TROJAN NUCLEAR PLANT

Defueled Safety Analysis Report, Revision 4

The following information is furnished as a guide for inserting new pages for Revision 4 into the Trojan Nuclear Plant Defueled Safety Analysis Report. This changes are denoted by the use of revision bars in the margin and revision number in the lower outside corner of the page.

Revised pages should be inserted as listed below:

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1.2 GENERAL PLANT DESCRIPTION

The Trojan Nuclear Plant site consists of approximately 623 acres located in Columbia County in NW Oregon on the Columbia River at River Mile 72.5 from the mouth. The distance from the reactor site to the nearest site boundary on land is 2172 ft. Major structures on the site include the Containment, Turbine Building, Auxiliary Building, Fuel Building, Control Building, and a single natural draft cooling tower.

The town of St. Helens, Oregon, the county seat of Columbia County, is located approximately 12 miles SSE of the site. The town of Rainier, Oregon, is located approximately 4 miles NNW and the town of Kalama, Washington, is approximately 3 miles SE of the site. There are three small unincorporated communities within a 5-mile radius of the site: Prescott, Oregon, located approximately 1/2 mile N of the site; Goble, Oregon, located approximately 1-1/2 miles SSE of the site; and Carrolls, Washington, located approximately 2-1/2 miles NNE of the site.

1.2.1 DESIGN CRITERIA

The principal design criteria for the Trojan Nuclear Plant are those fundamental architectural and engineering design objectives established for the Facility. The basis for development and selection of the design criteria used in this Facility were those which: (a) provide protection of public health and safety, (b) provide for reliable and economic Facility performance, and (c) provide an attractive appearance.

The essential systems and components of the Facility are designed to enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena. The designs are based on the most severe of the natural phenomena recorded for the vicinity of the site, with margin to account for uncertainties in historical data.

1.2.2 FUEL HANDLING SYSTEM

The Facility was designed to handle spent fuel under water from the time it leaves the reactor vessel until it is placed in casks for shipment from the site, although PGE is prohibited from moving fuel into the containment. Underwater transfer of spent fuel provides an optically transparent radiation shield, as well as a reliable source of coolant for removal of decay heat.

1.2.3 RADIOACTIVE WASTE TREATMENT SYSTEMS

The radioactive waste treatment systems provide equipment necessary to collect, process, monitor, and discharge radioactive liquid, gaseous and solid wastes that are produced at the Facility.

Liquid wastes potentially containing radioactive material are collected and monitored. Prior to discharge, equipment is provided for filtering and demineralizing the liquid as required. The treated water from the filters or demineralizers may be recycled for use in the Facility or may be discharged to the Columbia River. The miscellanecus dry waste, spent demineralizer resins and spent filters are shipped from the site for ultimate disposal in an authorized location.

Gaseous wastes are collected and discharged to the environment after filtration to keep the offsite dose within prescribed limits.

1.4 EXCLUSIONS FROM AND EXEMPTIONS TO CERTAIN PARTS OF TITLE 10 OF THE CODE OF FEDERAL REGULATIONS (10 CFR)

1.4.1 EXCLUSIONS FROM CERTAIN PARTS OF 10 CFR

In a letter dated March 15, 1993 from J. E. Cross, Portland General Electric (PGE) notified the Nuclear Regulatory Commission that on receipt of the Possession Only License (POL), the following Parts of Title 10 of the Code of Federal Regulations (10 CFR) are no longer applicable to the Trojan Nuclear Plant (TNP):

1.4.1.1 10 CFR 26, Fitness for Duty Program

10 CFR 26 is applicable to licensees authorized to operate a nuclear power reactor. Since approval of the POL, PGE has not been authorized to operate the TNP as a nuclear power reactor. Although this regulation is not applicable to TNP with a POL, PGE has committed via letter dated January 27, 1993 to the NRC to maintain the current Fitness for Duty Program at TNP after receipt of POL.

1.4.1.2 10 CFR 50.44, Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors

This regulation is applicable to fueled nuclear power reactors. The TNP reactor has been defueled permanently. Thus, this regulation is not applicable to TNP. This was codified by the Final Decommissioning Rule which became effective August 28, 1996.

1.4.1.3 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors

This regulation is applicable to fueled nuclear power reactors. The TNP reactor has been defueled permanently. Thus, this regulation is not applicable to TNP. This was codified by the Final Decommissioning Rule which became effective August 28, 1996.

1.4.1.4 10 CFR 50.48, Fire Protection

This regulation applies to operating nuclear power plants. TNP has been defueled and is not authorized under the POL to operate as a nuclear power plant again. Thus, this regulation is not applicable to TNP. The Final Decommissioning Rule, which became effective August 28, 1996, modified this rule to make it applicable to plants that are permanently shut down, and altered the rule to recognize the limited threat of fire for shutdown plants. The modified final rule is, therefore, now applicable to Trojan.

1.4.1.5 10 CFR 50, Appendix R, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979

This regulation applies to operating nuclear power plants. TNP has been defueled and is not authorized under the POL to operate as a nuclear power plant again. Thus, this regulation is not applicable to TNP.

1.4.1.6 10 CFR 50.49, Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants

This regulation applies to operating nuclear power plants. TNP has been defueled and is not authorized under the POL to operate as a nuclear power plant again. Thus, this regulation is not applicable to TNP. This was codified by the Final Decommissioning Rule which became effective August 28, 1996.

1.4.1.7 10 CFR 50.55a, In-Service Inspection Requirements

10 CFR 50.55a Paragraphs (f) and (g) specify requirements for in-service testing of pumps and valves and in-service inspection of components (including supports), respectively. 10 CFR 50.55a Paragraphs (f)(4) and (g)(4) indicate that Paragraphs (f) and (g) of the rule are applicable throughout the service life of the plant. Since TNP is no longer in service, the regulation is not applicable.

1.4.1.8 10 CFR 50.60, Acceptance Criteria for Fracture Prevention Measures for Light-Water Nuclear Power Reactors

10 CFR 50.60 requires licensees to meet the requirements of 10 CFR Appendices G and H. Appendices G and H are not applicable to TNP since receipt of POL. This was codified by the Final Decommissioning Rule which became effective August 28, 1996.

1.4.1.9 10 CFR 50.61, Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events

This regulation applies to operating nuclear power plants. TNP has been defueled and will not be authorized under the POL to operate as a nuclear power plant again. Thus, this regulation is not applicable to TNP. This was codified by the Final Decommissioning Rule which became effective August 28, 1996.

1.4.1.10 10 CFR 50, Appendix G, Fracture Toughness Requirements

10 CFR Appendix G specifies fracture toughness requirements of reactor coolant pressure boundary to provide adequate margin of safety during any condition of normal operation over its service lifetime. The TNP Reactor Coolant System is no longer operating and has completed its service life. Thus, this regulation is no longer applicable to TNP.

1.4.1.11 10 CFR 50, Appendix H, Reactor Vessel Material Surveillance Program Requirements

10 CFR Appendix H provides the requirements of a reactor vessel material surveillance program. The data obtained from this program are utilized as required by Sections IV and V of Appendix G. Appendix G is no longer applicable to TNP and thus Appendix H is also no longer applicable.

1.4.1.12 10 CFR 50.62, Requirements for Reduction of Risk From Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants

The ATWS mitigation system for a pressurized water reactor is intended to actuate auxiliary feedwater and initiate a turbine trip under conditions indicative of an ATWS. The issuance of the POL precludes power operation at Trojan, and thus this regulation is not applicable to TNP. This was codified by the Final Decommissioning Rule which became effective August 28, 1996.

1.4.1.13 10 CFR 50.63, Loss of All Alternating-Current Power

This regulation applies to operating nuclear power plants, ensuring that the reactor core is cooled and containment integrity is maintained in the event of a loss of all alternating-current power. The reactor core at TNP has been offloaded to the spent fuel pool and PGE is not authorized under the POL to operate Trojan as a nuclear power plant again. Thus, this regulation is not applicable to TNP.

1.4.1.14 10 CFR 50.71(e), Maintenance of Record, Making of Reports

This regulation applies to operating nuclear power plants. TNP has been defueled and is not authorized under the POL to operate as a nuclear power plant again. Thus, this regulation is not applicable to TNP. The Final Decommissioning Rule, which became effective August 28, 1996, modified this rule to make it applicable to plants that are permanently shut down, and altered the rule to recognize the limited safety importance of Safety Analysis Updates for shutdown plants. The modified final rule is, therefore, now applicable to Trojan.

1.4.1.15 10 CFR 70.24, Criticality Accident Requirements

By letter dated February 16, 1993 from S. H. Weiss (NRC) to J. E. Cross (PGE), the staff has notified Trojan that with the absence of new fuel, the design of the storage racks, and the procedural controls over fuel handling activities with Certified Fuel Handler training, adequate assurance against the occurrence of an accidental criticality
exists that this regulation no longer applies to Trojan in its permanently shutdown and defueled condition.

1.4.2 EXEMPTIONS TO 10 CFR RELATED TO THE PERMANENTLY DEFUELED CONDITION

There are also certain 10 CFR regulations to which the NRC has granted specific exemptions to PGE for the op[illegible]tion of the Trojan Facility.

1.4.2.1 10 CFR 50.54(o) and Appendix J, Primary Reactor Containment Leal[illegible] Testing for Water-Cooled Power Reactors

By letter dated April 12, 1993 from M. T. Masnik (NRC) to J. E. Cross (PGE) the NRC will permit PGE to cease all testing required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J.

1.4.2.2 10 CFR 50.54(y), Conditions of Licenses

By letter dated June 23, 1993 from M. T. Masnik (NRC) to J. E. Cross (PGE) the NRC will permit PGE to depart from a license condition or technical specification in an emergency as described in 10 CFR 50.54(y) provided that the emergency action is approved, as a minimum, by a Certified Fuel Handler prior to taking the action.

1.4.2.3 10 CFR 50.54(q) and Certain Sections of 10 CFR 50.47, "Emergency Plans," and 10 CFR 50, Appendix E, "Content of Emergency Plans"

By letter dated September 30, 1993, from M. T. Masnik (NRC) to J. E. Cross (PGE), the NRC authorized the Trojan facility to discontinue offsite emergency preparedness activities and reduce the scope of onsite planning.

1.5 MATERIAL INCORPORATED BY REFERENCE

Certain program manuals and topical reports have been incorporated into the DSAR by reference and are listed in the last section of each chapter. The reports include topical reports written by PGE as well as by Westinghouse, Bechtel and other organizations.

Some documents that are incorporated by reference continue to be updated to assure that the information presented is the latest available. These documents include those listed below:

(1) PGE-1060, "Permanently Defueled Emergency Plan."

(2) PGE-1012, "Fire Protection Plan."

(3) PGE-1017, "Security Plan."

(4) PGE-1020, "Report on Design Modifications for the Trojan Control Building."

(5) PGE-1037, "Trojan Nuclear Plant Spent Fuel Storage Rack Replacement Report."

(6) PGE-8010, "Nuclear Quality Assurance Program."

(7) PGE-1052, "Quality-Related List Classification Criteria for the Trojan Nuclear Plant."

(8) PGE-1021, "Offsite Dose Calculation Manual."

(9) PGE-1057, "Trojan Nuclear Plant Certified Fuel Handler Training Program."

(10) PGE-1024, "Trojan Nuclear Plant Security Force Training and Qualification Plan (Defueled Condition)."

(11) PGE-1061, "Decommissioning Plan."

(12) PGE-1063, "Supplement to Applicant's Environmental Report - Post Operating License Stage."

field are a DeHavilland 8 corporate plane, a Siddely Hawker, a Cessna Citation, and a Falcon Jet[13]. The Portland International Airport is located 33 statute miles south of the site, and is the only major airport within a 60-mile radius of the site. Portland inbound and outbound air traffic is controlled for a distance of 30 miles from the airport by Portland Air Traffic Control. Area-wide inflight traffic control is regulated by Seattle Air Traffic Control[14].

2.2.3 EVALUATION OF POTENTIAL ACCIDENTS

This section provides an evaluation of the capability of the Trojan Nuclear Plant to safely withstand the effects of an accident at, or as a result of the presence of, industrial, transportation and military installations or operations within 5 miles of the site. Potential accidents considered include impacts to and blockage of the cooling water intake structure; explosions of chemicals, flammable (including natural) gases or munitions; industrial and forest fires; accidental releases of toxic gases; accidental releases of corrosive liquids or oil into the Columbia River upstream of the cooling water intake structure; and collapse of the cooling tower.

2.2.3.1 Explosions

Shipments of commercial cargo past the Plant create the possibility of nearby explosions. Safety-related portions of the Plant are protected from such explosions.

Explosions unrelated to transportation are not considered significant. The quarry operations south of the site are located in the hills west of the Columbia River. Presently, there is no storage of explosives at the operating quarry which is 2 miles from the site. The quarry is not a large operation and only a limited amount of explosives are used. Because of the distance from the Plant and the protection afforded by the hillside and ridge between the quarry and the Plant, the quarry operation does not present a hazard to the safety of the Plant. A natural gas main extending to Wauna, Oregon,

downriver of the site runs along the hillside west of the site, approximately 1-1/2 miles from the site. The main is a 16 inch, 3-million foot/hour line, buried a minimum of 3 feet[10]. In addition, there is an odorizer station on the line at Goble, 2 miles south of the site, a river crossing at Deer Island, 4-1/2-miles south of the site and a river crossing at Rainier, 5-miles north of the site. The operation of these lines will not present a hazard to Trojan from explosion because of the relatively low explosive capacity of the gas and the distance from the Plant proper.

Even though there is some evidence that transportation accident rates involving hazardous cargo are below the overall average because of precautionary measures taken with such shipments[15], no credit is taken for this factor.

On the other hand, it may be assumed that the rate of accidents caused by the nature of the hazardous material itself (explosion of cargo not caused by traffic accident) is comparatively low and may be neglected. However, it is realized that the transportation accidents down to a low degree of severity must be considered here as the explosion may be triggered by a minor cause (shock, spark, heat).

The effect of a transportation accident involving explosive cargo is based on the assumption that an explosion does occur.

The major distance effects of a surface explosion are:

(1) Atmospheric effects (shock wave and momentum effects).

(2) Ground effects (direct and air-induced ground shock).

(3) Other effects (missiles, waves, noxious fumes or incendiary effects).

Studies of these effects have led to the conclusion that the atmospheric shock wave is the severest effect (most limiting) with regard to safety-related structures of power plants[16,17].

The value of the limiting (maximum permissible) overpressure generated by the atmospheric shock from the explosion is 2.2 psi (p_c = 2.2 psi). This limit is chosen in recognition of the fact that the actual overpressure experienced by parts of the structure may be up to twice this value (4.4 psi) due to reflected wave effects[18].

It is noted that test structures generally similar to those employed for safety-related structures at Trojan have demonstrated that overpressures of 5 psi can be endured without any significant damage to these structures[19]. Thus, it appears that the 2.2 psi overpressure used in this analysis incorporates substantial conservatism. Furthermore, it is noted that safety-related structures are designed to withstand pressure differentials of at least 3 psi due to tornado effects.

A missile caused by an explosion at a distance greater than that at which 2.2 psi would occur would not be as severe as the assumed tornado missiles. If the range were less than that corresponding to 2.2 psi, the blast damage would be more critical than the missile damage. Missiles from the cooling tower failure are considered comparable to the tornado missiles. If the probabilities of the explosion occurring within the range corresponding to 2.2 psi are shown to be remote, the effects of the explosion can be neglected.

The method of analysis employed subsequently is common to all types of transportation. This is due to the fact that the means of transportation considered here (train, barge) are confined to a rather narrow track (rail, channel) which leads past the Plant nearly as a straight line. Trains and barges are assumed to be moving along these lines at normal

speed and are therefore subject to accident rates per mile of track (taken to be constant along the track). The pertinent accident rates are discussed below.

An approximate relation for the peak (positive) overpressure of the shock wave from an explosion can be written as:

$$p = a \times g^{-b} \qquad (2.2\text{-}1)$$

where

p = peak overpressure (psi)

g = scaled distance from center of explosion ($ft/lb^{1/3}$)

a, b = constants.

A good approximation in the range of pressure $0.1 < p < 10$ psi is obtained by setting:

$$a = 200$$

The scaled distance is defined as:

$$g = \frac{P}{W^{1/3}} \; (ft/lb^{1/3}) \qquad (2.2\text{-}2)$$

where

P = distance from center of explosion (ft)

W = weight of explosive, TNT equivalent (lb).

2.4.6 PROBABLE MAXIMUM TSUNAMI FLOODING

Historically the evidence demonstrates that the mouth of the Columbia River is relatively insensitive to tsunamis when compared to Crescent City, California, 310 miles south of the Columbia River entrance.

The tsunami effects at the mouth of the Columbia River are further dissipated inside the river due to the characteristics of the estuary, as was demonstrated during the tsunami generated by the Alaskan earthquake of March 28, 1964.

The elevation of the Plant site is 45 feet MSL. Because of the large margin between the Plant elevation and the river surface and because of the insensitivity of the river to tsunami effects, tsunamis are not considered in the design criteria for the Trojan Plant.

2.4.7 ICE EFFECTS

The general climate in the lower Columbia River Basin is not conducive to ice formation. In addition, the flow of the river during periods of freezing temperatures is sufficiently large (200,000 to 400,000 cfs) that ice formation is impossible in the main streamflow. During extended periods of freezing temperatures, some icing is experienced along the banks of sloughs and inlets where the water is slow moving or stagnant.

Any surface ice formation in the vicinity of the intake structure would not affect the operation of the intake structure, as the water entrance is below the water surface.

The lowest recorded river temperature at the site was 34.1°F on February 6, 1971 (period of record 1967 to 1972)[17].

2.4.8 COOLING WATER CANALS AND RESERVOIRS

The cooling water for the Trojan Plant is drawn in through an intake structure located on the bank of the Columbia River and oriented perpendicular to the flow of the river.

The intake channel provides a water channel to the intake structure during low flows. During a flood, the intake channel would be kept clear by the scouring action of the water flow.

2.4.9 CHANNEL DIVERSIONS

The Columbia River, for most of its lower reach, flows through a relatively narrow gorge. Diversion or rerouting of the river is impossible because of this gorge and the mountain ranges through which the river flows. Blockage of flow due to a catastrophic landslide or some other incredible event upstream of the Plant would not present a hazard. Should flow in the river reach zero (which would require blockage of all tributaries also), the river channel would act as a extension of the Columbia River estuary. The service water pump inlets are located at -8 feet MSL.

2.4.10 FLOODING PROTECTION REQUIREMENTS

All facilities/equipment required for the storage of irradiated fuel are located above or are protected for water levels to Elevation 45 feet MSL. The intake structure is the only structure exposed to the river flow below Elevation 45 feet MSL. In the unlikely event of an intake structure failure resulting in the loss of service water supply for SFP cooling, at least 10 days are available to establish makeup flow to the SFP to maintain required cooling.

Trojan site. This maximum potential earthquake that could affect the site is called the Seismic Margin Earthquake (SME).

These studies determined a value for the SME peak horizontal ground acceleration to be 0.38 g[11]. The SME impact on facility design is discussed in Section 3.2.2.6.

2.5.3 SURFACE FAULTING

Regional and site geology are presented in Section 2.5.1.

2.5.3.1 Geologic Condition of the Site

The site is in the Willapa Hills geomorphic province, a part of the Coast Range. Most of the region is below 2000 feet in elevation. The descent from the hills to the Columbia River is rather precipitous, but elsewhere the hills merge gradually into the surrounding lowlands.

The bedrock in the area is comprised of a series of moderately folded tertiary formations of both sedimentary and volcanic origin. The folding in the area conforms generally to the northwest Coast Range structural trend. The Eocene formations north of the Columbia River are folded as part of a syncline. Pliocene sediments in the vicinity of the site are only slightly warped and the Pleistocene and Recent deposits appear to be flat lying and undisturbed. A detailed discussion of the regional geology is presented in Section 2.5.1.

Faulting is minor in the structural development of the area, and is generally of small displacement. Many of the mapped faults in the area are based on topographic lineations in the pre-Pleistocene strata. No evidence of post-Pleistocene surface displacement has been found in the area.

An extensive investigation was made as part of the geologic evaluation of the Trojan site to locate all faults which might be significant to the site. A special effort was made to detect any lineations or indications of offset in the alluvium, terrace deposits, or the Pleistocene alluvial deposits. As a result of the study, the following conclusions were reached:

(1) The "Kelso Fault", as indicated in Bulletin 54 of the Washington Division of Mines and Geology[12], does not exist.

(2) The available geologic evidence indicates that there is not a fault in the old stream channel west of and adjacent to the site.

(3) The field evidence indicates that the Clatskanie fault does not extend farther east than indicated on the Oregon State Geologic Map.

(4) The available geologic evidence does not indicate that a fault exists along the Columbia River adjacent to the Trojan site.

(5) The fault zone exposed in the road cut southeast of Kelso and 4.7 miles from the site is apparently not extensive and probably has not experienced movement since deposition of the Troutdale formation during lower pliocene time. This fault zone is not significant to the Trojan site.

(6) There is no fault within 5 miles of the Trojan site which has experienced movement since Pleistocene time.

2.5.3.2 Investigation for Capable Faults

An extensive literature study was made as part of the geologic evaluation of the Trojan site to locate all faults that might be significant to the site. The bibliography for this

and reporting of radioactive effluents meet the requirements of Regulatory Guides 1.21 and 4.1.

Criterion 64 is met by Trojan design.

3.1.2 CLASSIFICATION OF STRUCTURES, COMPONENTS AND SYSTEMS

The classification of structures, components and systems for the facility will be per the quality related definition as described in Topical Report PGE-8010, "Nuclear Quality Assurance Program". Implementation of the classification is described in PGE Report PGE-1052, "Quality-Related List Classification Criteria for the Trojan Nuclear Plant"[9].

For the permanently defueled condition, the Control, Auxiliary and Fuel Building Complex; SFP, including the spent fuel racks; and the fuel transfer tube are the only structures, systems, or components that are classified as Seismic Category I (safety-related). The Containment Structure and the Fuel Building Steel Superstructure are classified as Seismic Category II/I.

3.1.3 WIND AND TORNADO LOADINGS

The Trojan Facility is capable of withstanding the effects of severe winds or tornadoes without loss of capability of the safety systems to perform their safety functions. The following sections provide the basis for the design wind and tornado parameters and methods used in meeting the wind and tornado criteria for the Facility.

3.1.3.1 Wind Loadings

3.1.3.1.1 Design Wind Velocity

The design wind velocity for all Category I and Category II structures is 105 mph at 30 ft above the nominal ground elevation of 45 feet MSL. Section 2.3 provides meteorological data for the site.

3.1.3.1.2 Determination of Applied Forces

Shape factors, variation of wind velocity with height, gust factors and methods of converting wind velocities into loads to be resisted by the structures are based on ASCE Paper No. 3269, "Wind Forces on Structures", 1961. Table 3.1-1 indicates the design loads. These loads are considered in the design of all Category I and Category II structures. However, since wind and earthquakes are not assumed to act simultaneously, for structures where seismic forces exceed the wind loads, no further analysis is performed. For structures where wind load governs, the load is combined with other appropriate loads as required by the various load equations.

Wind loads are applied to the structures as uniform static loads on the vertical and horizontal projected areas of the structure walls and roof. Roof loads due to wind are treated the same as roof dead and live loads with the direction of loading taken into account. Only dead load is considered as resisting uplift. Horizontal wind loads are distributed by the walls to the floor and roof diaphragms which, in turn, transfer the loads to the lateral load carrying elements of the structures.

3.1.3.2 Tornado Loadings

In the continental United States, west of the Rocky Mountains, the occurrence of tornadoes is unlikely. However, to ensure that any damage which may be sustained by

3.2 SPENT FUEL STORAGE

This section describes the design features of the Control, Auxiliary, and Fuel Building Complex and the SFP which are utilized for the long-term safe storage of spent fuel at the site.

3.2.1 CONTROL, AUXILIARY, AND FUEL BUILDING COMPLEX

The Fuel Building, which contains the SFP, is part of the Control, Auxiliary, and Fuel Buildings Complex. This section describes the design bases, criteria and analytical techniques for the design of the Complex.

Figures 3.2-1 through 3.2-9 indicate typical construction details of the Seismic Category I structures and the Fuel Building steel superstructure, which is Seismic Category II/I. Figures 3.2-23 through 3.2-30 indicate typical construction details for the Containment Structure which is also Seismic Category II/I.

3.2.1.1 Description

The Complex is interconnected by their foundation systems and floor and roof slabs. Therefore, in the seismic analysis all three buildings were considered as a unit. The seismic analysis of the Complex is covered in Section 3.1.6.

The Complex is composed of a structural steel framing system with steel beams and columns supporting reinforced concrete floor slabs, with shear walls designed to resist lateral seismic forces of an earthquake. Most of the shear walls are of a composite-type construction consisting of a reinforced or unreinforced concrete core between two layers (wythes) of reinforced grouted masonry block. The two block wythes generally sandwich the structural steel frame so that the steel frame members are embedded in the concrete core.

3.2.1.1.1 Fuel Building

The Fuel Building at and below Elevation 93 feet 0 inches is a Seismic Category I structure. It contains facilities for the handling and storage of spent fuel and systems used for the processing of liquid, solid, and gaseous radioactive wastes generated by the Plant. It consists of four floors above grade (Elevation 45 feet), the SFP, cask loading pit, new fuel storage pit, cask wash pit and three reinforced concrete vaults enclosing the three Chemical Volume Control System (CVCS) holdup tanks.

The Fuel Building steel superstructure above Elevation 93 feet is classified as the Seismic Category II/I structure. The Fuel Building steel superstructure houses a 125 ton overhead crane capable of handling heavy loads, such as a spent fuel assembly shipping cask, and a fuel-handling crane which runs on rails mounted on the operating floor. The travel of the overhead crane is restricted over the SFP by means of mechanical stops and administrative controls. These stops and limit switches may only be bypassed or removed while following an approved procedure, and then only with the Shift Manager's approval.

Mechanical anti-derailing devices mounted on the wheel axles of the overhead crane bridge and trolley prevent the crane from being dislodged from its rails due to horizontal motion during an earthquake. The vertical acceleration due to an earthquake is not large enough to overcome the crane's downward load due to gravity.

At the west side, the Fuel Building is structurally connected to the Auxiliary Building. Expansion joint bellows at the fuel transfer tube provide for the relative movement between containment, containment internals, and the SFP. The bellows consider all loading conditions including the SSE and maximum hydraulic pressure. The design of the expansion bellows considers the maximum computed relative axial and lateral displacement of the fuel transfer tube occurring simultaneously. Access for inspection and maintenance of the bellows is provided.

3.2.1.1.3 Control Building

The Control Building is a Seismic Category I structure and consists of four floors above grade (Elevation 45 feet). The Auxiliary Building is located to the east of the Control Building and the two buildings are structurally connected. The Turbine Building, located to the west, is separated from the Control Building by a gap which is adequate to prevent interaction between the two structures during an earthquake.

The framing members for the Control Building are structural steel. Floor slabs are reinforced concrete and precast prestressed concrete panels. Concrete block masonry and reinforced concrete are used for walls.

3.2.1.1.4 Modification to the Complex

The Control Building was modified during 1980 and 1981 to increase the seismic capability of the overall Complex. The modification program was developed with two basic objectives: first, to add strength to the building; second, to strengthen the connections among certain wall panels to achieve a better group action and improve overall seismic performance.

Strengthening of the Control Building was done by the addition of four new structural elements: three parallel walls running in the north-south direction and a steel plate added to one wall. Two of the new walls closed the previous railroad bay openings in the east and west walls on column lines N and R, and the third wall is an interior shear wall crossing the previous railroad bay on column line N'. The upper portion of the west wall of the Control Building along column line R was further stiffened and strengthened by the addition of a 3-inch-thick steel plate. The previous railroad bay in the Control Building was thus totally enclosed, and the railroad spur that used to run through the Control Building was terminated in the Turbine Building.

In addition to the four new structural elements, structural improvements were also made at several locations to strengthen connections. Four structural improvements involved strengthening the existing bolted beam-column connections of column 46-N beneath Elevation 61 feet, Elevation 77 feet and Elevation 93 feet, and column 46-R beneath Elevation 77 feet by welding additional plates. The remaining structural improvements consisted of connecting the horizontal reinforcing steel at the following walls to make the steel continuous:

(1) The 41 wall at column 41-Q, Elevation 45 feet to Elevation 65 feet

(2) The 46 wall at column 46-N', Elevation 45 feet to Elevation 61 feet

(3) The 55 wall at column 55-Q, Elevation 45 feet to Elevation 61 feet

(4) The 55 wall at column 55-N', Elevation 45 feet to Elevation 61 feet.

Figures 3.2-10 through 3.2-14 show the structural modifications to the Control Building.

3.2.1.2 Design Bases

The design bases for the Complex are defined within the following classifications:

(1) Support - in addition to their own weight, the structures are designed to support the reactions and weight of the components of systems shown in Figures 3.2-15 through 3.2-22. Figures 3.2-15 through 3.2-22 reflect the structures and components of systems that existed during Trojan Plant operation, and are included in the DSAR for general information. The components of systems and the structures shown on the figures are being de-activated, decommissioned, or removed in conformance with the Trojan Nuclear Plant Decommissioning Plan,

PGE-1061, which was approved by the NRC, by letter dated April 15, 1996. These figures are, therefore, historical in nature and are not required to be updated as the decommissioning activities are completed. Control of formal engineering drawings is maintained in accordance with PGE-1061.

(2) Biological shield - the shielding design features are based on potential radiation sources during normal Plant operation, shutdown, and emergency operations[1]. The shielding design conservatively bounds the current defueled state.

(3) Missile shield - the missile shielding design bases are given in Section 3.1.5.

(4) Administrative control of access - the design bases are given in Section 7.7.

(5) Atmosphere control - the design bases include structural leak-tightness, external and internal pressures from usage of ventilation systems, and equipment shelter from the elements[1].

(6) Liquid control - protection from the efflux of water is required for the SFP and in areas where liquid radioactive wastes are handled by radioactive waste treatment systems[1].

(7) Seismic protection - design bases are given in Section 3.1.6.

(8) Wind protection - design bases are given in Section 3.1.3.

3.2.1.3 Applicable Codes, Standards, and Specifications

3.2.1.3.1 Applicable to Original Procedure

This section lists the codes, specifications, regulatory guides, and other documents used in the structural design of the structures listed in this section except for the modifications of the Complex and the Fuel Building steel superstructure. Section 3.2.1.3.2 lists the codes, specifications, regulatory guides, and other documents used in the design of the modifications to the Complex. Section 3.2.1.3.3 lists the codes, specifications, and other documents used in the assessment of the Fuel Building superstructure.

(1) American Concrete Institute (ACI):

ACI 315-65 "Manual of Standard Practice for Detailing Reinforced Concrete Standards".

ACI 318-63 "Building Code Requirements for Reinforced Concrete".

(2) American Institute of Steel Construction (AISC), "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings", Sixth Edition, 1963.

(3) American Society of Mechanical Engineers, "ASME Boiler and Pressure Vessel Code", Sections III, V, VIII, and IX, 1968 Editions.

(4) Uniform Building Code (UBC), 1967 Edition.

(5) AEC Publication TID-7024, "Nuclear Reactors and Earthquakes".

(6) American Society of Civil Engineers (ASCE), "Paper No. 3269, Wind Forces on Structures".

(7) American Iron and Steel Institute (AISI), "Specifications for the Design of the Light Gauge Cold-Former Steel Structural Members", 1961 Edition.

(8) American Water Works Association (AWWA), AWWA M-11.

(9) American Railway Engineering Association (AREA), "Manual of Recommended Practice".

(10) U. S. Army Corps of Engineers, Regulations with respect to dredging and construction of offshore structures for the Columbia River.

(11) State of Oregon, Bureau of Labor, Elevator Safety Act, Chapter 330.

3.2.1.3.2 Applicable to Procedure Used for the Complex

The following codes, standards and specifications were utilized in design, procurement and implementation of the Control Building modifications.

3.2.1.3.2.1 Codes and standards.

(1) Building Code Requirements for Reinforced Concrete, ACI 318-77, American Concrete Institute.

(2) Specifications for the Design, Fabrication and Erection of Structural Steel for Bulidings, American Institute of Steel Construction (AISC), November 1, 1978.

(3) Code of Standard Practice for Steel Buildings and Bridges, AISC (September 1, 1976).

(4) Standard Code for Arc and Gas Welding in Building Construction, AWS D1.1-79, American Welding Society (AWS).

3.2.1.3.2.2 NRC Regulatory Guides.

(1) Regulatory Guide 1.10, Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures, Rev. 1, January 2, 1973. This regulatory guide was withdrawn by the NRC in July 1981. Withdrawal does not alter prior licensing commitments based on its use.

(2) Regulatory Guide 1.15, Testing of Reinforcing Bars for Category I Concrete Structures, Rev. 1, December 28, 1972. This regulatory guide was withdrawn by the NRC in July 1981. Withdrawal does not alter prior licensing commitments based on its use.

(3) Regulatory Guide 1.28, Quality Assurance Program Requirements (Designs and Construction), June 7, 1972.

(4) Regulatory Guide 1.33, Quality Assurance Program Requirements (Operation), Rev. 2, February 1978.

(5) Regulatory Guide 1.38, Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants, March 16, 1973.

(6) Regulatory Guide 1.39, Housekeeping Requirements for Water-Cooled Nuclear Power Plants, March 16, 1973.

(7) Regulatory Guide 1.55, Concrete Placement in Category I Structures, June 1973. This regulatory guide was withdrawn by the NRC in July 1981. Withdrawal does not alter prior licensing commitments based on its use.

(8) Regulatory Guide 1.58, Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel, August 1973.

(9) Regulatory Guide 1.64, Quality Assurance Requirements for the Design of Nuclear Power Plants, Rev. 2, June 1976.

(10) Regulatory Guide 1.88, Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records, Rev. 2, October 1976.

(11) Regulatory Guide 1.94, Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants, Rev. 1, April 1976.

(12) Regulatory Guide 1.123, Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants, October 1976.

3.2.1.3.2.3 Specifications. Specifications for the Control Building modifications were prepared to cover the following areas:

(1) Furnishing and delivery of concrete.

(2) Forming, placing, and curing of concrete.

(3) Furnishing and delivery of reinforcing steel.

(4) Placing of reinforcing steel.

(5) Furnishing and delivery of miscellaneous steel items.

(6) Erecting miscellaneous steel items.

(7) Material testing services.

(8) Furnishing and installing of Cadwelds.

3.2.1.3.3 Applicable to Procedures Used for the Fuel Building Steel Superstructure

The following codes, standards, specifications, and other documents were used in assessment of the Fuel Building steel superstructure.

3.2.1.3.3.1 Codes and Standards.

(1) Specification for Structural Steel Buildings - Allowable Stress Design and Plastic Design, American Institute of Steel Construction (AISC), June 1, 1989.

(2) Code of Standard Practice for Steel Buildings and Bridges, AISC, September 1, 1986.

(3) Structural Welding Code - Steel, AWS D1.1-90, American Welding Society.

(4) Steel Embedments Appendix in Code Requirement for Nuclear Safety-Related Concrete Structures, ACI 349-80, Appendix B, American Concrete Institute (ACI).

(5) Building Code Requirements for Reinforced Concrete, ACI 318-86, American Concrete Institute.

3.2.1.4 Loads and Load Combinations

This section gives the load combinations that are used in the analysis and design of the structures listed in this section, including modification of the Complex. Basically, all structures are investigated for all the load combinations to determine the critical loads. Therefore, a list of load combinations used, as they apply to a particular structure, is not given.

3.2.1.4.1 Notations

The following notations are used in the load combination equations:

U = required ultimate load capacity

D = dead load of structure and equipment plus any other permanent loads contributing stresses, such as soil or hydrostatic loads. An allowance is also made for future permanent loads

L = live load

R = force or pressure on structure due to rupture of any one pipe

T_o = thermal loads due to temperature gradient through wall during operating conditions

Ho = force on structure due to thermal expansion of pipes during operating conditions

T_A = thermal loads due to temperature gradient through wall during accident conditions

H_A = force on structure due to thermal expansion of pipes during accident conditions

E = OBE resulting from ground surface acceleration of 0.15

E′ = SSE resulting from ground surface acceleration of 0.25

A = hydrostatic load due to upstream dam failure

W = wind load (Wind velocity 105 miles per hour at 30 feet above ground)

0 = capacity reduction factor (Defined in ACI 318-71 Code, Section 9.2.1)

f_s = allowable stress for structural steel

F_y = yield strength for structural steel.

3.2.1.4.2 Seismic Category I Structure During Normal Operation

For loads encountered during normal Plant operation, Seismic Category I structures are designed in accordance with design methods of referenced codes and standards. Seismic design is in accordance with Section 3.1.6.

3.2.1.4.2.1 Concrete. Reinforced concrete structures are designed for ductile behavior, i.e., with steel stresses controlling.

Design of concrete structures satisfies the most severe loading combinations, based on the load factors shown below:

$$U = 1.5\,D + 1.8\,L$$

$$U = 1.25\,(D + L + H_o + E) + 1.0\,T_o$$

$$U = 1.25\,(D + L + H_o + W) + 1.0\,T_o$$

$$U = 0.9\,D + 1.25\,(H_o + E) + 1.0\,T_o$$

$$U = 0.9\,D + 1.25\,(H_o + W) + 1.0\,T_o$$

In addition, for ductile moment-resisting concrete space frames and for shear walls:

$$U = 1.4\,(D + L + E) + 1.0\,T_o + 1.25\,H_o$$

$$U = 0.9\,D + 1.25\,E + 1.0\,T_o + 1.25\,H_o$$

3.2.1.4.2.2 Structural steel. Steel structures are designed for the following loading combinations without exceeding the specified stresses:

$D + L$	Stress limit $= f_s$
$D + L + T_o + H_o + E$	Stress limit $= 1.25\ f_s$
$D + L + T_o + H_o + W$	Stress limit $= 1.33\ f_s$

In addition, for structural elements, carrying mainly earthquake forces, such as struts and bracings:

$D + L + T_o + H_o + E$	Stress limit $= f_s$

3.2.1.4.3 Seismic Category I Structure During Accident and SSE

The Seismic Category I structures are proportioned to maintain elastic behavior when subjected to various combinations of dead loads, thermal loads, seismic and accident loads. The upper limit of elastic behavior is considered to be the yield strength, F_y, for steel (including reinforcing steel) is considered to be the guaranteed minimum given in appropriate ASTM specifications. The yield strength for reinforced concrete structures is considered to be the ultimate resisting capacity as calculated from the "Ultimate Strength Design" portion of the ACI 318-71 Code. The allowable shear and axial stresses are also in accordance with the "Ultimate Strength Design" portion of the above code.

3.2.1.4.3.1 Concrete. Concrete structures satisfy the most severe of the following loading combinations:

$$U = 1.05\,D + 1.05\,L + 1.25\,E + 1.0\,T_A + 1.0\,H_A + 1.0\,R$$
$$U = 0.95\,D + 1.25\,E + 1.0\,T_A + 1.0\,H_A + 1.0\,R$$
$$U = 1.0\,D + 1.0\,L + 1.0\,E' + 1.0\,T_o + 1.25\,H_o + 1.0\,R$$
$$U = 1.0\,D + 1.0\,L + 1.0\,E' + 1.0\,T_A + 1.0\,H_A + 1.0\,R$$
$$U = 1.0\,D + 1.0\,L + 1.0\,A + 1.0\,T_o + 1.25\,H_o$$

3.2.1.4.3.2 Structural steel. Steel structures satisfy the most severe of the following loading combinations without exceeding the specified stresses:

$D + L + R + T_o + H_o + E'$ Stress limit[a] $= 1.5\,f_s$
$D + L + R + T_A + H_A + E'$ Stress limit[a] $= 1.5\,f_s$
$D + L + A + T_o + H_o$ Stress limit[a] $= 1.5\,f_s$

Stress in some of the materials may reach yield strength. If this is the case, an analysis is made to determine whether the energy absorption capacity of the structure exceeds the energy input. The resulting deflection or distortion is reviewed.

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[a] Maximum allowable stress in bending and tension is $0.9\,f_y$. Maximum allowable stress in shear is $0.5\,f_y$.

3.2.1.4.4 Design Loads

The structures are designed for applicable loads and load combinations given in the previous sections. This section defines the various loads.

3.2.1.4.4.1 Dead loads. Dead loads consist of the weight of the concrete, structural steel, interior walls, equipment, main piping, contained fluids, cable trays, and duct work.

3.2.1.4.4.2 Live loads. Live loads for the design of the structural framing members are consistent with the intended use of the structure and the recommendations of the Uniform Building Code, 1967 Edition. The live loads that were used in the design of the structures are given in Table 3.2-1.

3.2.1.4.4.3 Earthquake loads. Earthquake loads are in accordance with Section 3.1.6.

3.2.1.4.4.4 Wind and tornado loads. The wind and tornado loads considered in the design of the structures are in accordance with the criteria given in Section 3.1.3.

3.2.1.4.4.5 Hydrostatic loads. The exterior walls and base slabs of the structures are designated for hydrostatic loads associated with an upstream failure of Grand Coulee Dam. Water level at the site would be at Elevation 42.75 feet MSL. The structures would not become buoyant as a result of this water elevation.

3.2.1.4.4.6 Negative pressure. The Fuel Building and Auxiliary Building are maintained at a negative pressure. The pressure differential between the inside and outside does not exceed 0.1 inches of water. The control room in the Control Building is maintained at atmospheric pressure.

3.2.1.4.4.7 Blast loads. The pressure due to 3000 tons of TNT exploding at the river midstream opposite the Intake Structure has been factored into the design.

3.2.1.5 Design and Analysis Procedures

3.2.1.5.1 Analytical Methods for Seismic Category I Structures (Original Procedure)

Classical theory, empirical equations, and numerical methods are used as necessary in the analysis of the structures.

Loads and load combinations delineated in Section 3.2.1.4 are considered. For dead loads, live loads, wind loads, and tornado loads, all of the methods listed above are used. Wind loads and tornado loads are converted to equivalent static loads and are applied to the structure either as uniform or concentrated loads. This is discussed in Section 3.1.3. The following computer programs are used in the static analysis:

(1) "Structural Engineering Systems Solver" (STRESS) (Bechtel CE 309-3) - this program performs linear analysis of framed structure (plane and space) for static loads by the stiffness method.

(2) "Composite Beam Analysis" (Bechtel CE 585) - this program analyzes and designs simply supported composite beams.

(3) "Structural Design Language" (Bechtel CE 901) - a method formulated for digital computer solution and based on a program commonly called ICES-STRUDL.

Classical methods used in the analysis are standard textbooks and handbooks as used in universities and engineering practice.

The seismic analysis of the structures is covered in Section 3.1.6. The mathematical models used are also shown in the aforementioned section.

3.2.1.5.2 Design and Analysis Procedure Used for the Complex

The structures of the Complex were analyzed for seismic loads as described in Section 3.1.6. The computer program used in the seismic analysis is described in Section 3.1.6.2.1. For the seismic analysis of the modified Complex the actual equipment and floor loadings were used in place of the floor live loads specified in Table 3.2-1.

3.2.1.5.2.1 New structural elements. Design methods used for the new structural elements for the Control Building were in accordance with applicable portions of the codes and specifications listed in Section 3.2.1.3.2. Concrete and reinforcing steel were proportioned using the Strength Design Technique in accordance with ACI 318-77. Structural steel members and plate were designed in accordance with the AISC specifications.

Criteria for Bolts:

Bolts used to fasten the 3-inch-thick plate to the R wall and the new concrete walls to the existing N and R walls were in accordance with the requirements of ASTM A-354 Grade BD. The allowable design shear force resisted by each bolt was calculated using the following formula:

$$V = \frac{\mu L F}{FS}$$

where

V = allowable bolt shear force in kips

μ = coefficient of friction at the interface between old and new elements, taken as 0.7

F = maximum bolt load of 200 kips for 2 inch diameter bolts

FS = factor of safety, taken as 2[a]

L = factor to account for losses.

Losses from the following factors were considered:

(1) Creep and shrinkage of the concrete and masonry walls.

(2) Initial relaxation of the bolt.

(3) Long-term relaxation of the bolt.

(4) Temperature losses.

Criteria for Shear Studs:

The allowable design shear force for shear studs was 1/2 the values given in Table 15 of the Nelson Division of TRW, Inc., publication, "Design Data 10 - Embedment Properties of Headed Studs", TRW Inc., 1974.

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[a] The calculation in which this safety factor was applied did not include the potential parameter variations used to determine the low frequency floor response spectra peak broadening. A conservative consideration of these potential variations leads to determination of a lower bound factor of safety of 1.1. However, as discussed in Section 3.1.6.2.9, the vertical growth in the wall panels in an earthquake due to the development of flexural cracking would more than compensate for the potential reduction in dead load due to creep and shrinkage effects, and a lower bound factor of safety of 1.3 would result. Transfer of Shear Forces to Bedrock: The new concrete shear walls are supported by reinforced concrete grade beams placed on bedrock and connected to the existing structure. Therefore, they will participate with the existing foundation in the transfer of shear forces to the bedrock.

3.2.1.5.2.2 Original structural elements. The structural elements used in the originally designed and as-built Complex consist of reinforced concrete construction for walls and slabs, structural steel members, and of masonry construction for shear walls and divider or partition walls.

Design methods and evaluations of structural adequacy of the structural elements composed of reinforced concrete and structural steel were in accordance with applicable portions of the codes and specifications listed in Section 3.2.1.3.1.

The following sections describe the analysis procedure for single wythe, double wythe and composite walls having safety significance for consideration of all in-plane and out-of-plane loads and the interaction of these loads. The walls having safety significance are walls: (a) used as shear walls, (b) either supporting or within 2 feet of safety-related equipment, (c) used as train separation for common mode failure consideration, or (d) used as design basis equipment protection, e.g., tornado and flood.

In-Plane Loading Conditions:

In order to evaluate the masonry walls, the in-plane shear force in each horizontal load carrying wall element corresponding to the governing loading condition was first determined. The wall capacity was then determined in accordance with the governing behavioral mode and compared with the shear force.

Evaluation of Capacity:

For evaluating in-plane capacities of the masonry walls of the Complex, the governing capacities were determined from consideration of each of the following potential behavioral modes: (a) flexure, (b) sliding, and (c) diagonal tension. In addition, vertical shear capacity was examined. Equations for each of these behavioral modes are discussed below, with appropriate references. Capacities calculated by application of these equations are based on results obtained from the testing program described in Appendix A of PGE-1020[3].

Notations:

A_B = area of masonry portion of wall section, square inches

A_g = area of concrete including cell grout and core, square inches

$A_w = A_B + A_g$ = Total area of section, square inches

A_{sc} = cross-sectional area of steel column, square inches

N_c = total number of columns crossing the shear plane

V_b = shear force resistance of the frictional component of beam to column connections, lbf

V_s = shear force resistance developed by shear friction of reinforcing steel on vertical edges of wall panel, lbf

$V_1 = V_b + V_s$, lbf

f_c' = specified compressive strength of concrete, psi

f_b' = compressive strength of block, psi

f_g' = computed compressive strength of composite section, psi

f_y = specified yield strength of reinforcing steel, psi

f_{ys} = specified yield strength of column steel, psi

h = height of wall panel, inches

l_w = width of wall panel, inches

t = thickness of wall, inches

v_f = nominal shear stress capacity of wall panel, psi

μ = coefficient of shear friction

ρ_v = distributed vertical reinforcing steel ratio

ρ_c = concentrated end vertical reinforcing steel ratio

ρ_h = horizontal reinforcing steel ratio

σ_0 = wall compressive stress due to normal load, psi.

(1) Flexural mode:

(a) Double curvature:

The procedure for determination of capacity of a wall behaving in the double curvature mode is described in Section 3.4.2.2 of PGE-1020 and the equation developed therein is as follows:

$$v_f = (l_w/h)\,(1.94\,\rho_c f_y + 0.93\,\rho_v f_y + 0.94\,\sigma_0)$$

(b) Single curvature:

For behavior in the single curvature mode, the top edge is considered free and the bottom edge is considered restrained. The magnitude of this moment restraint depends on the vertical reinforcing steel, the effective panel normal load at the top edge, and the vertical shears as limited by the beam-column connection and the horizontal reinforcing steel across the two vertical edges.

The equation for the single curvature shear capacity is:

$$v_f = (l_w/h)(0.97\,\rho_c f_y + 0.46\,\rho_v f_y + 0.47\,\sigma_0) + V_1/ht$$

(2) Sliding mode:

The method for determining sliding resistance at a wall slab interface of the Complex walls is presented in Reference 4. The total sliding resistance is developed as the summation of the resistance offered by the embedded steel columns, with the shear strength of the steel taken as $f_y/\sqrt{3}$, and the shear friction developed by the normal force and the vertical reinforcing steel crossing the shear plane. The equation for the sliding resistance is:

$$v_f = \mu\sigma_0 + 1.4\,(\rho_v + \rho_c)\,f_y + (A_{sc}/l_w t)\,(f_{ys}/\sqrt{3})\,[(N_c - 1.5)/(N_c - 1)]$$

where the coefficient of friction, μ, for the normal force is taken as the weighted average based on the relative areas of bearing composed of mortar bed joints and concrete in the block cell and the wall core. In the absence of any code-specified value, the coefficient of friction for the mortar bed joints is taken as 0.75, which is a lower bound value used in Reference 5. The numerator in the multiplying factor for the column resistance, $(N_c - 1.5)$,

represents the number of columns available to resist sliding in the entire wall section and is ec to the total number of columns crossing the shear plane minus one end column and half the other end column. The term (N_c - 1) in the denominator represents the number of panels; therefore

$$v_f = (0.75\, A_B/A_w + 1.4\, A_g/A_w)\, \sigma_0 + 1.4\, (p_v + p_c)\, f_y$$

$$+ (A_{sc}/l_w t)\, (f_{ys}/\sqrt{3})\, [(N_c - 1.5)/(N_c - 1)]$$

(3) Diagonal tension (shear) mode:

For evaluating the diagonal tension related shear capacity of a wall panel, the ACI deep beam equation is used, which gives the shear stress, v_c, as

$$v_c = (3.5 - 2.5M/l_w V)\ \ 2\sqrt{f_g'}$$

where M and V are the moment and shear force at the section. For a cantilever shear panel, M = Vh.

$$v_c = (3.5 - 2.5h/l_w)\ \ 2\sqrt{f_g'}$$

The compressive strength of the grouted masonry composite weighted average of masonry and concrete, or

$$\sqrt{f_g'} = \sqrt{f_b'}\ (A_B/A_w) + f_c'\ (A_g/A_w)$$

Accounting for the normal stress, σ_0, on the wall section, the resulting shear capacity, v_c^*, is

$$v_c^* = \sqrt{[(v_c + \sigma_0/2)^2 - (\sigma_0/2)^2]}$$

The horizontal and vertical reinforcing steel provide additional contributions to the diagonal tension strength.

Test results from specimens with height-to-width ratio of 0.5 (Reference 6) and height-to-width ratio of 1.17 (though titled as 1.0 in Reference 7) indicated that the cracking plane engages both the horizontal and vertical reinforcing steel; therefore, these may be considered as equally effective in providing the shear resistance.

Thus the final equation for the diagonal tension mode is

$$v_f = v_c^* + 0.5(\rho v + \rho h\ h/l_w)\ f_y$$

(4) Vertical Shear:

The vertical shear at embedded columns resulting from earthquake motion was also checked. The vertical forces are obtained from the finite element model and used in the load combinations in Section 3.2.1.4.2.1. The resistance is obtained by combining the shear friction resistance of the horizontal reinforcing steel (ACI 318-77, Section 11.7) and the resistance of the embedded bolted beam-column connection to slipping.

$$R_v = 1.4\ A_s f_y + \sum v_f$$

A_s = area of horizontal reinforcing steel crossing a hypothetical vertical crack along the embedded columns, square inches.

f_y = yield stress of reinforcing steel, ksi

v_f = the resistance of the bolted beam-column connection to slippage, based on a clamping force of 0.7 of the minimum tensile strength of the bolts 120 ksi) and a coefficient of friction of 0.35 (based on Reference 2).

3.2.1.5.3 Design and Analysis Procedure Used for the Fuel Building Steel Superstructure

The Fuel Building steel superstructure is basically a steel-framed, braced structure supporting a 125-ton overhead crane. Lateral loads are resisted by frames in the East-West direction and by bracing in the North-South direction. The superstructure was designed and analyzed for seismic loads as described in Section 3.1.6.2.8 and for tornado loads as described in Section 3.1.3. The computer program used in the analysis is identified in Section 3.1.6.2.8.

3.2.1.6 Structural Acceptance Criteria

3.2.1.6.1 Original Procedure

Except as noted in Section 3.2.1.4.2 allowable stresses are as specified in the applicable codes. Tables 3.2-2 through 3.2-4 give calculated results for the principal structural members of structures other than the Containment and Control Building Complex. Seismic stress contributions are listed separately to indicate their portion of the total stress.

In order to prevent loss of function of the structures, values are established to control the maximum structural deformations to within defined limits.

The limiting values for deformation and strength are normally set by the need to maintain structural integrity, the need to prevent structural deformation from displacing the equipment to the extent that the equipment suffers a loss of function, and the need

to prevent deformations which would inhibit the structure's ability to control leakage. Structural deformations were never the controlling criteria in the design of structures listed in Section 3.2.1.1. The allowable strains for concrete are in accordance with ACI 318-63.

3.2.1.6.2 Procedure Used for the Complex

3.2.1.6.2.1 In-plane loading conditions. Concrete block walls used in Seismic Category I structures, including the Complex, are designed to the UBC requirements for masonry. The major shear walls of the Complex are constructed of a high-strength concrete core, either reinforced or unreinforced (nonmasonry structural element), sandwiched between two wythes of reinforced grouted concrete blocks (masonry units). Since the provisions of the UBC applicable to masonry construction do not address a combination of masonry and nonmasonry units, the shear walls of the Complex are not addressed directly by the UBC. However, Sections 106 and 107 of the UBC allow departure from certain detailed code formulae and quantifications where such departures are supported by substantiation through testing. In addition, testing is the basis upon which most code criteria have been established. Accordingly, application of test results in the determination of wall capacities is appropriate.

Such a testing program, conducted during the period September 1978 through February 1979, is described in Appendix A of PGE-1020. The testing utilized 23 specimens which were designed to simulate parameters of the walls in the Complex. The materials used in the construction of these specimens as well as their aspect ratios and thicknesses were similar to the wall panels in the Complex. The testing program was extensive for its purpose and was typical of testing performed to substantiate code compliance.

The test results indicated that the hybrid unit of steel column-composite wall will have a lower bound diagonal tension capacity of 300 psi for the large percentage of composite walls which have an aspect ratio of approximately 0.5. However, this behavior relies

upon bond between the embedded steel frame and the surrounding concrete. In the development of the capacity criteria, this important structural aspect is conservatively neglected.

The capacity values obtained in the tests are not factored directly into the computation of the capacities of the Complex walls. Rather, the characteristics of the composite walls demonstrated by the test specimens formed the basis for development of theoretical equations which predicted the shear capacities of the individual wall panels as functions of percentage of vertical and horizontal reinforcing steel, the embedded steel columns, and the vertical load bearing on the wall panels from the dead load of the wall above.

The formulae given in Section 3.2.1.5.2.2, based upon understanding of behavior gained from the testing program, reflect at least the same level of conservatism as code equations.

3.2.1.7 Materials, Quality Control and Special Construction Techniques

3.2.1.7.1 Original Construction of Seismic Category I Structures

The following major structural materials are used in the construction of the Seismic Category I and Seismic Category II structures:

Concrete:

Auxiliary Building	f´c = 3000 psi
Fuel Building	f´c = 3000 psi
Intake Structure	f´c = 3000 psi
Control Building	f´c = 3000 psi
MSSS	f´c = 3000 psi
Turbine Building	f´c = 3000 psi
Shop building	f´c = 3000 psi

Chlorine Building	f´c = 3000 psi
Administration Building	f´c = 3000 psi
Security Building	f´c = 3000 psi
TSC	f´c = 3000 psi
Cooling tower	f´c = 4000 psi (veil)
	5000 psi (columns)
Reinforcing steel:	ASTM A-615, Grade 40 & 60
Structural steel:	ASTM A-36
High-strength bolts:	ASTM A-325
Stainless steel liner plate:	ASTM A-167, Type-304
Concrete block masonry:	ASTM C-90, Grade PI

3.2.1.7.2 Control Building Modification

Construction relating to modification of the Control Building was performed by Bechtel as described in Section 3.2.1.1.4. Bechtel performed design engineering, prepared specifications, obtained PGE's approval of the construction documents, and was generally under PGE's supervision and surveillance during construction work. Bechtel also had a staff of qualified supervisory personnel and inspectors assigned to the jobsite to ensure that the work was accomplished in accordance with the specifications and drawings.

3.2.1.7.2.1 Construction materials and specifications. Basically, four types of materials were used in the construction of the Control Building modifications. They are:

(1)	Concrete:	f_c' = 3500 psi
(2)	Reinforcing steel:	ASTM A-615-78, Grade 60
(3)	Structural steel and 3-inch plate:	ASTM A-36

(4) Bolts:

(a) High-strength bolts for structural steel: ASTM A-490

(b) High-strength bolts for 3-inch plate: ASTM A-354-78, Grade BD

A comprehensive set of documents, including specifications and drawings, was issued by Bechtel and approved by PGE for the entire modification work. These documents contained the technical provisions plus the necessary general and special conditions. The technical provisions were specific as to the scope, detail and quality of work required and were used by the inspectors along with the appropriate drawings to determine the acceptability of the work. In general, the technical provisions referred to the applicable standards and codes as the basic governing documents.

3.2.1.7.2.2 Quality assurance and quality control.

PGE Quality Assurance:

As the principal owner and the operator of the Plant, PGE was responsible for the administration and control of the total QA program as described in this section. In carrying out this responsibility, PGE delegated to Bechtel the responsibility for the QA of the engineering, procurement, and construction activities. PGE audited this delegated responsibility in accordance with the PGE Nuclear Plant Quality Assurance Program for Operation (NPQAP/O), which describes the overall QA administration and control including the interface controls between contracted organizations and PGE.

Bechtel Quality Assurance:

The Bechtel QA Program plan for use by the Bechtel Thermal Power Organization during design, procurement and implementation phases of the Control Building

modification is described in Bechtel Topical Report BQ-TOP-1, "Bechtel Quality Assurance Program for Nuclear Power Plants", Rev. 2A[8].

Bechtel Quality Control:

The Bechtel quality control program requirements, procedures and instructions implemented by Bechtel construction quality control engineers during the modification of the Control Building are described in the Bechtel Construction Quality Control Manual. This manual was prepared for use by Bechtel construction quality control engineers to assure that the required quality verification and inspection activities were performed. Project quality control instructions were issued by the division chief construction quality control engineer and provided supplementary technical and administrative direction to the project construction quality control engineer at the jobsite. Two types of project quality control instructions were used to supplement the requirements, procedures, and instructions contained in the Construction Quality Control Manual. They were administrative and technical.

Administrative quality control instructions were used to convey special and unique project provisions that further specified the construction quality control program on this specific project by modifying or superseding the requirements, procedures and instructions contained in the Construction Quality Control Manual.

Technical quality control instructions were used to convey information which supplemented but did not modify or supersede the requirements, procedures and instructions contained in the Construction Quality Control Manual.

3.2.1.7.2.3 Special construction techniques. No special construction techniques were utilized in the course of carrying out the modification to the Control Building. However, during most of the construction period the Plant was in operation and, therefore, the performance of the work was sequenced in such a way as not to reduce the strength of the shear walls or the structural capacity of the Complex below the level required to withstand the loading combination involving the SSE.

Also, the details of work were developed to preclude potentials for creating hazardous conditions or disrupting operations of the Plant. Details of Bechtel construction procedures and the modification work, including sequence of construction, are described in Section 4 of PGE-1020[3] and Licensee's Testimony on Matters Other Than Structural Adequacy of the Modified Complex[9].

3.2.1.8 Testing and Inservice Inspection Requirements

3.2.1.8.1 Original Procedure

No structural preoperational testing of the Seismic Category I and Seismic Category II structures was performed. During the life of the Plant, periodic inspections of the structures are made to visually inspect for apparent structural deterioration such as large cracks and excessive deflection of structural members. All seam and plug welds in the SFP liner plate were vacuum box tested upon completion of the welding. Where vacuum box testing was not possible, liquid penetrant testing was performed. The SFP has a system that provides for leakage to be detected at any time in the life of the Plant. This system consists of 2-inch by 2-inch troughs under the liner plate which lead to a collection system where leakage can be observed.

3.2.1.8.2 Procedure Used for the Complex

Transfer of shear force from the existing west and east walls of the Control Building to steel plate and new reinforced concrete wall elements is accomplished principally by friction with a clamping (normal) force supplied by through-wall post-tensioned bolts. A surveillance program was employed to ensure that the tension force in the bolts did not fall below design levels.

3.2.1.9 Foundations

The foundation slabs for Category I structures are made of reinforced concrete. Section 3.2.1.3 lists those documents applicable to the foundations of structures. The loads and load combinations

described in Section 3.2.1.4 are applicable to the design of foundations. Design and analysis procedures applicable to foundations are described in Section 3.2.1.5. Criteria for foundations are the same as those listed for the structures in general (see Section 3.2.1.6). Description of the reinforced concrete and structural steel used in the construction of foundations is in Section 3.2.1.7. Floor slabs were coated to prevent possible absorption of contaminated fluids and to facilitate decontamination.

3.2.2 SPENT FUEL POOL AND FUEL STORAGE RACKS

The SFP is the storage space for irradiated fuel from the reactor.

3.2.2.1 Design Bases

The original design bases for the SFP and rack is designed for the following:[10, 11, 12]

(1) Store 1408 fuel assemblies with maximum enrichment of 4.5 wt% and an average region burnup of 55,000 MWd/MTU. Some of the 1408 storage cells are used to store fuel rod storage canisters, radioactive filters, debris or specimen assemblies associated with refueling evolutions.

(2) Maintain $k_{eff} \leq 0.95$ during normal conditions with SFP water between 40°F and 140°F and a boron concentration of 0 ppm (demineralized water).

(3) Maintain $k_{eff} \leq 0.95$ for abnormal conditions, including:

(a) Loss of normal SFP cooling (water temperature of 212°F at SFP surface).

(b) Drop accidents involving items which may be transferred or handled over the fuel racks. These items are shown in Table 3.1-4.

(c) Accidental drop of a fuel assembly in any position or orientation.

(d) Effects on fuel assemblies resulting from an earthquake or missile (See Table 3.1-4).

During abnormal events, a soluble boron concentration of 2000 ppm is assumed. Variations in rack dimensions, neutron absorber parameters, fuel parameters, and fuel location permitted by fabrication tolerances are included in the analysis. Calculations were performed to determine the sensitivity of k_{eff} to abnormal SFP water temperatures which may be encountered during a loss of cooling water incident.

(4) Maintain fuel cladding integrity in the event forced cooling is lost and cooling occurs by SFP boiling (212°F) at the water surface. Evaporative losses are made up by a variety of sources as discussed in Section 4.3.1.

(5) The structural design of the spent fuel racks is in accordance with NRC Standard Review Plan 3.8.4 Appendix D, "Technical Position on Spent Fuel Racks." Construction materials for the racks conform to the requirements of ASME Section III, Division 1, Subsection NF.

The governing code for rack design and analysis is ASME Section III, Division 1, Subsection NF for Class 3 component supports. The design loads are specified in Reference 10.

(6) The entire structure, including the spent fuel racks, has been designed to Seismic Category I requirements.

(7) The dose rates at the surface of the SFP from spent fuel assemblies do not exceed 2.5 mrem/hr during spent fuel transfer and storage. Dose rates at the outside surface of the walls of the SFP do not exceed the maximum radiation zone level for the area[1].

(8) A fuel handling accident involving dropping of a single spent fuel assembly in the SFP from its maximum attainable height will not result in offsite radiation doses to the public exceeding the values calculated in Section 6.2.

(9) A spent fuel shipping cask shall not be moved into the Fuel Building[11].

3.2.2.2 System Design

The SFP is a reinforced concrete structure with seam-welded stainless steel plate liners. The pool volume is approximately 51,900 ft^3 with a surface area of 1300 ft^2. The pool is filled with borated water which is maintained at a concentration of ≥2000 ppm boron.

The spent fuel assemblies are stored in storage racks in parallel rows having a center-to-center distance of 10.5 inches in both horizontal directions. The racks are freestanding modules containing a neutron absorber (Boroflex). Burnable poison rod assemblies, neutron-source assemblies and thimble-plugging devices removed from the reactor are also stored in the SFP.

Adjacent to the SFP are two small pools. One is the fuel transfer canal which is connected to the refueling cavity (inside the containment) by the fuel transfer tube. The other is the spent fuel cask loading pit. Leak-tight doors have been provided between the SFP and these two smaller pools to allow underwater movement of the assemblies between pools. The doors open into the SFP so that when the doors are closed with the adjacent pools drained, water pressure tends to seal the doors. Additionally, each door is equipped with an inflatable boot seal around its periphery which is inflated when the door is closed using the instrument and service air system.

The water level in the SFP is maintained to provide at least 23 feet of water above the top of a spent fuel assembly in the storage racks, and at least 9.5 feet above the active portion of the fuel assembly during fuel transfer operations. This water barrier serves as a radiation shield, enabling the gamma dose rate at the pool surface from the spent fuel assembly to be maintained at or below 2.5 mrem/hr.

Overflows from the SFP drain into the SFP ventilation system (AB-4) exhaust ductwork and are directed to the dirty waste drain tank.

Beneath the SFP liner is a network of monitoring trenches which will collect any leakage through the liner. The trenches drain through normally open valves to the dirty radioactive waste treatment system. The leak detection valves are arranged into manifolds that are inspected periodically to monitor for SFP liner leakage.

Ventilation systems remove gaseous radioactivity from the atmosphere above the SFP and discharge through the Plant vent. The ventilation systems are described in Section 5.2. These ventilation systems are monitored for radioactivity by Process and Effluent Radiation Monitoring Systems (PERMS) which are described in Section 5.5.

A SFP Area Radiation Monitoring System (ARMS) is provided for personnel protection and general surveillance of the SFP area. These ARMS are discussed in Section 5.6.1.

The SFP water chemistry is sampled and maintained in accordance with Section 4.1.2. The environment to which the SFP liner and spent fuel are exposed is not conducive to corrosion.

3.2.2.3 Design Evaluation

The center-to-center distance between the adjacent spent fuel assemblies and the fixed neutron absorber are sufficient to ensure a $k_{eff} \leq 0.95$ even if unborated water and fresh nondepleted fuel, enriched in U-235 to 4.5 wt%, are in the SFP.

The design of the spent fuel storage rack assembly is such that it is impossible to insert the spent fuel assemblies in other than prescribed locations, thereby preventing any possibility of accidental criticality.

Mechanical and electrical stops are provided on the Fuel Building bridge crane rails to prevent an inadvertent traverse of the pool with heavy loads.

The only requirement to assure adequate cooling of the spent fuel is to maintain the water level in the SFP above the spent fuel elements. Section 6.3 analyzed the loss of forced SFP cooling due to failures of the non-Seismic Category I piping and the fuel transfer doors and determined that the loss of SFP cooling and inventory would not lead to uncovering of the fuel. Sufficient time would be available to establish makeup flow to the SFP.

Lines entering the SFP which could siphon the pool to Elevation 76-feet 7-inches or below (approximately 10 feet above fuel elements) are equipped with siphon breakers at Elevation 83-feet 11-inches.

3.2.2.4 Tests and Inspections

Neutron absorber coupons are located in the SFP and used for periodic verification of absorber effectiveness. The verification surveillance is every 4 years. Coupon inspection includes visual inspection, hardness, weight, and dimensional measurements.

The SFP liner leak detection manifold is inspected periodically to monitor for SFP liner leakage.

3.2.2.5 Instrumentation Application

A level switch is provided in the SFP for the purpose of annunciating high and low water levels to the control room. Local level indication is installed in the SFP.

SFP water temperature indication and a high temperature alarm on the sequence of events recorder are provided in the control room.

3.2.2.6 SFP Structure Re-evaluation for Beyond Design Basis Seismic Motions

The SFP structure was evaluated for ground motions that would result from the occurrence of a conservatively postulated Seismic Margin Earthquake (SME).

Cascadia Subduction Zone earthquake source interface events in the magnitude range of M 8 to 9 and intraslab events in the range of M 7 to 7 1/2, and random crustal earthquakes in the range of M 6 to 6 1/2 were considered to envelop hypothetical earthquake ground motion exposures to the Trojan site.

The bounding case SME was determined to be a Cascadia Subduction Zone intraslab earthquake of magnitude range M 7 to 7 1/2 at a hypocentral distance (source to site distance) of between 55 and 60 km. The corresponding peak horizontal ground acceleration at Trojan would be 0.38 g.

The SFP structure was deterministically found to have large to very large capacity margins over SME demands. Peak horizontal ground accelerations on the order of 2 g would be required to approach the threshold of unacceptable damage to the SFP walls. There are no rational earthquake sources that could produce anywhere near this level of ground acceleration at the

Trojan site. There is, therefore, a very high level of confidence that the SFP structure would not be damaged by the postulated occurrence of a SME.

A probabilistic assessment was also performed that concluded that the annual probability of unacceptable SFP wall structural performance due to an earthquake producing several times the SME level of ground acceleration would be considerably less than $5x10^{-8}$.

3.3 AUXILIARY SYSTEMS

This section discusses the auxiliary systems that are used to support the storage of spent fuel at Trojan. This section includes discussions on the fuel handling system, SFP cooling and demineralizer system, component cooling water system, service water system, compressed air system, makeup water treatment system, equipment and floor drain systems, Plant discharge and dilution structure, primary sampling system, fire protection system and program, control room habitability, and seismic instrumentation. These systems do not perform any safety functions with the reactor defueled.

3.3.1 FUEL HANDLING SYSTEM

The fuel handling system consists of equipment and structures utilized for handling spent fuel assemblies during fuel transfer operations. This discussion is limited to fuel handling equipment used for transfer operations within the SFP. The transfer of fuel to the Containment Building or to a spent fuel shipping cask is prohibited under Trojan's current license.

3.3.1.1 Design Bases

The fuel handling system is designed to minimize the possibility of mishandling or maloperation that could cause fuel damage and potential fission product releases. The following design bases apply to the fuel handling system:

(1) Fuel handling devices have provisions to avoid dropping or jamming of fuel assemblies during transfer operations.

(2) The fuel handling equipment has been designed for the loading that would occur during a Safe Shutdown Earthquake (SSE). The fuel handling equipment will not fail so as to cause damage to any fuel elements should a SSE occur during fuel transfer operations.

(3) The hoist used to lift the spent fuel assemblies has a limited maximum lift height which is determined by the length of the long-handled tool, so that the minimum required depth of water shielding is maintained.

Environmental conditions of the fuel handling equipment, such as exposure to borated water and high humidity, are considered in the design and selection of the material.

3.3.1.2 System Description

3.3.1.2.1 General Description

Fuel assemblies are moved in the SFP using the SFP bridge hoist. When lifting spent fuel assemblies, the hoist uses a long-handled tool to assure that sufficient radiation shielding is maintained. Fuel assembly inserts, such as thimble plugs, burnable poisons rods, rod control clusters, and source rods, may also be transferred between positions within the SFP.

3.3.1.2.2 Component Description

3.3.1.2.2.1 Fuel Building bridge crane. The Fuel Building bridge crane is an indoor electric overhead travelling bridge crane complete with a single trolley and all the necessary motors, control, brakes, and control station. The main hoist of the crane is rated at 125 tons and the auxiliary hoist at 25 tons. The crane and accessories have been designed and constructed for indoor service and were designed to handle new and spent fuel containers between the railroad cars and loading and unloading pits. Movement of

heavy loads outside the approved load path discussed in Section 4.2.1 must be authorized prior to implementation. Neither the main hoist nor the auxiliary hoist are capable of being immersed in SFP water.

3.3.1.2.2.2 SFP bridge hoist. The SFP bridge hoist is a wheel-mounted walkway, spanning the SFP, which carries an electric monorail hoist on an overhead structure. The fuel assemblies are moved within the SFP by means of a long-handled tool suspended from the hoist.

The manufacturer's construction drawing details include the following information:

> The SFP bridge hoist has a 2000-pound capacity with a 21-foot maximum lift. The hoist uses stainless steel Type-304 cable and a safety load hook.

The SFP bridge hoist incorporates design features to minimize the probability of a fuel handling accident. These features include:

(1) The SFP bridge cannot be moved unless the hoist is in the full up position (with the exception of jogging). An interlock prevents simultaneous operation of the SFP bridge drive and hoist.

(2) Hoist travel in the up direction is terminated by a limit switch that also provides the "full up" interlock for the bridge drive with a hoist geared limit switch providing backup. Hoist travel is designed to maintain safe shielding depth of fuel assemblies by limiting maximum lift height with the spent fuel handling tool attached.

(3) A geared limit switch is provided to automatically stop downward motion when the hoist drum does not have a sufficient number of cable wraps remaining.

(4) Indexing of the SFP bridge crane to spent fuel storage racks is accomplished by means of a grid system laid out on the bridge rails and on the handrail.

3.3.1.2.2.3 Spent fuel handling tool. The spent fuel assembly handling tool is a long handle tool which is about 30 feet long that is used to move spent fuel assemblies in the SFP. The tool is mechanically operated by an operator on the SFP bridge. The tool may also be used to handle debris baskets and specimen assemblies. The length of the tool assures that a minimum depth of 10 feet of water will remain above the active fuel during fuel movements.

The spent fuel assembly handling tool employs four cam actuated latching fingers, which grip the underside of the fuel assembly top nozzle. The operating handle that actuates the fingers is located at the top of the tool. When the fingers are latched, a lock pin is inserted into the operating handle (held by a ball and detent) to prevent accidental unlatching.

3.3.1.3 Design Evaluation

3.3.1.3.1 Seismic Consideration

The maximum design stress for the structures and for parts involved in gripping, supporting or hoisting the fuel assemblies is one-fifth the ultimate strength of the material. This requirement applies to normal working load and emergency pullout loads, when specified, but not to earthquake loading. To resist design basis earthquake forces, the equipment is designed to limit the stress in the load bearing parts to 0.9 times the ultimate stress for a combination of working load plus design basis earthquake forces.

(9) Valves and piping.

(10) Instrumentation.

The SFP cooling and demineralizer system components are located in the Auxiliary Building and Fuel Building.

The SFP cooling and demineralizer system is a closed-loop system consisting of three subsystems: the cooling system, the purification system, and the skimmer system.

The SFP cooling subsystem utilizes two SFP cooling pumps installed in parallel and two SFP cooling water heat exchangers installed in parallel with a common pipe between the pumps and the heat exchangers. The SFP cooling pumps draw suction from the SFP through a common strainer located above the spent fuel assemblies and discharge back to the SFP through the SFP cooling heat exchangers.

The SFP purification subsystem utilizes the SFP purification pump to divert a small fraction of the total flow through the SFP purification filter and/or the SFP demineralizer. Both the filter and demineralizer have sufficient capacity to recirculate the entire SFP water inventory at least once every two days. This purification loop is adequate for removing fission products and other contaminants which may be introduced by a leaking fuel assembly. Portable vacuum filtration units may also be used to support special cleanup evolutions.

The skimmer subsystem consists of the SFP skimmer pump and two 1-1/2-inch diameter standpipe and valve assemblies with quick disconnects to which a floating skimmer can be connected by a flexible hose. The SFP skimmer pump can draw suction from the top water surfaces of the SFP and discharge through the skimmer filter or empty filter housing back to the SFP.

3.3.2.3 Design Evaluation

A single SFP cooling pump and a single SFP cooling water heat exchanger can maintain SFP temperatures ≤140°F.

All lines entering the SFP which could siphon the pool to Elevation 76 feet 7 inches (approximately 10 feet above fuel elements) or below are equipped with siphon breakers at Elevation 83 feet 11 inches.

The SFP cooling and demineralizer system does not perform any safety functions. Loss of forced SFP cooling will cause the SFP water temperature to slowly rise. The longest time interval between SFP cooling system failure and its detection is 24 hours since the system is inspected once per shift. The failure would be detected sooner if the SFP water temperature reaches the alarm setpoint. If forced cooling cannot be restored, then the SFP water temperature will continue to rise, increasing the evaporation rate and possibly resulting in boiling within the SFP. The only requirement to assure adequate cooling of the spent fuel is to maintain the water level in the SFP above the spent fuel elements. Section 6.3 analyzed the loss of forced cooling to the SFP and determined that sufficient time exists to effect repairs to the cooling system or to establish makeup flow prior to uncovering the spent fuel. Makeup water is available from a variety of sources as described in Section 4.3.1.

In the event of a design basis seismic event, the systems that were designed to Seismic Category II requirements may not be operable. In this case, sufficient time would be available to align a source of makeup water to maintain water level.

3.3.3 COMPONENT COOLING WATER SYSTEM

The Component Cooling Water System (CCWS), shown in Figure 3.3-2, is a closed cycle system designed to provide a monitored intermediate barrier between components

installed in the nitrogen supply lines between the nitrogen sources and the surge tanks. The pressure may also be adjusted manually using the solenoid valves installed in the nitrogen supply lines from the Plant nitrogen storage system. Each tank is equipped with two safety relief valves to prevent overpressurizing the system.

Provisions exist for adding corrosion inhibitor to the system.

Two identical CCWS makeup pumps are installed to furnish makeup water to the system. Normal makeup water is supplied from the demineralized water storage tank using the demineralized water transfer pumps.

For the defueled condition, train independence and automatic isolation of selected loads are not required. A single CCWS pump and a single CCWS heat exchanger provide excess heat removal capacity for the current heat loads.

Piping in the portions of the CCWS providing SFP cooling is seamless carbon steel, fabricated and installed in accordance with the requirements of ANSI B.31.1.0, Code for Power Piping.

3.3.3.3 Design Evaluation

The CCWS does not perform any safety functions. Loss of component cooling water to the SFP cooling water heat exchangers will cause the SFP water temperature to slowly rise. If the component cooling water cannot be restored to the heat exchangers, then the SFP water temperature will continue to rise, increasing the evaporation rate and possibly resulting in boiling within the SFP. The only requirement to assure adequate cooling of the spent fuel is to maintain the water level in the SFP above the spent fuel elements. Section 6.3 analyzed the loss of forced cooling to the SFP and determined that sufficient time exists to effect repairs to the cooling system or to establish makeup flow prior to uncovering the spent fuel. Makeup water is available from a variety of sources as described in Section 4.3.1.

3.3.4 SERVICE WATER SYSTEM

The SWS, shown in Figure 3.3-3, is designed to provide water from the Columbia River to cool equipment and to supply water to various systems and equipment. With the reactor permanently defueled, the primary functions of the SWS are reduced to providing cooling water to the component cooling water heat exchangers and selected room cooler units as well as provide makeup to the spent fuel pool.

3.3.4.1 Design Bases

The SWS is designed to deliver the minimum required flows of water to equipment assuming a minimum water level of 1.5 feet below MSL in the Columbia River. With the reactor permanently defueled, system design requirements are reduced substantially from the original design bases of the system.

Heat transfer equipment was selected based on a temperature of 75°F, which exceeds the highest recorded Columbia River water temperature.

The system design includes provisions for inhibiting long-term corrosion and organic fouling of the system water passages.

Loss of the SWS will cause the SFP water temperature to slowly rise due to loss of heat removal capability from the SFP cooling water heat exchangers (due to loss of component cooling water cooling). The only requirement to assure adequate cooling of the spent fuel is to maintain the water level in the SFP above the spent fuel elements. Section 6.3 analyzed the loss of forced cooling to the SFP and determined that sufficient time exists to effect repairs to the cooling system or to establish makeup flow from an alternate makeup source prior to uncovering the spent fuel.

3.3.5 COMPRESSED AIR SYSTEM

The compressed air system provides the Plant compressed air requirements for pneumatic instruments and valves and for service air outlets located throughout the Plant which are used for operation of pneumatic tools. The system does not perform any safety functions.

The system uses water-cooled aftercoolers and compressors. The air receivers are connected to a common compressed air header which connects to the air filter unit. The discharge of the air filter unit connects to the air-dryer unit inlet and the service air header. The instrument air header is connected to the air-dryer unit discharge. Each air header supplies branch lines which supply instrument air and service air to the required loads throughout the Plant. The instrument and service air system provides air to the inflatable seals for the SFP gates and to the CCWS air-operated isolation valves (CV-3303, CV-3287, CV-3304, and CV-3288).

Loss of instrument air to the CCWS isolation valves would cause them to fail closed causing a loss of forced cooling to the SFP (due to loss of component cooling water to the SFP cooling water heat exchangers). The only requirement to assure adequate cooling of the spent fuel is to maintain the water level in the SFP above the spent fuel elements. Section 6.3 analyzed the loss of forced cooling to the SFP and determined that sufficient time exists to effect repairs to the cooling system or to establish makeup flow prior to uncovering the spent fuel.

3.3.6 BORIC ACID BATCH TANK

The boric acid batch tank will normally be used to supply borated water to the SFP. Procedural controls will be used for this process.

3.3.7 MAKEUP WATER TREATMENT SYSTEM

The Makeup Water Treatment System provides demineralized water of the required quality to meet Plant needs. Makeup water is processed and then stored in the demineralized water storage tank where it is available to meet Plant needs. The DWST is a source of SFP makeup water. The water is transferred to the SFP using a demineralized water transfer pump.

3.3.8 EQUIPMENT AND FLOOR DRAIN SYSTEMS

The following equipment and floor drainage systems are provided for the Plant:

(1) Dirty Radioactive Waste Treatment System (DRWS) drains.

(2) Clean Radioactive Waste Treatment System (CRWS) drains.

(3) Oily waste system.

3.4 ELECTRIC POWER

The electric power system consists of the offsite and onsite power systems.

3.4.1 OFFSITE POWER SYSTEM

3.4.1.1 Description

Four overhead transmission lines transmit power to the Trojan switchyard from switchyards at Allston, Rivergate, and St. Marys. The four transmission lines, the Trojan switchyard, and the two startup transformers constitute the offsite power supply system.

The offsite power supply is divided into two independent sources. Source No. 1 includes startup transformer No. 1, the 230-kV bus No. 1, and its associated power sources: the Trojan-St. Marys (PGE) and the Trojan-Allston No. 1 (BPA) 230-kV transmission lines. Source No. 2 includes startup transformer No. 2, the 230-kV bus No. 2, and its associated power sources: the Trojan-Rivergate (PGE) and the Trojan-Allston No. 2 (BPA) 230-kV transmission lines.

The design of the Trojan switchyard and the separation of the transmission towers allows for two physically independent and redundant offsite power supply combinations:
(1) Trojan-St. Marys and Trojan-Allston No. 1, and (2) Trojan-Rivergate and Trojan-Allston No. 2.

High speed clearing of faults and selective reclosing assure maximum availability of power. System stability studies have been made to assure that a sudden system load drop will not adversely affect the Trojan facility or the connected electric system.

The offsite power system does not perform any safety functions.

3.4.1.2 Analysis

An analysis of the offsite power system shows that it provides reliability and flexibility for maintaining offsite power to the Trojan facility. Loss of power and its effects on maintaining SFP cooling is analyzed in Section 3.4.2.2.

3.4.2 ONSITE POWER SYSTEMS

The onsite power system consists of the sources of electric power and their associated distribution systems in the generating station. Included are the batteries. In the defueled condition, the onsite power systems no longer perform any safety functions.

3.4.2.1 Description

3.4.2.1.1 12.47-kV System

The system consists of three 12.47-kV buses, designated H1, H2, and H3. Bus H1 is energized from startup transformer No. 1 and bus H2 is energized from startup transformer No. 2. Bus H3 is normally energized from bus H2. Bus H3 may also be energized from the yard loop.

3.4.2.1.2 4.16-kV System

The 4.16-kV system is designed to provide electric power for operation of Plant systems and equipment. It consists of four 4.16-kV buses which are normally supplied from the 12.47-kV system through two unit substation transformers.

Unit substation transformer No.1 supplies buses A1 and A5 and unit substation transformer No. 2 supplies buses A2 and A6.

For defueled conditions, train separation is no longer required. The ability to cross-tie buses provides flexibility for maintaining power to station loads.

3.4.2.1.3 480-V System

The 480-V system is designed to provide sufficient electric power for operation of Plant loads from load centers and MCC buses. The system consists of load centers, MCCs, loads fed from these load centers and MCCs, interconnecting cables, and associated instrumentation and control circuits.

For defueled conditions, train separation is no longer required. The ability to cross-tie buses provides flexibility for maintaining power to station loads.

3.4.2.1.4 120-V A-C System

The 120-V instrument a-c and the preferred instrument a-c systems are designed to provide reliable electric power for control and instrumentation.

The 120-V instrument a-c system was designed to supply 208/120-V control and instrumentation power for equipment not required for the safe shutdown of the reactor. This system supplies power to the normally open service water isolation valves at the Seismic Category I (original design) interface with the Seismic Category II portion of the system. In the defueled condition this automatic isolation is no longer needed.

The 120-V preferred instrument a-c system consists of four 120-V bus sections which are supplied from 120-V instrument a-c buses. For defueled conditions, the system no longer performs any safety functions.

3.4.2.1.5 125-V D-C System

The 125-V d-c system consists of two systems. Each system is comprised of one 125-V battery, two battery chargers, one 125-V d-c control center with two distribution panels and a split bus connected together by a nonautomatic circuit breaker, interconnecting cables, and associated instrumentation and control circuit. The system is arranged so that the battery or any one charger can independently supply the system bus. Power to each 125-V d-c system is normally supplied from one battery charger which rectifies 480-V a-c obtained from an associated 480-V a-c MCC. For defueled conditions, the system no longer performs any safety functions.

3.4.2.2 Analysis

A variety of electrical distribution system faults can cause the loss of forced cooling to the SFP. The only requirement to assure adequate cooling of the spent fuel is to maintain the water level in the SFP above the spent fuel elements. Section 6.3 analyzed the loss of forced cooling to the SFP and determined that sufficient time exists to effect repairs to the cooling system or to establish makeup flow prior to uncovering the spent fuel.

Based on the results of the analysis and the flexibility of the offsite and onsite electrical distribution system, sufficient time would be available to perform one of the following options:

(1) Restore offsite power and forced SFP cooling.

(2) Provide alternate source of power to systems reuqired for SFP forced cooling.

(3) Provide an a-c independent source of makeup water.

TABLE 3.1-3

LOCATIONS OF GAS STORAGE TANKS

Equip. No.	Service	Location	Dimensions	psig
T-118 A, B, C	Compressed air receivers	Turbine Bldg. Elevation 45 ft. outside EDG rooms between column 51 -55	3-ft dia x 8-ft high	165 max
T-151 A, B, C, D, E, F	N_2 bulk gas tanks	Roof of Control Bldg.	24-in dia x 21-ft long	2450
T-159	Liquid CO_2 storage tank	Turbine Bldg. between column 77 - 83, Elevation 45 ft.	4-ft dia x 10-ft long	200
T-157 A, B, C, D	Air accumulator	Main steam support structure, Elevation 64 ft.	183 in^3 each	125
T-125	Hydro-pneumatic pressure accumulator (air)	Turbine Bldg. south end, Elevation 45 ft., between column U and T	11-ft dia x 19-ft long	125
T-204 A, B	CCWS surge tank (N_2)	Elevation 77 ft., column line E and 61, between Reactor and Auxilary Building	7-ft dia x 8-ft 3-in high	150

TABLE 3.1-4

ANALYZED SFP LOAD DROPS AND MISSILES

LOAD DROPS					
Item No.	Item Description	Dry Weight (lb)	Max. Drop Height(in.)	Max. Impact Energy (in.-lbs)	Impact Surface (in.)
1	Spent fuel assembly with reactor control rod	1,624	12	19,488	8.42 x 8.42
2	Spent fuel assembly handling tool	356	172	61,200	9.0 x 9.0
3	Spent fuel assembly with handling tool	1,976	12	23,700	8.42 x 8.42
4	BPRA handling tool	800	148	118,400	18.0 x 18.0
5	Thimble plug removal tool	290	192	55,700	8.69 x 8.69
6	Fuel assembly channel spacing tool	350	254	88,900	8.42 x 8.42
7	Radiation specimen transfer basket	600	12	7,200	8.42 x 8.42
8	New fuel tool	80	480	38,400	9.0 x 9.0
9	Electric chain hoist	90	530	47,700	8 diameter
10	Portable RCCA change tool	900	148	133,200	8.42 x 8.42
11	Miscellaneous hand tooling/portable vacuum system	500	300	150,000	36 x 48

TORNADO MISSILES			
Item No.	Item Description	Dry Weight (lb)	Vertical Velocity at SFP Surface (mph)
1	4-in. x 4-in. x 12-ft long wood plank	108	140
2	3-in. diameter x 10-ft long steel pipe	76	52.5

LIST OF PERTINENT REGULATORY GUIDES FOR DEFUELED CONDITION

Revision and Compliance Status

Regulatory Guide

1.8 - Personnel Selection and Training (5/77), Rev. 1-R

Compliance Status

Comply with exception.

(a) Technical Specifications require that each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 except for the Radiation Protection Manager, who shall meet or exceed the qualifications of Regulatory Guide 1.8, April 1987, and Independent Safety Reviewers, who shall have five years of professional level experience and either a Bachelor's Degree in Engineering or the Physical Sciences or equivalent in accordance with ANSI/ANS-3.1-1981.

(b) The qualification requirements of ANS 3.1-1981 are used as a goal for the Technician with one exception. Specifically the Technician requirements given in ANSI N18.1-1971 (2-year working experience and an optional 1 year of related technical training) will be used as a minimum requirement in lieu of the 3 years experience required for ANS 3.1-1981.

(c) Paragraph 4.2.2 of ANSI N18.1-1971 requires there be an operations manager who holds a Senior Reactor Operator (SRO) license. Paragraphs 4.3.1 and 4.5.1 discuss supervisors and operators required to have NRC licenses. Section 5.2 discusses training of candidates for NRC examinations. Paragraph 5.5.1 recommends retraining include startup and shutdown procedures and emergency shutdown systems. In lieu of the above, no personnel are required to hold NRC licenses, and Certified Fuel Handler (CFH) training will only be given on subjects applicable to the permanently defueled mode. In addition, CFH training need only maintain proficiency in allowed activities.

Regulatory Guide

1.13 - Fuel Storage Facility Design Basis (3/71), Rev. 0

Compliance Status

Comply with exception.

A seismic Category I makeup system is not provided. The reduced decay heat load after 3 ½ years since the reactor was defueled allows over 10 days with no makeup or cooling before manual actions are required to prevent fuel damage. Therefore, adequate time is available to operate or restore one of the diverse non-seismic makeup sources.

Regulatory Guide

1.21 - Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants (6/74), Rev. 1

Compliance Status

Comply with exception.

The use of a meteorological tower for real time measurement of meteorological conditions may not be used. Worst case meteorological values, based on previous evaluations, will be used to conservatively estimate releases when the meteorological tower is not used.

Regulatory Guide

1.24 - Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Gas Storage Tank Failure (3/72), Rev. 0

Compliance Status

Comply.

quality-related materials, parts, and components only when specific protection measures are required. In lieu of the detailed requirements of ANSI N45.2.2-1972, quality-related items will receive a thorough engineering evaluation to assure that adequate protective measures are specified for packaging, shipping, receiving, storage, and handling of items for nuclear power plants. These protective measures will be consistent with standard/commercial engineering practices and manufacturers' recommendations.

(c) Marking may be applied to bare austenitic stainless steel and nickel alloy metal surfaces provided that it has been established that the marking is not deleterious to the item rather than as stated in Paragraph A3.9 of ANSI N45.2.2. Proper chemical controls will be employed to ensure there is no adverse metallurgical impact on the steel or nickel alloy. This position has been adopted in the document ASME NQA-2, Addenda 2a.

Regulatory Guide

1.39 - Housekeeping Requirements for Water-Cooled Nuclear Power Plants (9/77), Rev. 2

Compliance

Comply with exception.

Regulatory Guide 1.39 endorses the application of the recommendations of ANSI Standard N45.2.3, "Housekeeping During the Construction Phase of Nuclear Power Plants." The cleanliness and housekeeping standards established in ANSI Standard N45.2.3 are designed to control work activities, conditions, and environments that can affect the quality of important parts of a nuclear power plant. These parts include structures, systems, or components whose satisfactory performance is required for the plant to operate reliably, to prevent accidents that cause risk to the health and safety of the public, or to mitigate the consequences of such accidents if they were to occur. Specifically, the last paragraph of Section 2.1, "Planning," requires a written record of the entry and exit of material be established and maintained for Zone III cleanliness areas. In addition, the second paragraph of Section 3.2, "Control of Facility," states that appropriate control measures shall be provided through the utilization of such items as log books and tethered tools.

In that the plant no longer operates and has a relatively long time available to respond to, and mitigate the consequences of, any accidents postulated in this DSAR, strict application of the requirements of ANSI N45.2.3 is no longer necessary. Specifically, the use of a log for control of material above the spent fuel pool will no longer be required, but the option for its

use will not be precluded. If a log for control of material is not used, a visual inspection of the tops of spent fuel assemblies will be performed.

Regulatory Guide

1.58 - Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel (9/80), Rev. 1 Withdrawn

Compliance

Comply with exception.

Position C.2 requires certification of non-destructive examination (NDE) personnel per SNT-TC-1A-1975. The NDE personnel are certified per SNT-TC-1A-1984 at the Trojan Facility.

Regulatory Guide

1.59 - Design Basis Floods for Nuclear Power Plants (8/77), Rev. 2

Compliance

Comply with exception.

During the design phase of Trojan, pertinent regulations dealing with floods were 10 CFR 100.1 and General Design Criterion 2. The probable maximum flood (PMF) for Trojan was determined prior to the issuance of this Regulatory Guide and has been reviewed and approved by the NRC. As stated in this Regulatory Guide, previously reviewed and approved detailed PMF analysis at specific sites will continue to be acceptable.

Regulatory Guide

1.60 - Design Response Spectra for Seismic Design of Nuclear Power Plants (12/73), Rev. 1

Compliance

Comply with exception.

Design response spectra for the OBE and SSE were derived and used in the seismic analysis

and design of the Trojan Nuclear Plant prior to issuance of this Regulatory Guide. The methods used in the original derivation are described in FSAR Section 3.7.1 and have been reviewed and approved by the NRC. DSAR Section 3.1.6 summarizes the seismic design parameters.

Regulatory Guide

1.61 - Damping Values for Seismic Design of Nuclear Power Plants (10/73), Rev. 0

Compliance

Comply with exception.

(a) Table 1 of this Regulatory Guide specifies a damping value from 1 to 3 for piping systems (dependent upon pipe diameter and type of earthquake). PGE used a critical damping value of 0.5 percent in the dynamic analysis of vital piping systems and equipment. Seismic analysis of Trojan piping systems using a damping value of 0.5 percent is more conservative than the criteria specified by this Regulatory Guide.

(b) Table 1 of this Regulatory Guide specifies a damping value of 4 and 7 percent for an OBE and SSE, respectively, for reinforced concrete structures. PGE used critical damping values of 2 and 5 percent for the OBE and SSE, respectively. Analysis of Trojan's Category I structures employing damping values of 2 and 5 percent is more conservative than the criteria specified by this Regulatory Guide.

(c) Principle constituents of the Control, Auxiliary, and Fuel Building complex are reinforced concrete, reinforced grouted masonry, and a combination of the two. Table 1 of this Regulatory Guide does not specify damping values for such heterogeneous structures. However, the complex's structural materials can be conservatively assumed analogous to reinforced concrete for damping considerations. Damping values of 2 and 5 percent were used for an OBE and SSE, respectively, and are adequate.

(d) Table 1 of the Regulatory Guide specifies damping values of 2 to 3 percent for equipment, 2 to 4 percent for welded structures, and 4 to 7 percent for bolted structures. For structures and equipment, and supports for other than vital piping systems, PGE used damping values higher than 0.5 percent where justified. A maximum of 2 percent for OBE load combinations and 5 percent for SSE load combinations was used by PGE in accordance with the original design approach. Alternatively, some damping ratios are based on calculated stress levels as provided in

DSAR Table 3.1-5. The use of 2 percent for OBE and 5 percent for SSE, where justified, or damping ratios based on stress levels as provided in DSAR Table 3.1-5 meets the intent of the Regulatory Guide and is adequate.

Regulatory Guide

1.64 - Quality Assurance Requirements for the Design of Nuclear Power Plants (4/76), Rev. 2 Withdrawn

Compliance

Comply with exception.

Under certain conditions, the Manager, Engineering/Licensing may perform design verification provided:

(a) This manager is the only technically qualified individual or has special technical expertise which would allow him to perform a more thorough design verification.

(b) The need is individually documented and approved in advance by the General Manager, Trojan Plant.

(c) Quality Department audits cover frequency and effectiveness of use of the Manager, Engineering/Licensing as a design verifier to guard against abuse.

This position is consistent with NUREG-0800.

Regulatory Guide

1.74 - Quality Assurance Terms and Definitions (2/74), Rev. 0 Withdrawn

Compliance

Comply with exception.

The terms and definitions of ANSI N45.2 daughter standards that are not in agreement with ANSI N45.2.10 will be as used in the daughter standard. In addition, the definitions of Permanent (Lifetime) Records and Nonpermanent Records have been modified for clarity in accordance with the definition in the QA Program Glossary. The definition of "repair" in

ANSI N45.2.10 is replaced by "modify", because the term "repair" as used in various sections of the ASME Boiler and Pressure Vessel Code has a different meaning. The term "modify" is used to avoid confusion.

Regulatory Guide

1.76 - Design Basis Tornado for Nuclear Power Plants (4/74), Rev. 0

Compliance

Comply with exception.

Design, construction, and licensing activities for the Trojan Nuclear Plant were in an advanced stage prior to issuance of this Regulatory Guide. The recommendations of this Regulatory Guide were therefore not used as a basis for characterizing a design basis tornado. For the Trojan Nuclear Plant, criteria to evaluate safety-related structures for potential effects of tornados were developed pursuant to AEC General Design Criterion 2. All Plant structures containing systems needed to achieve and maintain safe shutdown were determined to be capable of resisting 200-mph tornado loads and a pressure differential of 1.5-psi bursting pressure applied at 1 psi/sec. Many of these structures were evaluated to be further capable of resisting 300-mph tornado loads and a pressure differential of 3-psi bursting pressure at 1 psi/sec. Draft Standard ANS 2.3 specifies a 200-mph tornado at a probability of 10^{-7} yr^{-1} for sites in Trojan's region. The 200-mph and 300-mph tornado maximum wind speed criteria are conservative with respect to the wind speed associated with the maximum tornado that could reasonably be hypothesized for the site. These criteria and methods have been reviewed and approved by the NRC.

Regulatory Guide

1.78 - Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release (6/74), Rev. 0

Compliance

Not applicable.

Analysis shows that Control Room habitability is not a concern in the defueled configuration. See the discussion in Section 3.3.12 of the DSAR.

Regulatory Guide

1.88 - Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records (10/76), Rev. 2 Withdrawn

Compliance

Comply with exception.

(a) The QA records will be stored in a facility with a minimum 2-hour fire rating in accordance with ANSI N45.2.9-1979, or as NFPA Class 1 records in a Class B (350-2 hours) container in accordance with NFPA 232-1975. ANSI N45.2.9-1979 has superseded ANSI N45.2.9-1974 and requires a minimum fire rating of 2 hours for facilities in which QA records are stored. NFPA 232-1975 is endorsed in this Guide as an acceptable alternative to the fire protection provisions listed in Subdivision 5.6 of ANSI N45.2.9-1974.

(b) The retention requirements of ANSI N45.2.9-1979 will be followed in lieu of the requirements in ANSI N45.2.9-1974. The retention requirements of ANSI N45.2.9-1979 appropriately reference the requirements of the regulatory agencies having jurisdiction over these records.

(c) Trojan will use optical storage as allowed by NRC Generic Letter 88-18, "Plant Record Storage on Optical Disks."

Regulatory Guide

1.91 - Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants (2/78), Rev. 1

Compliance

Comply with exception.

An exception is taken to the value of 1 psi for the maximum permissible overpressure generated by the atmospheric shock from an explosion. The analysis given in DSAR Section 2.2.3.1 calculated the probability of occurrence of a shock wave with a 2.2-psi overpressure. Test structures similar to the safety-related structures at Trojan have demonstrated that overpressures of 5 psi can be endured without any significant damage to

these structures. Furthermore, it is noted that these structures are designed to withstand pressure differentials of a least 3 psi due to tornado effects. For credible accidents, however, the maximum permissible overpressure value used in the analyses of Trojan is 2.2 psi.

Regulatory Guide

1.92 - Combining Modal Responses and Spatial Components in Seismic Response Analysis (2/76), Rev. 1

Compliance

Comply with exception.

Seismic analysis of structures, systems and components in the Trojan Nuclear Plant were completed prior to the issuance of this Regulatory Guide. The methods of combining modal responses and spatial components as recommended by this Regulatory Guide are technically sound. For modifications to the Trojan Nuclear Plant and for analysis of existing structures and components, either the square root sum of the squares method or the methods of modal and spatial combination recommended by this Regulatory Guide may be used at the option of the engineer performing the analysis.

However, modal responses were combined in accordance with this Regulatory Guide for the modification of the Control, Auxiliary and Fuel Building complex, with one exception. Contrary to Position C.2 of this Regulatory Guide, the combination of effects due to three spatial components of an earthquake was not performed. Rather, a co-directional response value was determined. That value was the maximum value obtained by adding the response due to the vertical earthquake with the larger value of co-directional response due to one of the two horizontal earthquakes by the absolute sum method.

Regulatory Guide

1.94 - Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants (4/76), Rev. 1

Compliance

Comply

Regulatory Guide

1.102 - Flood Protection for Nuclear Power Plants (9/76), Rev. 1

Compliance

Comply with exception.

Construction of Trojan was completed prior to the issuance of this Regulatory Guide. The flood design criteria and bases for floods used for Trojan are described in DSAR Sections 2.4 and 3.1.4. The need for shutdown procedures and communications per Position C.2 does not apply to a permanently defueled facility that does not require operator action to maintain fuel integrity for several days.

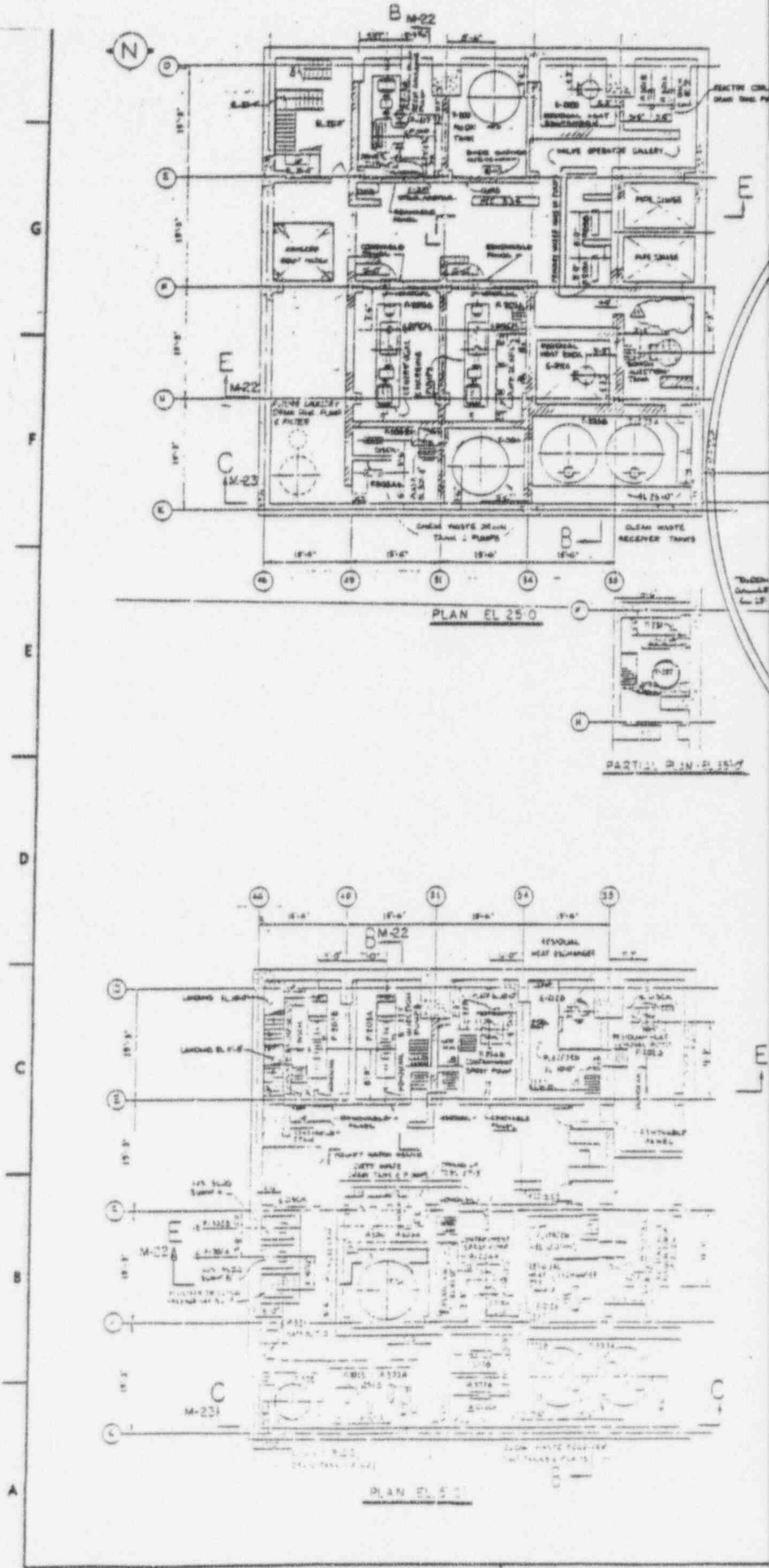
Regulatory Guide

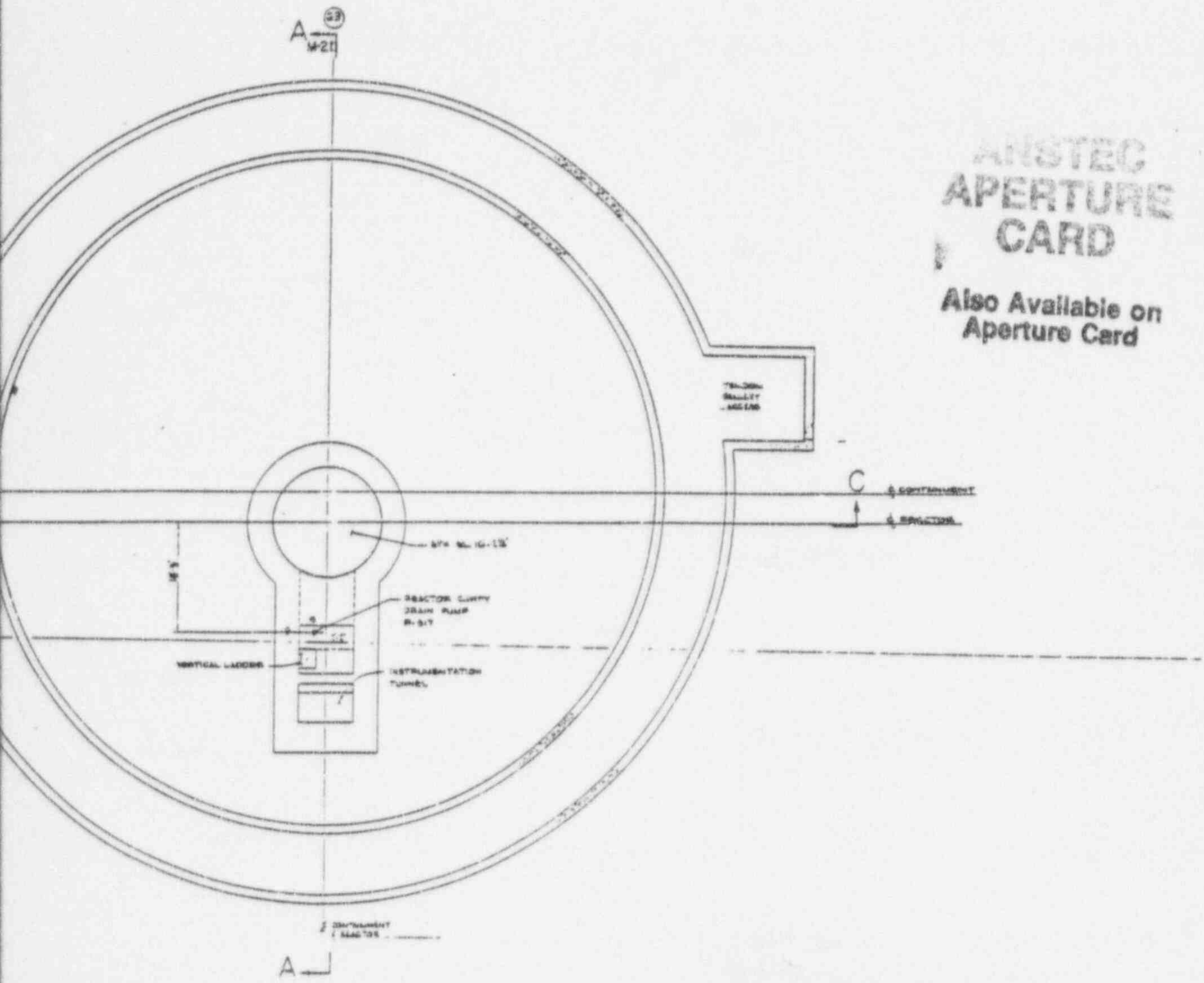
1.105 - Instrument Setpoints (11/76), Rev. 1

Compliance

Comply with exception.

Trojan does not conform to the regulatory position concerning the design verification of these instruments as part of the instrument qualification program recommended in Regulatory Guide 1.89. Trojan's setpoints meet the intent of this Regulatory Guide but not its specific format. Construction of the Trojan Nuclear Plant was completed prior to the implementation date of this Regulatory Guide. The method of selection of setpoints was previously approved by the NRC.


PLAN EL 25'-0
PARTIAL PLAN EL 35'-0
PLAN EL 5'-0
CHECK WASTE DRAIN TANK & PUMPS
CLEAN WASTE RECEIVER TANKS
VALVE OPERATOR GALLERY
PIPE CHASE
RESIDUAL HEAT EXCHANGER
G
F
E
D
C
B
A
8
7
6



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CARD

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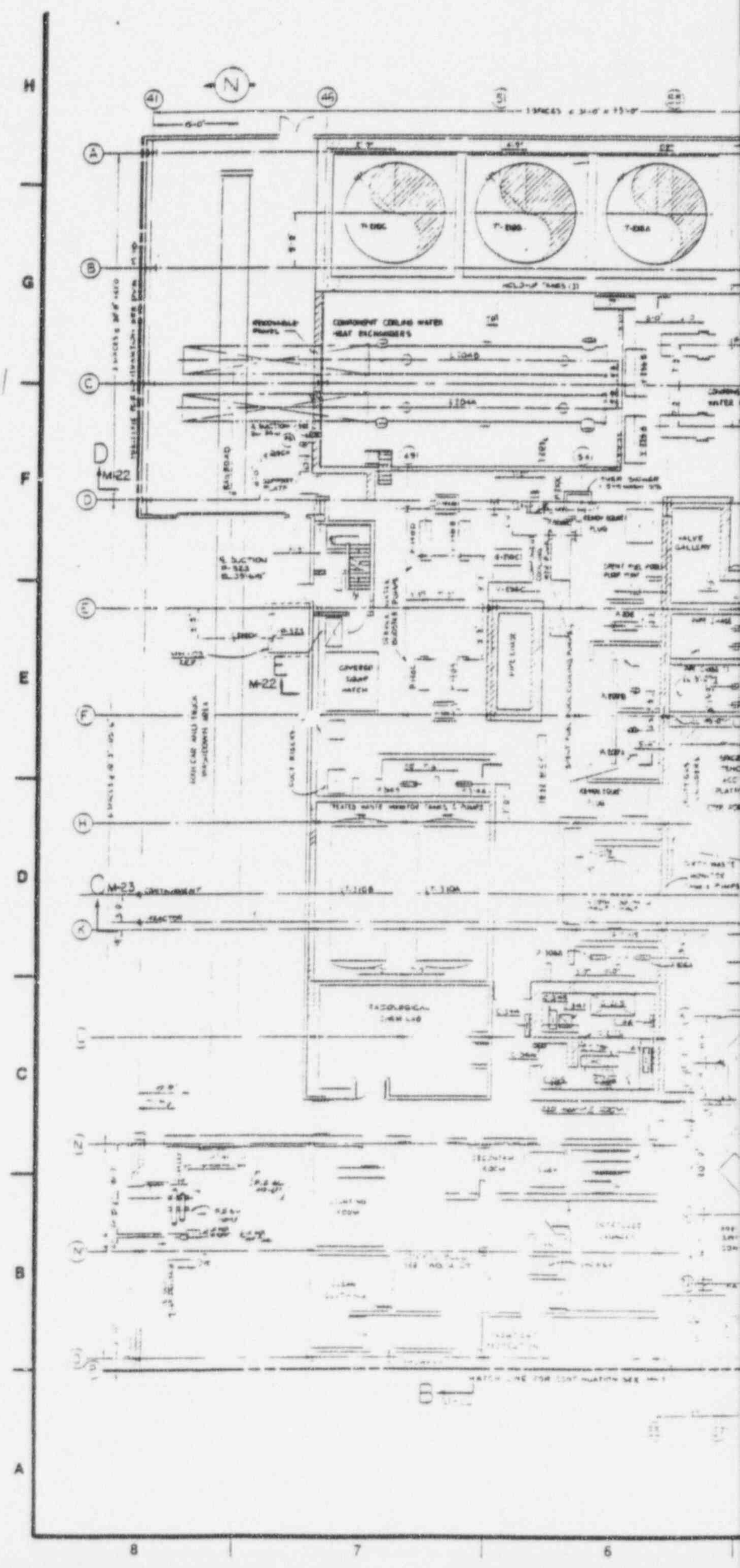
Aperture Card

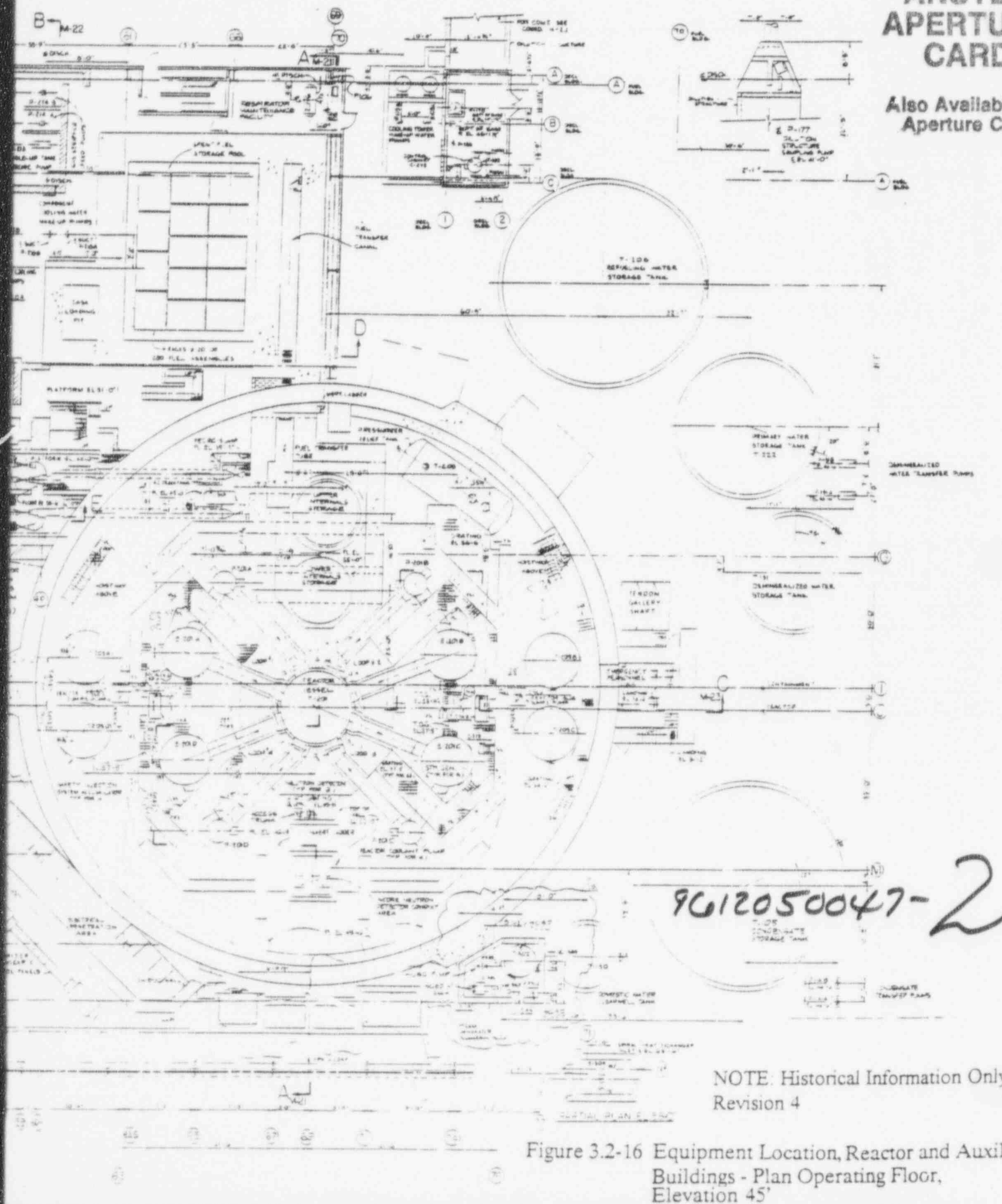
9612050047-1

Figure 3.2-15 Equipment Location,
Reactor and Auxiliary Buildings
- Plan Below Ground Floor

NOTE: Historical Information Only
Revision 4



M-22
M-23
COMPONENT COOLING WATER HEAT EXCHANGERS
HOLD-UP TANKS (3)
VALVE GALLERY
RADIOLOGICAL CHEM LAB



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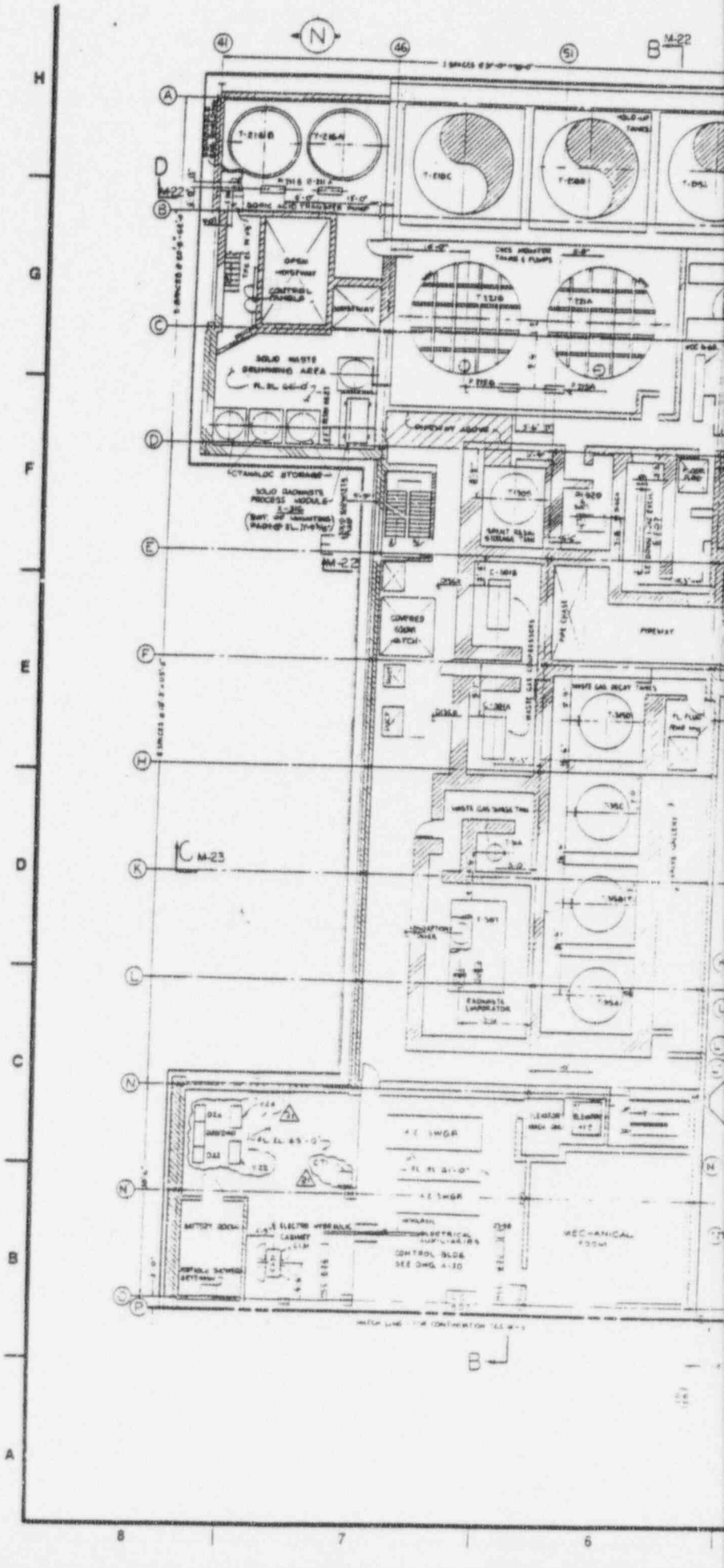
NOTE: Historical Information Only
Revision 4

Figure 3.2-16 Equipment Location, Reactor and Auxiliary Buildings - Plan Operating Floor, Elevation 45'



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N
41
46
51
M-22
M-23
T-216B
T-216A
SOLID WASTE
DRUMMING AREA
OPEN
HOISTWAY
SPENT RESIN
STORAGE TANK
PIPEWAY
WASTE GAS DECAY TANKS
WASTE GAS SURGE TANK
RADWASTE
EVAPORATOR
A.C. SWGR
BATTERY ROOM
MECHANICAL
ROOM
CONTROL BLDG
SEE DWG. A-20

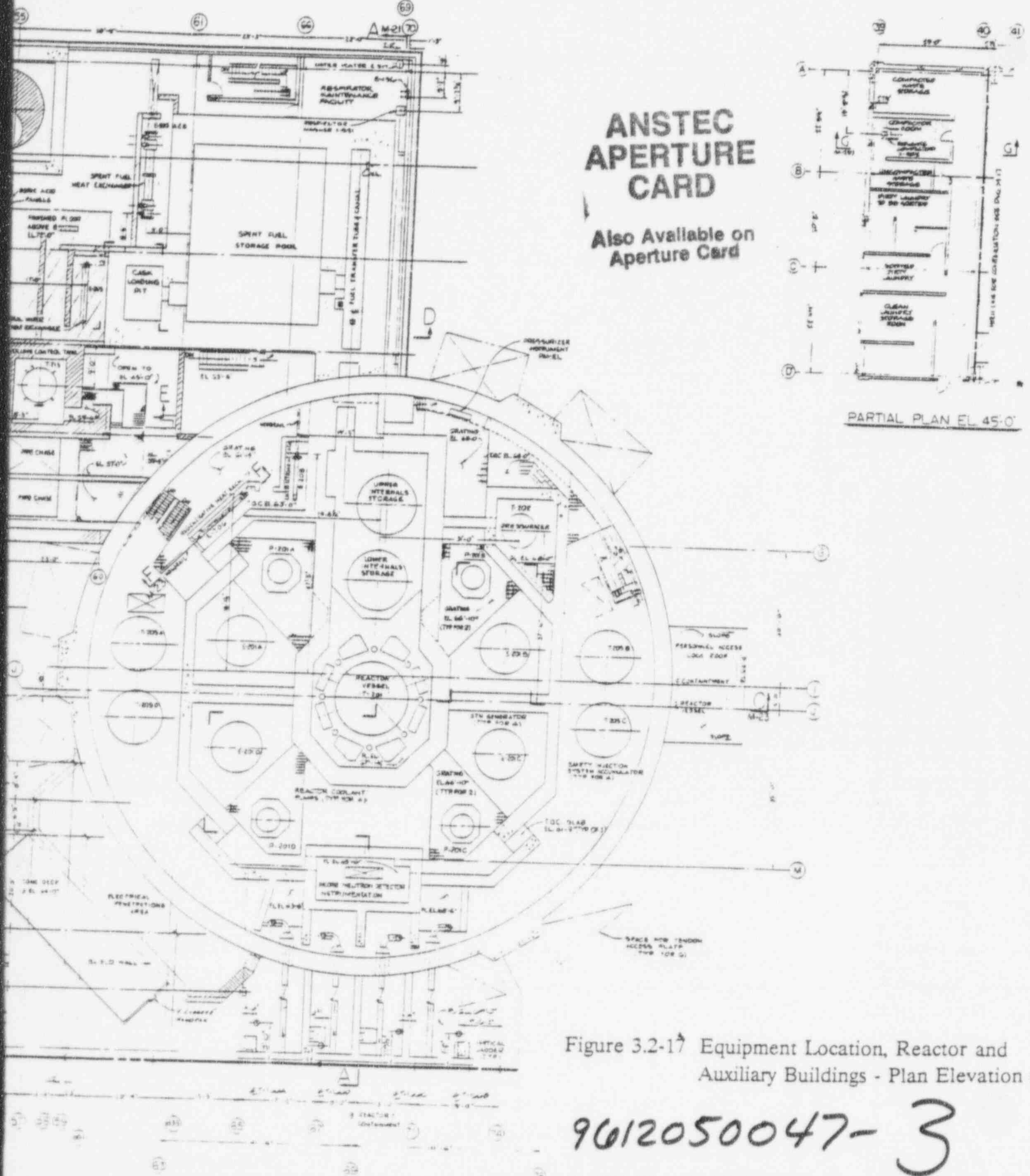
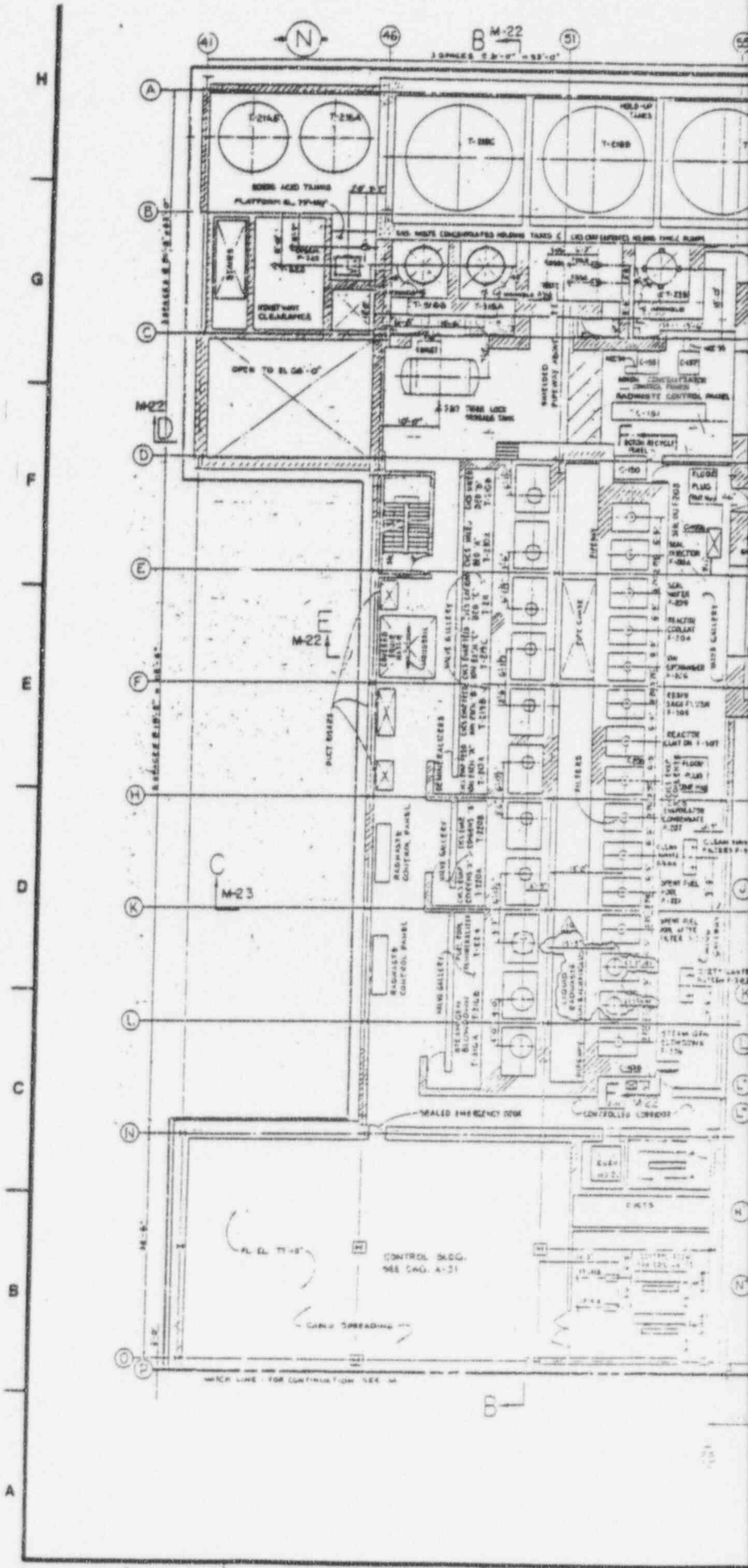


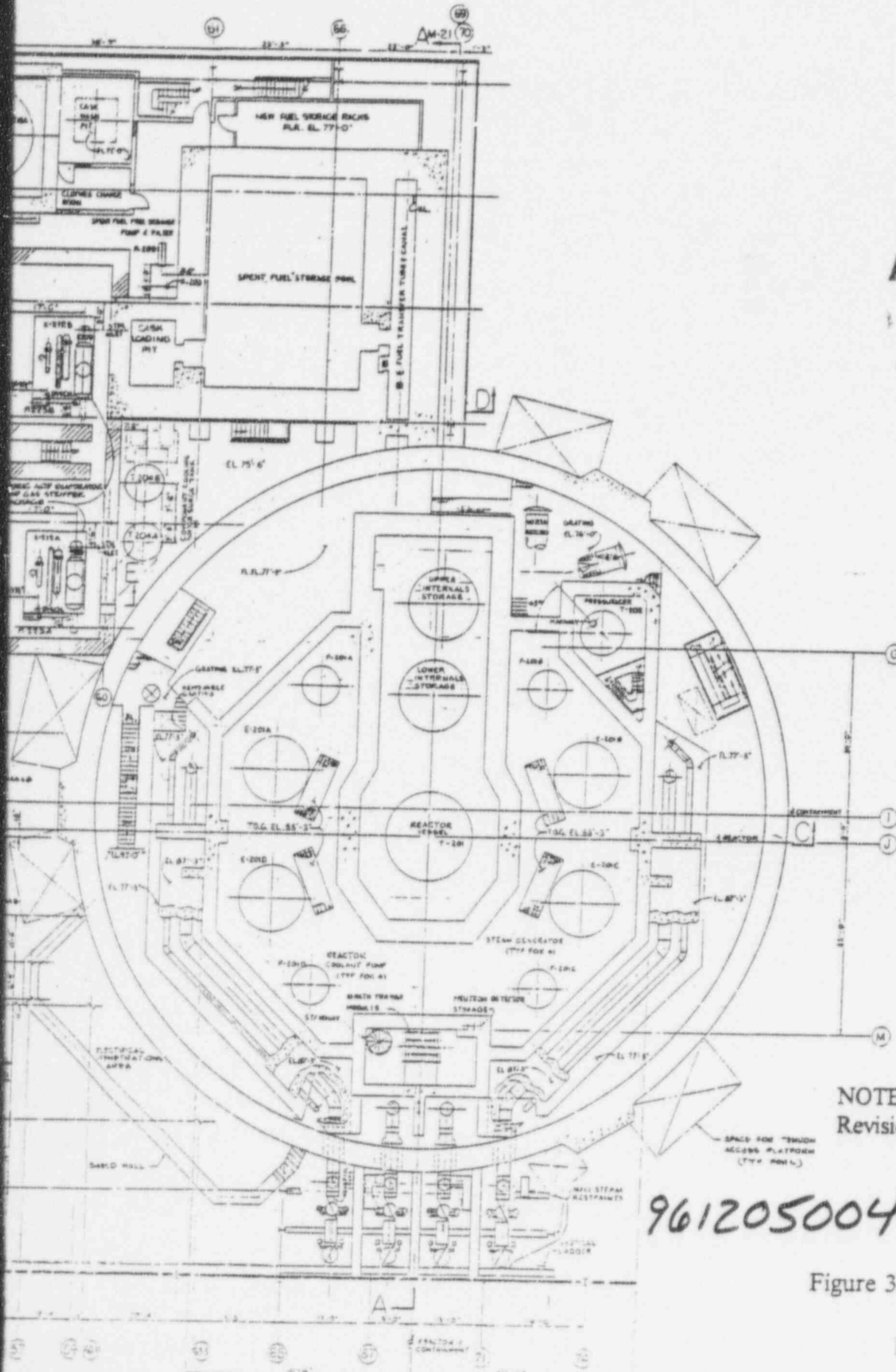
Figure 3.2-17 Equipment Location, Reactor and Auxiliary Buildings - Plan Elevation 6

9612050047- 3

NOTE: Historical Information Only
Revision 4



BORIC ACID TANKS
HOLD-UP TANKS
T-216B
T-216A
OPEN TO EL 58'-0"
RADWASTE CONTROL PANEL
SEALED EMERGENCY DOOR
CONTROLLED CORRIDOR
CONTROL BLDG.
SEE DWG. A-31
CABLE SPREADING
M-22
M-23



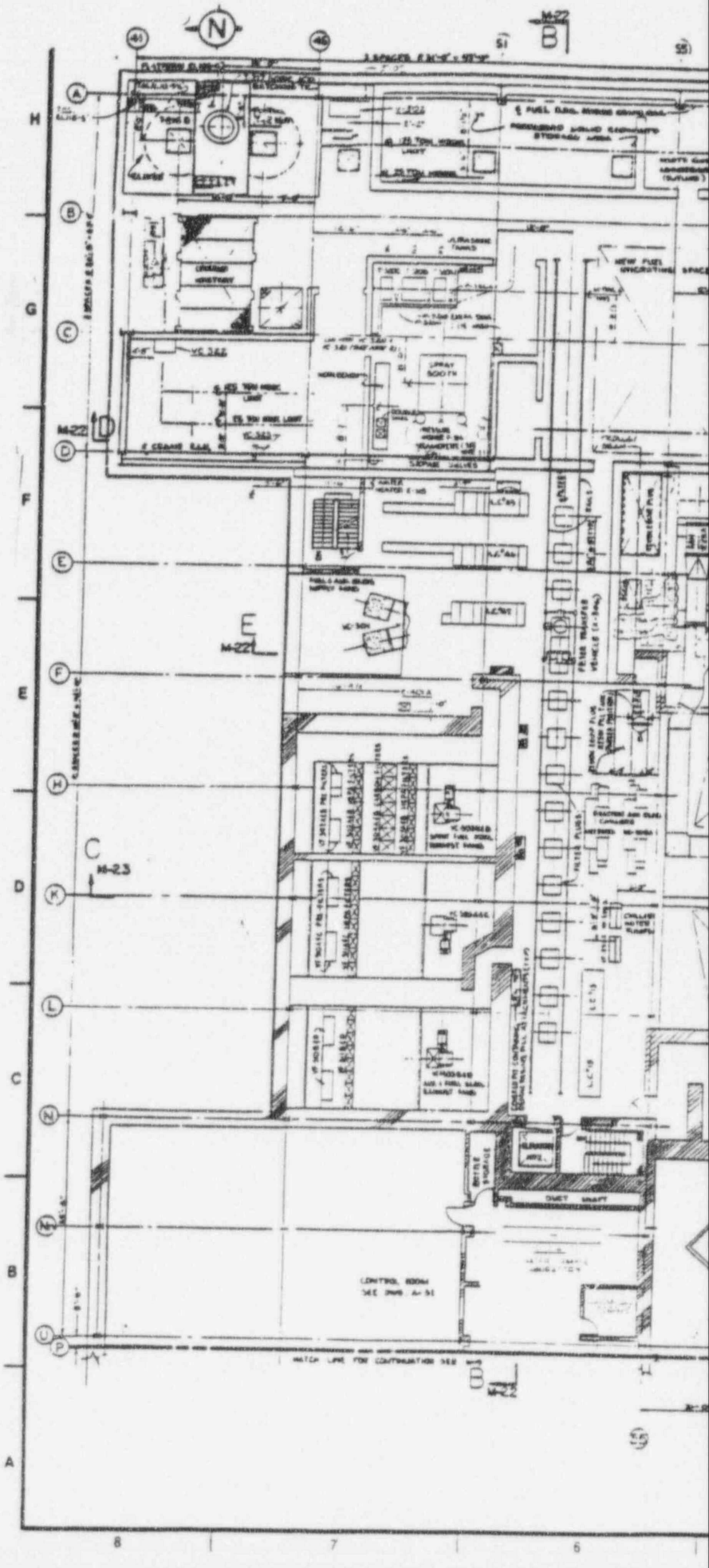
NOTE: Historical Information Only
Revision 4

9612050047- 4

Figure 3.2-18 Equipment Location,

Reactor and Auxiliary

Buildings - Plan Elevation 77'


N
M-22
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D
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C
M-23
CONTROL ROOM
SPRAY BOOTH
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6
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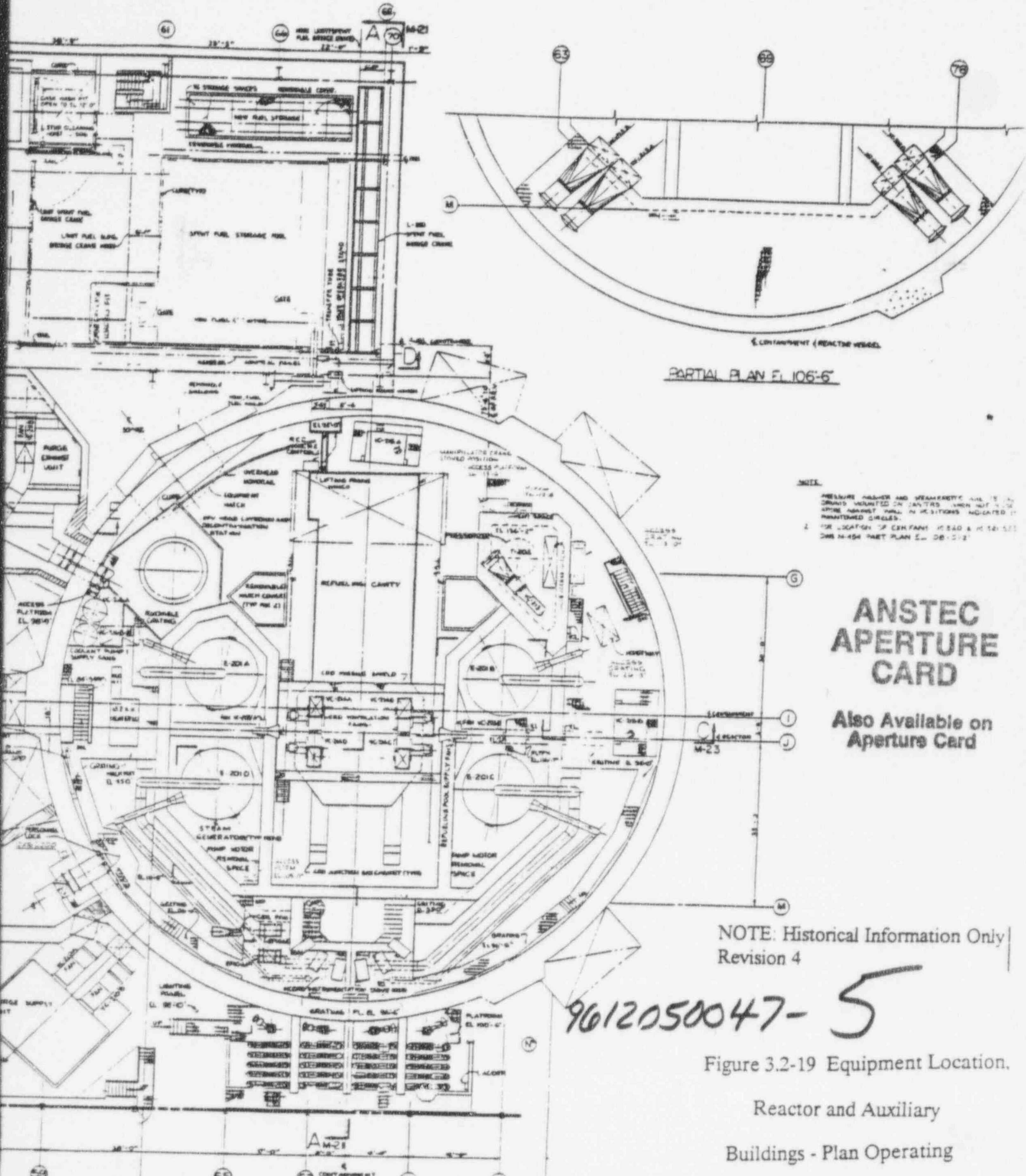
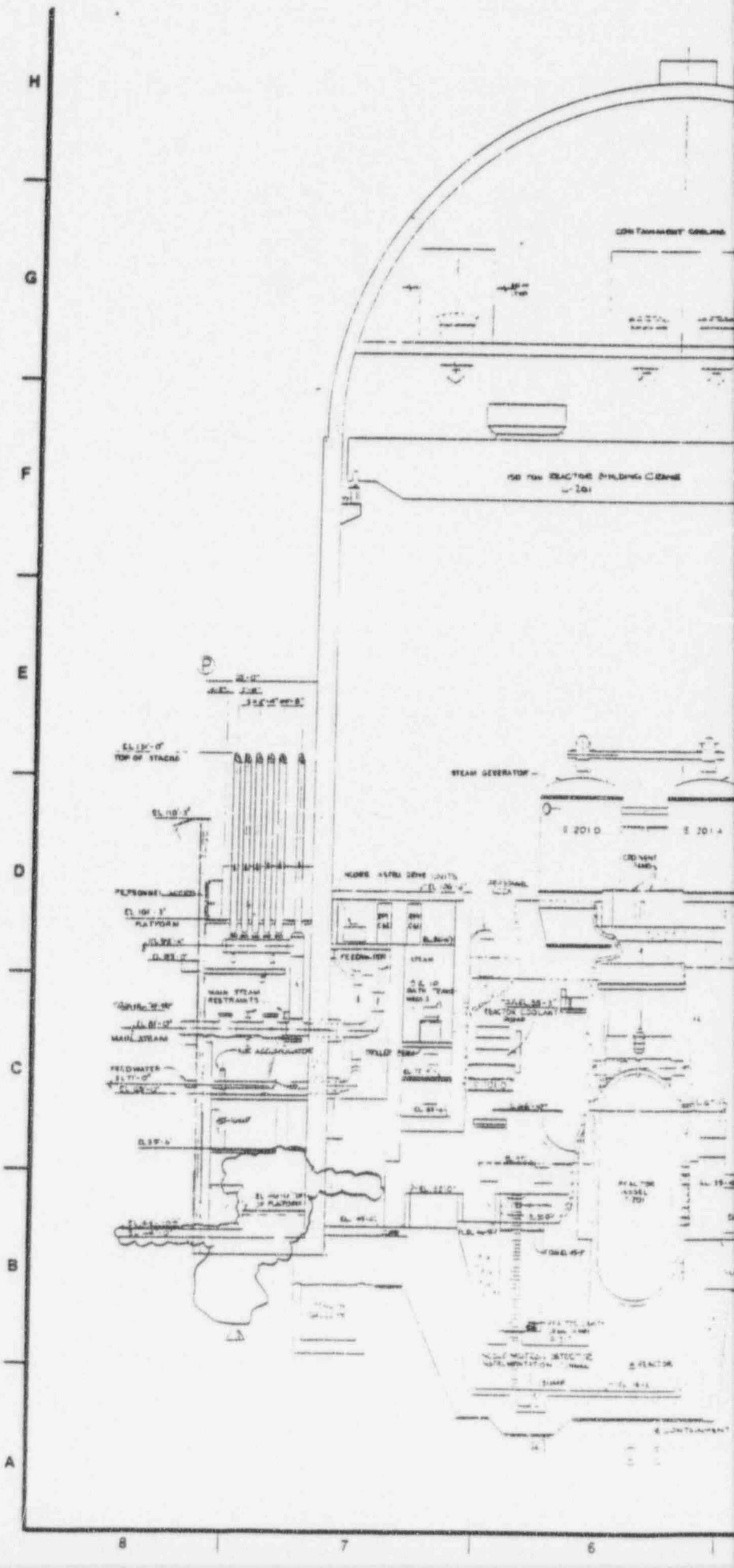


NOTE: Historical Information Only
Revision 4

9612050047- 5

Figure 3.2-19 Equipment Location, Reactor and Auxiliary Buildings - Plan Operating Floor and Above



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STEAM GENERATOR
MAIN STEAM
FEEDWATER
PLATFORM

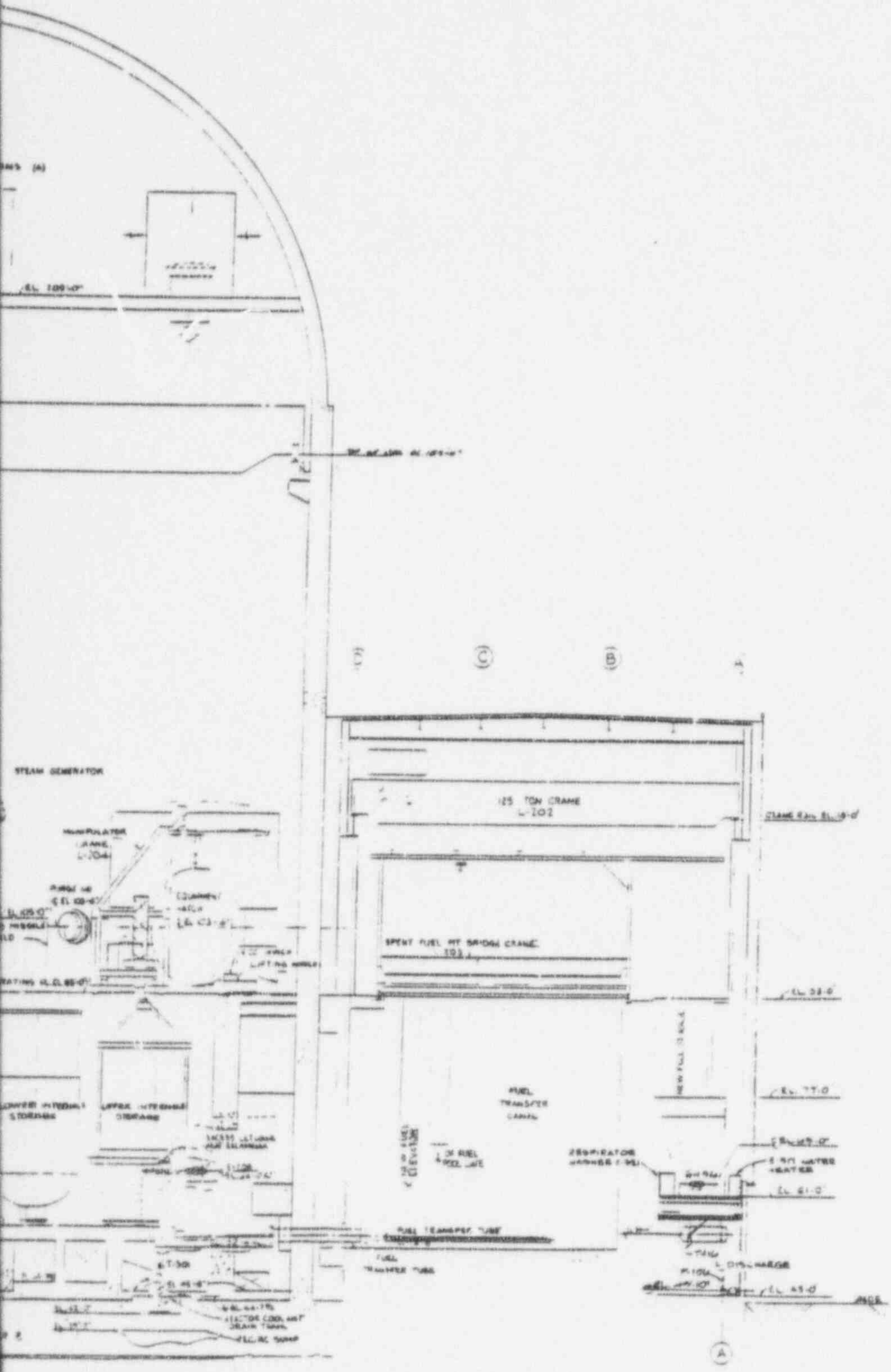

Figure 3.2-20 Equipment Location,

Reactor and Auxiliary

Buildings - Section A-A

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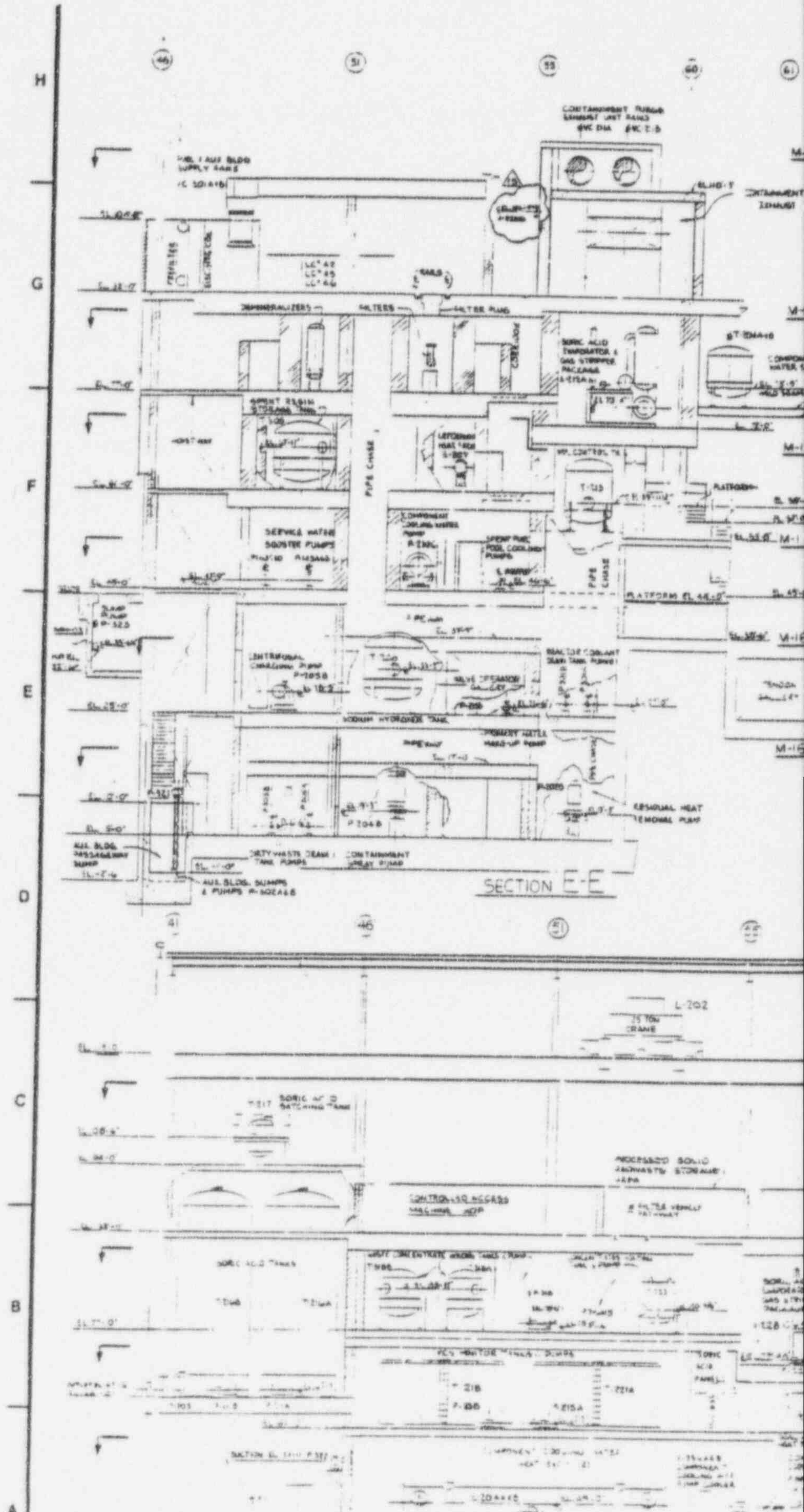
NOTE: Historical Information Only
Revision 4


H
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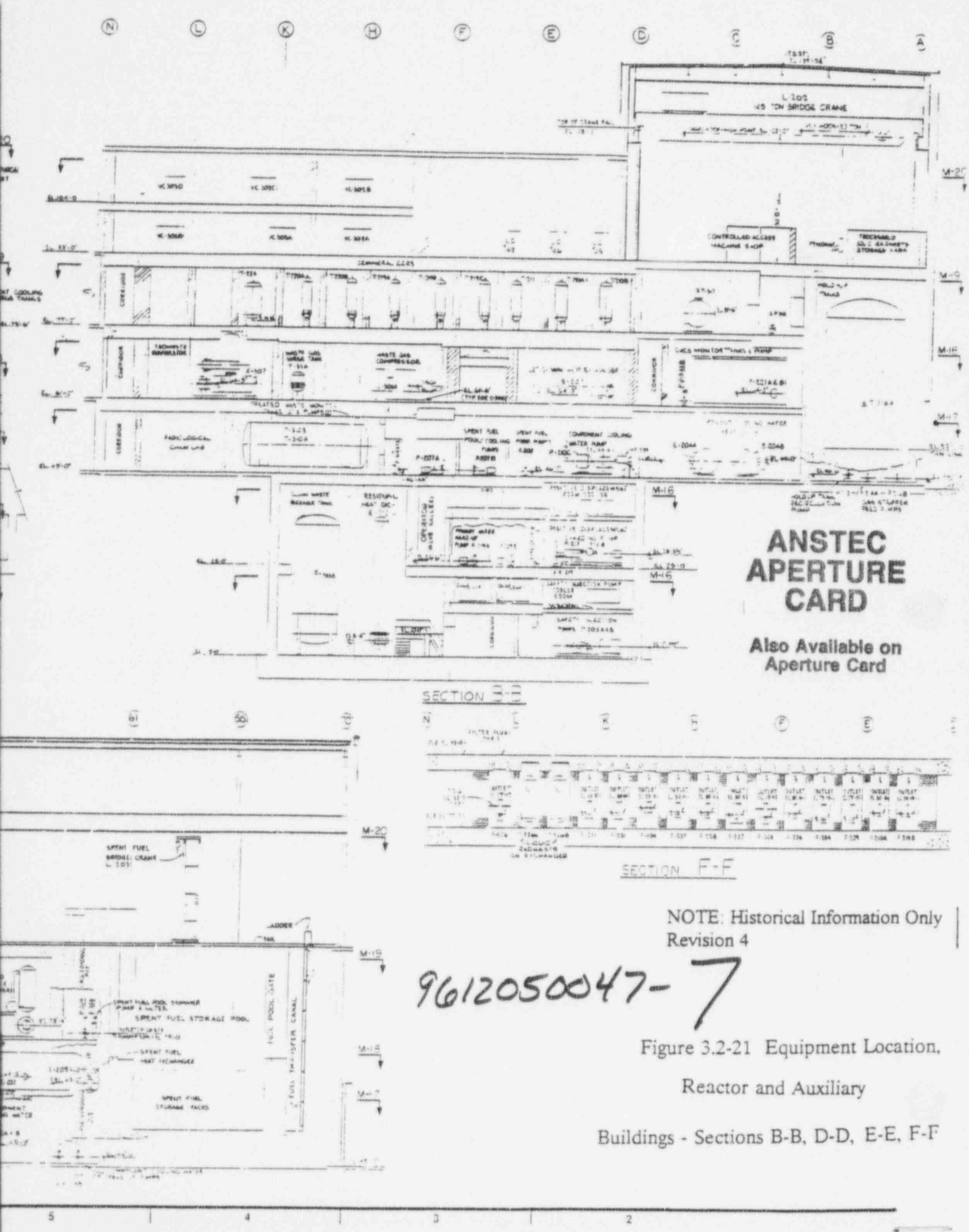
SECTION E-E
PIPE CHASE
SERVICE WATER BOOSTER PUMPS
SODIUM HYDROXIDE TANK
RESIDUAL HEAT REMOVAL PUMP
CONTAINMENT SPRAY PUMP
DIRTY WASTE DRAIN TANK PUMPS
AUX. BLDG. SUMPS & PUMPS
CENTRIFUGAL CHARGING PUMP
SPENT RESIN STORAGE TANK
DEMINERALIZERS
FILTERS
CONTAINMENT PURGE EXHAUST UNIT FANS
L-202
25 TON CRANE
BORIC ACID BATCHING TANK
CONTROLLED ACCESS
BORIC ACID TANKS
SECTION



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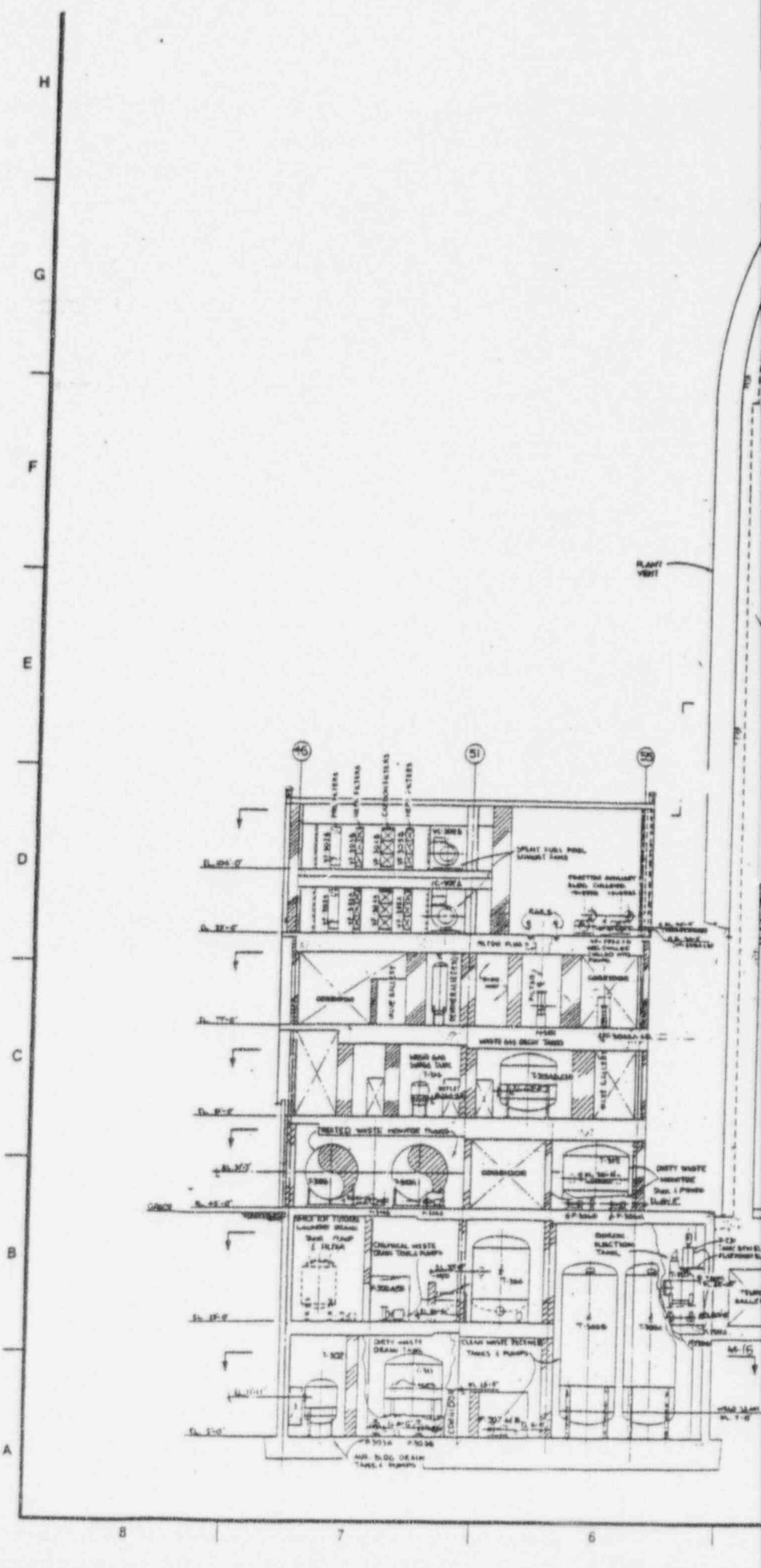
NOTE: Historical Information Only
Revision 4

9612050047-7

Figure 3.2-21 Equipment Location, Reactor and Auxiliary Buildings - Sections B-B, D-D, E-E, F-F



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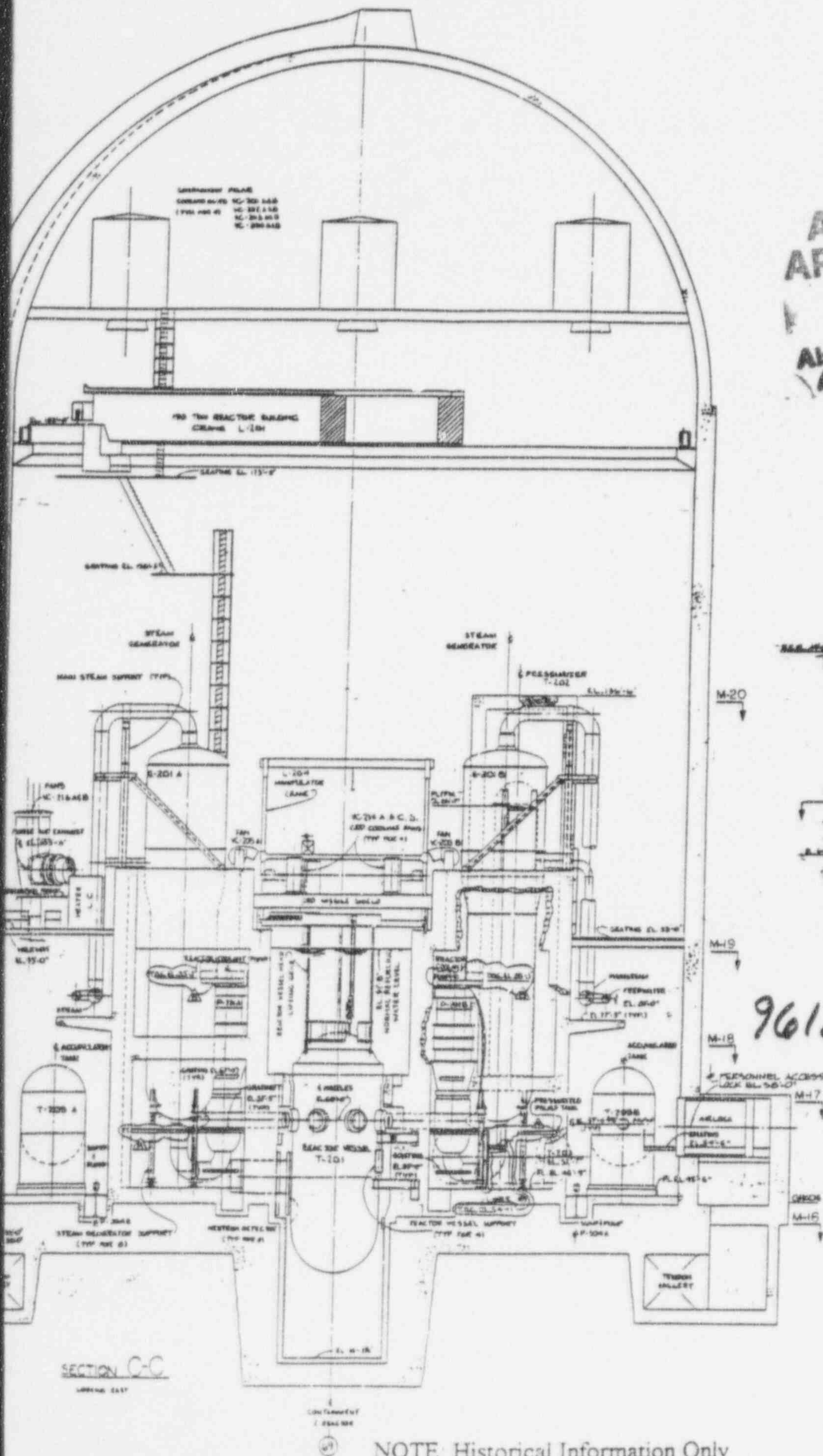
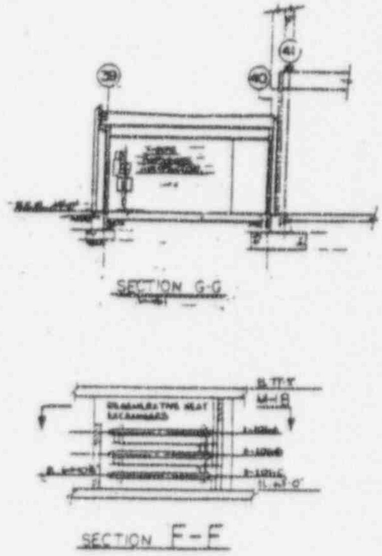
PLANT VENT
TREATED WASTE MONITOR TANKS
GRADE





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9612050047-8

Figure 3.2-22 Equipment Location,

Reactor and Auxiliary

Buildings - Sections C-C and F-F

NOTE: Historical Information Only
Revision 4

Note: Valve Positions are given for normal operation and are for information only

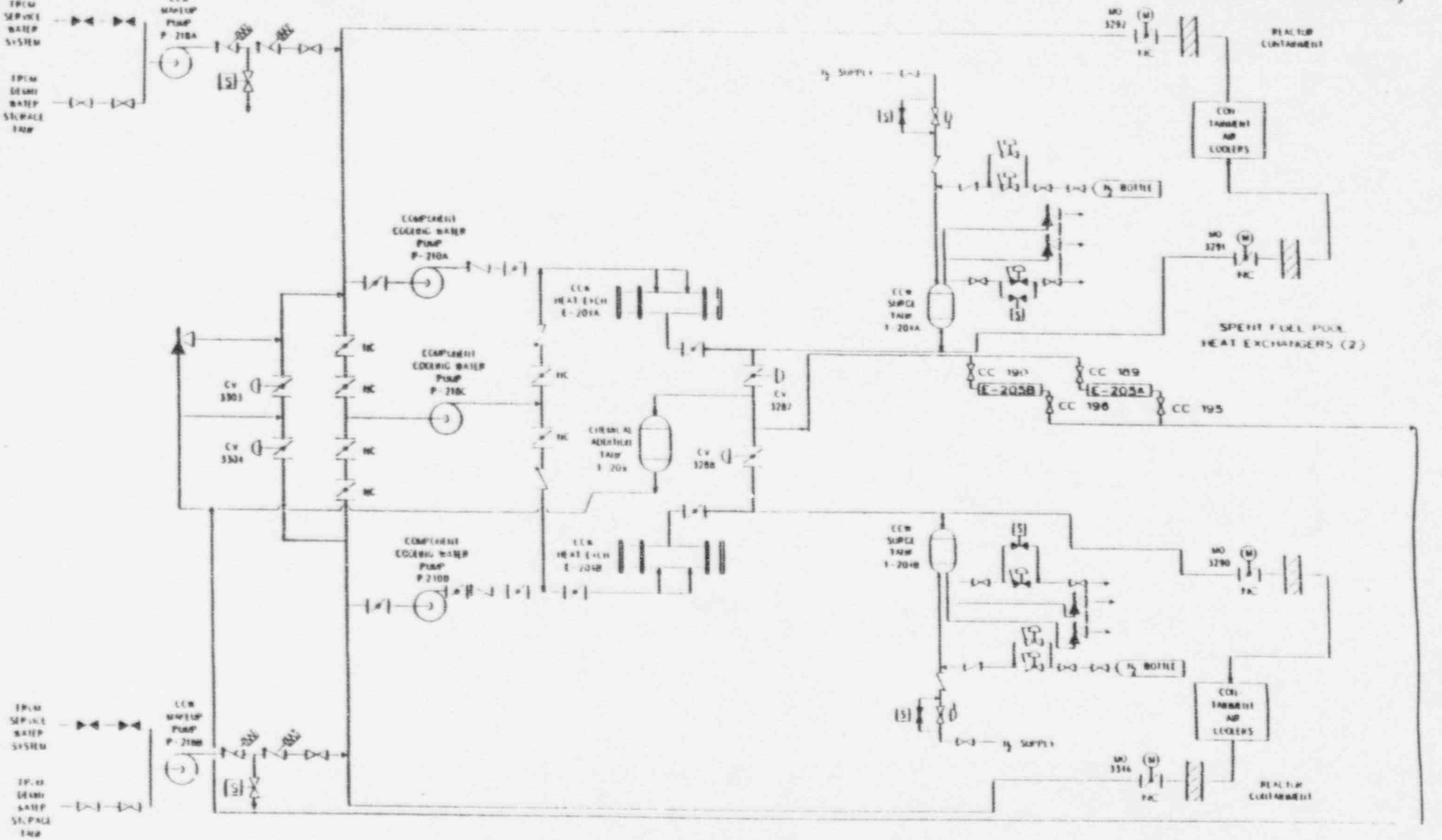


Figure 3.3-2 Component Cooling Water System

Note: Valve Positions are given for normal operation and are for information only

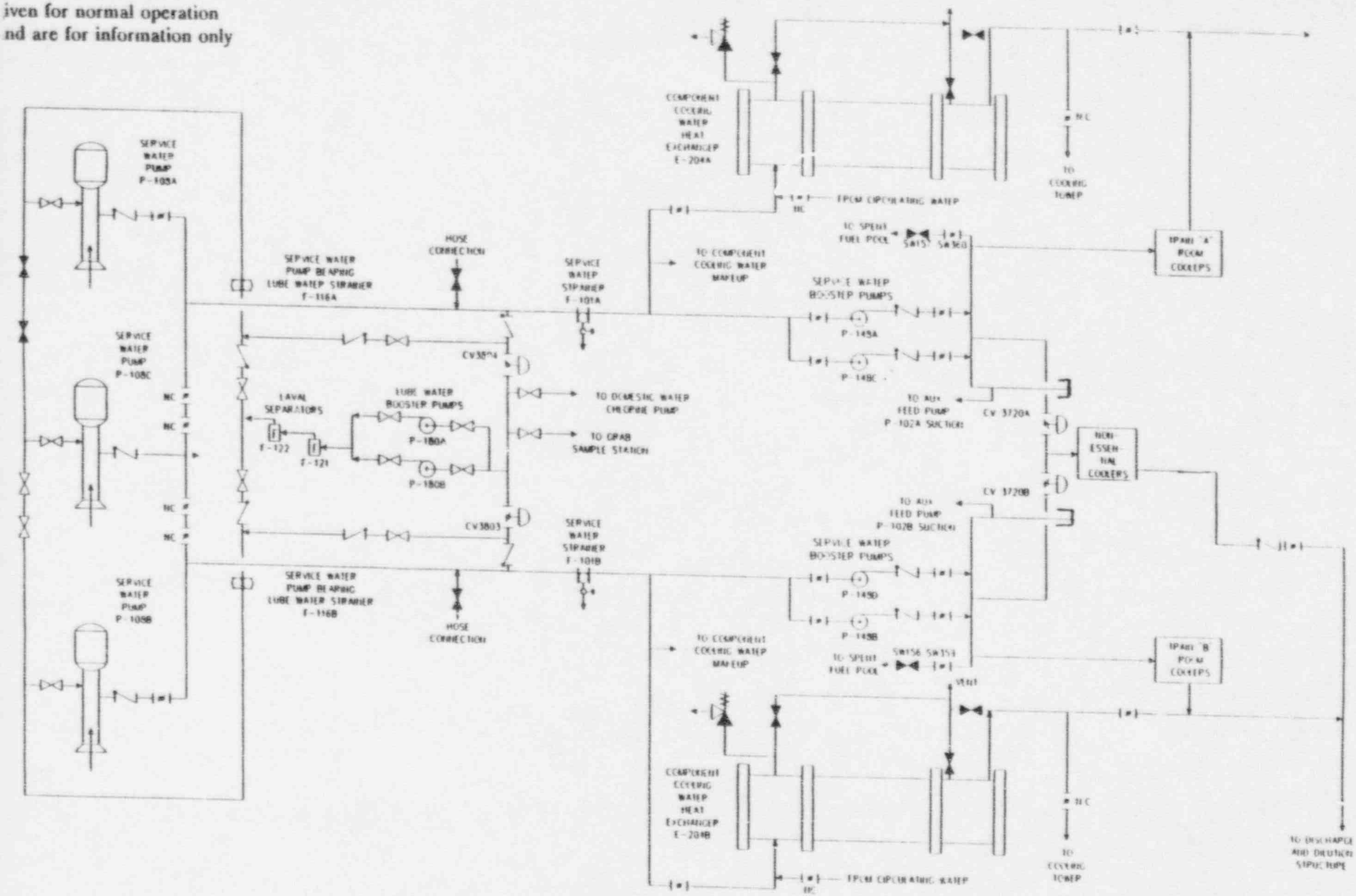


Figure 3.3-3 Service Water System

4.0 OPERATIONS

This chapter discusses facility operations related to the safe storage of irradiated fuel; including criticality prevention, chemistry control, instrumentation, maintenance activities, administrative controls of systems, spent fuel handling, and spent fuel cooling.

To focus proper attention on important equipment and activities during the defueled condition, certain structures, systems, components, and activities have been categorized as important to the safe storage of irradiated fuel.

The following structures, systems, and components are categorized as important to the safe storage of irradiated fuel:

(1) Safety-Related structures (includes the spent fuel racks; Spent Fuel Pool (SFP); fuel transfer canal; cask loading pit; fuel transfer tube; and the Control-Auxiliary-Fuel Building complex).

(2) Quality-Related installed monitoring equipment used to satisfy Trojan Technical Specification (TTS) surveillance requirements (includes SFP level monitoring and SFP temperature monitoring).

(3) Portions of the SFP Cooling and Demineralizer System that support the primary method of SFP cooling or SFP water purification.

(4) Portions of the Component Cooling Water System that support the primary method of SFP cooling.

(5) Portions of the Service Water System that support the primary method of SFP cooling.

(6) Portions of the Service Water, Primary Makeup Water, Demineralized Water, and Fire Water Systems that may provide makeup water to the SFP.

Activities that are important to the safe storage of irradiated fuel are controlled through use of appropriate approved procedures.

4.2 SPENT FUEL HANDLING

4.2.1 SPENT FUEL RECEIPT, HANDLING, AND TRANSFER

License NPF-1 prohibits the movement of a spent fuel shipping cask into the Fuel Building. This precludes the receipt or transfer of spent fuel. Movement of new or spent fuel into the Reactor Building is not authorized without prior NRC approval. The movement of spent fuel assemblies within the SFP is the only fuel handling activity authorized. This movement would be accomplished by use of the spent fuel pool bridge crane and spent fuel handling tool. Since spent fuel handling is limited to relocating assemblies to alternate storage rack locations, inadvertent criticality prevention is provided as described in Section 4.1.1.

4.2.1.1 Functional Description

Figure 4.2-1 provides a layout of the spent fuel storage area. Movement of spent fuel assemblies is accomplished by use of the spent fuel pool bridge crane and spent fuel handling tool.

The SFP bridge crane consists of a wheel mounted walkway spanning the SFP. The SFP bridge crane moves in the north-south direction on rails by means of a two-speed motor. The SFP hoist is a 1-ton capacity electrical monorail hoist that moves in the east-west direction on the overhead structure of the SFP bridge crane. The fuel assemblies are moved within the SFP by means of a long-handled tool suspended from the SFP hoist. The spent fuel handling tool is a manually actuated tool used to handle spent fuel in the SFP. Fuel assembly inserts, such as thimble plugs, burnable poison rods, rod control clusters, and source rods, may also be transferred between positions within the SFP.

Load movements outside of the SFP may be provided by the Fuel Building crane. The Fuel Building crane is a 125-ton pendant-operated crane and is provided with an auxiliary hoist rated at 25 tons. The Fuel Building crane is restricted from moving loads over the SFP. The Fuel Building crane is also limited to moving in a way that avoids the possibility of falling or dropping objects into the SFP. Mechanical stops installed on the rails prevent the crane hook from traveling beyond the centerline of the cask load pit. Electrical limit switches deenergize the bridge drive before the mechanical stops. These stops and limit switches may only be bypassed or removed while following an approved procedure, and then only with the Shift Manager's approval.

4.2.1.2 Safety Features

The SFP bridge crane incorporates design features to minimize the probability of a fuel handling accident. These features are discussed in Section 3.3.1.

Administrative controls in place to minimize the probability and consequences of fuel handling accidents include:

(1) Fuel handling operations can only be performed under the direct supervision of a CFH.

(2) Fuel handling operations must be performed in accordance with approved Plant procedures.

(3) Loads carried over the SFP and the heights at which they may be carried over storage racks containing fuel shall be limited to preclude impact energies over 240,000 in-lbs.

(4) The Fuel Building crane is restricted to operation within the cask movement envelope shown in Figure 4.2-2 for heavy loads, except when following an approved procedure, and then only with the Shift Manager's approval.

4.2.2 SPENT FUEL STORAGE

Section 3.2.2 provides a description of the SFP design including the storage racks. Although SFP storage rack capacity is 1408 fuel assemblies, only 781 fuel assemblies are stored.

The SFP cooling system provides forced cooling of the spent fuel assemblies. Bulk SFP coolant temperature is maintained between 40°F and 140°F. A more thorough description of the operation of the SFP cooling system is provided in Section 4.3.1.

Subcritical arrays are ensured by maintaining center-to-center distance between adjacent fuel assemblies and the fixed neutron absorber contained in the spent fuel storage racks. Additional discussion is provided in Section 4.1.1.

Radiation shielding is provided by the water level maintained in the SFP. A minimum of 23 feet of water is maintained above the top of a spent fuel assembly in the storage racks. During fuel transfer operations, at least 9.5 feet of water is maintained above the top of the active portion of a fuel assembly. This water barrier serves as a radiation shield enabling the gamma dose rate at the pool surface from the spent assembly to be maintained at or below 2.5 mrem/hr.

emergency makeup water source. Systematic guidance is provided by Plant procedures for leak identification and isolation.

The only requirement to assure adequate cooling for the spent fuel is to maintain the water level in the SFP so that the spent fuel elements are not exposed. All lines entering the SFP which could siphon the pool to Elevation 76 feet 7 inches or below (equivalent to approximately 10 feet above stored fuel assembly), are equipped with siphon breakers to limit SFP water loss to an elevation of 83 feet 11 inches. The SFP gates for access to the cask loading pit and refueling canal are equipped with inflatable seals to prevent level loss to these adjacent areas. For a postulated event wherein the SFP is drained to the Siphon Breaker level followed by failure of the SFP gates and subsequent spillage of SFP water to the fuel transfer canal and cask loading pit (initially assumed empty), SFP level would reach a minimum level of 76 feet 7 inches, or 10 feet above the top of the fuel assemblies. Additional discussion of the SFP design to prevent loss of level is contained in Section 3.2.2. Accident analyses for loss of SFP level are contained in Section 6.3.

4.3.1.3 Loss of Spent Fuel Pool Cooling

As discussed in Section 4.3.1, a high temperature alarm is provided in the control room with a maximum setpoint of 135°F. Plant procedures direct actions to restore operation of SFP cooling. As stated in Section 4.3.1.2, the only requirement to assure adequate cooling for the spent fuel is to maintain the water level in the SFP such that the spent fuel elements are not exposed. Procedure guidance directs the restoration of level by the use of makeup to the pool in the event of level loss from boil off. Restoration of SFP level is discussed in Section 4.3.1.2, Loss of Spent Fuel Pool Level.

4.3.1.4 High Spent Fuel Pool Level

A level switch has been provided in the SFP for the purpose of transmitting high and low water level annunciation signals to the control room. Plant procedures are in place that provide guidance for the systematic determination and isolation of the water source. Overflow from the SFP is directed to the dirty waste drain tank.

4.3.1.5 Safety Criteria and Assurance

The only requirement to assure adequate cooling for the spent fuel is to maintain the water level in the SFP so that the spent fuel elements are not exposed. The top of fuel seated in the spent fuel storage modules is located at Elevation 66 feet 7 inches. Dose calculations for fuel handling accidents assume a minimum of 23 feet of water above the top of spent fuel assemblies; therefore, SFP level less than Elevation 89 feet 7 inches requires operator action to restore level. This minimum required level also ensures dose rates at the SFP surface will be maintained <2.5 mrem/hr during fuel movements.

A SFP minimum boron concentration of 2000 ppm is assumed for certain design basis accidents to prevent inadvertent criticality. Sampling frequency and corrective actions to ensure boron concentration is maintained have been implemented.

A maximum temperature limit of 140°F was assumed for SFP bulk temperature. This limit ensures that for the worst case loss of SFP cooling accident, SFP boiling will not occur within 20 hours, and that at least 4 days are available to establish makeup to the SFP before boil off would reduce SFP level to 5 feet above the fuel assemblies. Maintaining 5 feet of water above the active portion of the fuel provides adequate radiation shielding to allow access for restoration of SFP level. Table 6.3-2 provides estimates of radiation dose rates for various SFP levels.

4.3.2 ELECTRICAL DISTRIBUTION

During normal Plant operation, Plant electrical demands are provided from the 230-kV switchyard. The design and reliability of the offsite power source are discussed in Section 3.4.1.

Offsite power is normally supplied to the Plant distribution system via the startup transformers.

In the event of loss of offsite power, forced cooling capability would be lost. Without restoration of forced spent fuel cooling, boiling of the SFP would not occur for at least 88 hours. SFP inventory is adequate to allow boiling for a minimum of 2[illegible].5 days without any source of makeup and still maintain >10 feet of level above the top of the spent fuel assemblies. Makeup capability to the spent fuel coolant system independent of Plant power sources can be provided by the fire main system (diesel-driven fire pump). A discussion of the fire main system is contained in PGE-1012, "Trojan Nuclear Plant Fire Protection Program."

4.3.3 SUPPORT SYSTEMS

Operation of the SFP cooling system is supported by the CCWS and SWS. Decay heat removed from the SFP cooling system is rejected to the CCWS via the SFP heat exchangers. The SWS provides heat rejection of the CCWS to the Columbia River. Decay heat may also be rejected to Containment via the CCWS and Containment Air Coolers. Other Plant support systems include ventilation, radiation monitoring, and radioactive waste systems, which are discussed in Chapter 5.

The CCWS interfaces with the SFP cooling system providing for heat rejection of the spent fuel decay heat via a SFP cooling water heat exchanger. Normal operation consists of one component cooling water pump in operation. As a result of Plant defueling, the in-service SFP heat exchanger is the only required component cooling water load. Component cooling

water flow to the SFP heat exchanger may be throttled or secured to allow for maintaining SFP temperature between 40°F and 140°F. Loss of the CCWS can result in a loss of SFP cooling which is discussed in Section 4.3.1.3.

The loads and design basis of the SWS are discussed in Section 3.3.4. In addition to providing heat rejection for the CCWS, the SWS provides a makeup water source for the SFP.

4.4 CONTROL ROOM AREA

The control room provides a centralized station to monitor operation of the SFP cooling and support systems. At least one person qualified to stand watch in the control room shall be present in the control room when irradiated fuel is stored in the SFP. A discussion of the activities normally performed by the control room operator is provided below.

SFP cooling instrumentation available in the control room includes: SFP temperature indication, SFP high temperature alarm, and SFP level alarm (high/low). In the event control room instrumentation becomes unavailable, the control room operator can utilize local monitoring of spent fuel pool temperature and level. All operation of the SFP cooling system, including normal and emergency makeup, is performed locally since remote operation of SFP components and makeup flow paths is not provided in the control room.

Operation of the Plant's electrical distribution system is normally controlled and monitored in the control room. Local operation of the electrical distribution switch gear and load breakers can be performed if required. Operation and monitoring of the CCWS and SWS is normally performed in the control room. In the event control room indication and control become unavailable, redundant control of the CCWS and SWS components is provided locally.

Additional systems that may be monitored or controlled in the control room include:

(1) Plant Ventilation.

(2) Area Radiation Monitors.

(3) Process Radiation Monitors.

(4) Fire Protection.

4.5 REFERENCES

REFERENCES FOR SECTION 4.1

1. Trojan Nuclear Plant, Final Safety Analysis Report, through Amendment 19 (December 1992).

2. PGE-1012, "Trojan Nuclear Plant Fire Protection Program."

REFERENCES FOR SECTION 4.2

1. Trojan Operating License, No. NPF-1, as amended through Amendment 191 dated May 6, 1993.

2. Trojan Nuclear Plant, Final Safety Analysis Report, through Amendment 19 (December 1992).

3. PGE 1037, "Trojan Nuclear Plant Spent Fuel Storage Rack Replacement Report."

4. PGE Calculation, Trojan Spent Fuel Pool Heatup and Boil Down Bases, TC-720 Revision 4, dated 11/12/96.

REFERENCES FOR SECTION 4.3

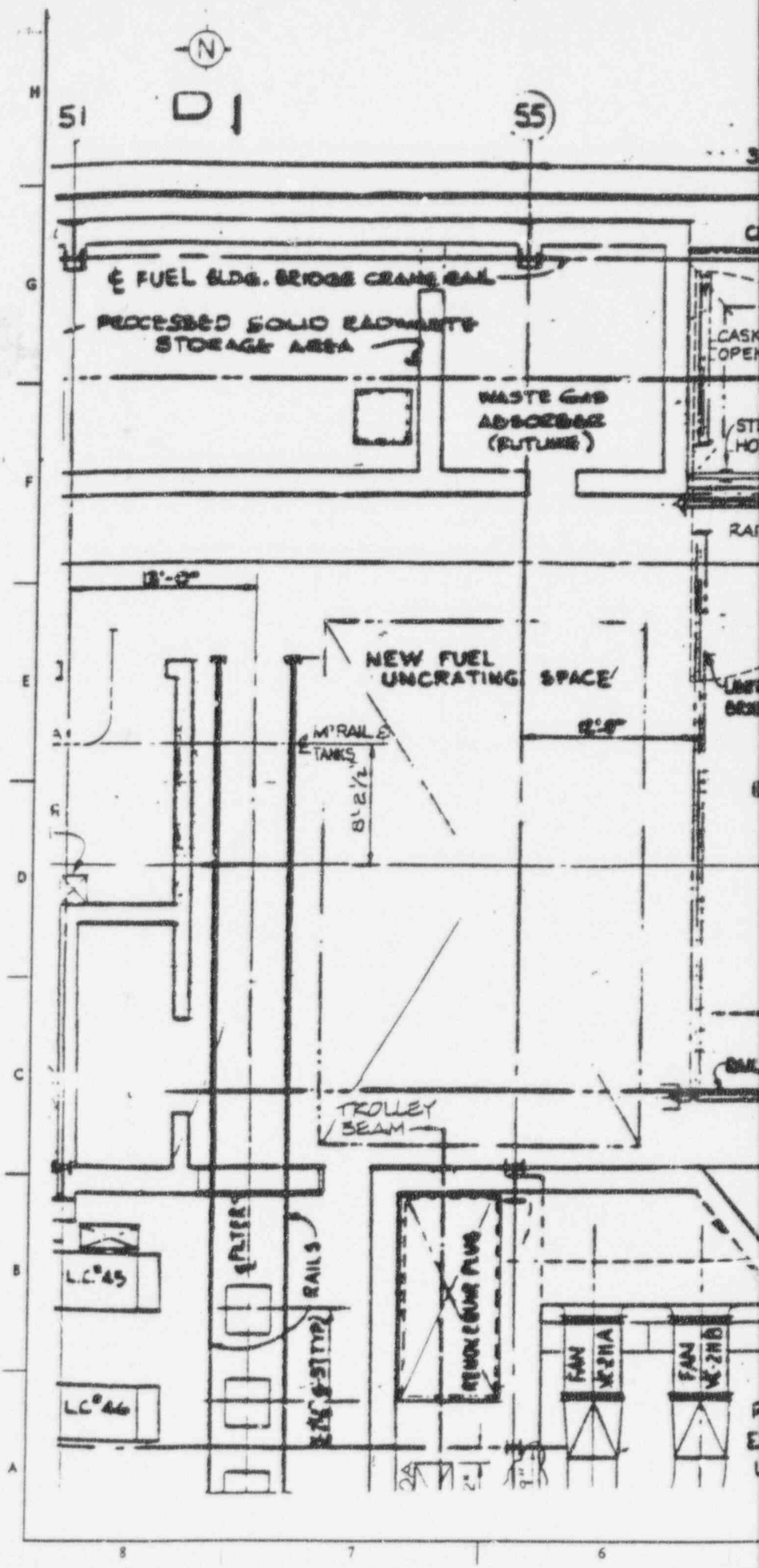
1. Trojan Nuclear Plant, Final Safety Analysis Report, through Amendment 19 (December 1992).

2. PGE-1012, "Trojan Nuclear Plant Fire Protection Program."

TABLE 4.1-1
SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS		MEASUREMENT RANGE	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	NOTES
TRIAXIAL TIME-HISTORY RECORDING ACCELEROGRAPHS						
ST-6336A	(Containment Base Slab)	0-2 g	M	18M	SA	1, 3, 4
ST-6336B	(Containment Wall)	0-2 g	M	18M	SA	1, 3, 4
ST-6336C	(Fuel Bldg, Elev. 93')	0-2 g	M	18M	SA	1, 3, 4
ST-6336D	(Cable Spreading Room)	0-2 g	M	18M	SA	1, 3, 4
ST-6336E	(Free Field)	0-2 g	M	18M	SA	1, 3, 4
TRIAXIAL PEAK ACCELEROGRAPHS						
SR-6340A	(Containment Base Inside, 45' North)	0-2 g	NA	18M	NA	None
SR-6340B	(Intake Building Roof, South)	0-2 g	NA	18M	NA	None
SR-6340C	(Top of Containment Inside, 201' 9" South)	0-2 g	NA	18M	NA	None
SR-6340D	(Control Building Mezzanine, Top of Ladder Above Secondary Sample Storage Room)	0-2 g	NA	18M	NA	None
SR-6340E	(Fuel Building, 93' Hot Shop, West, Behind the Wall)	0-2 g	NA	18M	NA	None
SR-6340F	(CCW Heat Exchangers, 45' Area 3 Base)	0-2 g	NA	18M	NA	None
SR-6340G	(West EDG Room)	0-2 g	NA	18M	NA	None
TRIAXIAL RESPONSE-SPECTRUM RECORDERS						
SR-6341	(Containment Foundation)	NA	M	18M	SA	2, 3, 5

NOTE: FIGURE 9.1-1 IS TAKEN FROM
M-20 SHT.1. REVISION 19.



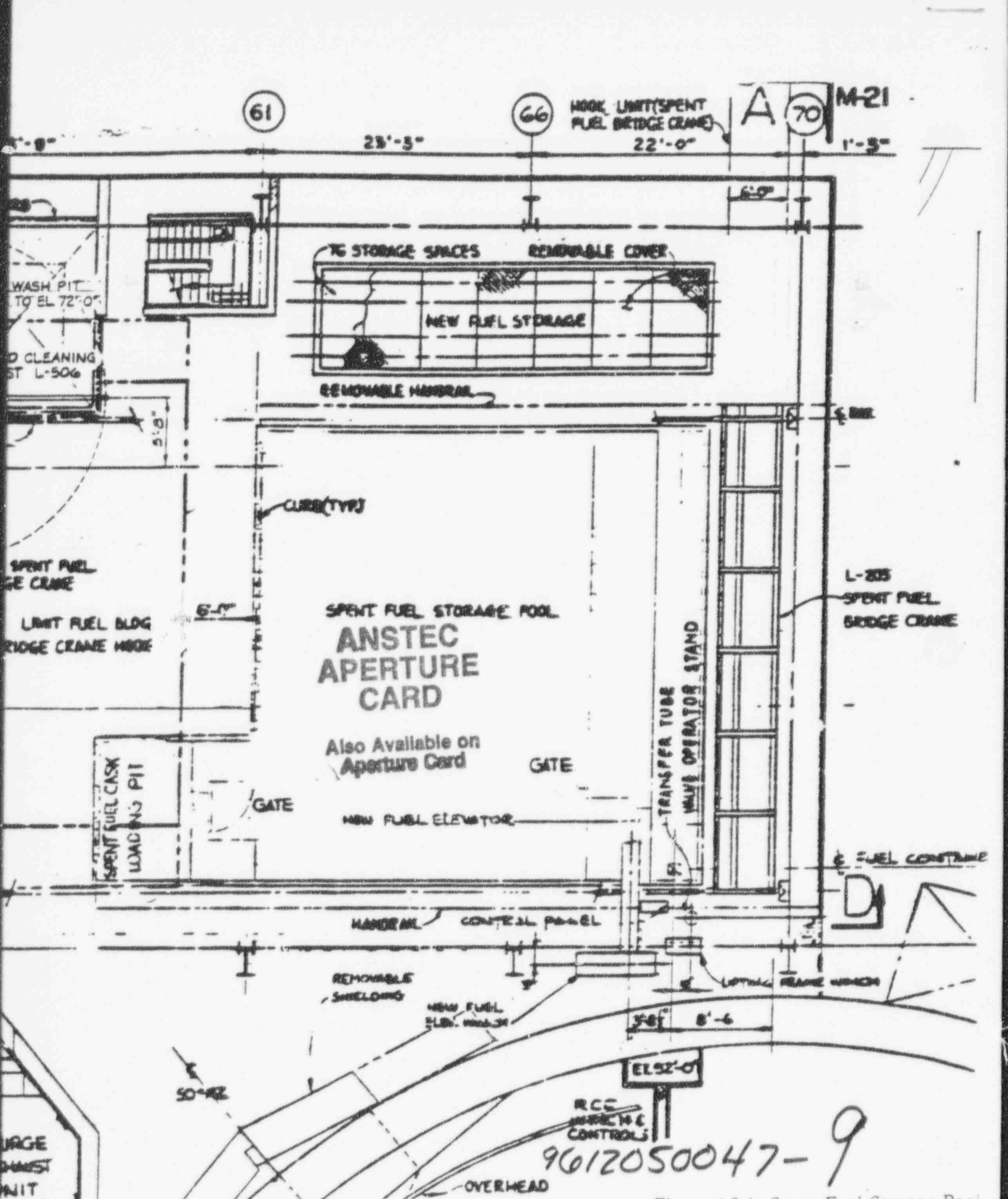


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Figure 4.2-1 Spent Fuel Storage Pool

Revision 4

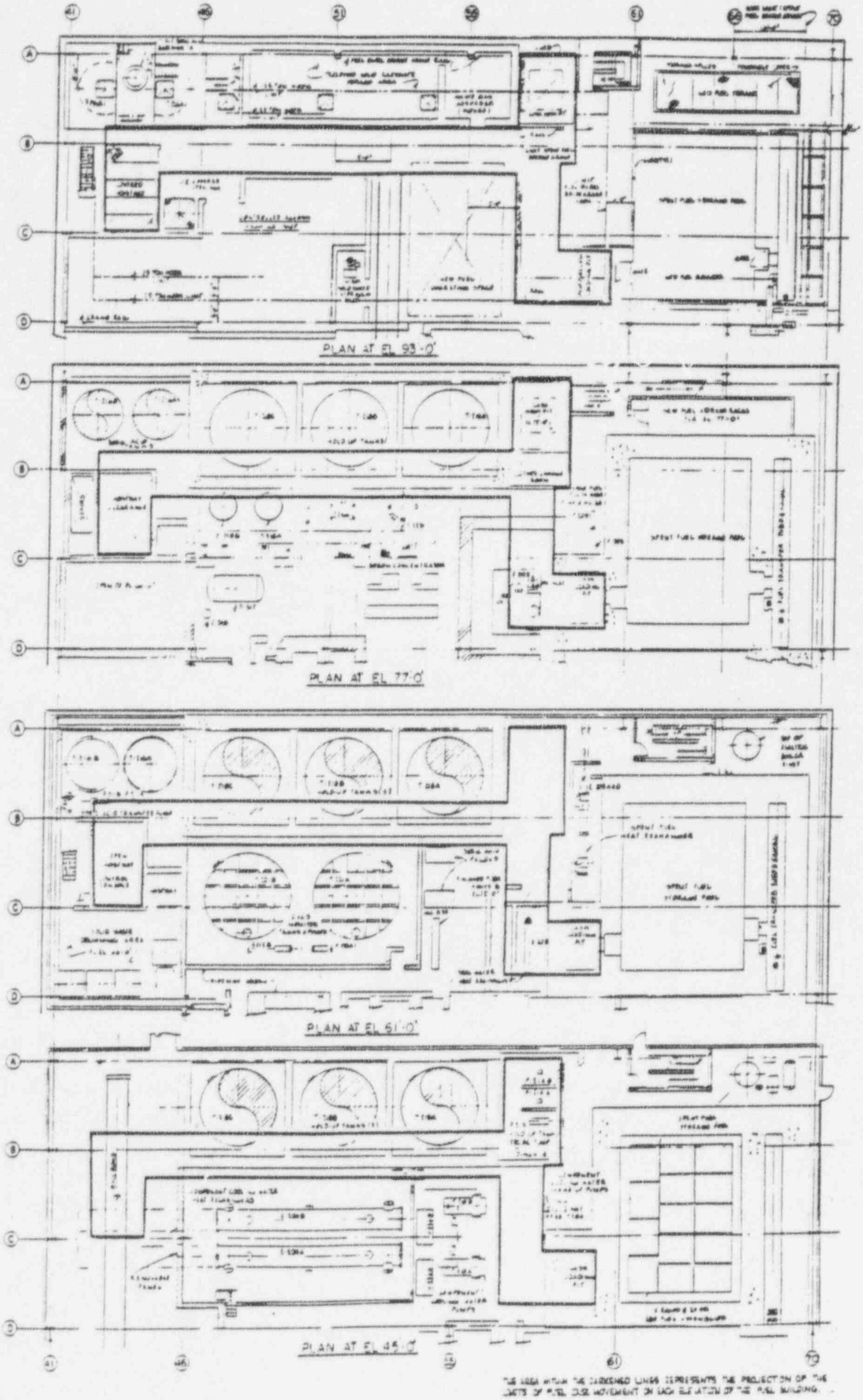

Figure 4.2-2 Plant Arrangement Diagram of Fuel Cask Movement Envelope

operating time at full power which was assumed to occur at the end of each month. This is conservative in that it places more fission events near the end of each month and allows less time for decay to occur during a month. SFP decay heat generation for a decay time of 40 weeks after shutdown was utilized in the analysis. Figure 6.3-1 demonstrates the effects of decay time on the heat generated in the pool which increases the safety margin with increasing time since reactor shutdown.

The evaluation showed that under the assumed conditions and at 3½ years after shutdown SFP boiling would not occur for at least 88 hours. Boil off rate would be no greater than 3.7 gpm and at least 540 hours (22.5 days) would be available to establish a make-up source and still maintain at least 10 feet of water level over the fuel. Figure 6.3-2 provides a graph of SFP heatup rate versus time after reactor shutdown. This graph demonstrates that additional response time becomes available as time from reactor shutdown increases.

Figure 6.3-3 provides a graph of time available prior to SFP boiling based on the time after shutdown. The time to boil is based on an initial SFP temperature of 140°F. Figure 6.3-4 provides a graph of calculated boil off rate based on time after reactor shutdown. Figure 6.3-5 provides the makeup rate (based on a makeup water temperature of 100 degrees F) that would be needed to maintain SFP level constant until forced SFP cooling could be restored. Time available to establish a makeup source prior to SFP pool level being reduced to 10 feet above the fuel due to boil off (i.e. no inventory loss from system leakage) is provided in Figure 6.3-6. For inventory losses due to boil off only, more than 3 weeks are available to establish a makeup water source.

6.3.3 LOSS OF FORCED SPENT FUEL COOLING WITH CONCURRENT SFP INVENTORY LOSS

Certain events exist which could result in loss of inventory of the SFP. The events that could result in loss of inventory are discussed in Section 6.3.1.1. Regardless of the initiating event a bounding set of conditions can be established and the bounding

consequences evaluated. Three events were identified in Section 6.3.1.1 that could result in a loss of inventory. An explosion, a seismic event, and SFP cooling system pipe failure were considered to have the potential to affect the integrity of the SFP cooling system. The seismic event is considered to be the most bounding since this event has the potential to affect all components associated with SFP cooling system including support systems. The following bounding initial conditions were assumed to exist at the time of a seismic event:

(1) Initial SFP temperature was 140°F.

(2) The fuel transfer canal is empty with gate closed.

(3) The fuel cask pit is empty with the gate closed.

The seismic event is assumed to cause a failure in the SFP cooling system which results in the instantaneous draining of the SFP to the level of the siphon breakers. The gates separating the fuel cask pit and fuel transfer canal then fail resulting in additional loss of level until equalized with the SFP level. This sequence of events, though unlikely, is conservative in that it maximizes SFP inventory loss. The resultant SFP level for this event would be 76 feet 10 inches or 10 feet 3 inches above the fuel. The SFP cooling system and SFP structure design are discussed in Chapter 3. Based on this design criteria this event is considered to be the maximum credible inventory loss prior to boiloff. The remaining discussion will demonstrate that adequate time is available to establish a source of makeup water to the SFP such that uncovery of the fuel and loss of spent fuel cooling is not credible.

Heatup rates were conservatively calculated assuming a starting water level of only 10 feet above the fuel. To account for the possibility that pool heat up may not be uniform additional conservatism was incorporated by reducing the available inventory by 10 percent. Credit was not taken for heat loss to the pool walls or for water available in

the fuel cask loading pit or fuel transfer canal. The time for the SFP to boil was conservatively calculated to be 49 hours. The boil off rate would then be no greater than 3.7 gpm. At least 10 days are available to establish a makeup water source to the pool prior to a level reduction to 5 feet above the fuel. This calculation showed that personnel access for recovery actions was possible with the coolant level reduced to 5 feet above the fuel.

Calculations were performed to determine maximum expected dose rates in the area above the SFP. Table 6.3-2 provides the results of these analysis.

Recovery of SFP level and cooling are discussed in Section 4.3.1.

6.4 REFERENCES

REFERENCES FOR SECTION 6.0

1. J. J. DiNunno, et al, Calculation of Distance Factors for Power and Test Reactor Sites, TID-14844 (March 1962).

2. M. E. Meek and B. F. Rider, Summary of Fission Product Yields for U-235, U-238, Pu-239, and Pu-241 at Thermal Fission Spectrum and 14 Mev Neutron Energies, APED-5398 (March 1968).

3. D. F. Toner, and J. S. Scott, "Fission-Product Release from UO2", Nuc. Safety 3, No. 2 (December 1961) pp 15-20.

4. J. Belle, "Uranium Dioxide Properties and Nuclear Applications", Naval Reactors, Division of Reactor Development United States Atomic Energy Commission (1961).

5. A. H. Booth, A Suggested Method for Calculating the Diffusion of Radioactive Rare Gas Fission Products From UO2 Fuel Elements, DCI-27 (1957).

6. "Report of ICRP Committee II Permissible Dose for Internal Radiation 1959", Health Physics, 3 (1960) p 30, 146-153.

7. C. M. Lederer, et al, Table of Isotopes, 6th ed. (1968).

8. Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors, Regulatory Guide 1.4, Directorate of Regulatory Standards, U.S. Atomic Energy Commission (June 1973).

9. J. J. DiNunno, et al, Calculation of Distance Factors for Power and Test Reactor Sites, TID-14844, U.S. Atomic Energy Commission (March 1962).

10. W. K. Brunot, EMERALD - A Program for the Calculation of Activity Releases and Potential Doses from a Pressurized Water Reactor Plant, Pacific Gas & Electric Company (October 1971).

11. W. K. Brunot, C. L. Beard and R. J. Lutz, PREL - A Program for the Calculation of Activity Release from a Reactor System, WCAP-7461, Westinghouse Electric Corporation, Proprietary (April 1971).

12. T. K. Shen, W. K. Brunot and R. J. Lutz, WEDOSE - A Program for the Calculation of Potential Off-Site Doses from a Reactor System, WCAP-7460, Westinghouse Electric Corporation, Proprietary (April 1971).

13. Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors, Regulatory Guide 1.4, Directorate of Regulatory Standards, U.S. Atomic Energy Commission (June 1973)

14. D. H. Slade, Meteorology and Atomic Energy, TID-24190, U.S. Atomic Energy Commission (1972).

REFERENCES FOR SECTION 6.1

1. Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure, Regulatory Guide 1.24 (Safety Guide 24), Directorate of Regulatory Standards, U.S. Atomic Energy Commission (March 23, 1972).

2. Basis for Fuel Handling Accident as Limiting - Case Accident, PGE Calculation RPC 93-025.

3. Seismic Design Classification, Regulatory Guide 1.29, Revision 3, U.S. Nuclear Regulatory Commission (September 1978).

REFERENCES FOR SECTION 6.2

1. Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors, Regulatory Guide 1.25, U.S. Atomic Energy Commission (March 23, 1972).

2. Basis for Fuel Handling Accident as Limiting-Case Accident, PGE Calculation RPC 93-025.

3. Doses at Existing & Proposed Site Boundary from Fuel Handling Accident after 6 Months Decay, PGE Calculation RPC 93-003 Revision 1 dated 3/9/93.

REFERENCES FOR SECTION 6.3

1. Trojan Spent Fuel Pool Heatup and Boil Down Rates, PGE Calculation TC-720 Revision 4 dated 11/12/96.

2. Dose Rate Vs. Water Height Over Spent Fuel in Pool, PGE Calculation RPC 93-24, Revision 0 dated 9/16/93.

7.1 PGE ORGANIZATIONAL STRUCTURE

7.1.1 MANAGEMENT AND TECHNICAL SUPPORT ORGANIZATION

This section describes the organization of Portland General Electric Company (PGE) as related to the operation of the Trojan Nuclear Plant.

7.1.2 NUCLEAR DIVISION

The Nuclear Division consists of Plant Organizations, Supporting Organizations, and Review and Audit Organizations.

7.1.2.1 Plant Organizations

Trojan Site Executive and Plant General Manager - responsible for the management of the Plant. The Trojan Site Executive and Plant General Manager (hereafter referred to as General Manager, Trojan Plant), reports to the Senior Vice President, Power Supply, and has responsibility for operation of the facility in a safe, reliable and efficient manner. He exercises administrative direction to personnel assigned to the Plant and coordinates the activities of Nuclear Oversight, Engineering/Decommissioning, Plant Support, Technical Functions, and Plant personnel, including Operations, Maintenance, and Personnel/Radiation Protection.

He is responsible for the operation of Plant equipment, proper maintenance, and establishment of the radiation requirements of the Plant.

Manager, Maintenance - reports to the General Manager, Trojan Plant, and is responsible for the proper maintenance of the Plant. He ensures that Maintenance implements the ALARA program in maintenance activities to maintain occupational doses ALARA and to minimize radioactive effluent and solid waste generation.

Manager, Operations - reports to the General Manager, Trojan Plant, and is responsible for the operation of Plant equipment. He ensures that the Plant is operated in accordance with the requirements of the operating license and the procedures of the Nuclear Division Manual. He ensures that Operations implements the ALARA program in work activities to maintain occupational doses ALARA and to minimize radioactive effluent and solid waste generation

Shift Manager - reports to the Manager, Operations, and is responsible to ensure the safe operation of the Plant equipment. He is responsible for the safety of shift personnel and assures Plant equipment is in a safe condition before allowing maintenance activities to proceed. He has the final approval authority for radioactivity releases.

Manager, Personnel/Radiation Protection - responsible to the General Manager, Trojan Plant, for chemistry, radiation protection, the ALARA program, emergency planning, nuclear fuel management, and safety. He is responsible for the onsite industrial safety program, and the Plant chemistry control and radiation protection programs.

7.1.2.2 Supporting Organizations

The Supporting Organizations consist of four main groups supporting Plant activities: Engineering/Decommissioning, Plant Support and Technical Functions, and Nuclear Oversight. The following summarizes the responsibilities and authority of major supporting organization positions:

General Manager, Engineering/Decommissioning - reports to the General Manager, Trojan Plant, and is responsible for engineering assistance, decommissioning, fire protection, and design control of Trojan. Under the direction of the General Manager, Engineering/Decommissioning, engineering support is provided for safe, efficient operation of plant equipment, drawing and design document preparation, and plant modifications.

Manager, Decommissioning Planning - responsible for the development of a decommissioning plan for the Trojan Nuclear Plant including the transition to dry fuel storage.

Manager, Engineering - responsible for engineering assistance, modifications, maintaining Plant configuration data and documentation, oversight and direction of the Fire Protection Program and Civil/Seismic Program. Also responsible for providing technical support for Plant Operations and Maintenance.

Project Manager, Component Removal Projects - responsible for component removal activities as they relate to facility decommissioning.

General Manager, Plant Support and Technical Functions - Reports to the General Manager, Trojan Plant, and has responsibility for the Plant Support and Technical Functions organizations, cost control, security, training, licensing, compliance and commitment management, and the overall administration and maintenance of records.

Manager, Nuclear Security - reports to the General Manager, Plant Support, and is responsible for the administration of the Plant Security Organization, implementation of the security program, and the Fitness-for-Duty Program. This responsibility includes interaction with state and federal regulatory agencies; communication and coordination with local law enforcement agencies; direct supervision of the security staff; administration of the contract with the security contractors; selection, training and staffing of the security organization; administering the security screening programs for personnel authorized unescorted access to the Plant or to Safeguards Information; preparation of procedures required to implement the security program; approval of security-related Plant Modification Requests (PMRs).

Manager, Licensing, Compliance, and Commitment Management - responsible for all activities required to maintain the permits and licenses required for the Plant including producing, maintaining, and interpreting licensing documents.

General Manager, Nuclear Oversight - reports to the General Manager, Trojan Plant, and is responsible for the QA Program, for audits that are carried out to verify compliance with the QA Program, for evaluating the effectiveness of the QA Program for each area that is audited, and for supporting line management in event analysis, review, and performance monitoring of activities that affect the safe and reliable operation of the facility. He has the authority and independence to identify quality problems; initiate, recommend, or provide solutions to quality problems through designated channels; and verify implementation of solutions to quality problems. He has the authority and responsibility to initiate stop work orders to responsible management, as necessary, for any condition adverse to quality.

7.5 DECOMMISSIONING PLAN

The Decommissioning Plan for Trojan Nuclear Plant is required by Title 10 of the Code of Federal Regulations, Part 50.82 and is contained in the "Trojan Nuclear Plant Decommissioning Plan," PGE-1061.