provision to be implemented to minimize exposure of station personnel in performing code inspection, such as removable insulation, smooth welds, etc. (C.2.(i)(11)

6. An adequate emergency lighting system can reduce potential exposures of station personnel by permitting prompt egress from high radiation areas if the station lighting system fails.

7. A staff member who is a specialist in radiation protection assigned the responsibility for contributing to and coordinating ALARA efforts in support of operation that could result in substantial individual and collective dose levels.

8. ...Station work areas to limit the average concentration of radioactive material in air to levels below... in Appendix B, Table 1, Column of 10 CFR Part 20. C.2.d

Provide justification for not implementing these provisions of Regulatory Guide 8.8, and demonstrate that alternative precautions you have taken will provide comparable levels of protection.

PALISADES PLANT SGRR

RESPONSE:

The philosophy of the radiation protection group is to maintain the occupational dose to all personnel as low as is reasonably achievable (ALARA). This has been stated as a policy in the Palisades Plant Radiation Protection Manual. In order to ensure that the various provisions of Regulatory Guide 8.8 are evaluated adequately, a third party ALARA review of the repair effort has been utilized. The reviewers' responsibilities are to develop ALARA "checklists" to be used in conjunction with the work package descriptions for each of the repair tasks. The "checklists" will evaluate and recommend any measure found appropriate to maintaining the personnel exposure ALARA as defined in Regulatory Guide 8.8 (Revision 3).

Although the detailed work packages are presently in the development stage, each of the following specific provisions will necessarily be given complete consideration for use during any replacement/repair activity:

1. The identification of exposure levels in any work zone is presently implemented per Palisades Plant health physics procedures. The Repair Report Section 4.3.5.4 describes the low background radiation waiting areas that will be used near each containment work area. As specified, special signs, tape, or rope-off areas will be utilized to designate these zones. Intermediate zones could be utilized, if found ALARA, as individual work packages are developed.

2. The effective streaming or scattering of radiation from installed shielding, such as plugs in open ended pipe lines, can be minimized through the use of local decontaminating of pipe stubs, use of temporary shielding, or exposure control by controlling ingress or egress to work areas. (See Repair Report Sections 4.3.5.2 and 4.3.5.3). All of these ALARA techniques are being evaluated for use with the individual work packages per third party ALARA review described above.

PALISADES PLANT SGRR

3. In addition to the temporary enclosure on the construction opening, airborne radioactivity inside containment during the steam generator repair effort will be controlled, monitored, and ultimately released via the plant vent stack. The air will be drawn through the hatches and construction opening and exhausted by the purge system via the plant ventilation stack, thus precluding airborne radioactive particles or gases from leaving containment openings utilized for construction activities. (See Repair Report Section 4.3.3).

4. Continuous air monitors, area radiation monitors, and portable survey instruments will be used in accordance with Palisades Plant health physics procedures. Daily and weekly operational checks, calibration, and response settings will be implemented and recorded as required per Palisades Plant health physics procedures.

5. Appropriate provisions will be implemented to minimize exposure of station personnel in performing code inspection, such as removable insulation, smooth welds, etc. Design features to improve maintenance and inspection are discussed in the Repair Report Section 2.2.2.

6. An emergency lighting system will be available for the steam generator replacement activities.

7. A staff member who is a specialist in radiation protection will be assigned to the responsibility for coordinating ALARA efforts.

8. Special measures will be implemented to minimize and control the average concentration of radioactive material in air to below those specified in Appendix B, Table 1, Column 1 of 10 CFR Part 29. In addition to temporary enclosures in areas where cutting will occur, containment air will be conditioned for the removal of airborne radioactivity by use of filters.

PALISADES PLANT SGRR

C-6 Explain what steps you plan to take to help maintain doses ALARA in this project. Indicate what use will be made of contaminations tents, lead wool blankets, gloveboxes, remote cutting and welding equipment, temporary shielding and ventilation systems. Also indicate what equipment will be mocked-up for training purposes.

RESPONSE:

An independent ALARA review of the steam generator repair effort has been utilized to make recommendations for maintaining doses ALARA (see question C-5). The ALARA recommendations made will be incorporated into the various work packages. Each of the following items will be used where appropriate to maintaining doses ALARA.

1. Contamination Tents - The cutting of primary coolant piping will be contained within specially designed contamination control envelopes. The envelopes will be provided with high efficiency filtration.

2. Temporary shielding in the form of; lead wool blankets, lead shield plugs and sheet lead will be used where effective to maintaining doses ALARA. Experience has shown that lead wool blankets can effectively reduce streaming around shield plug and sheet lead fittings.

3. Gloveboxes - As yet, there are no repair/replacement activities described, which can effectively use glovebox enclosures for maintaining doses ALARA. As work procedures develop, glovebox techniques will remain a viable option.

4. Remote cutting and welding equipment - Automatic welding machines with remote viewing will be used for welding operations made to interior located stainless steel cladding. Measuring equipment for determining the location of reactor vessel and steam generator during weld-up and cutting operations will utilize remote indicators.

   Although the cutting techniques have not been finalized, the ALARA considerations will be one of the determining factors for final selection.

PALISAGES PLANT SGRR

5. Ventilation Systems - Local construction related ventilation equipment will be used to supplement the permanent ventilation system and will remove fumes associated with welding and cutting operation as well as controlling the airborne contamination existing in the temporary enclosures.

6. Mock-Ups of reactor coolant pipe, and the steam generator primary head as well as the actual cutting and welding equipment will be utilized for training and work planning.

PALISADES PLANT SGRR

C-7 Provide a table showing the occupational collective whole body dose estimates for the following phases of the steam generator replacement repair acitivities: (1) preparation, (2) removal, (3) installation and (4) storage.

Discuss briefly your procedure for calculating these doses, taking into account the dose reduction measures proposed to maintain doses as low as reasonably achievable (ALARA), including local decontamination, temporary lead shielding, pre-job planning, pre-job training and use of remote tools where practicable.

RESPONSE:

Table C-7-1 provides the requested information. The table groups the various tasks into preparation, removal, installation, and storage phases of the replacement/repair effort. The man-rem is the product of (man-hours) X (average radiation field) for each task. It should be noted that each task consists of man-hours accumulated in several locations, and each location has a corresponding radiation field. A description of locations and average radiation fields is presented in Table C-1-5, credit taken for decontamination and temporary shielding is incorporated into the radiation field estimates. Reduction factors for shielding and/or decontamination are presented in the repair report, Table 4.3.2.

PALISADES PLANT SGRR
TABLE C-7-1
MAN-REM ESTIMATE
(Preparation, Installation, Removal, and Storage)

Preparation			Removal			Installation			Storage		
Task No.	Mils	Man-Rem	Task No.	Mils	Man-Rem	Task No.	Mils	Man-Rem	Task No.	Mils	Man-Rem
1	1,578	2.32	1	1,416	31.26	2	1,993	2.90	46	620	12.4
3	4,382	11.58	13	15	27.36	5	4,176	111.69	55	4,060	0.02
34	140	2.60	15	120	26.52	6	3,360	89.85			
35	21,500	0.11	16	324	79.38	7	6,008	32.13			
37	11,606	0.06	19	1,016	10.72	8	648	2.88			
38	8,304	0.01	21	115	3.02	9	8,224	44.88			
39	1,500	0.01	22	476	.912	10	1,184	5.28			
40	1,500	0.01	24	577	1.91	11	3,096	19.08			
41	744	0.01	26	240	4.03	12	1,944	13.44			
42	936	0.01	27	300	1.00	14	12	.42			
43	936	0.01	31	550	8.68	17	96	.78			
44	5,625	17.62	45	3,360	52.95	18	180	1.53			
47	100	0.01	56	80	.34	20	4,800	9.60			
48	3,160	0.01	57	48	.24	21	269	1.00			
49	480	0.01	58	320	1.30	22	4,762	14.27			
			59	48	.79	23	520	1.16			
(1)	(1)	165.28	60	180	3.02	24	1,579	3.33			
						25	260	16.18			
						26	560	5.37			
						27	700	1.25			
						28	381	.80			
						29	2,424	15.00			
						30	3,858	8.67			
						32	2,400	10.8			
						33	1,236	3.13			
						34	130	2.6			
						36	19,125	0.09			
						50	2,670	10.72			
						51	3,375	.01			
						52	3,750	11.73			
						53	945	0.01			
						54	750	0.01			
						61	120	0.81			
						62	600	4.05			
						63	1,440	3.24			
						64	480	1.08			
						65	1,920	4.32			
Subtotal	62,491	199.66		9,185	251.43		89,980	454.09		4,680	12.42
66	25,058	75.82	66	1,584	4.67	66	20,126	60.94			
Total Directs	87,549	275.48		10,769	258.10		110,106	515.03		4,680	12.42
Distributables 67	22,597	124		2,779	15.28		28,416	156.3		1,208	6.64
Nonmanual 68	49,301	73.86		6,064	8.98		62,000	92.33		2,635	3.95
Total	159.447	473.34		19,612	282.36		200,522	763.66		8,523	23.01

(1) Decontamination/Shielding Installation

PALISADES PLANT SGRR

C-8 Discuss your cutting and welding operations and cleanup of surface contamination in respect to "ALARA" guidelines (Section 4.3.3).

RESPONSE:

We note that a definite schedule has not been set for the cutting and welding, that experience is being gained by way of similar operations at other plants, and that detailed work plans have not been completed. The final operations will reflect applicable experience, and ALARA considerations. The following is an outline of operations under consideration at this time.

1. <u>Cutting of Reactor Coolant Pipe</u>

   Since the Palisades reactor coolant pipe is carbon steel w/stainless steel cladding, a plan is to utilize a track mounted oxygen-acetylene torch to cut the pipe. Consideration is also being given to mechanically cutting the pipe in order to minimize the pipe lost during the cutting process and to facilitate the machining operation. The cutting operation will be accomplished in an enclosure to limit spread of contamination.

2. <u>Handling of Pipe to Decontamination Area</u>

   After the pipe has been cut, temporary shield plugs will be secured to each open end of the reactor coolant pipe and the short pieces of pipe moved to a decontamination area. The temporary shield plugs will be mechanically attached to the pipe in order to reduce the number of welding and cutting operations.

3. <u>Field Machining of the RC Pipe Attached to Reactor and Reactor Coolant Pumps</u>

   Temporary shield plugs will be inserted a short distance into the pipe. The pipe between the shield plug and the end of the pipe will then be decontaminated to the extent practical. The pipe weld preparations would then be field machined.

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4. Field Machining of the Short Pieces of Reactor Coolant Pipe

After the new steam generator has been placed, the dimensions between the existing reactor coolant pipe and the new steam generator nozzles will be transferred to the short pieces of reactor coolant pipe and the pipe weld preparations machined.

5. Welding of RC Pipe

After set-up of the reactor coolant pipe the joints may be welded by one of the following methods:

1. Manually weld the carbon steel portion of the reactor coolant pipe and utilize an automatic welding machine with remote viewing for welding the stainless steel interior cladding.

2. Utilize an automatic welding machine with remote viewing to weld both the carbon steel and stainless steel interior cladding.

6. Clean-up of Surface Contamination

Loose surface contamination will be removed manually from the outside of reactor coolant pipe pieces, prior to cutting operations and again, prior to removal from contamination control envelopes. Plastic or other impervious sheeting will be used to cover pipe pieces before relocating to decontamination area.

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C-9 Your estimated dose range of 1,200 to 5,000 man-rem for the steam generator replacement/repair activities is too wide to assure that occupational exposure will be as low as reasonably practicable. Justify the high end of the range as being ALARA. Experiences with other designs indicate the feasibility of performing such work with substantially lower total doses than the high end of range you have predicted.

RESPONSE:

As presented in the task description (Question C-1), there are two welding techniques being considered for reactor coolant piping. One technique utilizes manual welding of reactor coolant carbon steel pipe with a machine weld-up of the stainless steel cladding. This technique would result in an estimated 1,547-2,808 man-rem for the repair. The second technique utilizes a machine weld-up of both carbon steel pipe and stainless steel cladding. This alternative results in an estimated 1,537-2,663 man-rem for the repair effort.

The high end of the range presented in Table 4.3.2, 5,000 man-rem, resulted from a manual weld-up of carbon steel pipe and cladding entirely from the inside. Due to the technical feasibility of the two welding techniques described above, this third alternative is no longer considered ALARA and has since been eliminated.

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C-10 Provide a rough breakdown of the activities, person-hour occupancies, and projected dose rates which are used in deriving the estimated total of 40,250 man-rems per unit for retubing in place (Section 8.7).

RESPONSE:

Refer to Appendix A of the SGRR response to Question A-2, as transmitted to the USNRC by the Consumer Power Company letter dated June 11, 1979. That analysis is currently under reevaluation.