Log # TXX-88200
File # 10010
Ref # 10CFR50.34(b)

TUELECTRIC

February 5, 1988

William G. Counsil
Executive Vice President

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NOS. 50-445 AND 50-446
ADVANCE DRAFT COPY OF FSAR AMENDMENT 68 CHANGES

Gentlemen:

Enclosed are advance copies of CPSES FSAR changes. The changes have been prepared and approved for the upcoming Amendment 68 to the CPSES FSAR. The pages which contain technical changes (and adjacent pages where needed to locate the context of the change) are being provided in order to facilitate your ongoing review. Although some of the pages may not match your existing copies (updated through Amendment 66) the section numbers and text are consistent, continuous and correct. Please note that the enclosed draft FSAR Amendment 68 changes should not be inserted into your FSAR. Our formal submittal of Amendment 68 will be forthcoming during early 1988.

Amendment 68 will provide updates, clarifications, revisions, corrections, additions and editorial changes to the FSAR as well as a revised Effective Page Listing for the affected portions of the report. Listed below is a summary of some of the more significant changes:

1.0 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

The 1A(N)/1A(B) sections have been revised to provide updates to the TU Electric position for the following Regulatory Guides:

1) Regulatory Guide 1.25
2) Regulatory Guide 1.38
3) Regulatory Guide 1.75

2.0 SITE CHARACTERISTICS

Sections 2.4 and 2.5, specifically relating to hydrology, geology and seismology, have been clarified or updated to include the information developed from the validation of civil/structural design criteria parameters (i.e. bearing capacity, lateral loads, settlement of Category I structures and all dynamic soil and rock properties).

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1/1

3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Section 3.6B has been revised to: 1) delete reference to arbitrary intermediate breaks, 2) redefine inside containment break exclusion area (BEA) piping to eliminate the break exclusion area, and 3) revise the "stress node and break location" figures to update the redefined containment break exclusion area piping and incorporate the revised break locations.

Section 3.7N has been revised to reflect the results of Westinghouse seismic qualification reports on the reactor internals.

Section 3.7B has been corrected to reflect the new seismic qualification for the Service Water Intake Structure (SWIS) and Category I Tanks based on reanalysis performed as part of the design validation for CPSES. For the SWIS and Category I Tanks a revised Amplified Response Spectra is required. Also included are various changes to the text, tables, and figures which reflect the reanalysis performed as part of the design validation for CPSES.

Section 3.8.1 has been revised to include several clarifications, corrections, and revisions. The applicable codes, standards, and specificaions were changed to provide the correct applicability dates and sections for 1) concrete materials, 2) punching shear for reinforced concrete, 3) thermal stresses, 4) structural steel, and 5) containment liner and penetrations. Changes for the liner seam welds allow the use of alternate non-destructive examination methods and acceptance criteria. The description of the design and analysis procedures was corrected to reflect the reanalysis performed as part of the design validation for CPSES. The requirement for ultrasonic testing of the liner plate in the vicinity of attachments was deleted.

Section 3.8.2 has been revised to correct ASME Code applicability dates and sections used for the construction of the containment liner and penetrations and to revise the load equations and acceptance criteria to be consistent with Regulatory Guide 1.57.

Section 3.8.3 has been revised to include several changes. The discussion of the polar crane derailment in regard to load combinations has been clarified. The description of the design and analysis procedures was corrected to reflect the reanalysis performed as part of the design validation for CPSES. The anchorage requirement for reinforcing steel (90^{o} hook) was changed to allow the required length to be based on testing.

Section 3.8.4 has been revised to include several changes. The description of the design and analysis procedures was corrected to reflect the reanalysis performed as part of the design validation for CPSES. The anchorage requirement for reinforcing steel (90^{o} hook) was changed to allow the required length to be based on testing. Alternative design criteria were added for brackets and corbels when considering shear stress.

Section 3.8.5 has been revised to clarify the seismic gap requirements between the foundation mats of seismic Category I structures.

Section 3.9N has been revised to correct the acceleration levels for Westinghouse supplied active valves, and to conservatively revise the damping used in the seismic analysis of the reactor coolant pump.

Table 3.9B-10 has been revised to expand the list of active valves.

Section 3.11N has been revised to describe the environmental design of mechanical and electrical NSSS equipment. Changes reflect 1) the effects of the revised plant service conditions (i.e. the effects of temperature, pressure, radiation etc.) on existing equipment, 2) the incorporation of the new definition of mild environment into the qualification program, 3) updates to the section to include the latest revisions of applicable references and 4) the present status of the CPSES qualification effort (i.e. work completed).

5.0 REACTOR COOLANT SYSTEM

Section 5.4 has been revised to reflect the revised RHR cooldown analysis performed as part of the CPSES design validation program.

6.0 ENGINEERED SAFETY FEATURES

Section 6.2 has been revised to: 1) update the NPSH curves for the containment spray pumps and provide associated text changes, 2) provide additional descriptions of the functioning of several containment isolation valves and 3) update the Tables associated with containment isolation valves to reflect as built conditions.

Section 6.3 has been revised to correct the method by which the accumulator motor operated isolation valves are locked closed during plant shutdown. Also revised the list of ECCS motor operated valves.

Section 6.4 has been revised to add a statement that a concurrent release of toxic gas due to a seismic event and a radiological release due to a LOCA is not considered in the design basis. Also a description of the analysis for the release of refrigerant in the control room has been added.

Table 6.4-6 has been revised based on Westinghouse "Radiation Analysis Design Manual, Standard Plant Model 412, " Revision 3.

Section 6.5 has been revised to provide a new opening time for the containment spray pump isolation valve.

7.0 INSTRUMENTATION AND CONTROLS

Chapter 7 has been revised to reflect the current plant design as a result of the design validation program. Changes also include updates, corrections and clarifications.

8.0 ELECTRIC POWER

Section has been issued with various updates, clarifications, typographical and editorial corrections. A major portion of the clarifications concern the separation of non-Class 1E and Class 1E circuits at CPSES based on the design validation program. A new table identifying non-Class 1E equipment connected to Class 1E power buses has been provided. The containment electrical penetration protection description and 125 VDC battery load tables have been updated. Two additional splice applications have been added to Appendix 8A.

9.0 AUXILIARY SYSTEMS

Chapter 9 has been revised to include; 1) corrections to the design criteria applicable to spent fuel pool cooling and purification, 2) revisions to the SSW and CCW temperatures and time to cooldown on RHR, 3) revisions to the description of the CCW control circuitry, 4) the deletion of the plant vent stack radiation monitor from the primary vent stack exhaust, and 5) revisions to the figures for the new fire water supply system.

10.0 STEAM AND POWER CONVERSION SYSTEM

Section 10.4 has been revised to: 1) provide new guidelines for chlorination, 2) correct the usable volume of the condensate storage tank, and 3) clarify and update the Auxiliary Feedwater Pump operation description.

11.0 RADIOACTIVE WASTE MANAGEMENT

Table 11.5-1 has been revised to correct the principal isotopes monitored in several process radiation monitoring detectors and to correct the bases for alarm set points in several other detectors.

13.0 CONDUCT OF OPERATIONS and,
14.0 INITIAL TEST PROGRAM

Chapters 13 and 14 have been revised to reflect the reorganization of the Startup organization to form the Test Department which is responsible for testing and initial startup activities.

17.0 QUALITY ASSURANCE

The changes to Table 17A-1 are technical or editorial in nature and do not affect the programmatic aspects for the establishment and implementation of the QA/QC programs.

NRC QUESTIONS AND RESPONSES

Changes to the following questions and responses (Section) have been made for clarification, correction, update, and addition:

1) 032.0 Instrumentation and Control Systems
2) 040.0 Power Systems
3) 130.0 Structural Engineering
4) 212.0 Mechanical Engineering
5) 421.0 Quality Assurance
6) 423.0 Quality Assurance

EDITORIAL

- Several editorial changes have been made which include the reissuance of Sections, Appendices, Tables and Figures utilizing the format allowing computerization of the amendment change bars and numbers. This computerization ensures consistency and accuracy of the FSAR.

A page-by-page description of the changes that will be included in Amendment 68 is attached (See Attachment 1). This attachment also serves as a listing of the pages which contain technical changes in this advance draft copy of Amendment 68. Pages which have only editorial changes (e.g. typographical corrections, repagination) are not discussed in the attachment.

In addition to the "upcoming" FSAR changes identified in Amendment 66 advance copy transmittal (TXX-6999 dated December 23, 1987), TU Electric anticipates further CPSES FSAR changes resulting from additional modifications to the CPSES design and/or physical plant. While the details of these changes are not sufficiently defined to include them in Amendment 68, we would like to supply the staff with advance notice of the potential for such changes. These potential changes are noted below with a brief description:

1. LOCA and Main Steam Line Break (MSLB) analysis in containment

   The LOCA and MSLB accident analyses and impacts upon the containment pressure-temperature transient are being revised to correct the error found in the CONTEMPT computer code.

2. Reanalysis of boron dilution event

   The boron dilution event is being reanalyzed for an increased boron dilution flow rate from the Volume Control Tank (VCT).

3. Containment spray pH

   The containment spray pH control is being reanalyzed. Design changes may be required as a result of the reanalysis.

If you have any questions concerning this letter or the enclosure, please do not hesitate to contact me or my staff.

Very truly yours,

W. G. Counsil

By: John W. Beck
John W. Beck
Vice President,
Nuclear Engineering

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Attachment

c- Mr. R. D. Martin, Region IV
CPSES Resident Inspectors - 3 copies

site in a properly cleaned condition. A Westinghouse process specification provides detailed cleaning requirements for equipment manufacturers, and is included as a procurement requirement, where appropriate.

Also refer to Appendix 1A(B).

Regulatory Guide 1.38

Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants

Discussion

For the CPSES whose application was docketed prior to April 1, 1973, the quality assurance procedures employed in the design and construction phases may vary in detail from the current position of Revision 1 (10/76) of the regulatory guide but meet its essential requirements in that they follow good business practices as defined in the applicable Westinghouse process specifications. 68

For activities initiated after January 1, 1975, for the CPSES, Reference [5] is applicable. This plan follows the guidance of ANSI N45.2.2-1972 (which is recognized by Regulatory Guide 1.38) in the design, procurement, fabrication and shipment of safety-related NSSS equipment. Measures are applied, as appropriate, to apply packaging requirements to procurement orders, to review supplier packaging procedures, to apply proper cleaning requirements, to apply proper marking and identification, to provide protection to equipment from physical or weather damage, to apply special handling precautions and to define storage requirements. A Westinghouse process specification incorporates detailed packaging and handling requirements for equipment manufacturers, and is included as a procurement requirement where appropriate.

59 | Refer to Section 2.3 and the response to Question 372.36 for a
| description of the design and siting of the primary meteorological
| tower.

Regulatory Guide 1.24

Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure

Discussion

The analysis of the radiological consequences of the radioactive gas storage tank failure accident presented in Section 15.7.1 complies with the requirements of Safety Guide 24 (3/23/72) except that only gamma radiation contribution is taken into account in the determination of whole body exposures.

Regulatory Guide 1.25

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

Discussion

The analysis of the radiological consequences of the fuel handling accident inside the Fuel Building presented in Section 15.7.4 complies with the requirements of Safety Guide 25 (3/23/72) except as follows:

68 | 1. No iodine adsorber efficiency has been considered.

2. Only gamma radiation contribution is taken into account in the determination of whole body exposures.

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Regulatory Guide 1.73

Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants

Discussion

Safety-related motor operated valves inside Containment comply with the guidance of Regulatory Guide 1.73, dated January 1974, with the exception that stem mounted limit switches are tested separately to the requirements of IEEE Standard 382-1972.

For details see Section 3.11B.

Also refer to Appendix 1A(N).

Regulatory Guide 1.74

Quality Assurance Terms and Definitions

Discussion

Q421.19 |

8 | This guide is not applicable to CPSES design and construction. The quality assurance provisions for operating phase activities are in accordance with the guidance of ANSI N45.2.10 - 1973, as endorsed by this regulatory guide dated February 1974.

Also refer to Section 17.2.

Regulatory Guide 1.75

Physical Independence of Electric Systems

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Discussion

CPSES design complies with the intent of Revision 1 (1/75) of this regulatory guide with the following comments: 60

Regulatory Position C.1 - The non-Class 1E security lighting circuits are isolated from their Class 1E power source with two separate Class 1E feeder breakers connected in series. These breakers are coordinated with their supply breaker and will be tested periodically to ensure that coordination is maintained. 68

The non-Class 1E AC essential lighting circuits are isolated from Class 1E power sources with two separate Class 1E breakers (i.e., main breaker and feeder breaker within the Class 1E lighting distribution panel) connected in series. These breakers are coordinated with their supply breaker and will be tested periodically to ensure that coordination is maintained. 66

The non-Class 1E AC essential lighting circuits use interconnecting cable (i.e., from the distribution panel feeder breaker to the lighting load) routed in conduit. The routing of the circuits in conduit ensures the physical and electrical independence from Class 1E circuits beyond the isolation breaker. 66

The non-Class 1E DC emergency lighting circuits connected to dedicated batteries are routed in conduit. The routing of the circuits in conduit ensures physical and electrical independence from Class 1E circuits. 66

The lighting circuits routed in conduit meet the separation criteria of FSAR Section 8.3.1.4. 66

66 Fiber optic cables used in non-Class 1E monitoring circuits carry no electrical energy by themselves and therefore are not required to maintain physical separation from Class 1E circuits.

66 Lesser internal wiring separation is being used between redundant safety systems and safety and non safety systems in BOP Analog Process Instrumentation Panels. This analysis is provided in Section 7.1.2.2.

66 The non-Class 1E diesel generator neutral grounding transformer is connected to the neutral of the Class 1E diesel generator. An analysis has been performed which demonstrates that a fault on the non-Class 1E portion of the circuit will not cause an unacceptable influence on the Class 1E system. In addition, the interconnecting cable is routed within the diesel generator room. The cable is routed in dedicated raceway and is inspected to Class 1E requirements.

65 Regulatory Position C.2 - For the purpose of electrical cable separation, acceptable enclosed raceway includes rigid metal conduit, electrical metallic tubing (EMT) and flexible metallic conduit. Ventilated tray covers are considered equivalent to solid non-ventilated tray covers. Cable bus enclosures are considered the same as enclosed raceway for separation purposes.

65 A wrap of woven silicon dioxide is equivalent to a metal enclosed raceway with respect to protection from electrical failures.

64 Regulatory Position C.6 - Lesser separations are being used in several locations between Class 1E wiring and non-Class 1E Area Radiation Monitoring detector wiring and Public Address System speaker wiring based on analysis. This analysis is provided in Section 8.3.

Regulatory Position C.9 - Splice type connections have been used to terminate field routed cables at equipment where the equipment is provided with pigtail cables and on field routed power cables spliced in manholes by means of in-line splices located in cable tray. Where an enclosure has been provided and space exists, the splices are located within the equipment enclosure, e.g., field cables for motor leads. Where this is not the case, the splices are located in raceways nearby. Such splices are utilized in CPSES design at: | 68 | 60 | 68

a. Electric penetration assemblies (EPAs) and Thermocouple Reference Junction Boxes | 68

b. Solenoid valves, limit switches, level switches, etc. (local mounted devices - LMDs) | 60

c. Connection of LMDs to Electric Conductor Seal Assembly (ECSA) pigtails. | 60

d. Equipment which can only accept smaller (than field cable) size cable. | 60

e. Manholes where field routed power cables use in-line splices located in cable trays. | 68

An analysis to justify cable splices in raceways is provided in Appendix 8A. | 60

Regulatory Position C.12 - Power circuits for the following equipment located inside the Cable Spreading Room/Control Room complex, are routed in exclusive conduits within the Cable Spreading Room/Control Room complex: | 66

8 Discussion

8 This regulatory guide is not applicable to CPSES; however, soils investigations are discussed in Section 2.5.

8 Regulatory Guide 1.139

8 Guidance for Residual Heat Removal

Discussion

8 Refer to Appendix 1A(N).

8 Regulatory Guide 1.140

8 Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants

8 Discussion

59 The CPSES design, maintenance and testing of the normal HVAC systems is in compliance with the requirements of this Regulatory Guide dated
60 March 1978. ANSI/ASME N509-1980 and ANSI/ASME N510-1980 shall be used for field testing activities in place of the older versions of these codes referenced in this Regulatory Guide.

59 Atmospheric cleanup trains installed at CPSES have two high efficiency filter banks in series. In-place testing of only one bank will be performed.

59 In-place testing of the high efficiency filter banks and adsorber will not be required for painting, fire and chemical release described in position 5.c and 5.d of this guide. Only laboratory testing will be performed for carbon efficiencies.

All streams in the SCR basin empty directly into SCR; therefore no channel routing coefficients were required. The applicability of the stream course response model to handle the PMF is discussed in Section 2.4.3.3.4. The ability of the SCR dam to withstand the PMF and coincident wave action is discussed in Section 2.4.3.6.

2.4.3.5 Water Level Determinations

The mass curve, the capacity-area-depth curves, and the spillway rating curves (Figure 2.4-9) are used in routing the PMF through the reservoir to evaluate water level. The resulting peak reservoir level is Elevation 789.7.

In routing, the reservoir water surface has been assumed to be nearly horizontal, and the volume of water in the reservoir has been assumed to be directly related to the reservoir elevation. These are reasonable assumptions in view of the shape and depth of the SCR. These assumptions allow the principle of continuity expressed as a storage equation ($\bar{I}t - \Delta s = \bar{O}\ t$, where $\bar{I}$ and $\bar{O}$ are the average rates of inflow and outflow for the time t, and s is the change in water volume during time t) to be applied directly to the routing problem [16].

2.4.3.6 Coincident Wind Wave Activity

The magnitude of the wind tide and wave runup are dependent upon the wind velocity, fetch and reservoir depth. The wind direction must coincide with the fetch direction. An overland wind velocity of 40 | 68
miles per hour has been approved by the USACE for use in determining |
freeboard requirements in the Fort Worth District. This 40 mph wind |
velocity is the highest that may reasonably be assumed to occur coincidentally with the probable maximum flood [17].

| Q371.7

The effective fetch length for wave generation was determined for the | 4
center of Squaw Creek Dam (fetch of 1.28 miles) and for the exposed |
side of the CPSES plant location (fetch of 1.25 miles). It was also |

2.4.5.2 Surge and Seiche History

There are no existing large bodies of water near the site that would allow development of either surge or seiche; therefore, there is no history of surge and seiches in the site vicinity.

2.4.5.3 Surge and Seiche

The small size, relatively shallow depth and irregular shape of Squaw Creek Reservoir indicates that there is a minimum probability of either surges or seiches occurring in the reservoir. Therefore, surge and seiche should not be considered significant at this site.

2.4.5.4 Wave Action

The effect of the maximum sustained wind on the reservoir surface has been evaluated in Section 2.4.5.1. This wind is considered coincident with a 10 year return period flood elevation in Squaw Creek (778.1 feet). Results of the wind wave activity calculation are presented in Table 2.4-14. With an effective fetch of 1.28 miles, computations indicate that the significant wave height will be approximately five
68 | feet with a period of 3.9 seconds. The maximum wave height will be
| about eight feet with a setup of 0.2 feet and runup of 6.8 feet [10].

68 | This will occur at Squaw Creek dam as illustrated in Table 2.4-14

The wind penetrated waves on the SSI are less than those on SCR due to the much shorter fetch available around the SSI area (see Figure 2.4-14, Figure 2.4-15 and Table 2.4-14).

2.4.5.5 Resonance

Due to the irregular shape and sloping sides of SCR and SSI, wave resonance will not have any significant effect on the maximum water elevation.

2.4.5.6 Runup

The maximum water elevation reached due to wave runup and setup at the plant site, Squaw Creek dam and SSI dam are 794.7 feet, 793.7 and 791.3 feet, respectively. All plant facilities will be above the maximum wave runup elevation of 794.7 feet. 68

The Service Water Intake Structure will be the only safety-related structure subject to wave action or wave runup. The operating deck will be approximately elevation 796', well above the maximum expected wave runup.

2.4.6 PROBABLE MAXIMUM TSUNAMI FLOODING

This site is nearly 300 miles from the Gulf of Mexico and the plant will be over 800 feet above sea level. Therefore, tsunami flooding will not occur.

2.4.7 ICE FLOODING

The Texas climate is too warm to allow the development of significant ice on any lake. Certainly there are no records of any major river in Texas freezing over at any time, so the possibility of ice flooding can be discounted.

2.4.8 COOLING WATER CANALS AND RESERVOIRS

2.4.8.1 Canals

No canals are involved.

2.4.8.2 Reservoirs

The reservoir side of the dam is protected by rip-rap and gravel blanket from the top of the embankment to elevation 760.0, which is ten feet below the minimum operating level. The top width of the embankment is 20 feet, exclusive of the gravel blanket and rip-rap.
Design for the rip-rap was based on an average over-water wind of 81 | 68
mph (Probable maximum wind - 200 year frequency) over the effective |
fetch distance. The method of computing the effective fetch distance |
set forth in Department of the Army Office of the Chief of Engineers ETL 1110-2-8, 1 August 1968, was adopted. Minimum layer thicknesses were determined using the requirements set forth in EM 1110-2-2300 (April 1959) and EM 1110-2-1601 (July 1970). The specific gravity of the rock was assumed to be 2.3. The result are as follows:

| Q371.7

a. Effective Fetch - 1.28 miles | 4

b. Significant Wave Hgt. - 5.0 ft.

c. Layer Thickness - 33 Inches

d. Average Rock Size - 22 Inches

2. Spillways

The service spillway is an uncontrolled structure (i.e., without gates), 100 ft wide, with a standard ogee crest at elevation 775.0. Additional discharge capacity for protection from extreme floods is provided by a broad-crest emergency spillway, 2,200 ft wide, excavated through the rock of the north abutment at elevation 783.0. A 12-inch diameter makeup water pipeline crosses the emergency spillway along its crest. This line is placed in a trench cut in limestone and is covered with a concrete cap. The top of this cap is at elevation 783.0. Locations of the spillways are indicated on Figure 2.1-2. A profile of the service spillway appears on Figure 2.4-18.

Design bases for the service spillway at Squaw Creek Dam are as follows:

a.	Spillway width:	100.0 ft
b.	Design Head:	14.2 ft

70,000 acre-feet per year by utilizing only the upper 51 percent of the available conservation storage volume.

2.4.11.2 Low Water Resulting From Surges, Seiches or Tsunamis

Not applicable (See Sections 2.4.5 and 2.4.6).

2.4.11.3 Historical Low Water

The extreme variability of flow in the Brazos River is depicted through a flow probability curve (Figure 2.4-24), which shows that the average discharge of 1,555 cfs at gaging station 8-0910 is equalled or exceeded only about 17 percent of the time [27]. This flow is modified by regulation of water by upstream reservoirs which tend to
68 decrease the variability. The impact that upstream control has had upon flow extremes is indicated from data which show that the Brazos River was known to dry up completely before construction of Possum Kingdom Reservoir and from Figure 2.4-25 which shows the lessening of annual flood events subsequent to Possum Kingdom Reservoir.

Squaw Creek has not been gaged long enough to allow a direct measure of flow variability, but indirect generalization of variability is gained by comparison with the Paluxy River. A flow probability curve for the Paluxy River is illustrated in Figure 2.4-25, which shows that the average discharge of 70.8 cfs was exceeded only about 11 percent of the time. Since the SCR catchment size is only about 16 percent that of the Paluxy watershed at gaging station 8-0915, this variability will be much more pronounced. Thus, inflow from Squaw Creek will be extremely variable.

Lake Granbury and SCR will serve to regulate the naturally variable flows and provide suitable minimum water levels on a dependable basis.

to the site. The postulated release was assumed to occur due to an accidental rupture of the waste holdup tank which is located in the Auxiliary Building near the Containment Building.

The volume of the tank is 30,000 gallons and at the time of rupture it was assumed that the tank was 80 percent full. The assumed quantities of radionuclides in the tank at the time of rupture are given in Table 2.4-20.

Q371.7 |

It was conservatively assumed that all the liquid radwaste (24,000 gallons, or 7.36 x 10^{-2} acre-feet) is spilled into Squaw Creek Reservoir. Minimum dilution in Squaw Creek Reservoir would occur at minimum pool elevation 770.00 feet (msl), corresponding to a storage
68 | volume of 135,062 acre-feet. Assuming complete mixing, the minimum
| dilution factor is 135,062/(7.36 x 10^{-2}) or 1.84 x 10^{6}.

The instantaneous concentrations in Squaw Creek Reservoir are calculated by dividing the concentrations in the tank by the dilution factor. Due to the decay characteristics of the radionuclides, the concentrations will decrease with time. The equation used to define the concentration of any radionuclide for certain periods of time is:

$$C_t = 2 \left(- \frac{t}{t_{\frac{1}{2}}} \right) \times Co \qquad \text{(Reference 42)}$$

where,

C_o = Concentration in time zero

t = Time interval considered

$t_{1/2}$ = Half life of radionuclide

C_t = Concentration at time t

The concentration of each radionuclide in Squaw Creek Reservoir at the end of the first day and at the end of the first month is shown in Table 2.4-21.

It also receives water from water-bearing units under greater hydraulic head which adjoin the Paluxy Formation. Figure 2.4-30 shows the outcrop area of the Paluxy Formation. South of the CPSES site, across the Paluxy River, the formation is confined by overlying fine-grained bedrock strata. These strata are not of significance to CPSES.

Groundwater discharges from the Paluxy Formation as springs and seeps in some outcrop areas. Where the formation is confined, there is limited water movement into overlying confining units if those units are at lower hydraulic head.

2.4.13.1.5 Onsite Water Table

68 Following the subsurface exploration program, a number of the borings were used to determine water levels. Of these borings, P-10 was completed in the Twin Mountains aquifer; the piezometric water level in that boring is elevation 670. The remainder of the borings monitored for groundwater were completed in the Glen Rose Formation.
68 Static water levels observed in these borings are presented in Figure 2.5.5-77 and range from elevation 749 to 830.

Water levels in the Glen Rose Formation are expected to show some variation in response to seasonal climatic changes; those in the Twin Mountains Formation will be much less influenced by seasonal
68 conditions because of the distance from the recharge area. A permanent system of piezometors will be installed in order to monitor ground water levels at the site. This program is described in Section 2.5.4.13.

2.5.4.13.

2.4.13.1.6 Water Quality

Potable groundwater occurs in the Twin Mountains, Glen Rose and Paluxy formations. The results of chemical analyses of groundwater obtained from wells drawing from these formations are summarized in Table 2.4-22. (Well locations are shown on Figure 2.4-33).

9 throughout the limestone as a function of time. This conservative analysis calculates the maximum concentration of any radionuclide anywhere in the groundwater to be 4.8×10^{-12} μCi/ml as a result of the postulated accident.

2.4.13.4 Monitoring or Safeguard Requirements

No planned releases to the ground water environment will take place at the plantsite; therefore, no monitoring is required. Pertinent information is provided in Section 6.1 of the Environmental Report.

2.4.13.5 Design Bases for Subsurface Hydrostatic Loading

The lateral pressure (σ) caused by the groundwater at a given point is equal to the unit weight of water (γ) times the vertical distance from the water table to the point at which the pressure is computed (H):

$$\sigma = \gamma H$$

Uplift pressures are similarly computed as γH, where H is the vertical distance from the water table to the surface on which the uplift is computed.

68 The design basis groundwater table is at elevation 775 ft (see Section 2.4.13.3.1), which is below the plant grade of 810 ft. The determination of the design basis groundwater table is consistent with the provisions of Section 2.5.- 6.

Safety-related plant structures located below this level are designed for the hydrostatic loads.

There is no dewatering at the site during or after construction.

TABLE 2.4-13

UNIT HYDROGRAPH CHARACTERISTICS

	Lower SCR Catchment	Upper SCR Catchment (Except Maximum Six Hours)	Upper SCR Catchment (During Maximum Six Hours)	
1. A (Sq. Mi.)	20.3	38.0	38.0	
2. L (Mi.)	4.4[a]	13.1	13.1	
3. Lca (Mi.)	2.7[a]	6.0	6.0	
4. $(LLca)^{.3}$	2.1[a]	3.7	3.7	
5. C_t	.6	1.1	1.1	
6. $t_p=C_t(LLca)^{.3}$ (hours)	1.26	4.07	4.07	
7. $t_a=t_p/5.5$ (hours)	.23	.74	.74	
8. t_r (hours)	3.00	3.00	3.00	
9. $t_{pr}=t_p+.25t_r-t_a)$ (hours)	1.95	4.64	4.64	| 68
10. $t_{pr}+.5t_r$ (hours)	3.45	6.14	6.14	
11. C_p640	420	440	440 x 1.3[b]	
12. $Q_{pr}=\frac{C_p640xA}{t_{pr}}$ (cfs)	4,370	3,600	4,680[b]	| 68

[a] Typical values for adjoining areas.

[b] For the most intense 6-hour rainfall period, the unitgraph ordinate is increased by 30 percent of the Upper SCR Catchment.

CPSES/FSAR

TABLE 2.4-14

COINCIDENT WIND WAVE ACTIVITIES

WIND & WAVE CHARACTERISTICS	PMF (elev. 789.7) and 40 mph Overland Wind			10 Year Return Period Flood (elev. 778.1) and PMW (81 mph Overland Wind)			
	PLANT SITE	SQUAW CREEK DAM	SAFE SHUTDOWN IMPOUNDMENT	PLANT SITE	SQUAW CREEK DAM	SAFE SHUTDOWN IMPOUNDMENT	
Effective Fetch	1.25 mi	1.28 mi	0.36 mi	1.25 mi	1.28 mi	0.36 mi	: 68
Average Depth	88 ft.	88 ft.	68 ft.	78 ft.	78 ft.	58 ft.	
Wind Ratio	1.155	1.18	1.05	1.155	1.158	1.05	
Set-up	0.04 ft.	0.06 ft.	0.01 ft.	0.29 ft.	0.21 ft.	0.06 ft.	
Significant Wave	2.25 ft.	2.5 ft.	1.1 ft.	4.7 ft.	4.8 ft.	2.3 ft.	
Maximum Wave	3.76 ft.	4.17 ft.	1.84 ft.	7.85 ft.	8.01 ft.	3.84 ft.	
Wave Period	2.8 sec.	3.0 sec.	1.9 sec.	3.8 sed.	3.85 sec.	2.6 sec.	
Wave Length	43.14 ft.	46.1 ft.	18.48 ft.	74.71 ft.	75.9 ft.	34.6 ft.	
Wave Steepness	0.094	0.C9	0.1	0.105	0.106	0.11	
Relative Runup	1.3 (Smooth 1:3 Slope)	0.95 (rip-rap 1:2 Slope)	0.85 (rip-rap 1:2 1/2 Slope)	1.3 (Smooth 1:3 Slope)	0.85 (rip-rap 1:2 Slope)	0.85 (rip-rap 1:2 1/2 Slope)	
Runup	4.9 ft.	3.96 ft.	1.56 ft.	10.20 ft.	6.81 ft.	3.26 ft.	
Runup + Setup	5.0 ft.	4.0 ft.	1.6 ft.	10.40 ft.	7.0 ft.	3.3 ft.	
Elevation Reached	794.7 ft.	793.7 ft.	791.3 ft.	788.5 ft.	785.1 ft.	781.4 ft.	

TABLE 2.4-20

CONCENTRATIONS OF RADIONUCLIDES IN POSTULATED ACCIDENTAL RELEASE*

Isotope	Floor Drain Tank Activity Ci	Half Life (Year)	Isotope	Floor Drain Tank Activty Ci	Half Life (Year)	
Br-83	8.63+0	2.73-4	Y-90	3.09-3	7.32-3	68
Br-84	4.27+0	6.05-3	Y-91m	3.00-1	3.64-4	68
Br-85	5.45-1	1.14-3	Y-91	5.18-2	6.68-3	68
I-130	1.91+0	1.41-3	Y-92	1.09-1	4.04-4	68
I-131	2.54+2	2.20-2	Y-93	3.45-2	1.16-3	68
I-132	2.54+2	2.63-4	Zr-95	5.90-2	1.75-1	68
I-133	3.82+2	2.37-3	Nb-95	5.90.2	4.00-3	68
I-134	5.18+1	1.00-4	Mb-99	6.81+1	7.54-3	68
I-135	2.09+2	7.55-4	Tc-99m	6.27+1	6.87-4	68
Rb-86	2.00+0	5.11-2	Ru-103	5.18-2	4.50-3	68
Rb-88	4.36+2	3.39-5	Ru-106	1.27-2	1.01+0	68
Rb-89	1.91+1	2.89-3	Rh-103m	5.18-2	1.07-4	68
Cs-134	2.09+2	2.06+0	Rh-106	1.27-2	3.39-7	68
Cs-136	2.63+2	3.59+2	Ag-110m	1.27-1	6.90-1	68
Cs-137	1.36+2	3.02+1	Te-125m	2.54-2	1.59-1	68
Cs-138	8.72+1	6.13-5	Te-127m	2.63-1	2.59-1	68
Ba-137m	1.27+2	4.85-6	Te-127	1.09+0	1.07-3	68
H3	3.18+2	1.23+1	Te-129m	1.73+0	3.82-3	68
Cr-51	5.00-1	7.59-2	Te-129	1.64+0	1.31-4	68
Mn-54	4.00-2	8.55-1	Te-131m	2.36+0	3.42-3	68
Mn-56	1.82+0	2.94-4	Te-131	1.09+0	4.76-5	68
Fe-55	1.82-1	2.70+0	Te-132	2.63+1	8.90-3	68
Fe-59	4.72-2	1.22-1	Te-134	2.73+0	8.00-5	68
Co-58	1.36+0	1.94-1	Ba-140	3.81-1	3.50-2	68
Co-60	1.73-1	5.27+0	La-140	1.27-1	4.60-3	68
Sr-89	3.91-1	1.38-1	Ce-141	5.72-2	8.90-2	68
Sr-90	1.09-2	2.86+1	Ce-144	3.54-2	7.78-1	68
Sr-91	5.63-1	1.09-3	Pr-143	3.72-2	3.72-2	68
Sr-92	1.18-1	3.09-4	Pr-144	3.54-2	3.29-5	68
			Ce-143	4.54-2	3.77-3	68

*E-5 = 10^{-5}

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TABLE 2.4-21

(Sheet 1 of 4)

MAXIMUM CONCENTRATIONS IN SURFACE WATER DUE TO | 68

POSTULATED RELEASES FROM THE LIQUID RADIOACTIV WASTE STORAGE TANK | 68

	Concentration in Squaw Creek Reservoir (uCi/cc)			Concentration in Lake Granbury		Concentration in Whitney Reservoir	
Isotope	Instantaneous	One Day	One Month	Conc. (uCi/cc)	Time (Hrs.)	(uCi/cc)	68
Br83	5.18-8*	4.90-11	0.0	6.57-12	1	**	68
Br84	2.56-8	6.13-22	0.0	7.55-13	1	**	68
Br85	3.27-9	0.0	0.0	2.20-20	1	**	68
I129	0.0	5.44-19	1.01-17	1.08-17	2 Yr	**	68
I130	1.15-8	2.99-9	3.16-26	7.66-12	24	**	68
I131	1.52-6	1.40-6	1.15-7	1.84-10	168	**	68
I132	1.52-6	1.33-7	2.77-10	1.90-10	1	**	68
I133	2.24-6	1.03-6	8.63-17	3.31-9	24	**	68
I134	3.11-7	2.08-15	0.0	2.00-11	1	**	68
I135	1.25-6	1.01-7	2.18-39	2.09-10	1	**	68
Rb86	1.20-8	1.16-8	3.94-9	4.06-11	1/4 Yr	**	68
Rb88	2.62-6	1.16-30	0.0	1.91-11	1	**	68
Rb89	1.15-7	3.49-36	0.0	4.97-13	1	**	68
Sr89	2.35-9	2.34-9	1.57-9	1.60-10	720	**	68
Sr90	1.14-10	1,14-10	1.14-10	5.42-11	2 Yr	**	68
Sr91	3.38-9	5.85-10	4.76-32	1.29-12	24	**	68
Sr92	7.08-10	1.52-12	0.0	9.44-14	1	**	68
Y90	1.85-11	4.04-11	1.14-10	5.43-11	2 Yr	**	68

Notes: * $5.18\text{-}8 = 5.18 \times 10^{-8}$ | 68

** Less than 1×10^{-3} times 10CFR20 MPC | 68

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TABLE 2.4-21

(Sheet 2)

MAXIMUM CONCENTRATIONS IN SURFACE WATER DUE TO | 68

POSTULATED RELEASES FROM THE LIQUID RADIOACTIV WASTE STORAGE TANK | 68

Isotope	Concentration in Squaw Creek Reservoir (uCi/cc) Instanianeous	One Day	One Month	Concentration in Lake Granbury Conc. (uCi/cc)	Time (Hrs.)	Concentration in Whitney Reservoir (uCi/cc)	
Y91m	1.80-9	3.72-10	3.02-32	8.19-13	24	**	68
Y91	3.11-10	3.27-10	2.39-10	2.45-11	720	**	68
Y92	6.54-10	2.20-11	1.71-70	1.24-13	1	**	68
Y93	2.07-10	3.97-11	6.54-32	9.09-14	24	**	68
Zr95	3.54-10	3.50-10	2.56-10	2.70-11	720	**	68
Nb95m	0.0	4.30-13	1.89-12	2.00-13	720	**	68
H695	3.54-10	3.54-10	3.29-10	4.66-11	1/4 Yr	**	68
Mo99	4.09-7	3.18-7	2.11-10	1.31-9	24	**	68
Te99m	3.76-7	3.05-7	2.04-10	1.26-9	24	**	68
Ru103	3.11-10	3.06-10	1.83-10	1.77-11	720	**	68
Ru106	7.62-11	7.60-11	7.20-11	1.97-11	1/2 Yr	**	68
Rh103m	3.11-10	2.76-10	1.65-10	1.59-11	720	**	68
Rh106	7.61-11	7.61-11	7.20-11	1.97-11	1/2 Yr	**	68
Te125m	1.53-10	1.51-10	1.07-10	1.11-11	720	**	68
Te127m	1.58-9	1,57-9	1.31-9	2.01-10	1/4 Yr	**	68
Te127	6.54-9	2.41-9	1.31-9	2.02-10	1/4 Yr	**	68
Te129m	1.04-8	1.02-8	5.60-9	5.19-10	720	**	68
Te129	9.84-9	6.63-9	3.64-9	3.38-10	720	**	68

Notes: * 5.18-8 = 5.18×10^{-8} | 68

** Less than 1×10^{-3} times 10CFR20 MPC | 68

TABLE 2.4-21

(Sheet 3)

MAXIMUM CONCENTRATIONS IN SURFACE WATER DUE TO | 68

POSTULATED RELEASES FROM THE LIQUID RADIOACTIV WASTE STORAGE TANK | 68

Isotope	Concentration in Squaw Creek Reservoir (uCi/cc) Instantaneous	One Day	One Month	Concentration in Lake Granbury Conc. (uCi/cc)	Time (Hrs.)	Concentration in Whitney Reservoir (uCi/cc)	
Te127m	1.58-9	1.57-9	1.31-9	2.01-10	1/4 Yr	**	68
Te127	6.54-9	2.41-9	1.31-9	2.02-10	1/4 Yr	**	68
Te129m	1.04-8	1.02-8	5.60-9	5.19-10	720	**	68
Te129	9.84-9	6.63-9	3.64-9	3.38-10	720	**	68
Te131m	1.42-8	8.15-9	8.42-10	2.92-11	24	**	68
Te131	6.54-9	1.82-9	1.88-16	6.52-12	24	**	68
Te132	1.58-7	1.28-7	2.69-10	5.94-12	168	**	68
Te134	1.64-8	7.22-19	0.0	7.51-13	1	**	68
Cs134	1.25-6	1.25-6	1.22-6	3.99-7	1 Yr	**	68
Cs136	1.58-6	1.50-6	3.23-7	2.90-8	168	**	68
Cs137	8.16-7	8.16-7	8.15-7	3.89-7	2 Yr	**	68
Cs138	5.23-7	1.77-20	0.0	1.57-11	1	**	68
Ba137m	7.62-7	7.72-7	7.71-7	3.68-7	2 Yr	**	68
Ba140	2.29-9	2.17-9	4.51-10	1.38-11	168	**	68
La140	7.62-10	1.26-9	5.19-10	1.58-11	168	**	68
Ce141	3.43-10	3.36-10	1.81-10	1.66-11	720	**	68

Notes: * 5.18-8 = 5.18 x 10^{-8} | 68

** Less than 1 x 10 $^{-3}$ times 10CFR20 MPC | 68

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TABLE 2.4-21

(Sheet 4)

MAXIMUM CONCENTRATIONS IN SURFACE WATER DUE TO POSTULATED RELEASES FROM THE LIQUID RADIOACTIV WASTE STORAGE TANK | 68

	Concentration in Squaw Creek Reservoir (uCi/cc)			Concentration in Lake Granbury		Concentration in Whitney Reservoir	
Isotope	Instantaneous	One Day	One Month	Conc. (uCi/cc)	Time (Hrs.)	(uCi/cc)	68
Ce143	2.73-10	1.65-10	7.28-17	6.04-13	24	**	68
Ce144	2.13-10	2.13-10	1.98-10	4.47-11	1/4 Yr	**	68
Pr143	3.43-10	3.37-10	8.08-11	6.99-12	168	**	68
Pr144	2.13-10	2.13-10	1.98-10	9.47-11	1/4 Yr	**	68
H3	1.91-6	1.91-6	1.90-6	8.65-7	1 Yr	**	68
Cr51	3.00-9	2.93-9	1.42-9	1.24-10	720	**	68
54	2.40-10	2.40-10	2.25-10	5.18-11	1/4 Yr	**	68
Mn56	1.09-8	1.72-11	0.0	1.43-12	1	**	68
Fe55	1.09-9	1.09-9	1.07-9	3.80-10	1 Yr	**	68
Fe59	2.83-10	2.79-10	1.78-10	1.77-11	720	**	68
Ce58	8.16-9	8.08-9	6.09-9	6.79-10	1/4 Yr	**	68
Co60	1.04-9	1.04-9	1.03-9	4.29-10	1 Yr	**	68
Ag110m	7.62-10	7.60-10	7.01-10	1.58-10	1/2 Yr	**	68
Ag110	0.0	9.88-12	9.12-12	2.06-12	1/2 Y4	**	68

Notes: * 5.18-8 = 5.18 x 10^{-8} | 68

** Less than 1 x 10^{-3} times 10CFR20 MPC | 68

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2.5 GEOLOGY AND SEISMOLOGY

2.5.1 BASIC GEOLOGICAL AND SEISMIC INFORMATION

The site of the Comanche Peak Steam Electric Station (CPSES) is 68
located on the Comanche plateau, a subdivision of the Central Texas
section of the Great Plains physiographic province (F2.5.1-1).
Gently dipping Lower Cretaceous limestone and sandstone directly
underlie the site.

Structurally, the site is located on the southern flank of the Fort 68
Worth Basin (F2.5.1-3), a sedimentary depositional trough formed in
mid-Pennsylvanian time. The trough is filled with Pennsylvanian and
Permian sediments. A regional unconformity separates these Paleozoic 68
sediments from the Lower Cretaceous sediments underlying the site
(F2.5.1-5).

Q361.6

Two major fault systems, the Balcones and the Luling-Mexia-Talco fault 68
zones, occur within 200 miles of the site (F2.5.1-3 and F2.5.1-4).
Both of these fault systems can be observed on the surface. 2
Subsurface faults have been identified within seven miles of the site.
These faults appear to die out in sediments over 270 million years
old. (See Section 2.5.1.2.4.)

Seventeen seismic events have been reported with epicenters within 200 68
miles of the site (F2.5.2-2). The closest large event was in
Intensity VII and occurred in 1882 near Paris, Texas, 155 miles
northeast of the CPSES (F2.5.2-3). The nearest event was the
Wortham-Mexia Intensity V-VI shock of 1932, occurring 90 miles
southeast of the site (F2.5.2-6). The region within 200 miles of the
CPSES site has been divided, based on geologic structure, into
tectonic provinces and subprovinces (F2.5.2-8). There have been no
reported epicenters within the subprovince in which the site is
located. An evaluation of the seismic history of the surrounding

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68 Topographic elevations in the site region range from about 550 feet to 1000 feet above sea level.

2.5.1.1.2 Geologic History

Figure 2.5.1-3 shows the regional tectonic structures within a two hundred mile radius of the CPSES site. The major features shown on this figure are discussed in this section. Each of the major structural features has undergone a distinct geologic history. By far, the greatest tectonic activity on a regional scale occurred prior to Mesozoic time.

Seven tectonic provinces have been delineated within a two hundred mile radius of the CPSES site. These provincial boundaries are shown on Figure 2.5.2-2 and their use in developing the Safe Shutdown Earthquake is discussed in Section 2.5.2.6.

1. Central Texas

   The Llano Uplift (Figure 2.5.1-3) is post-Early Ordovician and pre Late Pennsylvanian in age. In late Pennsylvanian time, the Precambrian surface was 10,000 feet higher in the western part of the Llano uplift than beneath the flanking Fort Worth Basin. Erosional losses indicate that uplift accounted for one-third and basinal subsidence for two-thirds of the vertical movement in these structural adjustments [2].

   The northeast trending horsts (Figure 2.5.1-4) of the region developed at the close of the mid-Pennsylvanian contemporaneously with the Ouachita Orogeny which intensely folded the equivalent and older beds in Oklahoma and Arkansas.

West of the Station site and beneath the veneer of Lower Cretaceous formations (Figure 2.5.1-5) lies a succession of Permian, Pennsylvanian, Mississippian and Cambro-Ordovician strata consisting predominantly of resistant limestone and shale with some sandstone. Basement granite likely underlies this Paleozoic section.

North of the Station site, in southern Oklahoma, the Cretaceous section is similarly composed of the Gulf and Comanchean series (Figure 2.5.1-6). However, the Gulf series here is thinner. In Oklahoma, the Comanchean is comprised of the same three groups as in Texas: the Washita, Fredericksburg and Trinity groups. The Washita is thicker and contains more sand in this region. The Paleozoic section in southern Oklahoma consists mainly of limestone, dolomite, marl and shale dating from the Cambrian through the Permian. One sandstone unit appears in the lower Cambrian. The Precambrian basement (over 570 million years in age) consists mainly of quartzite, granite, and gabbro.

2.5.1.1.4 Structure

1. General

Q361.30 |

A large number of varied regional geologic structures are present
7 | within 200 miles of the site. These are described in the following
| paragraphs, subsection 2.5.2.3 and shown on Figures 2.5.1-3, -4, -4a,
| -5 and -6. Tables 2.5.1-1 through 2.5.1-4 list the principal fold
| and fault structures in Texas and in the adjoining states of Oklahoma,
| Arkansas and Louisiana.

1.1 North Central Texas (Great Plains Physiographic Province)

Q361.30 |
Q361.6 |

The Llano Uplift, located approximately 100 miles southwest of the site, is structurally a large dome. Precambrian rocks dating more

than 570 million years in age are exposed in the center of the dome
over an area roughly 40 to 70 miles and are surrounded by formations
of Paleozoic age (225 to 570 million years old) and Cretaceous rocks
deposited 65 to 136 million years ago. The Precambian of the uplift
consists of metamorphic rocks (schist, gneiss and marble), batholithic
intrusions (granite) and late dike intrusions (felsite). Extensive | 7
faulting is associated with the Llano Uplift (Figures 2.5.1-3, -4, -4a |
and -6) and extends to the northeast and southwest under a covering of |
late Pennsylvanian and Cretaceous sediments. Evidence for extension |
of this faulting into the site vicinity could not be found in the | 2
field through stereoscopic study of areal photographs, however, data |
from recent hydrocarbon exploration in the site vicinity has indicated |
this faulting is present in the subsurface to the west of the site. |
(See Section 2.5.1.2.4.) Several major structural arches also |
originate in the Llano Uplift, and they, too, extend under the |
Cretaceous and Upper Pennsylvania formations [8]. One of these |
features, the San Marcos arch, plunges southeastward into the Coastal
Plain and coincides in position and trend with known Precambrian
folding. Other structures, namely the Lampasas Arch trending
northeast, the Edwards Arch trending southwest and the Bend Arch
trending north, possess an axial orientation approximately at right
angles to the lines of Precambrian folding.

North of the Llano Uplift, differential warping and uplift associated with faulting and subsidence in Pennsylvanian time formed the Fort Worth Basin and associated structurally high areas. The structural highs include: the Bend Arch to the west, the Red River Uplift and its continuation, the Muenster Arch to the north, and the Lampasas Arch to the south.

The Forth Worth Basin or Syncline has a northwest-southeast axis and extends as far to the northwest as southern Clay County. Because most of the strata in the basin were deposited during the Strawn time of the Pennsylvanian Period, the basin is commonly referred to as the

"Strawn Basin". the full thickness of Pennsylvanian sediments in the Forth Worth Syncline is not known. No faults are known or suggested
68 in the Basin. The Pennsylvanian section is comparatively thinner over the Bend Arch, Red River Uplift, and Muenster Arch structural highs.

The Bend Arch is a northward plunging anticline and is the most pronounced of the arches originating from the Llano Uplift. Because it originates from the north side of the uplift, there is a possibility that its southern extension may be connected with the
68 southwest trending Edwards Arch. The Bend Arch has probably resulted from two earth movements: a homoclinal tilt to the southwest in upper Pennsylvanian time, and a Post-Permian tilt of the entire region to the northwest, resulting in the northwest dip of the Pennsylvanian and Permian sediments.

The Muenster Arch, together with the Wichita Mountains Uplift comprise an uplift some 350 miles long. The trend has a Precambrian core and is flanked with truncated lower and middle Paleozoic strata. Geophysical evidence indicates a structural relationship between the Muenster Arch and the Wichita Mountain Uplift (Figure 2.5.1-8). The arch is completely blanketed by Cretaceous sediments and is shown in Figure 2.5.1-6.

The Lampasas Arch is a broad arch originating in the northeast portion of the Llano Uplift and is shown on Figure 2.5.1-6.

The Hardeman Basin is a depression at the western end of the Red River Uplift, filled with Pennsylvanian and Permian sediments.

1.2 Southern Oklahoma (Wichita-Arbuckle-Ouachita Mts.)

North of the Red River Uplift, in Oklahoma, are several major uplifts. From west to east, they are the Wichita Mountains, the Arbuckle

The final Coastal Plain structure to be considered is the San Marcos Arch. It is a broad, gentle, structural nose extending southeast from the Llano Uplift. It separates the East Texas and Rio Grande embayments and plunges southeast along its trend.

2. Geophysical Surveys

Two types of geophysical investigation have been utilized to examine regional geologic structure and assist the identification of tectonic provinces. These methods include gravity and natural gamma aeroradioactivity surveys.

2.1 Gravity Survey

The regional Bouguer gravity map (Figure 2.5.1-8) depicts large gravity anomlies, providing an indication of the rock densities associated with the known regional tectonic features and some insight into less well known basement features. The Bouguer reduction commonly yields negative values for areas which are located in continental interiors. Positive Bouguer anomaly valves (observed Bouguer gravity minum regional Bouguer gravity) generally indicate emplacement of high-density rocks near the surface [68] such as Llano Uplift.

Major lithologic-structural divisions of the basement agree well with the regional Bouguer gravity trends and regional gravity configuration is controlled largely by basement phenomena. In Texas, the general westward decrease in Bouguer values is caused by the isostatic effect of the High Plains.

The site, as shown on the regional Bouguer gravity map is located within the Fort Worth (Strawn) Basin and west of the Ouachita tectonic belt [19]. The features are shown on Figure 2.5.2-2. Basic data for construction of the map came from field measurement of the total gravitational force (commonly called absolute gravity), using the pendulum, and from conventional gravity-meter surveys which have been correlated with the absolute gravity determinations.

The updip limit of the Ouachita facies coincides with a series of gravitational maxima trending in the same general direction and also with both the Luling-Mexia-Talco fault system and the Choctaw fault system farther north (Figure 2.5.1-8, 19).

Conspicuous positive Bouguer gravity anomalies are apparent from a series of maxima which coincide with the Muenster and Wichita uplifts and the Amarillo Uplift (Figure 2.5.2-2). These maxima begin rather abruptly in northwest Collin County, Texas, (immediately north of Dallas) and extend northwest through Oklahoma. The maximum of +35 milligals on the Wichita trend in Kiowa and Greer counties of Oklahoma (approximately 200 miles north-northwest of the site) is contrasted to a -81 milligal minimum to the south in Wilbarger County, Texas (approximately 150 miles northwest of the site). The amount of basement uplift and the density contrast between granite and flanking sedimentary rocks is inadequate to account for the magnitude of the anomaly. Either there is a greater amount of gabbroic rock in the Precambrian of the Wichita-Amarillo trend than is apparent from examination of surface exposures and drilling records, or the answer lies in the distribution of mafic material in the subcrust.

A crustal model for Central Texas [65, 66] is shown on Figure 2.5.1-8. Calculated and observed Bouguer gravity anomalies show good correlation and agreement is also found with crustal seismic models proposed by others for Texas. Crustal thickness in the site vicinity is between 40 and 50 km. (25 and 31 miles) according to the model.

All of the major structural uplifts of the southern Mid-Continent correspond to regional isostatic maxima and the basins with gravity minima [67]. The overall isostatic anomaly pattern (see Figure 2.5.1-8, inset) is broadly coincidental with basement structure; however, some broad-wavelength anomalies, one of which is also of high amplitude, apparently result from density variations deep within the crust rather than from structural relief. The most prominent of these anomalies is a 200-mile-long minimum which is aligned east-west across Central Texas from near the New Mexico border to near the Ouachita tectonic belt [67].

The structural grain of the Marathon-Ouachita orogenic belt (stable Ouachita, see Figure 2.5.2-2) is represented by a series of disconnected gravity minima and a parallel nearly continuous, lineal gravity maximum. This elongated gravity maximum can be traced for about 500 miles as a continuous feature from the Ouachita Mountains to near the Marathon area and corresponds closely with the axis of greatest metamorphism in the Ouachita trend.

A major 90 milligal Hayford-Bowie isostatic gravity minimum is centered in the Ouachita Mountains and includes the adjacent Arkoma basin.

Isostatic and Bouguer gravity anomalies produced by the Waco Uplift
68 | are superimposed on the broader rimming gravity maximum. This
| structure has been shown to be faulted and anticlinal, and it probably
| represents a crustal block partially involved in late Paleozoic
| orogenic thrusting related to the Ouachita belt [67].

The site is situated on the central stable continental land mass and exhibits a Bouguer gravity value averaging -48 miligals [19] with low lateral gradients. These gravitational trends indicate that the site is far removed from tectonic boundaries and confirm the regional tectonic provinces developed in Section 2.5.2.2.

The Glen Rose Limestone (Kgr) constitutes the bedrock of the Station site and reservoir area. The outcrop of this formation, its general topography and location of principal station facilities are shown on Figure 2.5.1-12. The Glen Rose Limestone was originally called the "Alternating Beds" because the outcrop pattern is characterized by stair-step topography resulting from differential weathering of impure, nodular limestones, softer claystone beds, and resistant, sparry-cemented, medium-to-thick bedded, hard limestones. The local order of occurrence or dominance of these lithologic types, plus other physical or paleontologic aspects, permits locally useful zonation as shown on Figure 2.5.1-13 and on the lithologic cross section of the station vicinity shown on Figure 2.5.1-14.

68 The graphic lithologic log and the gamma-ray/resistivity log of boring P10 (Figure 2.5.1-13) illustrate a zonation of strata based on cores of all site boring logs, and related field information in the site vicinity.

Zone I, about 80-90 feet thick, is the bench-and-slope, uppermost outcrop unit indicative of major soft claystone beds alternating with hard, ledge-forming limestones. The basal claystone in contact with the underlying massive-bedded limestone of Zone II is the most prominent field contact visible throughout the reservoir area.

68 Zone II, 60-70 feet thick, is a massive unit of mostly nodular limestone with thin claystone partings best developed in the basal portion.

Zone III, 60-65 feet thick, is the most consistent unit
68 recognizable in electric logs in the region. The upper part, IIIa, is an alternating claystone-limestone section transitional with the more massive units above and below; the lower part, IIIb, is the massive-bedded, bluff-forming Thorp Spring Member which is well exposed in the Paluxy River Bed at Glen Rose and further upstream.

Zone IV, at the base of the Glen Rose is about 30-40 feet thick; and consists of an alternation of claystone and limestone, both sandy or with thin sand lenses. The unit is transitional with the fine grained, calcareous, silty sandstone of the underlying Twin Mountains Formation. 68

The regional persistence of the Glen Rose zonation is indicated on the 25-mile correlation cross section (Figure 2.5.1-14) from the plant site east to Cleburne, Texas.

Where the Glen Rose beds are not exposed at the surface, data obtained from borings indicate that it is generally manteled by a few feet (range of a few inches to 15 feet) of surficial soils. These soils consist of mixtures of clay, silt and sand with some gravel and cobble-size rock fragments. In valley bottoms, the Glen Rose beds are overlain by alluvial sediments and residual soils which range to approximately 10 to 15 feet in thickness. As evidenced in many of the boring locations, weathering processes have produced a weathering zone on the Glen Rose up to several feet in thickness. This zone has been chemically altered, partially oxidized, and in some areas over 40 top 60 percent of the interval decomposed to a soil consistency.

The upper limit of bedrock, below which no soil-like inclusions were encountered in borings, is indicated on the vicinity Bedrock Contour Map, Figure 2.5.1-15. Bedrock contours in the immediate plant site are shown on Figure 2.5.1-16. Borings logs are presented in Section 2.5.5.3.

The conditions encountered at Category I facilities including the overburden and top of rock and moderately to severely weathered rock zone where it overlies the better rock are shown on geologic cross sections in Figures 2.5.4-40, 2.5.4-41, and 2.5.6-10. The section locations are shown on Figure 2.5.5-5. Pertinent information regarding foundation grades, existing groundwater conditions and characteristics of subsurface materials are summarized on these figures. The surficial rock generally is of a tan to orange-tan limestone which is occasionally dolomitic and fossiliferous. The remainder of the Glen Rose in this area consists of light to dark gray argillaceous limestone with lenses and/or zones of gray to greenish-gray calcareous claystone. It is sporadically fossiliferous, dolomitic, pyritic and chrystalline. The unweathered Glen Rose is consolidated rock, medium hard to soft and thin-bedded to massive, becoming generally massive with fewer claystone zones below at approximately 770 feet in elevation. Oolitic zones are common in the massive portion of the formation. There are occasional zones which consist of limestone fragments enclosed in a claystone matrix, indicative of slump brecciation during deposition. In outcrops, the claystone lenses weather out to produce a thin-bedded appearance.

68 The Twin Mountains Formation underlies the Glen Rose Formation with a gradational contact. It was encountered in borings in the Station area at approximately elevation 610 to 615 feet. The sandstone beds of the Twin Mountains Formation constitute the aquifer for domestic water supplies and are collectively referred to in the region as the "Trinity Sand" [20].

68 In the postconstruction S/L cluster borings (Fig. 2.5.1-16), drilled for seismic crosshole survey, the Twin Mountain Formation was found to be about 250-ft thick and unconformably overlying the Mineral Wells Formation of the Strawn Series (Pennsylvanian age) at elevation 366 ft (Figure 2.5.4-30A). The Twin Mountains Formation is composed of sandstone and claystone, with occasional limestone layers [131]. The upper portion of this formation (above elevation 495 ft), as encountered in the borings, consists of interbedded sandstone, claystone, and argillaceous

limestone. The sandstone is the dominant lithology. It is very 68
soft to hard, friable in places, poorly cemented, light brown to light gray, and fine to medium-grained. The claystone and argillaceous limestone are medium hard to hard, dark gray to greenish-gray, occasionally green or maroon, and silty or sandy in places. The middle portion of the Twin Mountains formation (between elevation 495 and 458 ft) consists primarily of generally hard and brittle, reddish-brown, and highly slickensided claystone. The lower portion (below elevation 458 ft) is composed of interbedded sandstone and claystone. The sandstone is generally coarse-grained and occasionally conglomeratic, with infrequent fractures and common clay-rich intervals. Hard sandstone and claystone with up to 3-ft thick beds comprise the lowermost 20 ft of the formation. The presence of relatively unconsolidated sandstone and soft claystone in the middle and lower portions of the formation was indicated by the caliper logs of the S/L clusterly at elevation 489 ft (middle portion) and elevation 428 ft (lower portion). Poorly cemented sandstone was also encountered in borings P-9 and P-10 in which the drill rods rapidly penetrated some sandy zones due to easy cutting and/or jetting erosion by high-velocity drilling fluid.

The Mineral Wells Formation [130] was encountered in the S/L cluster 68
Borings at elevation 366 ft, and was penetrated for 65 ft, exposing predominantly claystone beds. These beds, being of Paleozoic age, are distinguished from the similar overlying Lower Cretaceous beds by a higher degree of induration manifested in greater hardness and slickenside frequency, and by the occasional presence of vertical fractures filled with limestone. They are also characterized by their massive nature, dark color, and intercollated limestone seams and layers [131].

Figure 2.5.1-14 illustrates the approximate relationship of stratigraphy at the site to the stratigraphy found in borings within approximately five to 25 miles of the site.

The extensive limestone strata which are present at the site and vicinity are not subject to solutioning because of argillaceous impurities. There is no evidence of solutioning observed in the core obtained from the numerous borings. Fractures or joints occur infrequently.

2 A more detailed description of subsurface conditions is presented in
Section 2.5.4.

2.5.1.2.4 Structure

Dip of the sediments in the site vicinity is to the east at approximately 25 feet per mile. Slight, gently warping of the sediments is occasionally evidenced in road cuts and natural exposures in the area surrounding the site. These are minor local features which are very limited in a real extent and degree of flexure. The infrequent slicken sides encountered in the Glen Rose Formation appear to be the result of differential settlement which occurred during compaction and diagenesis of the formation. The extensive investigations conducted at the site vicinity revealed no evidence of faults, shear zones or other anomalies.

2 Examinations of electric logs from recent gas exploration wells in the
4 site vicinity suggests the presence of subsurface fault. The
locations of these wells are presented on Figure 2.5.1-17. The
faults locations are presented on Figure 2.5.1-23. Larger scale maps
2 of the fault nearest the site are presented in Figures 2.5.1-23A, 24
and 26. This fault has no well cuts and is inferred from trends in
4 subsurface data. The fault exhibits normal movement and is
approximately 65 miles long (Figure 2.5.1-23). The throw on the
fault appears to decrease upward until normal conditions are reached
at a subsea datum of approximately 2,000 feet.

Q361.6
Q361.7
2 The faulting is associated with basement faulting. The structure
model which accommodates this faulting is one of older faulting from
basement movement which faulted the Ellenburger (see Section
2.5.1.1.4).

Later movement renewed the fault to cut the Marble Falls and Big Saline formations and continued upward into the Strawn but died out as the Strawn formation was deposited. There is no evidence of renewed movements after middle to late Strawn time (over 270 million years ago). | 2

| Q361.8
| Q361.9

Figures 2.5.1-23A, 24A and 26 are the revised structure maps of the Ellenburger, Marble Falls and Big Saline formations. They are based on more data than was available for the maps filed with the initial FSAR. The current information does not indicate any faulting within five miles of the site. A structural nose trending southwest to northeast is present on the Ellenburger, Marble Falls and Big Saline horizons. No faulting is interpreted with this nose nor is any anticipated within five miles of the plant site. If by further hydrocarbon exploration any faulting is found within the five-mile radius, it is not expected that it will extend above a datum of approximately 2,000 feet subsea (approximately the middle or upper portion of the Strawn formation) [.24]. On the Strawn marker structural map (Figure 2.5.1-28), no faulting is depicted within five miles of the plant site. Where a fault is mapped on older beds to the northwest of the plant (approximately seven or eight miles from the site) there is no faulting at the level of the Strawn marker. This indicates the faulting has died out in the Strawn Formation between the Big Saline Formation and the Strawn marker. Cross section A-A' (Figure 2.5.1-30) illustrates this upward limitation of the faulting. Cross section B-B' (Figure 2.5.1-31) illustrates the absence of faulting near the site. | 68 | 2

| Q361.21

Figure 2.5.1-32 (Cretaceous Structural Map) presents the structural contours on the Paluxy Sand-Glen Rose Limestone contact over the 5-mile radius area and, over a smaller area, on the prominent marker bed about 75 feet below the top of the Glen Rose. Contouring on the | 4

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4 marker bed is confined to the central portion of the site area in a 2- to 3-mile strip adjacent to the NW-SE trending valley of Squaw Creek. Elevations for datum control of contouring were taken from published USGS topographic quadrangles. The principal points of the Paluxy-Glen Rose contact were established in the field through road traverses; the upper Glen Rose marker bed points were in large part transferred directly from areal photographs to the topographic base (Figure 2.5.1-10, Vicinity Geologic Map).

4 Figure 2.5.1-32 shows a consistent north-south regional strike and constant dip to the east at a rate of about 25 feet per mile for both of these horizons. The mappable continuity of the Paluxy-Glen Rose contact and of the upper Glen Rose marker bed plus the structural contouring indicate no surface anomaly in the site area.

68 Structural cross sections C-C' and D-D' (Figures 2.5.1-34 and -35) supplement the structural maps and likewise indicate no evidence of post-Paleozoic faulting, known or inferred, west or northwest of the 5-mile radius area. The most significant subsurface Glen Rose datum
4 is the massive Thorp Spring Limestone Member. The top of this high resistivity unit marks the boundary between Glen Rose zones IIIa and IIIb of the lithologic section at the plan site (Figure 2.5.1-13). All correlation lines extended from well to well, based on gamma-ray logs updip and self-potential/resistivity logs downdip, are essentially parallel to the Zone IIIa - Zone IIIb boundary (= top of Thorp spring Member). No faulting is indicated in either of the cross sections within the Cretaceous rocks.

2.5.1.2.5 Groundwater

Detailed information regarding site groundwater is described in Section 2.4.13.

permeability. In addition, there was no evidence of styolites, pressure solution of grains, or collapse structures of any scale noted in the thin section analysis. The rarity of dolomite as reported from petrographic and X-ray analysis suggests that collapse structures have not occurred from the effect of volume change in the transformation of Mg-calcite to dolomite and calcite. The low amount of magnesium shown in the chemical analysis indicates that the magnesium is already in the dolomite phase and that future dolomitization as an isochemical reaction is impossible.

These laboratory tests indicate that the Glen Rose Limestone should have a very low permeability, an indication which is confirmed by the packer test data presented in Table 2.5.6-1. The laboratory data also indicates that future solutioning of the limestone should not occur.

Considering the various material types, chemistry, distribution and site environment, it is concluded that solutioning activity in the limestones beneath the site does not exist. Results of detailed petrographic and chemical analysis of the limestone below the site indicate that the potential for solutioning is very low. Detailed results of these tests are given in Section 2.5.4.2.3.

At a considerable distance east of the site vicinity, anhydrite and gypsum and present in Glen Rose Formation [10]. Under certain conditions such material could be subject to solutioning. However, these conditions are not known to occur near the site.

3.1.2.2 Oil and Gas Wells | 68

The Paluxy and Glen Rose formations are major lower Cretaceous hydrocarbon producing reservoirs in the East Texas Embayment. These reservoirs are mostly east and south of the Luling-Mexia-Talco Fault Zone [44] and are nearly 100 miles east of the site. Erratic sands in the Pennsylvanian Strawn, and the Ordovocian Ellenburger limestone | 68 are major producing zones in the Bend Arch area 25 to 50 miles or more to the west.

68 | No exploratory or producing oil or gas wells existed within five miles
| of CPSES prior to 1974. At that time, three exploratory wells have
been drilled in that area and all three were reported as dry and
abandoned. The first area production was reported early in 1977. In
light of new data, a study of mineral potential was performed by H. J.
Gruy and Associates, Inc. and is presented here.

Q361.11 |

68 | Using the subsurface geologic terminology established by Gulf Energy
| and Minerals Co. -U. S. No. 1 Otis Rollins well in the Louis
68 | Boatwright Survey A-33, Hood County, there are four Paleozoic
| formations having the potential for hydrocarbon production in the area
| of the CPSES. In ascending order, these are the Ellenburger
Formation, the Marble Falls Formation, the Big Saline sandstone and
the Strawn sandstones. The size of a possible blowout from the above
listed formations is in the order they are listed (i.e., maximum in
the Ellenburger and minimum in the Strawn sandstones). The
probability of finding hydrocarbons is in the reverse order as listed.
In the following discussion, wells which fall within a five-mile
radius of the plant site are listed with a well identification number
keyed to Tables 2.5.1-5 through 2.5.1-10 and to Figure 2.5.1-17
entitled "Natural Gas and Oil Pipelines and Wells Within Five Miles of
2 | CPSES". As noted on this Table, all hydrocarbons produced within five
| (5) miles of the site are believed to be from stratigraphic traps not
| structural traps [124]. The outline on the net hydrocarbon
| isopachous (Figures 2.5.1-25, 27 and 29) maps which represent the size
| of the hydrocarbon accumulations do not appear to be structurally
| controlled.

A structure map (Figure 2.5.1-23) on the top of the Ellenburger formation was constructed for Somervell, Erath, Hood, Johnson, Hill overburden, permitting groundwater to reach the anhydrite and convert it to gypsum. Mean annual precipitation is less than 30 inches where outcrops of gypsum and gypsiferous material occur. The site is distantly removed from all historic uplifts of this type.

The areal distribution of expansive soils in the region is illustrated on Figure 2.5.1-20. It is not anticipated that materials of a swelling character will have any adverse effect upon structures at the site whose foundations are constructed below the soil zone. 68

3.6 Effects of Man's Activities

Except for the removal of minor quantities of sand, gravel and dimension stone in the site vicinity, no mining has occurred. In connection with plant operation, it is anticipated that a water supply will be developed within wells on the order of 600 gallons per minute from the Twin Mountains Formation. This pumpage may eventually reduce the artesian pressure in the area, but withdrawal of groundwater will have no deleterious effect on rock deformation or stability. Extraction of subsurface fluids is discussed in Section 2.5.1.2.6.

4. Physical Properties of Geologic Materials

The results of onsite geophysical surveys and laboratory testing are provided in Section 2.5.4.4.

5. Safety Criteria

Safety criteria are discussed in detail in Sections 2.5.4.10 and 3.7.

2.5.2 VIBRATORY GROUND MOTION

Vibratory ground motion and the design earthquake (Safe Shutdown Earthquake) are evaluated in consideration of:

The most recent intensity assigned to this earthquake is MM V-VI by Coffman and Von Hake [70]. The felt area given is 1,000 square miles in agreement with the area contained within Sellards' "limit of felt area" of radius 17 miles. A higher intensity of MM VII is given to this earthquake by Docekal [69]. It apprears that the Docekal value is based on chimney damage alone (see previous description of chimney damage). The felt area given by Docekal is "2,000 sq. mi." [69]. According to Brazee's [88] (Figure 2.5.2-7), and Nuttli and Zollweg's [117] recent work on felt areas for earthquakes in the eastern and central U.S., both of these assigned intensities are too large for the Wortham-Mexia Earthquake. Even a MM V maximum intensity for this earthquake yields a felt area five times that given by Sellards. | 68

The possibility exists that this event may be related to petroleum withdrawal in the area. The first major Texas oil discovery occurred in June, 1894, when the city of Corsicana was drilling a water well. The Mexia Field was discovered in 1920, and the Wortham Field in 1924-25. The Mexia field is shown in Figure 2.5.2-6. The Mexia Field covers an area of about 3800 acres. Figure 2.5.2-7 shows the annual amounts of crude oil taken from the Mexia field from 1920 up to and including 1932 (data for the years 1927-1929 could not be obtained). For each barrel of crude oil taken out, there are roughly two and one-half barrels of saltwater removed [89].

Two years earlier, Sellards [90] had studied subsidence in the Goose Creek Oil Field in Harris County, Texas. Sellards writes that, unlike the Goose Creek Field where "the geological section includes only relatively incoherent strata, under conditions as at Mexia where the section includes resistant strata, as the limestone of the Midway and Austin formations, subsidence, if such occurs might result in a jar locally producing a tremor of considerable force." Sellards then calculates the total volume of fluid withdrawn from the Mexia and Wortham Fields. This amounted to some four billion cubic feet.

Sellards concludes by saying: "Whether this tremor therefore proves continued activity in the Mexia-Wortham line of faulting or records merely local subsidence in these oil fields incident to the removal of oil is at present undetermined."

In addition to the Corsicana and Mexia-Wortham Fields, there are a number of other oil fields along the Mexia-Talco Fault System. Some of these fields still producing are listed below:

Cumulative Oil Withdrawn
From Discovery to 1/73
by Counties

Discovery Date	Field	County	Area (Acres)	(10^6 BBLS)
1894	Corsicana	Navarro	4000	192
	Darst Creek	Guadalupe	1900	157
1925	Lytton Springs	Caldwell	2000	
1928	Salt Flat	Caldwell	7000	227
1936	Talco	Titus & Franklin	9500	298

The combined total area of these fields is about 25,000 acres. The counties in which the above and smaller Mexia-Talco Fault Zone fields are located have produced about 875 million barrels of oil to date compared with the 110 million barrels withdrawn from the Mexia-Wortham Fields as of January 1932, as calculated by Sellards [33]. If the Wortham-Mexia event of 1932 was caused by petroleum withdrawal, it was certainly anomalous in light of the far greater volumes withdrawn from other fields along the fault zone (underlain by essentially similar geological conditions) without any recorded or reported seismic activity.

Sellards [33] states that as of January 1, 1932, 704 x 106 cubic feet of gas had been taken out of the Mexia and Wortham Fields. While the report in the Houston Chronicle [19] states that no blasting operations were in progress at the time of the earthquake, the possibility of an underground gas explosion cannot be ruled out as a possible cause of the disturbance. If faulting were the cause, according to Brazee [88], an area within a radius of some 35 miles should have felt the earthquake at intensity V level if the maximum intensity of the Mexia quake is assigned an MM V rating.

2.5.2.1.6 1952, April 9 - El Reno, Oklahoma Earthquake

The El Reno, Oklahoma event in one of the largest historic earthquakes in Oklahoma. Cloud and Murphy [27] indicate that the event was felt over Oklahoma (except for the panhandle), eastern Kansas, central and northern Texas, western Iowa, Missouri, Arkansas and the southern tip of Nebraska. Slippage along the Nemaha fault about five miles southwest of Oklahoma City is credited by Cloud and Murphy as the source of the event and the felt data conforms to the known trace of the fault. Figure 2.5.2-8 outlines the areas in which the various intensity levels were observed. Maximum intensity was VII although damage was not extensive. Portions of chimneys fell in El Reno and Ponca City and bricks loosened from a building wall and tile facing of commercial buildings bulged at Oklahoma City.

Intensity I-III was reported from Dallas and Fort Worth, Texas for this event. Aftershocks were observed on April 11, 16; July 16, and August 14.

2.5.2.1.7 1959, June 17 - Lawton, Oklahoma Earthquake

Eppley and Cloud [93] report that this shock was felt over an area of about 12,000 square miles in Oklahoma and Texas. Maximum intensity

The Stuart City Reef marked the Comanchean shelf edge, separating the lagoonal depositional environments of the Mesozoic Shelf from the deep water environment of the ancestral Gulf of Mexico.

In Texas, this reef trend is now buried under more than 10,000 feet of younger sediments but its younger equivalent in Mexico (El Abra) is exposed in outcrop.

7. Gulf Coast Tectonic Province

The Gulf Coast Tectonic Province is divided into an east and west division by the Sabine Arch similar to the Mesozoic Shelf Province. This division separates the two major depocenters of the Gulf Coast Tertiary geosyncline. During the Eocene and Oligocene, the area of maximum deposition (over 50,000 feet) was central and southern Texas but, during the Miocene and Pliocene-Pleistocene, the depocenter (over 40,000 feet) had shifted to a position off southern Louisiana [61].

2.5.2.2.2 Geologic Structures and Faults

In this section the locations of geologic structures and descriptions of the significant faulting in each tectonic province are presented.
68 | The significant geologic structures and faults are shown in relation
| to the tectonic provinces on Figure 2.5.2-2.

1. West Central Texas Tectonic Province

The West Central Texas province is part of the central stable region of the United States founded upon Precambrian crystalline rock and mantled by a thin veneer of sedimentary rock. The crystalline basement is comprised of both igneous and metamorphic rocks similar to rocks of the Canadian Shield.

4. Ouachita Tectonic Province

Four independent thrust sheets form the principal tectonic structures of the Ouachita Tectonic Province. From northwest to southeast these are the Choctau, the Pine Mountain, the Ti Valley fault and the Windingstair faults. Each of these sheets has been thrust from south to north and has been broken by numerous smaller high-angle reverse faults that probably join the main thrusts at depth. Larger cross faults are occasionally present with numerous minor cross faults. The major thrusts of the Ouachita system have been identified in eastern Oklahoma but appear to die out to the east in the shale sequences. None of these thrust faults are presently known to be active. A zone of open folds lies to the north and comprises the Arkoma Valley basin.

5. Balcones-Mexia-Talco Province

The Balcones-Mexia-Talco faults are a complex assembly of faults that follow approximately the border of the Tertiary and Cretaceous formations of the Gulf Coastal Plain and are probably related to collapse of the continental margin on the seaward side of Paleozoic orogenic belts [47, 113].

It is believed that the fault systems, generally categorized as the Balcones, Luling-Mexia-Talco, and Charlotte-Fashing fault zones, have basement control or deep tectonic involvement, and that thege fault systems are formed over the original zone of weakness that allowed the Ouachita Geosyncline to form along the edge of the Texas Craton [112].

Wells drilled in the general area of the Balcones fault system encountered beneath the Cretaceous a sequence of steeply-dipping clastic sedimentary rocks showing varying degrees of weak metamorphism [3]. Near the beginning of the Miocene period (about 22.5 million years ago) the Coastal Plain strata faulted and slumped towards the Gulf at an increased rate of speed [15]. This was due to a readjustment of the earth's crust caused by sinking of the Gulf Coast region under the heavy load of Cretaceous and Tertiary sediments. The Balcones fault acted as the hinge over which the sediments attempted to stretch [63]. The movement on the Luling-Mexia-Talco fault system was likely due to readjustment incidental to the Balcones faulting and the consequent dip towards the Balcones zone.

The Balcones fault zone forms a great arc, convex gulfward. Its strike is easterly in Kinney, Uvalde and Medina Counties and changes to northeasterly and then north-northeasterly across Bexar, Comal, Hays, Travis, Williamson, Bell, and McLennan Counties.

Faults in the zone are all normal and trend mostly down to the coast, with antihetic faults that form grabens. Distinct faults within the zone are partially an echelon. The average dip of the fault is 45 degrees which generally decreases with depth [15, 47]. It is believed that these normal faults penetrate the basement, beneath Cretaceous and Tertiary sediments. A typical cross section through the Balcones Fault Zone is shown on Figure 2.5.2-11.

Displacements of up to 1,500 to 1,700 feet occur across the Balcones fault zone. The latter figures are for Bexar County in which maximum displacement has been reported. Northeast and southwest of Bexar County the displacement diminshes in magnitude. Individual faults may have any amount of displacement up to 700 feet. Topographic relief along the fault also varies from extensive to none. A prominent escarpment exists throughout most of the area where more resistant Lower Cretaceous strata have been uplifted [47].

The Luling-Mexia-Talco Fault Zone probably represents complimentary (antithetic) up-and-down-to-the-coast movements that developed in response to the down-to-the-coast movement along the Balcones Fault Zone [64]. A typical cross-section through the fault zone is shown in Figure 2.5.2-12. The Balcones and Luling-Mexia-Talco fault zones,

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although related to the action of the Gulf Coast Geosyncline, are not considered growth faults as are the younger more coastward faults. Although they exhibit some of the same characteristics as growth faults (i.e., decreasing dip with depth) they are believed to have basement control as opposed to the more shallow and younger faults in the thicker sediments of the Gulf Coast Geosyncline [47].

Both the Balcones and the Luling-Mexia-Talco fault systems pass within
200 miles of the site. The Balcones system, the nearest, passes
within 50 miles of the site and is situated just east of the Ouachita
68 Tectonic front. The Luling-Mexia-Talco Fault system passes within 80
miles of the site to the east and generally is parallel to the trend
of the Balcones system; both fault systems are shown on Figure 2.5.2-
13.

The possibility of the Balcones and the Luling-Mexia-Talco fault systems being active according to 10CFR100, Appendix A, has been carefully considered. Details in the literature, field observations by prominent earth scientists, the opinions of other experts and our own assessment of field conditions have all lent appropriate weight in making a judgement. Insufficient evidence exists to warrant classification of these fault systems as active along any portion of their known limits.

The 300-mile-lone Balcones fault zone passes closest to the site in Hill County at which location it is 50 miles southeast of the site.

An examination both of areal photographs and field conditions was conducted in order to evaluate the possibility of activity along the nearest portion of the fault zone. These efforts revealed no sound evidence indicating any degree of activity. In many cases, faults within the zone have brought earth materials of contrasting physical properties into juxtaposition. Problems of civil structures stem from this cause or others, rather than from displacement along the failure (fault) plane.

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Consideration also has been given to on-site ground motion related to larger historical events originating beyond 200 miles and extrapolated to the site. The largest known events would not have shaken the site at intensities higher than III or IV. This would be the level of the historical events of a New Madrid shock [115, 116] migrated closer along related structures southward, or a Valentine event migrated closer southeastward along related structure.

In summary, the maximum ground motion which conservatively should be postulated for a Maximum Potential Earthquake would result from the occurrence of an earthquake equal to the 1882 Intensity VII shock near Bonham, Texas. Ground motion from this event, minor local events, or large distant shocks would produce horizontal ground motion at the site less than or equal to 0.10 g.

2.5.2.5 Seismic Wave Transmission Properties of the Site

68 To determine seismic wave transmission and other physical properties of the subsurface material, the following geophysical studies were conducted at the Station and Safe Shutdown Impoundment Dam Areas:

1. Seismic refraction surveys to evaluate the compressional waver velocities of bedrock and the materials overlying bedrock and to evaluate the depth to bedrock;
2. Uphole velocity surveys to provide additional data regarding compressional wave velocities of bedrock and materials overlying bedrock;
3. A surface wave survey to evaluate types of surface waves and their characteristics;
4. 68 Preexcavation crosshole shear wave surveys to evaluate shear and compressional wave velocities of bedrock and the materials overlying bedrock;

5. Postconstruction crosshole seismic wave survey to evaluate shear and compressional wave velocities of a 500-ft thick bedrock interval below plant grade, with a particular emphasis on claystone interbeds; 68
6. Ambient noise studies to evaluate the predominant characteristics of groundmotion due to background noise level; and, 68
7. Gamma ray and electrical resistivity logs in the Squaw Creek Dam site borings, and gamma-ray, gamma-gamma (density), electrical resistivity, and caliper logs in borings S-1A, L-1, and L-2 at the Station site to evaluate strategicgraphic features and obtain rock physical properties at various depths. 68

Details of the geophysical surveys and results are tabulated and discussed in Section 2.5.4.4.

The properties of the materials underlying the site are described in Section 2.5.4.2. All major structures will be founded on sound bedrock. The low level of anticipated as well as past seismic activity at the site is further improved by the conservative assumptions applied to identify the maximum potential earthquake for design purposes. Therefore, detailed analysis of the seismic wave transmission characteristics of the site is not considered necessary.

2.5.2.6 <u>Safe Shutdown Earthquake (SSE)</u>

A conservative Safe Shutdown Earthquake having a peak horizontal ground acceleration at the top of bedrock of 0.12 g has been selected for design.

Design response spectra and artificial time-history records are described in Section 3.7.1. SSE selection has been based on a

2.5.4.1 Geologic Features

Geologic considerations relating to stability of subsurface materials are described in Section 2.5.1.2.6.3. In support of the discussion presented in Section 2.5.1.2.6.3, no evidence of solutioning or past solution activity has been observed in any excavations made in the unweathered Glen Rose Limestone.

The geological history of the site area is presented in Section 2.5.1.2.2.

The information pertaining to the site foundation medium, particularly 68
the data generated from the postconstruction S/L cluster borings (S-1, S-1A, L-1, and L-2) by a detailed seismic crosshole survey, geophysical logging, continuous coring, and dynamic testing of representative core samples [131, 132], was used to divide the subsurface materials underlying the site to a depth of 500 ft into 10 intervals on the basis of their lithologic characteristics and, consequently, engineering properties. These intervals are identified in Table 2.5.4-5E by stratigraphic units, prevalent lithologies, and ranges of thickness, depth and elevation. The depths and elevations shown on Table 2.5.4-5E apply to the vicinity of the power block area. Due to the slight eastward dip of the strata (Section 2.5.1.2.4), these depths and elevations differ slightly for other locations.

2.5.4.2 Properties of Subsurface Materials

The subsurface materials of Cretaceous age, which underlie the site to 68
a depth of apporximately 444 ft (elevation 366 ft) have been sampled and laboratory tested for a wide range of physical and chemical properties. The materials of the Glen Rose Formation, extend from elevation 810 ft (plant grade) to elevation 610 ft and consist of light-to-dark gray argillaceous limestone with lenses and zones of gray to greenish-gray calcareous claystone. The materials of the Twin Mountains Formation,

68 extend from elevation 610 ft to elevation 366 ft and consist of interbedded claystone and sandstone sequences.

68 The static engineering properties of the subsurface materials underlying the site to a depth of about 50 ft (from plant grade at elevation 810 ft to elevation 760 ft) were determined in the preexcavation stage of the site geotechnical investigations. These properties are discussed in Section 2.5.4.2.1. The dynamic engineering properties of the subsurface materials underlying the site were studied in both the preexcavation and the postconstruction stages of the site geotechnical investigations. The preexcavation investigations were conducted at a number of locations, and were effectively limited to a depth of 200 ft (although some boreholes were drilled to a maximum depth of 400 ft). The postconstruction investigations reached a depth of 500 ft at the location represented by the S/L cluster borings. The dynamic engineering properties are discussed in Section 2.5.4.2.2.

2.5.4.2.1 Static Engineering Properties

68 The static strength and other physical properties of the upper portion of the Glen Rose Formation (Zone I and the top of Zone II) were determined in the laboratory by means of the following types of tests:

1. Unconfined compression tests.

2. Bulk modulus of elasticity determination.

3. Double ring shear test.

consolidation and reduce permeability. A typical stress-strain curve (Lab Number 4) is shown on Figure 2.5.4-2 and a Mohr Diagram for each of the samples tested is shown as Figure 2.5.4-3.

The ratio of lateral strain to vertical strain, Poisson's Ratio, was determined on two selected typical samples of claystone and three selected typical samples of limestone. Both directions of strain were measured using SR-4 Resistant Wire Strain Gauges attached to the surface of the sample by means of appropriate adhesives. As many as four cycles of axial load application were made in order to fully investigate stress-strain characteristics of the typical materials. The results, in graphical form, of these determinations are shown by Figures 2.5.4-4 through 2.5.4-13, inclusive.

The double ring shear strength of selected typical rock samples was determined in pre-chosen direction essentially parallel to the bedding planes. The double ring shear test apparatus consists of three parallel steel plates with openings of a size very slightly greater than the sample diameter. Two of these steel plates are supported on horizontal base and, in turn, support the rock specimen. The third plate or ring, which is sandwiched between the outer plates, acts as a loading device and is allowed to react with the specimen. The results of these determinations are shown on Table 2.5.4-2.

The tensile strength of selected and typical rock materials were determined by the Brazil tension test. These tests are accomplished by placing a rock specimen horizontally between two bearing plates and loading in compression until failure occurs in tension. The results of these determinations are shown in summary form on Table 2.5.4.-3

68 | Table 2.5.4-4 is a summary of the static rock properties for the upper | portion of the Glen Rose Formation on which the major plant structures | were found.

2.5.4.2.2 Dynamic Properties of Subsurface Materials

The dynamic properties of the subsurface materials were determined by 68
field geophysical surveys conducted in both the preexcavation and the postconstruction stages of the site geotechnical investigations, and by laboratory testing of rock core samples obtained from deep postconstruction borings S-1, S-1A, and L-2 which were drilled together with boring L-1 for a detailed crosshole seismic survey. The scope and the results of the geophysical surveys are presented in Section 2.5.4.4. Discussion of the laboratory testing is included in this section.

The laboratory testing program included measurement of unit weights 68
and performance of resonant column testing (which provided shear moduli and damping properties). Out of 79 core pieces delivered to the laboratory, 64 were tested; all of them for unit weight, and 17 were subjected to resonant column testing. These samples were taken from the S/L cluster borings between elevation 809 ft and elevation 428 ft. Within this depth interval of 381 ft, 30 samples were obtained from the Glen Rose Formation (17 claystone, 12 limestone, and 1 sandstone sample), and 34 samples were obtained from the Twin Mountains Formation (11 claystone, 3 limestone, and 20 sandstone samples). A summary of the laboratory test results [132] is shown on Table 2.5.4-5G. Reference 132 discusses the details of the test program and presents data on the variation of shear modulus and critical damping with strain levels.

Considering the range of stiffness indicated by Tables 3.7B-24 through 68
29, and by inspection of the sensitivity of the elastic expressions listed in Tables 3.7B-5 and 6 to variation in shear modulus (G) and Poisson's ratio ( ), it is evident that the profile stiffness represented on Table 2.5.4-5H lies within the ranges of stiffness indicated by Tables 3.7B-25 through 29. On this basis, the additional information provided by the postconstruction geophysical survey

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68 generally confirms the validity of the profile stiffness represented on Table 2.5.4-5.

68 2.5.4.2.3 Rock Core Compositional Analyses

Petrographic, solubility, x-ray and chemical analyses were performed on ten cores of Glen Rose and Twin Mountains formations. These tests indicate that both the Glen Rose and Twin Mountains formations are not susceptible to solutioning which verifies field observations and experience.

1. Petrographic Analysis

The cores were slabbed lengthwise and one slice was kept intact and prepared for observation. It was ground on 220, 400, and 600 grit grinding powder, washed, then etched for 3 seconds in 10% HCl solution, washed, and dried. The procedure was completed for all cores as well as for the rock chips remaining from the thin sections that were prepared. The etched slabs were examined under binocular microscope.

Thin sections were made for each core and were ground to a thickness of approximately 18 microns, after which petrographic modal analyses were completed for each thin section. All work was done on Leitz Dialux-Pol polarizing research microscopes. The description terminology is standard Fold Carbonate Classification with the textural packing classification of R. Dunham added as a supplement. Both classifications are described in detail in American Association of Petroleum Geologists Memoir 1 on the Classification of Carbonate Rocks (ca 1962). Three thin sections were prepared from the cores

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The locations of the borings, wells and piezometers are shown on Figures 2.5.5-5 and 2.5.5-6. The methods used in drilling the borings are discussed in Section 2.5.5.3. The description of the geophysical surveys is presented in Section 2.5.4.4.

Geological cross sections are presented in Figures 2.5.4-40, 2.5.4-41, 68
and 2.5.6-7 through 2.5.6-11. The limits of excavations are shown in
Figure 2.5.4-27.

2.5.4.4 Geophysical Surveys

In the preexcavation period, the following geophysical surveys were 68
conducted at the Station and Safe Shutdown Impoundment Dam areas,
providing data to depths of 200-ft and 100-ft, respectively:

1. Seismic refraction surveys to evaluate the compressional wave velocities of bedrock and the materials overlying bedrock and to evaluate the depth to bedrock;

2. Uphole velocity surveys to provide additional data regarding compressional wave velocities of bedrock and materials overlying bedrock;

3. A surface wave survey to evaluate types of surface waves and their characteristics;

4. Crosshole shear wave surveys to evaluate compressional and 68
shear wave velocities of bedrock and the materials
overlying bedrock;

5. Ambient noise studies to evaluate the predominant characteristics of ground motion due to background noise level; and.

6. Gamma ray and resistivity logs (at Squaw Creek Dam also) to evaluate stratigraphic features.

68 The locations of surveys 1 through 5 are shown on Figure 2.5.4-14. Locations of borings P-10, M-1, D1-2, D1-8A and D1-9 in which survey 6 was performed are shown on Figures 2.5.5-5 and 2.5.5-6. In the postconstruction period, two methods of geophysical exploration were conducted in the deep S/L cluster borings. Both provided detailed information on rock properties from a 500-ft-thick sedimentary section. These methods were:

68 - Crosshold measurement of seismic wave velocities on which rock shear modulus and Poisson's ratio were based; and

68 - Complex borehole logging including gamma-ray, electrical resistivity, gamma gamma, and caliper surveys. The geophysical logs contributed to the straitigraphic refinement of the drilled column. The gamma-gamma logs also provided in situ densities.

68 The geophysical explorations are discussed in detail in the following sections.

2.5.4.4.1 Seismic Refraction Survey

A seismic refraction survey, 4,000 lineal feet in length, was conducted at the Station location along two seismic profiles. A seismic refraction survey (1,500 lineal feet) also was performed at the Safe Shutdown Impoundment (SSI) Dam site. Adjustments were made | 68
to seismic line locations where pipelines were present. Seismic energy was produced by detonation of explosive charges (five to 15 pounds of Nitromon-S; a Du Pont product) placed in drilled holes. These holes ranged in depth from five to 15 feet. The energy released by the charges was detached by vertically-oriented geophones spaced at 50-foot intervals along the seismic profiles. The geophones, manufactured by Electro-Tech Labs, have a natural frequency of 14 Hertz (cycles per second) and are fitted with a spike or metal plate to obtain proper coupling with the site materials. The energy impulse received by the geophones was recorded by a Dresser SIE seismic amplifier system coupled with a Dresser SIE R6 recording oscillograph.

The compressional wave velocities and the corresponding depths to bedrock were evaluated by plotting first arrival times of seismic energy at each geophone against the distance of each geophone from the seismic energy source. The time-distance data for each profile and the corresponding subsurface cross sections of the profiles are shown on Figures 2.5.4-15 through 2.5.4-18. Representative compressional | 68
wave velocity data determined from the seismic refraction survey (and the uphole survey) are summarized on Tables 2.5.4-5B and 2.5.4-5C. The accuracy of the calculated depths to the bedrock surface is considered to be 10 to 20 percent. This estimate is based on a number of factors. The velocity and characteristics of the near surface materials can introduce some error into interpreted depths. Uphole surveys, an examination of soil and core samples, and an examination of the boring logs were used to reduce this type of error. The timing accuracies of the equipment and the seismic caps also lead to the selection of tolerance which acts as a guide in determining the accuracy of the estimate.

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2.5.4.4.2 Uphole Velocity Survey

Uphole velocity surveys were performed at the station and SSI dam locations. These provided a comparison with compressional wave velocities obtained by the seismic refraction surveys at each location. Borings P-1 at the Station location and M-5 on the SSI dam were selected for these surveys. A geophone array was placed on the ground surface around each boring. Small explosive charges were fired at 10-foot intervals from the bottom to the top in each boring. The seismic energy released by the explosives was detected by the geophones in each array. This energy was recorded by the same system
68 that was used in the seismic refraction surveys. The results of the uphole velocity surveys are presented on Figures 2.5.4-19 and 2.5.4-20, as well as on Tables 2.5.4-5B and 2.5.4-5C.

2.5.4.4.3 Surface Wave Survey

A surface wave survey was conducted at the station and SSI dam locations to evaluate the types and characteristics of any observable surface waves.

Two Sprengnether Engineering Seismograph VS-1200, three-component geophones were placed 300 to 350 feet apart on the ground. The output of these geophones was fed into a VS-1100 amplifier with a gain characteristic of 100, and from the amplifier into a special attenuator circuit. The resultant output was recorded on the Dresser SIE R6 recording oscillograph.

Explosive charges were placed in drill holes at varying distances from, but in-line to, the geophones. Explosive charges used in this survey varied from five to 40 pounds of Nitromon-S.

In addition to this procedure, recordings were made of the energy produced by horizontal impacts of an eight-pound sledgehammer. This energy was detected by six one-component (horizontal) geophones spaced five to 20 feet apart.

The compressional and shear wave velocities developed from the surface 68
wave survey are presented on Tables 2.5.4-5B and 2.5.4-5C.

2.5.4.4.4 Crosshole Shear Wave Surveys

Crosshole shear wave surveys were performed at two plant locations and 68
at the SSI dam site utilizing borings drilled into bedrock. These surveys were conducted in two separate time periods; initially, prior to excavation, and subsequently, after construction was essentially completed. The preexcavation survey was made in somewhat different manner than the postconstruction survey. The difference included: the scope and the location of the survey, the techniques applied, and the thickness of the stratigraphic column involved.

During the preexcavation crosshole survey, stationary subsurface shots 68
were used as a source of the seismic energy. Wave propagation from two shot points was recorded in borings P-1 and P-2 at distances ranging from 370 ft to 1,175 ft, and from a single shot point in two preconstruction SSI dam borings M-5 and M-6 at distances of 400 ft and 700 ft, respectively. Compressional and shear wave arrival times were recorded in each of these borings by using 4 to 6 geophones vertically arranged within the upper three zones of the Glen Rose Formation between elevation 650 ft and elevation 818 ft.

For the preexcavation survey, two three-component, low-frequency 68
geophones (Mark Products L-1-3DS) were placed at corresponding elevations deep in each recording borehole. Explosive charges were detonated in drilled holes at fixed distances from each of the

borings. The resultant seismic energy was recorded by the Dresser SIE system. After each shot (energy detonation), each geophone was raised
4 25 feet. Details of the crosshole shear wave survey at both the plant site and the Safe Shutdown Impoundment Dam are presented in Table 2.5.4-5A. In addition to this data, seismic energy developed by a technique of producing horizontally-generated waves was also recorded by each of the geophones at 25-foot intervals. The technique for producing horizontally-generated waves was developed by the Dames & Moore geophysical staff. It involves the use of explosives (Primacord) placed in trenches at the ground surface adjacent to a boring. Two trenches are used, one on either side of a boring. Depending on the site conditions, the trenches vary in length from three to eight feet long, and in depth from six inches to two feet deep. The trenches used on this site were five feet long, and one foot deep. The trenches are "v" shaped with one face cut vertical and the other sloped at some angle to the vertical. This angle will vary for different conditions, but for this survey, an angle of 60 degrees was used. The vertical face of the trenches on either side of a boring are opposed to each other as shown in Figure 2.5.4-21. Primacord is placed in the bottom of each trench and fired simultaneously by the use of seismic caps. The trenches are not backfilled, allowing for a backblast. The resultant detonations produce a horizontal torque, centered around the boring, which is recorded in the boring by use of a three-component geophone.

68 The results of this survey are plotted as an uphole shear survey, with proper calculations for the shot offset and depth. Primary and secondary arrivals are plotted on a time-depth plot. The plot for the survey at boring M-5 is shown on Figure 2.5.4-21.

Shear wave arrivals often are found to be masked by relatively large-amplitude motions caused by multiple reflected and refracted compressional waves and the surface waves. To overcome this difficulty high-energy and low-energy recordings were made at corresponding depths in each of the borings.

The compressional and shear wave velocities, determined in the initial 68
survey by the crosshole techniques and by the surface wave surveys, are presented on Tables 2.5.4-5B and 2.5.4-5C.

During the postconstruction crosshole survey [131] conducted in four 68
closely spaced, up to 500-ft-deep S/L cluster borings (Figure 2.5.4-14), seismic waves were generated by a vertically mobile hammer inserted in "source" boring S-1A, and these waves were detected by three horizontally separated geophones installed at the same elevation in each of the three "listening" borings: S-1, L-1, and L-2. These three "listening" borings were 5 ft, 8 ft, and 9.5 ft, respectively, away from the "source" boring. The measuring points were located within the interval from elevation 797 ft (12 ft below plant grade), which is within Zone I of the Glen Rose Formation, to elevation 317 ft (492 ft below plant grade), which is beneath the unconformity separating the Cretaceous Twin Mountains Formation from the underlying Paleozoic Mineral Wells Formation.

The postconstruction survey was performed at a location immediately 68
west of the Unit 2 turbine building area (Figure 2.5.4-14A). The survey equipment consisted of a four-channel Nicolet Oscilloscope (Model 4094-2), a Bison shear-wave hammer (Model 1465-1), and three Geospace Geophones (Model HS-J-LP3D). The Bison downhole shear-wave hammer was the energy source. This hammer consists of a stationary body with a sliding mass, and is hydraulically locked in the "source" boring. Using a hoist cable, the mass is lifted or dropped to produce impacts against the stationary hammer body. The hammer, designed to permit either upward or downward impacts, generates vertically polarized shear waves of opposing polarities. These polarities allow identification of the shear wave arrivals in the presence of refracted or reflected seismic arrivals in layered formations. Crosshole

68 measurements were normally taken at 3-ft intervals from 0 to 45 ft, at 5-ft intervals from 45 to 300 ft, and at 10-ft intervals thereafter. Additional measurements were performed to determine the seismic wave velocities in claystone layers at depths of approximately 20 to 30 ft and 100 ft. Crosshole measurements were taken at a total of 90 elevations, resulting in a total of about 440 individual records. The survey procedures are described below:

68 - The Bison hammer was suspended in boring S-1A ("source boring") at the desired depth. The hammer shoes were hydraulically extended to clamp the hammer in-place at the elevation.

68 - Geophones were suspended in borings S-1, L-1, and L-2 ("listening" borings) at the same elevation, and the pneumatic tubes were inflated to couple the geophones to the casing wall.

68 - Typically, 10 impacts (upward or downward) were performed with the shear-wave hammer.

68 - The wave motion produced with each individual impact was stored in the Nicolet Model 4094-2 oscilloscope. The motions produced by 10 impacts were averaged to yeild one record for that combination of depth, impact direction, and "listening" boring.

68 - Upon completion of each series of impacts, the data were written onto a floppy disk as a permanent record and labeled to indicate depth, impact direction, and "listening" boring.

68 - This procedure was repeated such that two records for upward impacts and two records for downward impacts were obtained for each "listening " boring at each elevation.

- Upon completion of the recording sequence, the hammer and geophones were released and advanced to the next recording elevation where the procedure was repeated. 68

The borings used for the postconstruction crosshole geophysical surveys were cored either continuously (S-1, and S-1A below 325 ft) or partially (L-1 and L-2). They were logged geophysically (except for S-1), cased by PVC pipe which was grouted (except for S-1), and subjected to a verticality survey to allow accurate determination of interhole separation at all depth levels where crosshole seismic velocity measurements were made. 68

Generalized subsurface conditions at the location of the S/L cluster borings are shown in Figure 2.5.4-30A which includes a generalized geologic column, the geophysical logs of boring S-1A, and a profile of seismic wave velocities. The mean compressional and shear wave velocities, was well as the Poisson's ratio for various depth intervals between plant grade and the bottom of the S/L cluster boreholes (at a depth of 503 ft), are listed on Table 2.5.4-5F. 68

2.5.4.4.5 Ambient Vibration Measurement

Measurements of ground motion due to background (ambient) vibrations were made at each site when drilling rigs or other equipment were not operating.

The three-component VS-1200 Sprengnether Engineering seismograph was used to record ambient ground motions. This seismograph has gain characteristics of 20 inches per inch per second in the velocity mode, 12 inches per inch per second in the acceleration mode and 200 inches per inch in the displacement mode. A VS-1100 amplifier with a gain characteristic of 100 was used for all recordings. The resultant maximum gain level is 2,000 for the velocity mode, 1,200 for the acceleration mode and 20,000 for the displacement mode.

The three components of ground motion measured were radial, vertical and transverse. The seismometer was oriented facing either north or 4 east. Table 2.5.4-5D presents the results of the ambient ground motion measurements.

68 2.5.4.4.6 Borehole Geophysical

68 Gamma ray and resistivity surveys were made in preexcavation borings P-10, DI-2, DI-9, M-1 and DI-8A to aid in evaluating the correlating on-site stratigraphy. These boring locations are shown on Figures 2.5.5-5 and 2.5.5-6. The survey results are illustrated on Figures 2.5.4-22 through 2.5.4-26.

68 Gamma-ray, electrical resistivity, gamma-gamma, and caliper surveys were made in postconstruction boreholes S-1A, L-1, and L-2 as an aid to correlation of the borings and for refinement of the site stratigraphy. The gamma-gamma survey also provided a continuous record of in situ densities. The location of the borings are shown in Figures 2.5.4-14A and 2.5.5-5. The geophysical logs of these borings are almost identical due to closeness of the boreholes; the survey results are well represented by the logs of boreholes S-1A, which are shown in Figure 2.5.4-30A.

2.5.4.4.7 Interpretation of Geophysical Data

The refraction survey data indicated slight variations in the compressional wave velocity of the bedrock at the Station location. This variation in compressional wave velocities is most prevalent at shallow depth near the edges of the peninsula on which the Station is located.

The compressional wave velocities from the uphole velocity survey are different from the compressional wave velocities from the refraction survey. This is because a refraction survey develops an average compressional wave velocity over a large lateral distance, whereas the uphole velocity survey records the compressional wave velocity around the isolated point (i.e., the boring). It is felt that the compressional wave velocities obtained by the seismic refraction survey are a more appropriate measure of the actual dynamic properties of the site materials.

In the preexcavation stage of the site geophysical investigation, the 68
geophysical surveys did not penetrate into the Twin Mountains
Formation. However, indirect geophysical information from the
surface and shear wave surveys indicates that both the compressional
and shear wave velocities of the Twin Mountains Formation are lower
than the corresponding velocities of the Glen Rose Formation. This
was supported by the geologic borings which obtained basically 100
percent core recovery within the Glen Rose Formation and a much lower
percentage recovery and Rock Quality Determination (RQD) in the
underlying Twin Mountains Formation. The velocity inversion between
the Glen Rose and Twin Mountains Formations was assumed to result in a
seismic energy response characterized by the Glen Rose Formation
acting as a seismic energy channel, or in essence, a plate of
limestone. All these conclusions and assumptions have been 68
substantiated by the detailed postconstruction crosshole seismic
survey that penetrated through the Twin Mountains Formation into the
uppermost part of the underlying Mineral Wells Formation.

Surface waves at this site have unusual characteristics because of the seismic velocity inversion. The uppermost higher velocity Glen Rose Formation which acts as a plate or seismic channel, produces several discrete, but overlapping Rayleigh wave trains. These wave trains vary from 15 to 25 hertz, and from 5,000 to 2,500 feet per second (respectively) apparent velocity. The wave trains are essentially trapped waves within the plate of limestone and indicate an average shear wave velocity greater than 5,000 feet per second in the limestone. These waves are apparently second of higher mode vibrations within the plate. The maximum amplitudes and longest wave train have a frequency of 25 hertz. The maximum amplification of seismic energy, from an explosive source detonated at shallow depth is, therefore, at 25 hertz.

Some of the seismic records show a low-frequency wave train of about five to seven hertz. This wave is not well defined, but is probably the first mode Rayleigh wave vibration in the limestone plate.

68 The refraction and surface wave surveys are considered to give reasonably accurate data for the determination of compressional wave velocities while equally reliable date for the determination of the shear wave velocities was obtained by the Primacord technique. The surface wave surveys also produced a set of plausible values, with the initial crosshole surveys producing the least reliable set of values. However, the postconstruction crosshole survey produced more detailed measurements of both compressional and shear wave velocities. In comparision, the preexcavation geophysical explorations were not so detailed, systematic, and deeply penetrating. The postconstruction velocity measurements, together with the gamma-gamma log-derived densities, were the basis for the final design values of shear modulus and Poisson's ratio for the generalized profile (Table 2.5.4-5E). The initially selected values of the parameters based on the preexcavation geophysical surveys, are shown on Table 2.4.

68 The dynamic rock properties based on both the preexcavation and postconstruction geophysical surveys are presented in Table 2.5.4-5H. It should be noted that in the final determinations of compressional and shear wave velocities the information from all the techniques are integrated and used, with an understanding of the reliabilities, to produce the reported results.

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2.5.4.5 Excavations and Backfill

The limits of required excavation for the power plant is shown on 68
Figure 2.5.4-27. Figure 2.5.4-28 shows the location of the plant
site and all cut and fill areas that occur adjacent to the plant site.

Figure 2.5.4-30 shows a 43-ft thick geologic section of the rock 68
encountered in the excavations for Category I structures. This
section is typical for the Category I portion of the power plant and
related adjacent Category I support facility excavations. The
limestone bed identified as bed "R" in Figure 2.5.4-30 is a reference
bed in the plant site. The elevation of the top of bed "R" in Unit 1
reactor excavation was 782 feet 7 inches and 780 feet 9 inches in the
Service Water Intake Structure excavation. This difference is due to
the undulating nature of the beds and also because the Service Water
Intake is downdip of Unit 1 Containment. Figure 2.5.4-30A is a 68
general subsurface profile which includes a 500-ft thick geologic
section of the materials constituting the foundation base of all plant
structures.

Structural backfill was required on the north side of the Service 68
Water Intake Structure. Figure 2.5.4-31 is a cross section of this
area.

All Seismic Category I Electrical Duct Banks and the Service Water 63
Pipe Trench require Category I backfill and bedding and are delineated
in Figure 2.5.4-27. Figure 2.5.4-32 shows typical sections of the
Service Water Pipe Trench and a Seismic Category I Electrical Duct
Bank.

68 Geologic sections and blast-fracture maps were constructed as each
excavation was completed for Category I structures. Figures 2.5.4-
33, 2.5.4-34 and 2.5.4-35 are the fracture maps developed for the Unit
68 1 Containment and reactor excavation. Figure 2.5.4-36 is the photo
grid for photographs taken of the Unit I Containment and Reactor
excavation walls and Figure 2.5.4-37 (21 sheets) are the photographs
depicted in Figure 2.5.4-36.

Excavation methods consisted of blasting, utilizing the presplit, primacord method, or heavy equipment demolition tools where blasting was not an acceptable method. Removal of the broken rock was accomplished by heavy equipment.

Protection of the exposed rock against weathering was required due to the argillaceous nature of some of the limestone beds and the claystone beds which are present. The wall portion of the excavations was protected by the application of shotcrete. The floor portion was protected by a minimum three inch thick concrete seal slab.

No groundwater was encountered in the primary Glen Rose Limestone, and therefore only normal pumping equipment and procedures were required to remove storm runoff and concrete curing water which collected in the open excavations.

Upon completion of removal of material from an excavation, a preliminary inspection was performed by the site geologist. After the final cleanup, just prior to placement of shotcrete or seal slab, photographs were taken and blast fracture maps were developed by the site geologist. At the time the geologist approved an excavation, the QA/QC program became involved, assuring cleanliness prior to application, curing and testing of protective coatings, and the inspection of these protective coatings prior to being covered by structural concrete.

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To measure the rebound of the foundation rock due to the excavation on 68
the plant site, two extensometers were installed. Figure 2.5.4-28
shows the locations of the extensometers. Monitoring results are
presented in section 2.5.4.13.

The source for Category I backfill material is on site. Figure 68
2.5.4-29 shows the location of the quarries.

During the preliminary grading and excavation of the plant site area,
down to elevation 810', a bed of very hard, tan, fossiliferous,
crystalline limestone, was intersected at an elevation of 825'. This
particular bed is present throughout the site area and an initial
quarry site northwest of the plant (Pit "A", Figure 2.5.4-29) was
selected for sampling and testing. Material was quarried from Pit "A" 2
and processed through the crusher plant on site, a sample pulled and
graded to the proper gradation, and then subjected to cyclic
laboratory testing to verify that the material met specification
requirements. Also sampled for testing was the concrete agregate 68
sand which is being used as a bedding material around the pipe in the
Service Water Pipe Trench and under the Seismic Category I Electrical
Manholes. A complete discussion of this testing is covered in
Section 2.5.4.7.

The compaction requirement for the backfill materials, placed against
the Service Water Intake Structure, is a minimum of 95% of the maximum
dry density as determined in accordance with AASHTO Designation T-99,
Method D. In-place density was determined in accordance with ASTM
D1556 (Sand Cone), ASTM D2167 (Balloon), or ASTM D3017 and D2922
(Nuclear). These density requirements apply to all Category I 68
backfill material. Figure 2.5.4-38 shows the gradation requirements
for the backfill material.

68 Based on the results of in-place density tests for Category I
backfill, the following average backfill densities were obtained:

68 $\gamma Dry = 130$ pcf
68 $\gamma Wet = 135$ pcf
68 $\gamma Sat = 140$ pcf

68 The compaction requirement for the bedding material placed in both the
Seismic Category I Electrical Duct Banks and Service Water Pipe
Trench, is a minimum density of not less than 80% of the relative
density as determined by ASTM Test Designation D2049, latest revision
effective prior to September 4, 1975. In-place density was
11 determined in accordance with ASTM D1556 (Sand Cone), ASTM D2167
(Balloon), or ASTM D2922 (Nuclear). Figure 2.5.4-38 shows the
68 gradation requirements for the bedding material. The gradation,
relative density, and percent compaction summaries are presented in
Figures 2.5.4-44 thru 53.

68 Based on the results of in-place density tests for Category I bedding,
the following average densities were obtained:

68 γ Dry = 120 pcf
68 γ Wet = 130 pcf
68 γ Sat = 135 pcf

2.5.4.6 Groundwater Conditions

Q371.7

4 A detailed description of groundwater is presented in Section 2.4.13.
No groundwater was encountered during excavation for the plant
foundation. Groundwater observations from piezometers installed at
the site are provided on Figure 2.5.5-77.

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2.5.4.7 Response of Soil to Dynamic Loading 68

The plant structures are founded on rock, however, granular material 68
was required as backfill, pipe bedding and for the dam filters. The
response of these materials to dynamic loading was assumed during the
initial design phase. Later, an investigation of the actual dynamic
strengths of these materials was performed by means of a cyclic
triaxial testing program.

During design, adopted values for the cyclic shear strength for 68
Category I backfill and bedding material were based on the published
data for granular soils. A discussion of the published data is
presented in Section 2.5.4.3.4. Since the D_{50} of the bedding 68
material was approximately the same as for Filter "A" for the SSI Dam,
the material was assumed to have essentially the same cyclic shear
strength characteristics. Category I backfill material has a D_{50}
of approximately 15mm and since this material is coarser than the
published data utilized, the cyclic shear strength characteristics of
this material should be greater. For design, the cyclic shear
strength criteria was assumed based on the data presented on Figure
2.5.6-48. The specific criteria was that the material with the
gradation limits as shown on Figure 2.5.4-38 shall not allow
development of double amplitude strain larger than 5% under specific
corresponding stress conditions shown on Table No. 2.5.4-11.

In order to verify the adopted cyclic shear strength criteria for 68
Category I bedding, backfill and filter materials, a test program
utilizing cyclic triaxial tests was developed. A minimum of four
cyclic triaxial tests were performed for each type of material
utilized; one at each confining pressure as listed on Table 2.5.4-11.
The applied stress given for the development of 5% double amplitude 2
strain in five cycles was utilized since this stress is more critical
than the stress combination specified for ten cycles of

2 loading. Thus any specimen that developed 5% strain at ten cycles of load under a higher stress level automatically meets the criteria specified at ten cycles.

2 Four (4) different materials have been tested to determine if they meet the cyclic shear strength criteria as described above. The materials which have been tested are as follows:

	Sample No.	Material
2	1	Filter A Material
2	2	ASTM C-33, Fine Aggregate
2	3	Glen Rose "Gray" Limestone
2		Crushed Stone
2	4	Glen Rose Limestone "Crowder
2		Quarry" Crushed Stone

68 The gradation limits of these materials, as well as the actual gradation curve of the tested materials, are shown are Figures 2.5.4-38A through 2.5.4-38D, respectively. The densities of the four (4) materials tested are summarized on Table 2.5.4-12.

68 For conventional laboratory triaxial testing of soils, it is generally recognized that the diameter of the test speciman should be at least six times the maximum particle size to avoid possible effects of sample geometry on test data. For Samples 3 and 4, the maximum size for the crushed stone was on the order of 2 inches, which would require triaxial cells capable of taking 12-inch test specimens. Such large cells, especially for dynamic testing, are not available for cyclic triaxial testing; therefore, modeling techniques have been utilized to model the field gradation of Samples 3 and 4.

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One technique recommended by John Lowe [134] describes a technique of 68
preparing a model soil with a grain size distribution curve parallel
to the field material. Research on laboratory modeling of soils
utilizing the parallel grading curve method has shown that the method 2
can be used to predict the static and dynamic strength and deformation
characteristics of soils. For Samples 3 and 4, a laboratory 68
gradation range parallel to the field gradation requirements of the
material has been drawn and an artificially blended material has been
prepared (within these limits) and utilized for the cyclic triaxial
testing. (See Figures 2.5.4-38 C & D). For Samples 1 and 2,
samples falling within the field gradation curves were obtained and 68
were utilized for the cyclic triaxial testing. The test program
followed guidelines set up in Report No. NUREG-31 entitled 2
"Laboratory Triaxial Testing Procedures to Determine the Cyclic
Strength of Soils", which was prepared under Contract No. NRC-E (11-
1) 2433 for the U.S. Nuclear Regulatory Commission.

The results of the initial tests are summarized in Table 2.5.4-13. 2
One plot for each material tested at various confining pressures, 68
relating double amplitude axial strain to the number of cycles of
constant cyclic shear stress, are shown on Figures 2.5.4-39, Sheets 1
through 4. All of the materials tested were of sufficient strength
to meet the cyclic strain criteria at 5 cycles of load. The ASTM C-
33, Fine Aggregate material (Sample 2) and the "Crowder Quarry" 2
Crushed Stone (Sample 4) also met the strain criteria at 10 cycles of
applied load the Glen Rose Limestone Crushed Stone (Sample 3) met
the strain criteria at 10 cycles for confining pressures of 2000 and
4000 pounds per square foot. However, the test results showed that 68
the Glen Rose crushed limestone at 6000 and 8000 psf confining
pressure, (Sample 3) and Filter "A" (Sample 1) did not meet the strain
criteria in 10 cycles at the stress ratios

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68 utilized in the test program. Therefore, additional tests were run on Sample 1 and Sample 3 at reduced stress levels corresponding to stress ratios required to develop 5 percent strain in 10 cycles of
68 load. The results of these tests are summarized in Table 2.5.4-13A.

68 For Sample 3, additional tests at reduced stress levels showed that the material did not meet the cyclic strain criteria at effective normal pressure of 6000 and 8000 psf in 10 cycles of load. The
2 figures relating double amplitude strain to number of cycles for Sample 3 show that the cyclic strains were relatively low until about 6 to 8 cycles, after which large strains developed quite rapidly in only a few additional cycles and failure developed due to necking of the specimen.

68 For Sample 1, Filter "A" material, additional tests at reduced stress levels showed that the material did not meet the specification requirements at 10 cycles of loading. Utilizing all the test results
2 for Filter "A" material, a liquefaction curve has been developed for Filter "A" material and is shown on Figure 2.5-16. From this liquefaction curve, a new curve has been developed relating the cyclic shear stress required to cause 5 percent strain to normal effective stress and is presented as Figure 2.5A-17.

68 The cyclic shear strength criteria of Filter "A" material for 10 cycles of load should be as follows:

	Effective Normal Pressure Kips per Sq. Ft.	Cyclic Shear Stress Required to Cause 5×10^{-2} strain
2	0	0
2	2.0	0.90
2	4.0	1.60
2	6.0	2.20
2	8.0	2.80

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In summary, the following points should be made: 2

1) Filter "A" material met the cyclic shear criteria for 5 2
cycles of load. The criteria for 10 cycles of load
described in the previous paragraph has been adopted for
design of pipe bedding material, where Filter "A" material
is utilized in construction as a bedding material. How 4
these results affect the stability of the SSI Dam is
discussed in Appendix 2.5A.

2) ASTM C-33, Fine Aggregate met all of the cyclic shear 68
strength criteria as shown in Table 2.5.4-11.

3) Glen Rose Limestone Crushed Stone met all the cyclic shear 68
strength criteria as shown in Table 2.5.4-11 except at
normal effective stresses of 6 and 8 KSF at 10 cycles of
load. This material can be utilized as backfill material
up to a confining pressure of 4 KSY (i.e., backfill to 2
approximately 33 feet in depth.).

4) "Crowder Quarry" Crushed Stone met all the cyclic shear 68
strength criteria as shown in Table 2.5.4-11.

To date, all Category I backfill has been "Crowder Quarry" crushed 68
stone which meets all specifications for cyclic testing when compacted
to the required density and the gradation of the materials meets the
gradation requirements. Bedding material for Category I pipe has
been ASTM C-33 fine aggregate sand which also meets all cyclic shear
strength criteria. As stated previously the effect of the cyclic
strength tests results on the stability of the SSI Dam are discussed 2
in Appendix 2.5A.

2.5.4.8 Liquefaction Potential

The entire Nuclear Power Plant foundation consists of firm, unweathered, Glen Rose Limestone with no liquefaction susceptible soils present.

68 The cyclic shear strength of all Category I backfill and bedding materials used show that there is no liquefaction potential; all materials used meet or exceed the design criteria for cyclic strain for the ground acceleration of 0.12 g adopted for the Safe Shutdown Earthquake.

68 2.5.4.9 Earthquake Design Basis

68 A conservative Safe Shutdown Earthquake (SSE) having a peak horizontal ground acceleration at the top of bedrock of 0.12 g has been selected for design (Section 2.5.2.6). The Operating Basis Earthquake (OBE) is equal to 1/2 the SSE. Design response spectra and artificial time-history records are described in Section 3.7B.

2.5.4.10 Static Stability

68 2.5.4.10.1 Bearing Capacity

68 The major structures at the plant have been constructed on individual mat foundations (except individual spread footings under the Turbine Building columns) founded in the Glen Rose Limestone Formation. Properties of the limestone as well as the claystone material were determined and are presented in Section 2.5.4.2.

The ultimate bearing capacity of the Glen Rose Formation located under 68
each structure has been determined from equations derived for local shear failure from the theory of elasticity [133] as follows:

$$q\ (\text{ultimate}) = (\text{shear strength}) \times (3.14)$$

The shear strength has been taken as one-half of the unconfined compressive strength. The above equation does not take into account 68
the effects of overburden or the size of the foundation which would increase the safety factor against a bearing capacity failure. A study of the founding elevations indicated that some upper structures are founded on the claystone. However, when considering the anticipated pressure bulbs due to the applied loads, it is unrealistic to assume that claystone alone is subject to the applies stress. In fact, any shear failure occurring in the claystone zones would also have to pass through the thick underlying limestone layers.

68

From the results of the unconfined compression tests presented in 68
Table 2.5.4-1, a reasonable average value of unconfined compressive strength for the claystone below elevation 810 is 40 ksf. This results in a very conservative value of 60 ksf for the ultimate bearing capacity of rock beneath Category I structures. The factor of safety against a local bearing capacity failure is the ultimate bearing capacity divided by the maximum foundation edge pressure. Table 2.5.4-6 summarizes the static bearing capacity factors of safety for the Category I structures. The safety factors against local shear failure exceed the minimum allowable of higher than the values presented in Table 2.5.4-6 due to the effects of embedment and foundation size.

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68 The dynamic bearing capacity factors of safety were calculated for the maximum loads caused by the SSE. Table 2.5.4-6a presents the results of this analysis. The safety factors exceed the minimum allowable of 2.

68 The maximum edge pressure resulting from the overturning effects of earthquake or tornado loadings is 24.9 Kips per square foot under the Containment structure. The safety factor against overturning of the Containment is 2.0. It is calculated by taking moments about the lower edge of the foundation mat using the overturning forces generated by the safe shutdown earthquake. Vertical pressure distribution under the Containment during the SSE is shown in Figure 2.5.4-42.

2.5.4.10.2 Settlement of Foundations

68 The foundations of all structures in the plant area are founded on the Glen Rose formation. The removal of the overburden to elevation 810 resulted in a slight elastic rebound measured by extensometers to be 0.02 inches (Section 2.5.4.13). Due to the nature of the supporting rock and plant loads, a very small elastic settlement of the structures is anticipated. Time-dependent consolidation settlement of the claystone layers will not be considered due to the fact that these layers are heavily overconsolidated and are minor in proportion to the limestone over an approximate 200 feet thick depth of influence.

68 The settlements of the Category I structures were calculated using the theory of elasticity for a uniformly loaded area on an elastic half-space. The following equation [138] was utilized:

68

$$s = C_B \; p \; B \; \frac{(1 - \nu)^2}{E}$$

68

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where: s = elastic settlement | 68
C_B = shape and rigidity factor | 68
p = uniform foundation pressure | 68
B = width of rectangular footing or diameter of a | 68
circular footing | 68
ν = static Poisson's ratio, 0.25 | 68
E = static elastic modulus, 4.8×10^5 psi. | 68

The rock mass deformation modulus was estimated based on site seismic surveys (Table 2.5.4-5) with a correction for rock mass quality (RQD) [139]. The modulus was also checked using the results of moduli determined by laboratory methods (Table 2.5.4-4). Uncertainty of the effect of blasting and construction activities on the in-situ modulus was considered when evaluating an appropriate modulus for the use in the analysis. The modulus used for the settlement analysis was 4.8×10^5 psi. | 68

A summary of the settlement of the Category I structures is presented in Table 2.5.4-7. The maximum predicted settlement is 0.26 inches at the center of the reactor containment. Since the settlement is elastic, it should occur immediately after final load application. | 68

| Q362.4

Expansive characteristics of the claystone were measured by conducting an absorption-pressure-swell test. The results of the test indicate | 68
that introduction of water to the sample could generate a pressure of 1140 pounds per square foot in the upward direction when swell is prevented. Therefore, is the applied load equals or exceeds this value, no swell should be anticipated. However, if the loads on the | 68
claystone layer are less than 1140 pounds per square foot, some swelling will take place. Figure 2.5.4-43 relates the total potential swell to the thickness of a claystone layer. The interpretation of this graph is shown by use of a hypothetical example | 2
on the same sheet. The small amount of swell calculated indicates no problems are anticipated due to the swell potential of the claystone.

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2 During construction, additional samples of the claystone layers were obtained and tested to evaluate the swell potential of the claystone beneath Category I Pipe Structure.

Q362.4

2 The tests showed a maximum of free-swell of 3.0 percent and ranged from 0.7 to 3.0 percent. The maximum pressure to restrain swelling was 1940 pounds per square foot and ranged from 517 to 1940 pounds per square foot. These values are directly comparable with values reported during design (3.0 percent free-swell and 1875 pounds per square inch absorption pressure).

Q362.12

2 Along the alignment of the Category I Piping, an analysis shows the
swell potential to be in the range of 0.01 inches, which is considered
4 negligible. Therefore, in areas where intact rock has been exposed
above the 15 foot line, the removal of claystone beds was not
warranted. In areas where pre-splitting of the rock had already been
carried down to the 15 foot depth, all fractured rock has been removed
to reach the intact sound Glen Rose Limestone and Category I backfill
material has been placed. Figure 2.5.4-37A is a plan view of the
Surface Water Intake and Discharge pipe trench excavation showing the
intact rock elevation along the alignment.

2 2.5.4.10.3 Lateral Forces

68

68 The magnitude and distribution of lateral earth pressures is a function of the allowable yielding of the wall, the backfill or rock characteristics, water pressure, surcharge loads from adjacent structures, and, for seismically designed structures, the earthquake loading. The concrete foundation walls were conservatively assumed to be rigid, unyielding walls. Therefore, the coefficient of earth pressure at rest, K_O, has been used in evaluating lateral soil pressures on these walls. A value of $K_O = 0.47$ was used based on an assumed friction angle of [illegible] for backfill at the site.

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It was conservatively assumed that the groundwater table is at plant grade (elevation 810) except at the service water intake structure where plantside water level was taken as elevation 780. The actual groundwater level conditions will be verified by piezometric data (Section 2.5.4.13). 68

Q362.4

For structures cast against excavated rock wall, lateral pressure against the walls has been taken as: 2

$$\sigma_h = \sigma_v \left(\frac{\mu}{1-\mu}\right) = 0.33\ \sigma_v$$

4

Where: σh = horizontal stress or pressure, psf 4
σv = vertical stress or pressure, psf 4
μ = Poisson's Ratio - 0.25 4

Q362.4

The vertical stress used for the individual structures should also include any surcharge load which exists. 2

68

This is an estimate of rock pressure which does not take into account the relative stiffness of the wall and the rock mass. In some cases, a more detailed evaluation of rock pressures based on the unique geological discontinuities was performed. 68

Dynamic loadings include pressures due to the soil or rock mass, water, and surcharge, accelerated in the vertical and horizontal directions. Figure 2.5.4-54 graphically depicts the static and dynamic lateral earth pressures. 68

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2.5.4.11 Design Criteria

68

68 Methods used to evaluate bearing capacity, settlement and lateral earth pressures are discussed in Section 2.5.4.10. Soil and rock properties used in the analyses are provided in Sections 2.5.4.4 and 2.5.4.5. The slope stability of the SSI and Squaw Creek dams is discussed in Section 2.5.6.

68 The minimum acceptable design factor of safety for bearing capacity is 3.0 under static loading conditions and 2.0 for dynamic conditions. The minimum allowable factor of safety for the SSI dam slope stability is 1.5 for static conditions and 1.1 for dynamic conditions.

2.5.4.12 Techniques to Improve Subsurface Conditions

No special techniques were required to improve foundation conditions except where blasting shattered or fractured the rock adjacent to or within the plant foundation.

68 In order to improve these areas of exception, dental concrete was used to replace fractured rock and to fill overexcavated areas. In areas where fractures not requiring removal were encountered, grout pipes were installed. Grouting operations are to be performed at the appropriate time, i.e., when the foundation affected was completed to the point that the grout was confined.

2.5.4.13 Subsurface Instrumentation

2.5.4.13.1 Extensometers 68

Two extensometers were installed in locations shown on Figure 2.5.4-28. Extensometers were installed during removal of common material from the plant site to measure the rebound of the foundation material and were monitored during the excavation phase of plant construction. At the end of the excavation phase the extensometers were terminated and the data evaluated. The measured rebound was 0.02 inches which is well within the anticipated amount of rebound.

2.5.4.13.2 Post-construction Groundwater Monitoring 68

A system of 15 piezometers will be installed in the plant area in 68
order to monitor groundwater conditions. The location of the piezometers is shown in Fig. 2.5.4-55 and the typical piezometer detail is shown in Fig. 2.5.4-56.

Design of plant structures initially assumed groundwater at elevation 68
775 ft. (Section 2.4.13). Installation and observation of these piezometers will verify the actual groundwater condition at the site. The piezometer boreholes will be completed at test borings, with continuous NX rock core and water pressure tests throughout the cored interval. Upon completion of piezometer installation, groundwater data will be collected to determine seasonal variations. The groundwater level will also be correlated with weekly rainfall.

2.5.4.14 Construction Notes

Required excavation operations, including blasting, at times resulted in fracturing or shattering of the foundation rock adjacent to or within the plant foundation.

Section 2.5.4.12 describes the treatment of shattered or fractured foundation rock.

Frequent changes in construction techniques were made in the early stages of excavation in the plant area. All such changes were within specification requirements, were made to reduce fracturing of rock, and included the following techniques:

1. Line drilling to isolate the proposed excavation area

2. Pre-splitting

3. Rock sawing (isolation)

4. Blast hole spacing

5. Amount and type of explosives

6. Time-delay of multiple blasts

No significant construction problems, other than rock fracturing, have been encountered.

68 During construction small volumes of water in-leakage at various structures was observed. The in-leakage problem exists at the turbine building and various Category I structures. This condition was solved at the turbine building by installing a permanent drainage system.

68 The in-leakage is caused by perched water collected at the foundation exteriors. The perched water source is mainly the result of surface water, such as rainfall runoff or construction and maintenance runoff. The water migrates to the foundation exterior surfaces through the utility trenches, filled with permeable bedding and backfill material, that intersect the structures.

Fifteen piezometers (Section 2.5.4.13.2) will be installed and will be 68
monitored as permanent piezometers. The perched water condition around the plant structures will be evaluated and remedial actions implemented.

2.5.5 STABILITY OF SLOPES

2.5.5.1 Slope Characteristics

There are no slopes, either natural or man-made, in the vicinity of 68
the plant, whose failure could adversely effect the plant.

The existing slopes along the creek bed that form the reservoir behind the Safe Shutdown Impoundment Dam are comparatively gently slopes and are characterized by a very thin mantle of soil underlain by limestone of the Glen Rose Formation. In many areas, all soil cover has been eroded and the rock outcrops. The steepest slope present along the reservoir boundary occurs on the south side of the Safe Shutdown Impoundment and approximately 600 feet upstream from the axis of the SSI Dam (Figure 2.5.5-2). Current topographic data indicates that at this location, the ground elevation increases from elevation 750 to elevation 800 over a horizontal distance of approximately 150 feet, or a slope of approximately 3 horizontal to 1 vertical. All other slopes along the total impoundment shoreline are in the range of 4 horizontal to 1 vertical, or flatter.

Topography of a part of the reservoir in the plant site vicinity, 68
shown on a 10 foot contour interval is presented on Figure 2.5.5-5 together with the location of Sections Numbered 1-1, 2-2, 3-3, and 4-4. These sections include the steepest slope condition along the shoreline of the SSI reservoir.

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Subsurface profiles at each location were obtained by trench excavation. A wedge-type stability analysis using moist, saturated and submerged unit weights, appropriate strength properties for the materials present, and a seismic coefficient of 0.12 g was made for Sections 1-1, 2-2, 3-3 and 4-4. The results indicate a minimum safety factor against a slope slide of greater than 3. The Sections, including the results of the wedge-type stability analysis, are shown on Figures 2.5.5-1, 2.5.5-2, 2.5.5-3, and 2.5.5-4.

Some degree of weathering of the rock outcrops has occurred, and if one assumes that the weathering has extended to such a degree as to be the equivalent of a granular material with a 0 angle of 35 degrees, the existing slope will have a safety factor against a shallow near-surface slide greater than 2.0, a value considered entirely adequate. Any slide deeper than that of the weathered rock will have a larger safety factor.

2.5.5.2 Design Criteria and Analyses

4 | The method of dynamic analysis of the SSI Dam is described in Section | 3.7B.2.13 and the details of the analysis are presented in Section | 2.5.6.5.

68 | 2.5.5.3 Log of Borings

The locations of borings made for the plant and dam explorations are shown in Figures 2.5.5-5 and 2.5.5-6. The logs of these borings are presented in Figures 2.5.5-7 through 2.5.5-76.

Borings were drilled with truck-mounted rotary-wash or auger drilling equipment. Overburden soil samples suitable for testing were obtained from some of the borings, and the underlying rock has been continuously cored utilizing standard NX coring equipment. The soils encountered have been classified in accordance with the Unified Soil

Classification System. Geologic rock classifications have been based on macroscopic and hand lens examination of cores. Percent core recovery and R.Q.D. (Rock Quality Designation) information are presented on the boring logs.

The borings were drilled in the overburden soils by advancing a 3 or 4 inch diameter hole to the desired sampling depth. Undisturbed soil samples were obtained by hydraulically pressing a 2-inch Shelby tube or a 2.5-inch diameter thinwall sampler. Some disturbed samples were obtained from the auger flight, and disturbed Standard Penetration Test samples were also taken.

Drilling mud and casing were used when necessary during the drilling of overburden soils. Prior to coring the underlying rock, each boring was flushed thoroughly with water.

Two-inch diameter plastic pipes were installed in several borings to enable observation of groundwater levels (Figure 2.5.5-5). Because the Glen Rose formation is essentially impermeable, water levels reach static equilibrium slowly. Therefore, observations were continued for an extended period to confirm static levels. The observations made in borings at Category I facilities are summarized in Figure 2.5.5-77.

The static water level in the Twin Mountains formation was observed in Boring P-10, one of two borings penetrating the stratum. The piezometric level is indicated to be Elevation 670.

2.5.5.4 Compacted Fill

A detailed discussion of the areas requiring fill and the sources of fill materials required is given in Section 2.5.4.5. The extent of these fill areas are shown on Figure 2.5.4-28. | 68 | 4

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2.5.6 EMBANKMENTS AND DAMS

2.5.6.1 General

2.5.6.1.1 Squaw Creek Dam

The purpose of Squaw Creek Dam (SCD) is to impound a cooling lake for
68 CPSES. The location and configuration of the reservoir are shown in
Figures 2.5.6-1 and 2.4-1. Under normal conditions, the reservoir
will remain in the five-ft range between a normal minimum operating
level at elevation 770.0 and the crest of the service spillway at
elevation 775.0

The layout of Squaw Creek Dam is shown in Figure 2.4-16. A typical cross-section of the embankment is shown in Figure 2.4-17. The dam is designed to pass the Probable Maximum Flood with sufficient freeboard to prevent overtopping by wave action generated by a sustained 40 mph overland wind. The top of the dam is a elevation 796.0 with overbuild to elevation 798.4 to compensate for estimated settlement. The central section is constructed of select, impervious material, with a cutoff extending down to impervious foundation material. The outer zones of the embankment are of less select material with the outer portion of the downstream shell being rock fill. A filter system separates the impervious central zone from the less select outer zone on the downstream side and extends outward to the downstream toe to provide drainage and piping protection for the core.

The reservoir side of the dam is protected by rock riprap and a gravel blanket from the top of the embankment to elevation 760.0, which is 10.0 ft. below the minimum operating level. The top width of the embankment is 20 ft., exclusive of riprap and blanket. Design of the riprap was based on an average over-water wind of 95 mph.

The service spillway is an uncontrolled structure 100 ft wide, with an ogee crest at elevation 775.0 Additional discharge capacity for protection from extreme floods is provided by a broadcrest emergency spillway, 2,200 ft. wide, excavated in rock of the east abutment at elevation 783.0. The sides of the discharge channel from the stilling basin to Squaw Creek are protected with rock riprap and blanket. The floor of the channel is native limestone. A 12-inch makeup water pipeline crosses the emergency spillway along its crest. This line is placed in a trench cut in limestone and is covered with a concrete cap. The top of this cap is at elevation 783.0.

Service outlet facilities are located near the right abutment of the dam and consist of an intake tower and outlet conduit. The tower has a 6 ft. by 6 ft. control gate at its base leading to a 6 ft. diameter conduit through the dam that discharges to Squaw Creek. The tower has ports at various levels to allow selective taking of water from the reservoir. Backup closure facilities are provided to protect against loss of storage due to inability to close a valve or gate.

2.5.6.1.2 Safe Shutdown Impoundment Dam

A portion of the arm of the reservoir that is formed by the channel of Panther Branch is utilized as a Safe Shutdown Impoundment, which holds water for emergency cooling use. The secondary reservoir is separated from the main body of the reservoir by a rock fill dam. An open spillway channel was excavated through the narrow ridge to the southwest of the SSI Dam, to connect the SSI with the main body of the reservoir. The floor of the channel is at elevation 769.5, six in. below the normal minimum operating level, and water will pass back and forth to keep the large and small reservoir surfaces at the same elevation under normal operating conditions. If the level in the main reservoir should drop due to some emergency, the SSI Dam will hold back 367 acre-ft of reserve water to allow continued cooling and safe shutdown of the plant.

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The location and layout of the SSI Dam are shown in Figures 2.5.6-1 and 2.4-23. Details of the SSI Dam embankment are shown in Figure 2.4-21. The middle zone is of select, impervious material, wetted and rolled, and carried down to impervious foundation material for effective cut-off. The outer zones are rock fill. Two-stage filters were placed between the core and the rock fill sections. The embankment crest is 40 feet wide at elevation 796.0. The spillway channel is 40 feet wide, with side slopes of 1-on-3 and a channel slope of 0.003 upstream and downstream from a concrete wall approximately flush with the channel bottom.

The SSI Dam is designed to pass the Probable Maximum Flood with or without the presence of SCR, with sufficient freeboard to prevent overtopping by waves generated in either direction, i.e. from the SSI or SCR, by a sustained 40 mph overland wind.

The embankment is designed to be stable during the rapid drawdown caused by the sudden and total loss of the SCR.

Seismic Design Criteria for the SSI are discussed in Section 3.7.

The ability of the SSI to meet the criteria of Regulatory Guide 1.27 is discussed in Section 9.2.5.

2.5.6.2 Exploration

The subsurface investigation conducted for Squaw Creek Dam and SSI Dam during design consisted of the undisturbed sampling of alluvial overburden materials as well as the undisturbed sampling of the primary sediments present.

68 | The cohesive soils present along the centerline of the dam site were sampled at vertical intervals of 5.0 ft or at a change in material type, whichever was the lesser distance. These samples of cohesive

materials were obtained by means of a thin-walled, seamless, Shelby tube sampler forced into the soil by means of a rapid thrust from two balanced hydraulic rams. The samples were then field extracted, preserved in moisture-proof polyethylene plastic and sealed to prevent changes in moisture content and accompanying changes in physical characteristics, identified as to boring number and depth, and packaged for transport to the laboratory in Dallas, Texas, for further tests and analyses.

Primary sediments encountered were sampled continuously utilizing a double tube, bottom discharge, conventional coring tool fitted with an appropriate cutting bit, using water as a drilling fluid.

Field infiltration tests were conducted in selected zones of the alluvial material in order to establish the permeability of the soil. | 68
In addition, pressure testing of the primary sediments was accomplished by the insertion of an expandable packer at the location selected while water pressure inside the boring, between the packer and the bottom of the boring, was regulated by a water pump at the ground surface. The depth of the tests pressure in pounds per square in., and the quantity of water consumed during the tests are recorded, where applicable, on the individual field boring logs, and are summarized on Table 2.5.6-1. In addition, selected samples of primary sediments penetrated at the site were preserved in polyethylene plastic, in the manner described above, and all samples were transferred to the laboratory for further evaluation and tests. | 68

Detailed descriptions of materials penetrated as well as all field testing conducted in conjunction with this investigative program are shown, where appropriate, on the boring logs which are discussed in Section 2.5.5.3.

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2.5.6.2.1 Squaw Creek Dam

A series of NX-size vertical core borings were drilled in connection with the exploration for the foundations of the Squaw Creek Dam and appurtenances. Figure 2.5.5-6 shows the location of core borings in relationship to the centerline of the dam.

Borings identified as the D-I series and the D-II series, a total of 28 NX-size core borings, represent the foundation exploration along the centerline of the dam. A total of nine NX-size core borings, identified as Series ES borings, have been utilized to explore the emergency spillway location. Borings identified as SO-1 through SO-11 were drilled for the purpose of exploring two potential alignments for the service outlet works to be located in the valley section of the proposed construction. The service spillway, located on the right abutment of the proposed dam site, has been explored by a series of 16 NX-size core borings numbered SS-1 through SS-16. Additional foundation data was obtained through the drilling for two, four-inch diameter core borings located within the proposed spillway and identified herein as Borings Numbered SS-17 and SS-18.

In addition, further investigation of the service spillway area was accomplished by means of four NX-size, inclined, core borings drilled at the locations shown on the Figure 2.5.5-6 and identified as core borings numbered SSA-1 through SSA-4.

Geologic Profiles representing cross-sections along the Embankment Centerline, the Service Outlet Works, and the Service Spillway are shown on Figure 2.5.6-7 through 2.5.6-9, respectively.

2.5.6.2.2 Safe Shutdown Impoundment Dam (SSI Dam)

68 | A series of NX-size vertical and angle core borings were drilled in connection with the exploration for the foundation of the SSI Dam,

the equalization channel and rock excavation areas. Figure 2.5.5-5 shows the location of the core borings in relation to the centerline of the dam and equalization channel. | 68

Borings identified as the M Series, a total of ten NX-size core borings, represent the foundation excavation along the alignment of the SSI Dam. Borings MS-7 and MS-8 are vertical, NX-sized core borings drilled in the equalization channel area. Borings BA-1 and 2 are angle borings drilled in the balancing channel area. | 68

A geological profile along the centerline of the SSI Dam is presented in Figure 2.5.6-10. A geological section was developed for the equalization channel using the information from MS-7 and 8 and BA-1 and 2. This geological section is shown on Figure 2.5.6-11. | 68

2.5.6.2.3 Borrow Material

Several potential borrow areas were explored and evaluated for use as potential impervious core and random fill material for Squaw Creek Dam and SSI Dam. Figure 2.5.4-29 shows the approximate locations of the different borrow areas investigated. Each of the borrow areas was investigated by a series of six-inch diameter continuous-flight auger borings or excavated pits. Laboratory analysis and engineering evaluation have been conducted on the materials lifted from the auger borings in order to develop design parameters utilizing such materials. Additional core borings with field testing were drilled during construction in the area of the Service Water Outlet alignment and in the West Abutment to help verify depths to competent rock. | 2

| 2

2.5.6.3 Foundation and Abutment Treatment

2.5.6.3.1 Squaw Creek Dam

Squaw Creek Dam is an earth and rockfill embankment structure founded on existing valley alluvium. To minimize leakage through the foundation, a core trench was excavated through the alluvium and weathered rock to impervious unweathered limestone and claystone. This core trench extends well into the abutments. Figure 2.5.6-2 is a profile of the completed excavation.

All transverse cracks were removed by excavation or intersected by a keyway trench, except in the vicinity of the service outlet where removal would undermine the structure. Here a rock saw was employed to cut an 11-in.-wide trench across the cracks to the depth of the crack. The trench was then backfilled with dental concrete. Water was encountered in vertical cracks exposed in the keyway trench, especially on the right side of the creek and at the horizontal contact of the alluvium and limestone. This water could cause the exposed claystone seams to weather rapidly causing overhangs of the limestone layer above the seams. To prevent this weathering, pneumatically placed concrete (shotcrete) was used to seal the claystone seams. The inflow of water from the vertical cracks was controlled by inserting a steel pipe at the base of the crack and then sealing the crack with shotcrete. When the shotcrete had gained strength and it was desired to begin backfill, the pipes were sealed with wood plugs. Potholes and overhangs were filled with dental concrete. Dental concrete and shotcrete were used to cover rough rock surfaces to provide a smooth surface to place backfill against. Figure 2.5.6-5C(1) is a photograph showing shotcreting activities.

Prior to excavation, the foundation for the random fill and rockfill zones of the embankment was cleared and grubbed of trees and roots and then stripped to rock or a maximum depth of 9 in. to remove topsoil,

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rubbish and vegetation. The foundations for the service spillway an service outlet structures were then excavated to solid limestone.

Claystone seams were sealed with a coating of shotcrete. Over excavation was corrected with 1500 psi concrete. Anchor bars were installed as required and the foundation drains were drilled after placement of the structural concrete. Water encountered in the excavation of the foundation for the service outlet intake tower was controlled by sumping and pumping.

The effectiveness of the cutoff trench was judged visually. Mason-Johnston & Associates, Inc., Geotechnical Consultants, determined when the weathered zone had been penetrated. The effectiveness of the saw cutoff trench was tested by observing the depth of the crack and, in some cases, the stoppage of the flow of water.

Weathered material extended to a greater depth in the right abutment than anticipated, causing the cutoff trench to be extended some 250 ft. into this abutment. The trench was terminated prior to reaching the service spillway. The water tightness of the formation between the service spillway structure and the termination of the deep core trench was verified by drilling both vertical and horizontal holes and pressure testing. These hole were filled with grout.

Construction procedures used for the foundation and abutment treatment were as follows:

Soil material was removed by scrapers.

Limestone and claystone were removed by blasting which was prededed by pre-splitting the limits of the excavation. Spacing and loading of the blast holes were adjusted as conditions dictated. Blasted material was 68

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68 removed by front end loaders and end dump trucks. Final shaping of the excavation was accomplished with demolition tools and hand-operated pavement breakers.

Dental concrete had a 28 day design strength of 3000 psi. The concrete was centrally batched and placed in accordance with the procedures of Section 8, ACI 301-72.

The pneumatically-placed concrete was a wet pre-mixed mixture conveyed by air slugs through a flexible tube and deposited in place by air pressure. The material had a minimum 7 day strength of 2400 psi and a minimum 28 day strength of 3000 psi.

All excavations were pumped dry, cleaned with high velocity jets of air and/or water and inspected prior to backfill.

Estimates of the construction quantities involved are included in Section 2.5.6.9.1.

2.5.6.3.2 Safe Shutdown Impoundment Dam

Safe Shutdown Impoundment Dam is a rockfill dam with an impervious
earth core. The entire foundation was excavated to <u>unweathered</u>
limestone to provide suitable support for the rockfill shells and to
68 provide an impervious base for the core. (The term unweathered is
interpreted with respect to engineering properties and not in a purely
geological sense.) Figure 2.5.6-3 is a profile of the completed
excavation.

All transverse cracks under the core zone were removed by excavation or sealed by making a 10-in.-wide saw cut across the crack for its full depth and backfilling with dental concrete. Some seepage was encountered on the upstream side of the excavation in rock strata above the foundation base. Pneumatically placed concrete (shotcrete)

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was used to seal claystone seams to prevent weathering and possible overhangs. This shotcrete was removed just prior to placement of core and filter material in their respective zones.

The abutments against which the core material would be place were cut to a slope not steeper than one on one in order to provide "wedging" as the core consolidated, thus increasing the watertightness of the contact between core and abutment. Vertical faces not in excess of 30 in. (the largest size rock in rockfill) were allowed under the rockfill zone. Verticals in excess of 30 in. were brought to a slope of one on one with fill concrete. Potholes were leveled with dental concrete.

The spillway is an open cut channel between SCR and the SSI. This channel was bottomed on limestone. The crest elevation is maintained by a concrete wall across the channel that is excavated into the limestone foundation.

The effectiveness of the impervious cutoff was further assured by excavating a trench 2 ft. wide, and ranging in depth form 6 to 8 ft., across the base of the foundation on the centerline of the dam. This trench was visually inspected by the Mason-Johnston Geotechnical Consultant to assure that all cracks had been intercepted prior to its backfill with dental concrete. All excavation to unweathered limestone was accepted by the Engineering Geologist prior to covering with shotcrete, dental or fill concrete. Upon completion of all excavation, the entire foundation was jointly inspected and accepted by the Freese and Nichols Project Engineer and the Mason-Johnston Project Geotechnical Consultant.

Construction procedures used for the foundation and abutment treatment were the same as described for the SCD in Section 2.5.6.3.1. Estimates of the construction quantities involved are included in Section 2.5.6.9.2.

2.5.6.4 Embankment

4 2.5.6.4.1 Squaw Creek Dam

A typical cross-section of Squaw Creek Dam is shown on Figure 2.4-17. The top of the dam is at elevation 796.0. The central core is constructed of select, impervious material wetted and rolled, with a cutoff trench extending down into the impervious rock foundation material. The outer zone of the embankment are of a less select material with the outer zone of the downstream side being a rock fill. The filter system separates the impervious central zone from the less select outer zone on the downstream side and extends outward to the downstream toe to provide drainage and protection of the core.

The upstream side of the dam is protected by riprap and gravel blanket from the top of the embankment to elevation 760.0, which is ten feet below the minimum operating level. The crest width of the embankment is twenty feet.

The sides slopes of the dam as shown on Figure 2.5-17 are three horizontal to one vertical on the downstream slope and two horizontal to one vertical to elevation 775 on the upstream slope and below 775 a
2 three horizontal to one vertical slope. An upstream berm is located at elevation 760.0.

2.5.6.4.2 Safe Shutdown Impoundment Dam

2

A typical section of the SSI Dam is shown on Figure 2.4-21. The middle zone of the SSI Dam is composed of select impervious material, wetted and rolled, and carried down to the underlying firm, unweathered Glen Rose Limestone Formation. The outer zones of the SSI Dam consist entirely of the firm Glen Rose Limestone selectively quarried and processed to remove all claystone particles from the rock shell. The two zones of the SSI Dam are separated by two-stage filters. All components of the SSI embankment extend vertically downward to contact with the firm, unweathered, Glen Rose Limestone.

There are many aspects of the design construction procedure of this Class I Dam which make it much more conservative than most other existing dams. Primarily, these consist of the following:

1. The outside slope of the SSI Dam is flatter than slopes commonly used for a fill rock dam. Dams considerably higher than the SSI have been constructed in more seismically active areas with outside slopes in the range of 1.75 horizontal to 1 vertical.

2. The rock shell fill of the SSI Dam was placed in layers and | 4
compacted with multiple passes of a vibratory 10 ton roller. This compaction of the fill rock will provide a material with great strength and low compressibility.

3. The foundation of the entire SSI embankment is supported by the firm, unweathered Glen Rose Limestone.

4. The crest width of the dam (elevation 796) is 40 ft., which is | 68
considerably greater than most existing dams of 70 ft. height.

5. The width of the core at normal water surface is 20 feet.

6. The core material, while considered practically impervious, is not highly plastic. (CL in the Unified Soil Classification System).

7. Two stage filter, both upstream and downstream of the core, are present to control seepage, prevent piping, and control the effects of any cracks that may develop during the occurrence of the SSE.

A comparison of the SSI Dam with other existing fill rock dams in areas much more seismically active than the CPSES site is presented in Table 2.5.6-2. Each of the dams in Table 2.5.6-2 has been constructed with slopes, both upstream and downstream, steeper than those for the SSI Dam at the CPSES Project and constructed in a much more seismically active area. In summary, this SSI Dam is considered most conservative in design and construction and is conservatively appropriate for its intended purpose of providing a reservoir to function as the Ultimate Heat Sink during SSE conditions.

2.5.6.4.3 Material Properties Used for Design

A discussion of the laboratory procedures and tests used for adoption of both static and dynamic material properties for the rock shell, filter and clay core materials is included in the following subsections.

1. Shell Material

2 The laboratory modeling technique used during design for shell material consisted of the following sequential steps:

a. A probable range of gradations resulting from drilling, blasting, loading, placing, and compacting of the Glen Rose Limestone was estimated from prior experience, taking into account the known gradation of core material. The limits of probable design gradation of the shell material are shown on Figure 2.5.6-12.

2 b. Laboratory model techniques were used to obtain engineering properties of the rockfill. Actual NX (2 1/8" diameter) cores of the limestone from the plant site exploration borings, including a statistical average amount of claystone seams, were crushed in the laboratory to conform to the estimated gradation on a fraction-by-fraction basis; i. e., ultimate 10 inch diameter stones were

reduced in the laboratory to particles 3 mm in diameter; one-quarter inch particles were reduced to 0.2 mm particles, etc. The gradation curve of the shell gradation model is shown on Figure 2.5.6-13.

c. The material representing the shell was then remolded, placed in a triaxial chamber, and a multi-stage "Q" test was conducted. The Mohr Diagram obtained from this type of test is shown as Figure 2.5.6-14. Engineering properties of the shell material utilized for design purpose, are as follows:

   Cohesion = 0
   ϕ = 37°
   Tan ϕ = 0.754
   Unit Dry/Moist Weight = 115 pcf
   Unit Saturated Weight = 135 pcf

2. <u>Core Material</u>

Borrow areas upstream of Squaw Creek Dam and downstream from the SSI Dam were investigated as potential borrow to form the impervious clay core of the two dams. Materials available for use in the core are CL in classification (Unified Soil Classification System) and have Liquid Limits ranging from 31 to 44 percent, Plastic indices ranging from 11 to [illegible] and Specific Gravities ranging from 2.65 to 2.70. Strength properties were determined on remolded materials prepared at moisture contents ranging from one percent below optimum to 3 percent above optimum and at density ranges equal to approximately 95 percent of the maximum density by the Standard AASHO Compaction Procedure. | 2

Strength properties were determined by means of the Direct Shear test, where the rate of strain was computed to avoid a build-up of pore pressure and by triaxial testing in the form of Q test and R tests where complete saturation was obtained using back pressure and where pore pressures were measured.

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Settlement and drainage characteristics of the impervious clay core were estimated from data obtained during consolidation testing with the coefficient of permeability, in the range of 10^{-8} centimeters per second, being computed from consolidation characteristics.

Typical results of laboratory tests conducted during design analysis of the material that has been used in the clay core of the SSI Dam are represented by Figures 2.5.6-15 through 2.5.6-17.

Cyclic strength determinations were not an important consideration in the dynamic analysis as explained in Section 2.5.6.5.2.

Based on a detailed analysis of the clay material, the following properties have been adopted for design of the core material:

Liquid Limit = 30-40%
Plasticity Index = 12-20
Optimum Moisture = 12%
Embankment Unit Dry Weight = 108 pcf
Embankment Unit Saturated Weight = 128 pcf
Strength Properties, Considolidated-Undrained
(CU) Cohesion = 432 psf
$\phi = 18^{\circ}$
Tan $\phi = 0.325$

3. Filter Zones

Materials that were used to form the upstream and downstream filters, separating the core from the shell, conformed to the gradation requirement of both the shell and core materials and was obtained offsite from Brazos River alluvial deposits. A two stage filter system has been utilized with the gradation limits of each zone shown on Figure 2.5.6-12.

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4. Dynamic Strength Properties of the Filters and Rock Shell | 2

For design purposes the seismic shear strength of the filter materials | 2
and rock shell for the dam has been estimated based on published data
for granular soil, field embankment tests, and adopted criteria for
construction. Factors considered in the assessment of the seismic
strength characteristics included; (a) relative density; (b) grain
size; (c) gradation; and (d) effect of initial static shear stresses.

Published data on the effect of relative density on seismic strength | 2
of filter materials have shown an increase in strength with increase
in relative density. Based on this data, a relative density of at
least 80 percent has been specified for the filters.

The effect of mean grain size on the cyclic shear strength of granular | 68
soils have been studied by Lee and Fitton [130], Seed and Peacock
[131], and Wong [132]. The results of Lee and Fitton indicate a
substantial increase in seismic strength as D_{50} increases from about | 2
0.1 to 4 mm, and a very large increase in strength as D_{50} increases
above 4 mm. The data presented by Seed and Peacock confirms Lee and
Fitton's results The results of Wong indicate only a slight increase
in strength as the mean grain size increases from 0.6 to 10 mm, but a
rapid increase in strength for mean grain sizes larger than 10 mm.

The mean grain size of the fine filter is 0.6 mm (average) and the | 4
coarse filter 8.0 mm (average) as shown by Figure 2.5.6-12. Although | 68
the available published data would indicate a slight to substantial
increase in strength as the mean grain size increases, this effect has
been conservatively ignored in assessing the cyclic shear strength
characteristics of the filter. Rather, the cyclic shear strength of
the filter has been assessed on the basis of data for Sacramento River
sand [133]. The data on this particular sand (a uniformly graded
fine sand with D_{50} = 0.2 mm) has been utilized because of the | 4

4 comprehensive cyclic test data published, covering a wide range f confining pressures, relative densities, and seismic stress-strain levels.

2 Although the filter material is a well-graded material, whereas Sacramento River sand is a uniformly-graded sand, it is felt that little, if any, difference in the strength can be detected between well-graded and uniformly graded material for the same mean grain
68 size. Research by Lee and-Fitton [136] confirm this conclusion.
2 Wong (1970) tested a well-graded sandy gravel and found that the strengths were lower than uniformly graded gravels at low strains (approximately 5 x 10^{-2}); however, the well-graded gravels were more resistant to the development of large strain; in other words, 10 x 10^{-2}. Therefore, the published cyclic triaxial tests data for Sacramento River sand provide a reasonable basis for assessing the cyclic strength characteristics of the filters. Figure 2.5.6-48, prepared from this data, shows the cyclic shear stresses required to cause 5 x 10^{-2} strain in 5 and 10 cycles for filter materials compacted at a relative density of 80%.

2 The dynamic strength of the rock shell has been estimated in a similar manner as for the filters. Based on data by Wong (1970), the effect of mean grain sizes larger than 10 mm results in a rapid increase in cyclic strength. For a mean grain size of 100 mm, an increase in the cyclic strength of 83% may be expected and for a mean grain size of 152 mm at least 100% increase may be expected. The gradation of the rock in the actual test embankment had a mean grain size of 152 mm and the average of the rock shell in the actual dam is estimated to be no lower than 100 mm. Thus, the cyclic strength may be expected to be
68 increased between 83% and 100%. To be conservative in analysis of the SSI Dam, the relationship shown in Figure 2.2.5-47 was derived.
2 This figure presents the cyclic shear stresses required to cause 5 x 10^{-2} strain in 10 cycles for the compacted rock shell.

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Initial static shear stress effects which occur in the filters and 68
rock shell, and would result in an increase in material strength, have
been ignored.

In developing the cyclic shear strength characteristics for design, it 68
was assumed that pore pressures do not dissipate during earthquake
motions. Because of the large particle size and high permeability of
the rockfill, it is certain that any pore pressures would dissipate to 2
some degree during the ground motions, and this would result in higher
rock shell strengths than are being used in this analysis.

Lee and Fitton (1969) found that clay materials have cyclic shear 68
strengths greater than cyclic shear strengths for sands; however, the
clay core of the SSI Dam has not been evaluated for cyclic shear
strength but has been analyzed using the conservative assumption of
liquefaction as described in Section 2.5.6.5.2.

During construction a test program was set up to evaluate the cyclic 2
shear strengths of backfill material used at CPSES to verify design
assumptions. Included in the specimens tested were the fine filter
material, Filter "A". The details of the investigation are discussed
in Section 2.5.4.7. How the results of Filter "A" tests affect the
stability of the SSI Dam is discussed in Appendix 2.5A.

2.5.6.4.4 Rock Field Tests 2

A series of field tests for fill rock properties was conducted during 68
design at the site of an abandoned quarry located at the CPSES site.
The location was selected to ensure that materials tested would be
representative of those actually used in the construction of the SSI
Dam. Core borings were made at the proposed quarry site and a
careful study was made of all cores to insure that the test quarry
rock was representative of that which would be used in construction.

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The primary objectives of the rock field tests included:

1. Determination of excavation procedures.

2. Evaluation of several wall-protection schemes.

3. Determination of compacted fill rock properties by construction and evaluation of the test embankment section.

68 Excavation operations at the quarry site consisted of the removal of
all residual soils by the use of suitable-equipped crawler equipment,
the drilling of blast holes to depths considered necessary to provide
the desired quantity of materials, presplitting holes and blasting so
as to provide a near vertical quarry face, and the removal of all
excavated materials by loading equipment. The material was then
transported to the test-embankment site and compacted in various layer
thicknesses with various roller passes. The completed test-
embankment was evaluated by means of a test trench. The test trench
68 showed the presence of claystone particles. Since the test-embankment
obtained with this type of quarry operation was considered totally
unsuitable for high quality, pervious, rock fill section due to the
presence of claystone particles, this source of material was
abandoned.

Making full use of he results obtained from the first test quarry program, a new area immediately adjacent to the first area was prepared for excavation by removing all residual soils, drilling and shooting pre-splitting holes to establish a good quarry wall surface, then drilling and shooting blast holes on several patterns. Blast hole depths were controlled so as to bottom immediately above the elevation of known claystone layers. Shot hole patterns of 7 ft. by 7 ft. and 9 ft. by 9 ft. were used; in each case, holes were 3.5 in. in diameter. All holes were bottom loaded with DuPont Pour-vex, column loaded with HD-1 primer, and shot with primer cord. Water gel was used when some of the holes were noted to contain water.

Shot hole spacing, depths, and powder were varied to provide proper maximum size rock, well-graded rock with a minimum of spalls, and little or no claystone particles. With the controlled blasting operation, the limestone portion of the quarry rock was loaded out to the processing area and the remaining claystone areas and zones, readily identified, were removed by crawler equipment and wasted.

Quarried rock was processed by use of a vibrating grizzly and a vibrating screen; the size of openings in the vibrating screen were altered, as required, until material of a desired gradation was obtained.

The test embankment construction consisted of the following basic steps:

1. An area was selected for the test embankment and stripped of all overburden soil and weathered rock until a smooth, hard, and level unweathered Glen Rose Limestone surface was obtained. Corners of a 50 x 100 foot rectangle were marked.

2. Processed quarried rock was then hauled to the test embankment site, dumped, and smoothed to an approximately two-foot thick lift with a dozer. A ten-ton smooth wheel vibratory roller with the vibration mechanism disengaged was then allowed to make one pass to further smoother the surface of the lift. A grid pattern of approximately 10 by 7 ft. was established (21 points) on the surface of the rock layer and the elevation at each grid point determined and recorded.

3. Each layer was then compacted with a self-propelled vibratory roller weighing 10 tons and creating a dynamic force of 40,000 pounds, vibrating at 1400 cycles per minute, and having a drum diameter of 60 inches with a drum length of 104 inches. The

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elevation of the marked grid pattern points was determined after each pass of the vibratory roller; the number of passes varied up to six.

4. After the construction of the embankment was complete, a trench was excavated in the central portion of the test embankment and at right angles to the direction of roller passes to determine conditions of the embankment cross section.

Following the completion of the embankment, a designated area of the embankment was sluiced with water to visually determine the permeability and drainage characteristics of he rock fill and to provide an approximation of the moisture content retained in the fill.

Two large-scale density tests were made on the embankment in the following manner:

1. Following the placement of a particular lift and the application of a particular number of passes of a vibratory roller, a four-foot diameter steel ring was placed on the rock surface and secured in place by sand bags on the perimeter flange of the rings.

2. The rock material inside the ring and for a depth equal to the lift thickness, in the range of two feet, was hand excavated and saved for further use.

3. The excavation was lined with polyethylene film very loosely, including the bottom and sides.

4. Water levels from calibrated barrels were noted and water from these barrels was introduced into the polyethylene lines excavation until the level was within one in. of the steel ring.

5. The steel ring was then removed.

6. Additional water was added until the excavation was full; water levels in the calibrated barrel were noted and the volume of he excavation determined by the volume of water used.

7. The weight of the rock excavated was determined.

8. The "moist" density was determined by dividing the weight of the rock by the volume of the hole.

9. The rock from the excavation was "quartered", piece-by-piece by hand, to obtain a smaller but representative sample and the moisture content was determined.

10. The "dry" density was then determined.

The results of all field observations and compaction tests, including | 2
grain size distribution, have been analyzed and are provided in Figure |
2.5.6-18 through 2.5.6-24 and Table 2.5.5.8. |

After removal of the desired quantity of rock from the test quarry, the existing floor was cleaned and the previously obtained vertical wall resulting from the pre-splitting operation was washed clean with a water nozzle. Shot crete panels were then constructed in order to produce a protecting surface against weathering on the vertical rock face. Two panels were selected; on the first panel a shot crete thickness of approximately one in. was constructed while on the second panel a surface approximately two in. in thickness was constructed. A seven sack cement mix was utilized in the construction of both panels.

Two shear tests were made in the field. The first shear test was run | 68
to determine the angle of internal friction between the rockfill |
embankment material and the in-place, firm, unweathered Glen Rose |

Limestone material. After determination of the friction angle between the rockfill embankment material and the underlying relatively smooth rock surface, a second direct shear test was made of the rockfill embankment material itself.

The shear box consisted of a reinforced, metal wall, six-ft cube, open on top and bottom, and separated on the bottom from the underlying relatively smooth limestone base by a system of small diameter rollers. Loads were applied to the box by means of a calibrated hydraulic ram acting through a suitable yoke arrangement and a ball-type joint. Deflections, or movements, were measured by means of dial micrometers placed at appropriate locations along the bottom and top edge of the shear box. Schematic details of the physical arrangement are illustrated on Figure 2.5.6-25.

Boulders larger than 12 in. in diameter were removed from the rockfill material so as to be compatible with the physical dimensions of the large scale shear box; all materials were weighed prior to placing in the shear box and then compacted with a "jumping jack" compactor to a density equivalent of that previously obtained from large scale density tests made after completion of the test embankment. The gravitational weight of the rock produced the normal force for four successive layer thicknesses.

The results of these tests are shown on Figures 2.5.6-21 and 2.5.6-22. The angle of sliding friction of rockfill material on in place limestone of approximately 42.9 degrees was determined as shown by Figure 2.5.6-23.

At the completion of the sliding resistance shear test, a strip of metal 4 in. in height was taken out of the shear box at about mid-height of the box and the hydraulic ram was removed so that the upper part of the box could be sheared (relative to the fixed lower portion of the box) to determine the angle of internal friction of the rockfill embankment material. The rock in the zone of the shear plane was limited to slightly less than 4 in. in size to correspond with the mechanics of the shear box. The results of these tests are shown on Figures 2.5.6-21 and 2.5.6-22 and indicate an angle of internal friction of 48 degrees.

Rockfill embankment material removed from the large scale field density test location was screened to determine the grain size distribution. The results of this determination are shown on the attached Figure 2.5.6-24. The results of the field density test are shown in summary form on Table 2.5.6-3.

2.5.6.5 Slope Stability

2.5.6.5.1 Squaw Creek Dam

Embankment stability studies for the Squaw Creek Embankment have been made for the condition immediately after construction of the embankment but prior to the filling of the reservoir; and for a condition after construction with the reservoir present and a steady seepage condition established. In addition, an upstream reservoir drawdown case has been analyzed. Selected cases have been analyzed for static conditions and for a pseudo-dynamic condition wherein an acceleration of 0.12g is applied to the driving forces.

Appropriate soil strength parameters, described in detail in the following sub-sections, were used in embankment slope stability analysis. By means of the Swedish method of slices, a series of circular slides was analyzed until the minimum safety factor representing the critical slide surface was obtained. Full use of a computer program was made for this analysis. The computer program used for this analysis is described in Reference 134. After the location of the critical sliding surface was determined, hand | 68

computational methods were used to further evaluate the effects of pore pressure build-up during construction, steady seepage conditions, and the effect of an upstream reservoir drawdown. A series of wedge type analyses were made, by hand computational methods, to further determine the stability of the embankments under both static and pseudo-dynamic conditions. The results of these studies are shown on
68 Figure 2.5.6-26. The safety factors presented on the table on Figure 2.5.6-26 are considered adequate and acceptable for the stability of the embankment under static loading conditions, wherein the minimum safety factor is 1.49, and under dynamic conditions, wherein the minimum safety factor is 1.04.

1. Strength Parameters

The primary formation present at the site, as previously described, is that of the Glen Rose Limestone and strength and physical properties of these materials were evaluated by means of unconfined compression tests, Brazil Tensile Tests, and double ring shear tests.

A portion of the embankment was constructed of rock obtained from required excavations on the site and this rock was quarried and run through a grizzly crusher so as to be well-graded, quarry run, with a maximum rock size limited to thirty inches in diameter. Strength and physical properties of this material were determined in the laboratory by a modeling technique. The modeling technique is described in Section 2.5.6.4.3. The strength properties of the impervious core and random fill material were determined by an evaluation of borrow material as described in Section 2.5.6.4.3.

The design strength parameters of the various materials that will comprise the embankment section used in the slope stability analysis are as follows:

Core:

γ = 117 pcf (95%-100% compaction)

C_u = 720 psf

ϕ_u = 15.5°

tan ϕ_u = 0.277

C = 2300 psf

ϕ = 16°

tan ϕ = 0.287

Random:

γ = 122 pcf (95%-100%)

C_u = 432 psf

ϕ_u = 19°

tan ϕ_u = 0.344

C = 2020 psf

ϕ = 19°

tan ϕ = 0.34

Rock:

γd = 115 pcf; $_s$= 135 pcf

$\overline{C}$ = 0

ϕ = 37.5°

tan ϕ = 0.767

Verification of design parameters is discussed in Appendix 2.5B. | 68

2.5.6.5.2 Safe Shutdown Impoundment Dam (SSI Dam)

Stability of the SSI Dam was evaluated for operating conditions when the embankment and foundation are subject to ground motion from the SSE and when there is a total and instantaneous drawdown of the downstream side of the SSI Embankment resulting from the catastrophic loss of the Squaw Creek Dam. The dynamic response of the dam was determined by the procedure stated in Section 3.7.2.13. The details of the analysis are presented in the following subsections.

The static slope stability analysis of the embankment was performed
68 and is shown on Figure 2.5.6-27. During the initial investigation, uniform upstream and downstream slope of 1.75 (horizontal) to 1 (vertical) and 2.5 (horizontal) to 1 (vertical) were evaluated.

2 The circular failure surface analysis utilized design strength
68 properties given in Section 2.5.6.4.3. The failure surface which cuts through the core, filter and rock shell was found to provide a safety factor of 1.73 for the 1.75:1 slope condition. However, the most critical failure surface corresponds to a relatively shallow shear surface near the outer face of the shell where the safety factor was found to be 1.32. When the condition of rapid drawdown with negligible drainage of the rock slopes was considered, the safety factor was found to be 0.91.

2 Since the effects of the postulated seismic loading will be applied simultaneously to the condition of rapid drawdown, the result in outer
68 slope stability would be reduced to a still lower value. The results of the stability studies, using design strength properties, for the uniform slope of 2.5:1 provided a safety factor of 2.43 for a critical sliding surface passing through the core, filter and rock shell. This factor decreased to 1.89 for a relatively shallow failure surface near the outer face of the rock shell. When rapid drawdown conditions were considered, again with negligible drainage of the shell, the safety factor was found to be 1.38. The effects of postulated seismic loading reduce the safety factors from those of static and total drawdown conditions; however, it appeared likely that the outer shell slopes in the range 2.5 to 1 would be adequate. The dynamic stability analysis was conducted utilizing the 2.5 to 1 slopes.

The liquefaction potential of the rock shell material and of the clay core material is considered practically non-existent; that of the

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filter zone materials is considered low. However, the effects of any possible liquefaction of the filter zone or core on the stability of the embankment has been analyzed by replacing the filter and core with a heavy liquid (static) and making a wedge type failure surface analysis where a seismic coefficient is applied. Filter zone gradation requirements are dependent on the gradation of the core material and shell material. The gradations for the rock shell material as well as the coarse and fine filters for the SSI Dam are
shown on Figure 2.5.6-12. The properties of the impervious core | 2
filters and rock shell are discussed in Section 2.5.6.4.3.

1. Finite Element Model

The structural integrity of the SSI Dam during the SSE was analyzed | 68
using a finite element method similar to that described by Clough and Chopra [135] and Chopra, Clough, Seed, et al. [136]. The finite element model consisted of a grid of 15 x 12 (180) nodal points (see Figure 2.5.6-28) connected in a regular fashion with 14 x 11 (154) quadrilateral elements. The general quadrilateral elements were formed by assembling the stiffness matrices of four constant triangles
(CST). For the rockfill materials, the formulation of the stiffness | 68
matrix was the standard plain strain CST as described by Clough [137]. The mass matrix was formed by lumping one-quarter of the mass of each quadrilateral at the corner nodes in a manner consistent with Subsection 3.7.2.3. Static condensation was then used on the two center node displacements to reduce the element to 8 degrees of
freedom. The clay core was modeled in a similar manner except that an | 68
additional hydrostatic variable was added to improve the accuracy in treating the nearly incompressible clay using the method described in

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68 Herrman [138]. The reformulated quadrilateral has two center node displacements and one pressure variable that are statically condensed to again give an 8 degree of freedom element. The clay core represents only a small portion of the cross-section and is modeled by the two central columns of elements in Figure 2.5.6-28.

2. Reservoir Dam Interaction

68 The hydrodynamic model of the SSI reservoir was accomplished using the added mass matrix described by Zienkiewicz [139] for the upstream and downstream faces of the SSI Dam. This model neglects the compressibility of the water but is a good approximation for dams below 100 feet in height [140]. The added mass coefficients were formed by solution of Laplace's equation in a semi-infinite plane reservoir and lead to a banded (non-diagonal) mass matrix. The added mass matrix coefficients are in the form of influence coefficients and decay rapidly away from the diagonal of the mass matrix. The band of the mass matrix was limited to 5 nodes.

3. Material Properties

68 The Shear moduli for the seismic analysis were based on the recommendations of Hardin & Drnevich [141], the results of the laboratory tests, the direct shear test, the average number of cycles expected during the duration of the SSE, and the average maximum shear strains predicted by the maximum spectral response for the lowest fundamental mode of vibration. The following properties were used:

ROCK FILL:

N = 10 cycles

e = 0.46 (void ratio)

γ = 0.05 (Shear Strain)

γ_{DRY} = 115 pcf

γ_{SAT} = 135 pcf

μ = 0.35 (Poisson's Ratio)

$\lambda_{avg.}$ = 0.08 (critical damping ratio)

G = 45000 σj psf

where σ is the effective vertical stress

CLAY CORE:

γ_{DRY} = 115 pcf

μ = 0.495 (Poisson's Ratio)

$\lambda_{avg.}$ = 0.08 (critical damping ratio)

G = 500,000 psf

4. Natural Frequency and Loads of Vibration

The natural frequencies and nodes shape were determined using the subspace interaction procedure described in Bathe & Wison [142]. The first 10 natural frequencies were found to be in the range of 2.5 to 10 cycles per second. | 68

5. Ground Motion History

An artificial ground motion history consistent with AEC Regulatory Guide 1.60 (revised) and the Newmark, Blume, and Kapur spectral envelope was used to perform both a step-by-step time motion history analysis and a spectral analysis of the seismic response of the SSI Dam. The artificial history was selected consistent with an anticipated structural damping of 8 percent. (The closest artificial history was for 7 percent structural damping). The 8 percent damping | 68

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was determined from the elevation of the maximum shear strain together with the average vertical stress of a number of cycles using the Hardin & Drnevich formula.

The spectral envelope and artificial ground motion acceleration history are shown in Figures 3.7-4 and 3.7-9 for the horizontal component and Figures 3.7-18 and 3.7-23 for the vertical component. The ground motions were normalized so that the maximum horizontal acceleration was 0.12 g (at time = 3.52 seconds) and the maximum vertical acceleration was 0.08 g (at time = 3.44 seconds). It is important to note that although the peak vertical acceleration is 2/3 of the peak horizontal, as given by the Newmark, Blume, Kapur formula, it lies outside and is as severe as that for the horizontal component for all but the lowest fundamental frequency for the dam. Table 2.5.6-4 summarizes the Spectral Velocities for horizontal and vertical ground acceleration is 2/3 the horizontal, the maximum vertical accelerations in pseudo one degree of freedom systems of comparable frequency content of the dam are equal to the maximum horizontal accelerations. This is a very conservative modeling of an earthquake.
68 A further comparison of the artificial acceleration history (Figure 3.7-9) with an actual 1925 earthquake at the site of Sheffield Dam [143] reveals that the artificial history has many more severe peaks (a > 0.05 g) than that of the Sheffield Earthquake.

The artificial time histories were developed and the dam analyzed for a duration of 10.24 seconds. Considering the very severe nature of the horizontal and vertical ground motion histories and since the number of cycles is not as significant a parameter for rockfill as for earth dams, this duration was deemed adequate for design purposes.

6. Step-By-Step Mode Superposition

Based on the natural frequencies and modes determined from the finite element model, a step-by-step mode superposition analysis was performed using the artificial histories for the combined horizontal and vertical components of the ground motion. The full 10.24 second duration was analyzed using a time increment of 0.01 seconds. The first 10 modes were analyzed using the Newmark generalized acceleration method. Figure 2.5.6-29 is a computer plot of the history of the acceleration of the nodal point at the downstream crest of the dam ("15-12"). The maximum acceleration of this node is 0.7 g and occurs at 5.85 seconds.

The response of the SSI Dam at times of 5.61 seconds, 5.70, 5.76, 5.85 (the maximum acceleration), and 5.91 seconds was selected for detailed study. At each of the stated time intervals, three drawings were prepared showing the nodal point acceleration vectors, the nodal point displacements, and the ratio of shear stress divided by the overburden stress. For time interval 5.61 seconds through 5.91 seconds, these relationships are shown on Figures 2.5.6-30 through 2.5.6-44, inclusive. The figures illustrating the acceleration vectors and the displacements at each nodal point and for each selected time interval also contain a vertical profile of acceleration or displacement from the top of the dam to the base of the dam along line 9.

Final displacements of the SSI Dam were estimated and have been plotted at a distorted scale to emphasize the permanent deflections and are shown on Figure 2.5.6-45. The procedure of determining permanent displacements consisted of the following:

a. Nodal point displacements were determined at the time of the maximum acceleration of nodal point "15-12", the time of 5.85 seconds.

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b. It was assumed that the artificial time history (Figure 2.5.6-29) for duration of 10.24 seconds may contain 15 peaks (all less than the maximum occurring at 5.85 seconds).

c. It was assumed that the number of such peaks over a 30 second time interval would be three times that occurring in the 10 second interval.

d. It was assumed that the computed displacements at each nodal point were in error by 100 percent and were multiplied by two (doubled).

Thus, the computed displacements at a time of 5.85 seconds were multiplied by a factor of 90 (15 times 3 times 2 = 90) to arrive at the final displacement configuration as shown on Figure 2.5.6-45. These displacements are considered extremely unlikely to actually occur in view of the pyramiding of the very conservative assumptions involved.

The procedure used in evaluating the seismic stability of the SSI Dam at local points and on horizontal planes consists of the following steps:

4 a. From the artificial ground motion study of the dam, the induced shear stresses at the various locations throughout the dam were evaluated.

b. The normal effective stresses at the various locations throughout the dam were determined.

c. Determinations of the cyclic shear stresses required to cause 5 x 10-2 strain were made at the various locations throughout the dam by use of Figures 2.5.6-47 and 2.5.6-48.

d. An evaluation was made of the ration of cyclic shear stresses required to cause 5×10^{-2} strain (τ_f) with the shear stresses induced by the artificial ground motion, (τ_d).

The ratio, τ_f/τ_d has been considered to represent a local factor of safety against the development of 5×10^{-2} strain. Figure 2.5.6-46 provides a graphic representation of the local factors of safety analyzed at Time=5.85 seconds (the time of the maximum acceleration). Table 2.5.6-5 presents the average of the local safety factors along the various horizontal planes. In this table, the average safety factors along horizontal planes decrease as the planes approach the crest elevation of the dam, as expected.

Local or point safety factors on or near the outer slopes near the crest of the dam have values less than one; these are not considered significant. 2

During construction, a test program to evaluate the cyclic shear strength of Class 1 backfill material was conducted. The results modified the curves on Figure 2.5.6-47 and Figure 2.5.6-48 as well as Figure 2.5.6-46. The effect of these changes on the stability of the SSI Dam are discussed in Appendix 2.5A. 2

7. Determination of Safety Factors

The evaluation of the effects of the SSE on the stability of the SSI rockfill dam has been made using two basic approaches. The first consisted of determining the dynamic response of the dam, the determination of the time of maximum acceleration, the magnitude of shear stresses at nodal points at this time, and the determination of the stress ration (τ_d/τ_f) or point safety factors. The details of this procedure are more fully described in Section 2.5.6.5.[illegible].6.

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68 The second procedure of determining the stability of the SSI rockfill dam during the conditions of the SSE consisted of a series of wedge analyses wherein the ultimate assumption was made that the core would liquefy, regardless of the value of acceleration, and would be resisted by the rock shell when the shell itself is subjected to the maximum value of acceleration as determined by the finite element procedure described in Section 2.5.6.5.2.6 for the time of 5.85 seconds.

The equation for determining the mean acceleration, $\overline{a}$, for any distance, z, below the top of the dam, as related to the height of the dam (H) and the magnitude of the peak acceleration at the top of the dam (ψ), was rigorously determined as:

$$\overline{a} = \psi \left(1 - \frac{2z}{3H}\right)$$

The derivation of the equation is shown on Figure 2.5.6-49.

The equation of forces for a wedge type of analysis at any horizontal plane through the SSI rockfill shell when the shell itself was subjected to acceleration was rigorously derived and, in terms of safety factor, is expressed as:

$$F = \frac{\gamma r \cot\beta \ \tan \emptyset}{K \gamma r \cot\beta \ \tan \emptyset + \gamma c}$$

The derivation of this equation is shown on Figure 2.5.6-50.

Thus, for any horizontal plane through the SSI rockfill dam at a depth z below the crest of the dam, the mean acceleration over the vertical distance H was determined using the maximum crest acceleration at T=5.85 seconds; this mean acceleration was redefined as K and the resulting safety factor determined for each of two cases. Case I

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No.	Station	Tip Elevation
P-I-1	6+60 C	746.95'
P-I-2	6+70 C	766.71'
P-II-2	9+50 C	746.90'
P-II-2	9+60 C	767.03'

All piezometers have been installed in accordance with MJ QAP-18.

The wellpoint-type piezometers consist of industrially accepted stainless steel jacket wellpoints approximately 30" X 1.25"; a #60 mesh wire screen forms the outside of the wellpoint. Each wellpoint is coupled to a 1-in. standard galvanized riser pipe which extends from the tip elevation to the crest elevation. At the crest, the riser pipe has been fitted with a cap which is vented to allow pressure equalization.

Installation of the piezometers was accomplished by advancing an NX-size core boring down to a predetermined depth utilizing the rotary drilling process. After the boring was completed and bailed, the bottom portion of the boring was filled with approximately 1 ft of Ottawa sand. This sand formed a cushion for the wellpoint piezometer which was placed on the sand. After the wellpoint was seated, the | 68
sand column was extended around the wellpoint to approximately 3 ft above the screen. The remainder of the boring annular space, extending through the clay core of the dam, was filled to approximately elevation 786 with a dry bentonitic-type drilling additive. The final step in the installation process was the application of contractor-furnished grout through the remainder of the rockfill section up to the surface. The grout was formed into a conical shape at the surface to provide a positive anchor for the riser pipe. The uppermost part of the riser was recessed within this grout cone so as to be slightly below the existing crest elevation.

The piezometers installed in the SSI Dam will be monitored periodically during reservoir impoundment and operation in order to gain a measure of the piezometric level within the core of the dam. The monitoring of the piezometers will be in accordance with MJ QAP-21 (Draft 1).

21 The SSI Dam will be inspected and piezometer readings taken annually, Results will be evaluated for compliance with design criteria and any deficiencies will be corrected in a timely manner.

2.5.6.8.2 Squaw Creek Dam Instrumentation

In order to verify design parameters, and to evaluate post construction surface deformations, alignments, piezometric levels and internal settlements within Squaw Creek Dam, the system of permanent instrumentation as shown on the attached Figure 2.5.6-55 and 2.5.6-56 has been installed. Permanent instrumentation has consisted of surface alignment monuments to monitor the horizontal and vertical surface movements at the dam crest, settlement plates installed within the interior of the embankment core in order to monitor internal vertical deformations and wellpoint and pneumatic piezometers in order to monitor porewater pressures within the embankment section.

Surface monuments have been installed along the crest of Squaw Creek Dam at the following locations:

No.	Station	Offset	Elevation
SCD-SM-1	39+00	12.0' D.S.	796.7
SCD-SM-2	39+00	12.0' U.S.	796.8
SCD-SM-3	41+00	12.3' D.S.	796.6
SCD-SM-4	41+00	11.8' U.S.	797.0
SCD-SM-5	43+00	12.0' D.S.	796.9

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SCD-SM-6	43+00	11.9' U.S.	797.1
SCD-SM-7	45+00	12.2' D.S.	797.0
SCD-SM-8	45+00	11.9' U.S.	797.8
SCD-SM-9	47+00	12.2' D.S.	797.3
SCD-SM-10	47+00	11.9' U.S.	797.8
SCD-SM-11	49+00	11.8' D.S.	798.3
SCD-SM-12	49+00	11.5' U.S.	798.5
SCD-SM-13	51+00	12.0' D.S.	798.5
SCD-SM-14	51+00	11.6' U.S.	799.0
SCD-SM-15	53+00	12.0' D.S.	798.6
SCD-SM-16	53+00	12.2' U.S.	799.3
SCD-SM-17	55+00	12.2' D.S.	798.5
SCD-SM-18	55+00	11.9' U.S.	799.3
SCD-SM-19	57+00	12.1' D.S.	798.7
SCD-SM-20	57+00	12.0' U.S.	798.2
SCD-SM-21	59+00	12.4' D.S.	798.3
SCD-SM-22	59+00	11.8' U.S.	799.0
SCD-SM-23	61+00	12.2' D.S.	798.3
SCD-SM-24	61+00	11.9' U.S.	799.0
SCD-SM-25	63+00	12.1' D.S.	798.8
SCD-SM-26	63+00	12.2' U.S.	799.3
SCD-SM-27	65+00	11.7' D.S.	798.9
SCD-SM-28	65+00	11.7' U.S.	798.8
SCD-SM-29	67+00	12.9' D.S.	798.8
SCD-SM-30	67+00	11.9' U.S.	799.1
SCD-SM-31	69+00	12.2' D.S.	798.7
SCD-SM-32	69+00	12.2' U.S.	799.0
SCD-SM-33	71+00	12.4' D.S.	798.7
SCD-SM-34	71+00	11.6' U.S.	798.9
SCD-SM-35	73+00	12.3' D.S.	797.6
SCS-SM-36	73+00	12.2' U.S.	797.8
SCD-SM-37	75+00	12.1' D.S.	797.7
SCD-SM-38	75+00	11.9' U.S.	796.5

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Installation of the surface monuments was accomplished in general accordance with MJ QAP-19. The installation procedure is similar to that described in Subsection 2.5.6.8.1.

The surface monuments installed on the crest of Squaw Creek Dam will be periodically observed for horizontal and vertical movements throughout the course of reservoir impoundment and operation.

Settlement place have been installed in the core of Squaw Creek Dam at the following locations:

Station	Offset	Elevation
50+49.2	0.6 Left of C	724.43
60+40.0	C	710.05
70+50.0	C	674.53

Each settlement plate consists of a rectangular piece of steel conforming to ASTM-A-36 with the following dimensions: 2.5' X 2.5' X 0.375'. Installation of the settlement plates was accomplished in the same general manner as that outlined in Subsection 2.5.6.8.1.

2 | If settlement readings are desired, a core boring will be advanced to penetrate through the core of the dam down to the elevation of the settlement plates; and a permanent observation casing will be installed rising from the elevation of the settlement plate up to the crest of the dam. Utilizing the casing as access to the settlement plate, elevations could be taken periodically on the settlement plates and monitored to observe the internal vertical deformations.

Various combinations of wellpoint and pneumatic transducer piezometers have been installed in Squaw Creek Dam as shown on Table 2.5.6-7.

The piezometers were grouped in three ranges. Typically, each range of piezometers consists of a set of piezometers upstream of the centerline spread uniformly throughout the core of the dam, a set of piezometers downstream of the centerline spread uniformly throughout the core, and a set of piezometers near the toe of the dam with piezometers in the random section and in the alluvial section. | 68

In order to facilitate the ongoing construction process, the piezometers contained within Range II were installed during the course of constructions. Two different types of piezometers were installed at Range II. Standard industrially-accepted stainless steel jacket wellpoint piezometers were installed at all locations for Range II. The wellpoints were nominally 30 inches in length by 1.25 inches and were encased by a No. 670 mesh screen. Additionally, companion pneumatic transducers were installed at the same elevation as the two deepest piezometers for each set (i.e., upstream, downstream, and near the toe). The pneumatic piezometer operates on a null balance principle whereby external pore pressure exerted on the transducer tip is exactly balanced b by an equal and opposite internal gas pressure supplied when reading the piezometer via an interconnecting tubing which connects the transducer to the surface readout station. By installing the pneumatic piezometers as companions to wellpoint piezometers at equal depths, their pore pressure readings may be compared to those obtained by the standard wellpoint type piezometers.

Similarly, wellpoint and pneumatic piezometers were installed near the toe of the dam at Station 46 and Station 69. The remainder of the piezometers, that is, the upstream (wellpoint) and near downstream (pneumatic) sets at Stations 46 and 69, were installed after completion of the Squaw Creek Dam. | 68

The upstream set of piezometers for both Stations 46 and 69 were installed by drilling down through the completed crest of the dam. In order to avoid any disturbance to the finished grades on the downstream slope, it was impossible to install the downstream set of piezometers for Stations 46 and 49 by using vertical rotary drilling. Therefore, it as decided to install these piezometers by angle-core drilling from the existing completed crest of the dam down to the appropriate elevation and offset as previously determined. By using the angle-hole method of installation, standard wellpoint piezometers could not be utilized. Therefore, pneumatic transducers were installed for the near downstream sets at Stations 46 and 69. Even though the hole where the piezometers has been installed is drilled at an angle, the pore pressure reading at the tip elevation is in no way affected by the angularity of the hole.

68 The installation of the wellpoint piezometers in Squaw Creek Dam was essentially similar to that utilized for the SSI Dam as described in Paragraph 2.5.6.8.1, except that grout was not utilized to backfill the holes. Instead, the holes were filled with a bentonitic drilling additive. Installation of the pneumatic transducers was accomplished in a similar manner to that of the wellpoints. In this case, the riser pipe was utilized to protect the readout tubings leading from the pneumatic transducers to the ground surface.

Regular readings of the wellpoint and pneumatic piezometers are being made during impoundment and will continue during operation of the reservoir. These readings will be used to verify design parameters and to alert the geotechnical consultants to any extraordinary rise in pore pressure within the core of the dam or to any blockage of the filter on the downstream toe of the dam.

2.5.6.9 Construction Notes

2.5.6.9.1 Squaw Creek Dam

Estimated construction quantities for Squaw Creek Dam are as follows:

1.	Core Trench Excavation - Common	141,166 C.Y.
2.	Core Trench Excavation - Rock	137,488 C.Y.
3.	Emergency Spillway Excavation	1,205,165 C.Y.
4.	Service Outlet Excavation	104,282 C.Y.
5.	Service Spillway Excavation - Common	191,177 C.Y.
6.	Service Spillway Excavation - Rock	205,452 C.Y.
7.	Impervious Fill	1,138,935 C.Y.
8.	Random Fill	3,733,191 C.Y.
9.	Rock Fill	526,149 C.Y.
10.	Filter Sand	178,555 C.Y.
11.	Riprap	41,274 C.Y.
12.	Riprap Bedding	13,150 C.Y.

Figure 2.5.6-5A(1) is a photograph of the SCD site after stripping activities.

Construction of Squaw Creek Dam commenced on November 17,1 974, with the clearing and grubbing of the dam's foundation. Details of and | 68
problems encountered during the foundation and abutment excavation and |
treatment are discussed in Section 2.5.6.3.1. Figure 2.5.6-5B(1) |
shows a photograph of the foundation preparation activities. Soil material removed from the core trench was placed in the random embankment zones. Rock material excavated was wasted since it was generally too large for the rockfill zone.

On February 14, 1975, excavation of the east core trench was completed. The core trench was backfilled in section; that is, the east side of Squaw Creek was excavated to the abutment then backfilled. The same procedure was used on the west side of the creek which was completed on March 25, 1975. The excavation of the core trench in the closure section was completed on November 3, 1975.

Upon completing the backfill of the east core trench, placement of material in the east embankment began. As mentioned before, soil material excavated from the core had been placed in the random zone.

Select and random materials were obtained from upstream borrow pits. These pits were irrigated prior to use to bring the moisture content close to optimum. Rockfill material was obtained from the emergency spillway excavation. The east embankment was completed to elevation 791 on March 4, 1976, and the west embankment on July 19, 1976, when both embankments were brought to elevation 793. Figure 2.5.6-5A through Figure 2.5.6-5J are progress photographs of the construction of SCD.

Completion of the west embankment was delayed due to the time required to construct the service outlet conduit and backfill adjacent to it. Additional delay was encountered due to the need to extend the core excavation deep into the right abutment to unfractured material. The embankment could not be placed higher than the backfill in the abutment without forming a pocket that would pond water.

The specifications required that the select zone have a minimum liquid
68 limit of 35. This requirement was relaxed to a minimum of 30 for fill placed above elevation 780 due to a shortage of material with a higher liquid limit.

It was originally planned to place all material excavated from the emergency spillway in the rock zone of the embankment. However, more random material containing weathered rock was encountered in the emergency spillway than anticipated. This material contained too much rock to be used as random fill and too much soil to be used as rockfill. It was elected to waste it rather than try to process it.

Later, a grizzly was used to separate the rock from the soil. The rock was processed for riprap and the soil stockpiled for use in the random zone. The remaining excavation was divided into two classifications. The upper zone contained weathered crystalline limestone of suitable quality to be used as riprap. The lower zone contained unweathered limestone and claystone seams not suitable for riprap, but adequate for rockfill. These zones were selectively excavated and used as stated.

The rockfill zone of the dam now required more material than would be available from the emergency spillway excavation. Rather than quarry additional rockfill, it was decided to reduce the rockfill zone to be compatible with the volume of material available. A minimum horizontal thickness of 16 ft. was maintained for workability. The purpose of this zone is to use materials required to be excavated from the emergency spillway and to act as slope protection for the back of the dam. Specifications were modified to allow quarry run material not exceeding 24 in. in size.

Difficulties with compacting material adjacent to piezometer clusters at Station 50+00 and 60+00 are discussed in Appendix 2.5B along with corrective action taken. | 2

Closure of the stream began on July 28, 1976. The details of this operation are described in Section 2.5.6.7.1. The dam was completed on June 16, 1977. | 68

2.5.6.9.2 Safe Shutdown Impoundment Dam

Estimated construction quantities for the Safe Shutdown Impoundment Dam are as follows:

1.	Excavation	447,800 C.Y.
2.	Impervious Fill	63,900 C.Y.
3.	Filter A	22,800 C.Y.
4.	Filter B	22,800 C.Y.
5.	Rock Fill - Graded	450,700 C.Y.
6.	Rock Fill - Quarry Run	97,600 C.Y.

Construction of the SSI Dam began on February 10, 1975, with the
68 | clearing and grubbing of the dam's foundation. Details of and
| problems encountered during the foundation and abutment excavation and
| treatment are discussed in Section 2.5.6.3.2 Figures 2.5.6-4A thru
| 2.5.6-4G are a series of photographs of the dam during construction.

The most significant problem in the foundation excavation was that weathered and unsound rock was encountered deeper than expected. It was originally estimated that 305,000 C.Y. of material would have to be removed to secure a proper foundation; however, 548,300 C.Y. were removed.

Specifications required all slopes in the abutments to be 3 (H) on 1(V). This was to be accomplished with dental concrete where necessary. Excavation of the rock ledges tended to break vertical, and the claystone ledges had to be excavated vertical. Sloping there verticals to 3 (H) on 1 (V) slope would require a considerable amount of concrete; therefore, the specifications were modified to allow a 1 on 1 slope, with a maximum of 30 inches of vertical rock face when in rock. This applied to the rockfill and filter zones only. In the core zone, the faces were hand trimmed to a 1 on 1 slope and all claystone layers were protected from weathering with 1-1/2 inches of pneumatically-placed concrete. The specifications for dental concrete required that it be consolidated to a dense impervious mass. When placed on a 1 on 1 slope, it could not be vibrated without flowing down slope. Since this material is used only under the rockfill and filter zones where being impervious is not important, the specifications were changed to allow the use of fill concrete. Fill concrete was consolidated by vibration insofar as possible then by tamping. Core test samples demonstrated that the compressive strength of this material was adequate to support the rockfill.

On April 13, 1976, a joint inspection of the foundation was made by the Freese and Nichols Project Engineer and the Mason-Johnston Project Geotechnical Consultant, resulting in permission being given to proceed with backfill. The presence of weathered (Fractured) layers of limestone was noted in the right abutment. Attempts to remove this material were unsuccessful and it was believed that this material extended through the abutment. Removal of this material was not required since it was structurally sound and located above elevation 769.5 where water tightness is not important. The fractures were judged to be tight enough not to cause piping of the core material during its limited exposure to flood water.

Sources of embankment materials were as follow:

1. Rock Fill - Borrow "B" and "C" quarries as defined in Appendix 2.5A, "Construction Records and Design Verification of SSI Dam."

2. Impervious Core - Selected borrow pits in Panther Branch valley below the SSI, and in Squaw Creek Valley above SCD.

3. Filter Materials - Commercial sources.

The time required to carry out effectively selective quarry operation in the plant site to obtain sufficient material to construct the SSI Dam was prohibitive when compared to the excavation schedule for the plant. Quarries of satisfactory quality were located alone Squaw Creek, two miles upstream from SCD.

Rock fill was selectively quarried from these sources to minimize claystone content and brought to a stockpile near the crusher. Sufficient material was stockpiled before impoundment of water in SCR since the quarry would be one of the first areas flooded. The rock material was hen run through the crusher and either placed as embankment or stockpiled.

Rock fill was dumped from end-dump trucks and spread into a two-ft-thick layer with a dozer. Segregation problems caused by spreading were corrected through modifying the dozer blade by adding "wings" to keep the larger stones from rolling to the side. Compaction was accomplished by four passes with a 10 ton vibrating roller.

The preirrigated impervious core material was processed with a mixer prior to loading in dump trucks and being transported to the fill. The material was dumped and spread into a loose lift not greater than 8 inches thick and compacted by a minimum of 8 passes of a sheepsfoot roller.

68 | Originally up to a 12-inch loose lift thickness was allowed. Considerable difficulty was experienced in compacting the filter zones to the 80 percent relative density required. This problem was solved by reducing the lift thickness from 12 inches to 8 inches and compacting with hand operated vibrating sleds.

In August 1976, it was determined that a portion of the filter material did not comply with gradation requirements. At this time, the top of the embankment was at approximately elevation 756. An investigation of the location and extent of the undesirable material was jointly conducted by Freese and Nichols and Mason-Johnston. It was determined that acceptable filter material was present in all locations below elevation 747.5. All of the undesirable filter material was removed down to elevation 747.5. The mechanical requirements of the removal process also required the removal of

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138. Perloff, W., Pressure Distribution and Settlement, in: Foundation Engineering Handbook, edited by Winterkorn, H.F. and Fang, H., Van Nostrand Reinhold Company, New York, 1975, pp. 148-196. | 68

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TABLE 2.5.4-1

(SHEET 1 of 8)

SUMMARY OF UNCONFINED COMPRESSION TESTS

BORING NUMBER	DEPTH FT.	ELEVATION FT.	ULTIMATE STRENGTH TSF	DESCRIPTION	
					68
					68
P-1	40.0 - 40.5	796.7 - 796.2	18.0*	Claystone, gray	68
P-1	69.5 - 70.5	767.2 - 766.2	405.8	Limestone, gray	68
P-1	90.0 - 90.7	746.7 - 746.0	214.8	Limestone, gray	68
P-3	39.8 - 40.7	784.9 - 784.0	11.2*	Claystone, gray	68
P-3	59.9 - 60.8	764.8 - 763.9	349.0	Limestone, gray	68
P-3	70.0 - 70.6	754.7 - 754.1	318.5	Limestone, gray	68
P-4	59.6 - 60.3	789.7 - 789.0	70.0	Limestone, gray, shaly	68
P-4	80.3 - 80.8	769.0 - 768.5	335.9	Limestone, gray	68
P-4	100.0 - 100.9	749.3 - 748.4	305.4	Limestone, gray	68
P-9	7.5 - 8.0	826.2 - 825.7	3.8	Claystone, tan, weathered	68
P-9	12.9 - 14.0	820.8 - 819.7	69.8	Claystone, tan, weathered	68
P-9	20.0 - 22.0	813.7 - 811.7	178.9	Limestone, gray	68
P-9	24.3 - 25.0	809.4 - 808.7	5.0*	Claystone, black	68
P-9	28.7 - 29.5	805.0 - 804.2	192.0	Limestone, gray	68
P-9	33.5 - 34.9	800.2 - 798.8	255.0	Limestone, gray	68
P-9	36.1 - 35.5	797.6 - 797.2	121.2	Limestone, gray	68
P-9	39.0 - 39.5	794.7 - 794.2	9.9*	Claystone, gray	68
P-9	42.2 - 42.5	791.5 - 791.2	124.7	Limestone, gray	68

* Denotes claystone below elevation 810

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TABLE 2.5.4-1

(SHEET 2)

SUMMARY OF UNCONFINED COMPRESSION TESTS

BORING NUMBER	DEPTH FT.	ELEVATION FT.	ULTIMATE STRENGTH TSF	DESCRIPTION	
P-9	50.7 - 51.8	783.0 - 781.9	12.1*	Claystone, gray	68
P-9	52.4 - 52.9	781.3 - 780.8	278.4	Limestone, gray	68
P-9	58.2 - 58.7	775.5 - 775.0	183.2	Limestone, gray	68
P-9	60.0 - 60.4	773.7 - 773.3	31.4*	Claystone, gray	68
P-9	64.6 - 64.5	769.1 - 769.2	88.6	Limestone, gray, shaly	68
P-9	70.0 - 70.8	763.7 - 762.9	155.4	Limestone, gray	68
P-9	81.1 - 81.5	752.6 - 752.2	192.0	Limestone, gray	68
P-9	91.1 - 92.4	742.6 - 741.3	606.0	Limestone, gray	68
P-9	106.0 - 107.2	727.7 - 726.5	253.0	Limestone, gray	68
P-9	112.1 - 113.5	721.6 - 720.2	331.6	Limestone, gray	68
P-9	128.3 - 129.0	705.4 - 704.7	530.1	Limestone, gray	68
P-9	134.9 - 136.8	698.8 - 696.9	88.1*	Claystone, dark, gray	68
P-10	11.2 - 11.7	831.8 - 831.3	20.5	Limestone, tan, weathered	68
P-10	14.1 - 15.0	828.9 - 828.0	87.3	Limestone, tan, weathered	68
P-10	15.6 - 17.3	826.4 - 825.7	325.0	Limestone, gray, (band)	68
P-10	22.0 - 22.8	821.0 - 820.2	133.1	Claystone, gray, calcareous	68
P-10	28.2 - 28.6	814.8 - 814.4	582.5	Limestone, white	68
P-10	31.5 - 33.5	811.5 - 809.5	152.7	Limestone, gray	68
P-10	43.3 - 44.0	799.7 - 799.0	222.5	Limestone, gray	68

* Denotes claystone below elevation 810

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TABLE 2.5.4-1

(SHEET 3)

SUMMARY OF UNCONFINED COMPRESSION TESTS

BORING NUMBER	DEPTH FT.	ELEVATION FT.	ULTIMATE STRENGTH TSF	DESCRIPTION	
					68
					68
P-10	45.7 - 46.2	797.3 - 796.8	240.0	Limestone, gray	68
P-10	48.7 - 49.3	794.4 - 793.8	15.0*	Claystone, gray	68
P-10	54.6 - 55.3	788.5 - 787.8	121.2	Limestone, gray	68
P-10	58.5 - 59.0	784.6 - 784.1	21.5*	Claystone, gray	68
P-10	60.9 - 62.0	782.2 - 781.1	7.1*	Claystone, gray	68
P-10	63.1 - 63.5	780.0 - 779.6	15.2*	Claystone, gray	68
P-10	70.0 - 70.8	773.1 - 772.3	48.5*	Claystone, gray (band)	68
P-10	70.8 - 71.6	772.3 - 771.5	139.6	Limestone, gray	68
P-10	74.6 - 76.0	768.5 - 767.1	329.2	Limestone, gray	68
P-10	77.3 - 78.0	765.8 - 765.1	220.2	Limestone, gray	68
P-10	81.3 - 82.0	761.8 - 761.1	270.5	Limestone, gray	68
P-10	88.8 - 89.7	754.3 - 753.4	200.7	Limestone, gray	68
P-10	97.6 - 99.4	745.5 - 743.7	235.6	Limestone, gray	68
P-10	107.0 - 108.1	736.1 - 735.0	445.0	Limestone, gray	68
P-10	115.5 - 116.1	727.6 - 727.0	196.3	Limestone, gray	68
P-10	143.6 - 144.7	699.5 - 698.4	261.8	Limestone, gray	68
P-11	20.5 - 21.2	827.5 - 826.8	149.8	Limestone, gray	68
P-11	26.2 - 27.0	821.8 - 821.0	69.7	Limestone, gray, shaly	68
P-11	30.0 - 31.4	818 - 816.6	198.5	Limestone, gray	68

* Denotes claystone below elevation 810

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TABLE 2.5.4-1

(SHEET 4)

SUMMARY OF UNCONFINED COMPRESSION TESTS

BORING NUMBER	DEPTH FT.	ELEVATION FT.	ULTIMATE STRENGTH TSF	DESCRIPTION	
P-11	36.3 - 36.7	811.7 - 811.3	147.3	Limestone, gray	| 68
P-11	47.0 - 48.0	801.0 - 800.0	261.8	Limestone, gray	| 68
P-11	54.3 - 54.8	793.7 - 793.2	381.3	Limestone, gray	| 68
P-11	61.2 - 61.6	786.8 - 786.4	57.9*	Claystone, gray	| 68
P-11	72.3 - 73.2	775.7 - 774.8	27.1*	Claystone, gray	| 68
P-11	73.9 - 74.5	774.1 - 773.5	161.4	Limestone, gray	| 68
P-11	82.7 - 83.7	765.3 - 764.3	266.1	Limestone, gray	| 68
P-11	91.0 - 92.0	757.0 - 756.0	161.4	Limestone, gray	| 68
P-13	13.5 - 14.1	837.3 - 836.7	6.3	Claystone, tan, weathered	| 68
P-13	24.0 - 25.5	826.8 - 825.3	48.5	Claystone, gray	| 68
P-13	28.5 - 29.1	822.3 - 821.7	39.0	Claystone, gray	| 68
P-13	39.5 - 40.1	811.3 - 810.7	327.2	Limestone, gray	| 68
P-13	50.1 - 51.0	800.7 - 799.8	301.0	Limestone, gray	| 68
P-14	12.6 - 13.2	837.8 - 837.2	10.9	Limestone, tan, weathered	| 68
P-14	30.6 - 31.1	819.8 - 819.3	698.1	Limestone, gray	| 68
P-14	34.7 - 35.2	815.7 - 815.2	130.9	Limestone, gray	| 68
P-14	54.9 - 55.7	795.5 - 794.7	15.7*	Claystone, gray	| 68
P-14	68.0 - 68.7	782.4 - 781.7	17.4*	Claystone, gray	| 68
P-14	75.7 - 77.1	774.7 - 773.3	114.6	Limestone, gray (band)	| 68

* Denotes claystone below elevation 810

TABLE 2.5.4-1

(SHEET 5)

SUMMARY OF UNCONFINED COMPRESSION TESTS

BORING NUMBER	DEPTH FT.	ELEVATION FT.	ULTIMATE STRENGTH TSF	DESCRIPTION	
P-14	83.3 - 83.9	767.1 - 766.5	240.0	Limestone, gray	68
P-16	8.2 - 9.0	833.3 - 832.5	15.2	Claystone, tan, weathered	68
P-16	15.2 - 15.9	826.3 - 825.6	89.0	Limestone, gray, shaly	68
P-16	25.2 - 26.0	816.3 - 815.5	179.9	Limestone, gray	68
P-16	39.2 - 40.0	802.3 - 801.5	117.8	Limestone, gray	68
P-16	44.6 - 45.1	796.9 - 796.4	20.6*	Claystone, dark gray	68
P-16	56.7 - 57.6	784.8 - 783.9	9.2*	Claystone, dark gray	68
P-16	64.5 - 65.6	777.0 - 775.9	15.6*	Claystone, dark gray	68
P-16	74.5 - 75.0	767.0 - 766.5	200.7	Limestone, gray	68
P-17	6.1 - 7.5	830.9 - 829.5	71.2	Limestone, tan, weathered	68
P-17	24.3 - 24.8	812.7 - 812.2	220.3	Limestone, gray	68
P-17	37.9 - 38.4	799.1 - 798.6	353.4	Limestone, gray	68
P-17	45.3 - 45.9	791.7 - 791.1	123.4	Limestone, gray	68
P-17	48.4 - 48.9	788.6 - 788.1	186.9	Limestone, gray	68
P-17	64.6 - 66.0	772.4 - 771.0	5.3*	Claystone, dark gray	68
P-17	73.6 - 74.6	763.4 - 762.4	280.0	Limestone, gray	68
P-25	43.0 - 43.8	807.7 - 806.9	112.0*	Claystone	68
P-25	49.3 - 50.0	801.4 - 800.7	183.2*	Claystone	68

* Denotes claystone below elevation 810

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TABLE 2.5.4-1

(SHEET 6)

SUMMARY OF UNCONFINED COMPRESSION TESTS

BORING NUMBER	DEPTH FT.	ELEVATION FT.	ULTIMATE STRENGTH TSF	DESCRIPTION	
					68
					68
P-25	56.0 - 56.9	794.7 - 793.8	77.1*	Claystone	68
P-26	42.2 - 42.8	809.0 - 808.4	5.5*	Claystone	68
P-26	69.5 - 70.4	781.7 - 780.8	21.2*	Claystone	68
P-26	70.4 - 71.3	780.8 - 779.9	21.2*	Claystone	68
P-26	75.0 - 76.0	776.2 - 775.2	21.6*	Claystone	68
P-27	34.4 - 35.6	809.7 - 808.5	2.1*	Claystone	68
P-27	37.4 - 39.4	806.7 - 804.7	54.6*	Claystone	68
P-27	49.0 - 49.9	795.1 - 794.2	23.5*	Claystone	68
P-27	59.0 - 60.5	785.1 - 783.6	9.6*	Claystone	68
P-27	64.3 - 65.6	779.8 - 778.5	15.6*	Claystone	68
P-27	69.0 - 69.7	775.1 - 774.4	23.8*	Claystone	68
P-27	71.7 - 72.6	772.4 - 771.5	12.4*	Claystone	68
P-28	31.0 - 31.5	813.1 - 812.6	73.5	Claystone	68
P-28	57.0 - 58.3	787.1 - 785.8	19.6*	Claystone	68
P-28	62.5 - 63.0	781.6 - 781.1	13.0*	Claystone	68
P-28	69.3 - 70.5	774.8 - 773.6	33.3*	Claystone	68
P-29	25.2 - 26.8	810.2 - 808.6	53.4*	Claystone	68
P-29	46.0 - 46.8	789.4 - 788.6	36.7*	Claystone	68
P-29	49.1 - 49.9	786.3 - 785.5	15.3*	Claystone	68

* Denotes claystone below elevation 810

TABLE 2.5.4-1

(SHEET 7)

SUMMARY OF UNCONFINED COMPRESSION TESTS

BORING NUMBER	DEPTH FT.	ELEVATION FT.	ULTIMATE STRENGTH TSF	DESCRIPTION	
P-29	53.5 - 54.9	781.9 - 780.5	17.9*	Claystone	68
P-29	60.7 - 61.4	774.7 - 774.0	15.5*	Claystone	68
P-30	41.8 - 43.3	810.7 - 809.2	46.3	Claystone	68
P-30	52.1 - 53.0	800.4 - 799.5	14.6*	Claystone	68
P-30	55.0 - 56.0	797.5 - 796.5	10.4*	Claystone	68
P-30	63.9 - 64.8	788.6 - 787.7	31.4*	Claystone	68
P-30	74.1 - 76.0	778.4 - 776.5	35.9*	Claystone	68
P-31	0 - 1.5	832.4 - 830.9	1.7	Sandy clay	68
P-31	5.5 - 6.3	826.9 - 826.1	49.4	Limestone, weathered	68
P-31	11.9 - 13.3	820.5 - 819.1	41.3	Claystone	68
P-31	16.2 - 17.0	816.2 - 815.4	167.8	Limestone with claystone	68
P-31	19.2 - 20.0	813.2 - 812.4	113.3	Limestone with claystone	68
P-31	22.8 - 23.9	809.6 - 808.5	4.9*	Claystone	68
P-31	30.1 - 30.7	802.3 - 801.7	104.6	Limestone with claystone	68
P-31	43.2 - 44.5	789.2 - 787.9	85.7	Limestone with claystone	68
P-33	8.3 - 10.0	797.9 - 796.2	166.9	Limestone with claystone	68
P-33	13.5 - 14.0	792.7 - 792.2	15.9*	Claystone with limestone	68

* Denotes claystone below elevation 810

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TABLE 2.5.4-1

(SHEET 8)

SUMMARY OF UNCONFINED COMPRESSION TESTS

BORING NUMBER	DEPTH FT.	ELEVATION FT.	ULTIMATE STRENGTH TSF	DESCRIPTION	
P-33	18.3 - 19.0	787.9 - 787.2	121.6	Limestone	68
P-33	37.5 - 38.3	768.7 - 767.9	72.9	Limestone with claystone	68
P-33	45.8 - 46.7	760.4 - 759.5	162.2	Limestone with claystone	68
P-33	48.3 - 49.2	757.9 - 757.0	176.6	Limestone with claystone	68
P-33	51.2 - 52.5	755.0 - 753.7	232.7	Limestone with claystone	68
P-33	58.6 - 59.4	747.6 - 746.8	149.2	Limestone with claystone	68

* Denotes claystone below elevation 810

TABLE 2.5.4-5

INITIALLY SELECTED VALUES OF | 68
PREEXCAVATION-DYNAMIC FOUNDATION DESIGN PARAMETERS | 68

Foundation Material	Compressional Wave Velocity, Vc (ft/sec)	Shear Wave Velocity (ft/sec)	Wave Density* (lbs/cu ft)	Insitu Wet Poisson's Ratio (u)	Shear Modulus (G) or Modulus of Rigidity (lbs/sq in)
Weathered Rock	2,600	1,000	144	.35	4.5×10^4
Moderately Weathered to Fresh Rock (Glen Rose Limestone above approximately elevation 770)	9,500 to 11,000	5,500 to 6,000	150	.30	8.0×10^5
Underlying Massive Fresh Rock (Glen Rose Limestone below approximately elevation 770)	11,000 to 12,500	6,000 to 6,500	155	.27	1.2×10^6
Underlying Twin Mountains Rock	7,000 to 8,000	3,200	135	.32	3.0×10^5

Young's Modulus, E = 2 (1+u) G.

Mass Density = $= \frac{\text{Insitu Density}}{32.2}$

$$E = v_c^2 \frac{(1 + u)(1 - 2u)}{(1 - u)}$$

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TABLE 2.5.4-5A

(SHEET 1 of 2)

DETAILS OF PREEXCAVATION CROSSHOLE SURVEYS | 68

Plant Site

Shot Station	Shot Station Ground Elevation	Shot Elevation	Distance From Boring P-1	Distance From Boring P-2	Geophone Elevations Boring P-1	Geophone Elevations Boring P-2
A	848	841	870	1175	693, 718, 743 768, 793, 818	743, 768, 793 818
B	[illegible]46	839	370	675	Same as A	Same as A

Safe Shutdown Impoundment Dam Site

Shot Station	Shot Station Ground Elevation	Shot Elevation	Distance From Boring M-5	Distance From Boring M-6	Geophone Elevations Boring M-5	Geophone Elevations Boring M-6
A	[illegible]5	748	400	700	650, 675, 700 725	650, 675, 700 725

NOTE: All elevations and distances are in feet. | 68

TABLE 2.5.4-5A

(SHEET 2)

DETAILS OF PREEXCAVATION CROSSHOLE SURVEYS | 68

Boring	Surface Elevation
P-1	837
P-2	827
M-5	738
M-6	730

NOTE: All elevations and distances are in feet. | 68

CPSES/FSAR

TABLE 2.5.4-5B

REPRESENTATIVE GEOPHYSICAL DATA* FROM PREEXCAVATION SURVEYS AT STATION LOCATION | 68

Approx. Depth (Feet)	Materials Description	Compressional Wave Velocity Ft/Sec	Method	Shear Wave Velocity Ft/Sec	Method	"Dynamic" Poisson's Ratio
0 to 12	Clayey, sandy silt (soil derived by rock weathering) and weathered limestone	2,700	Refraction	1,000-1,100	Surface Wave	0.42
		2,500	Surface Wave			
12 to 200 (bottom of boring)	Light to dark gray argillaceous limestone (Glen Rose Formation)	12,500	Refraction	5,500	Surface Wave	0.35 to 0.38
		9,600	Uphole	5,600-6,200	Crosshole	
		11,500	Crosshole			
		11,200	Surface Wave			

* Data from uphole and crosshole surveys at borings P-1 and P-2 and refraction survey in vicinity.

** Values are derived by computation using compressional and shear wave data.

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TABLE 2.5.4-5E | 68

(Sheet 1 of 2) | 68

LITHO-STRATIGRAPHY OF SUBSURFACE MATERIALS | 68
TO A DEPTH OF 500 FT | 68

Geologic Characterization						
Geologic Units			Superpositional Boundaries			68
Zone	Formation	Prevalent Lithologics	Depth Range (Ft)	Elevation Range	Thickness	68
I Upper	Glen Rose	Hard limestone with soft claystone interbeds	0 - 23	810 - 787	23	68
I Lower	Glen Rose	Soft claystone alternating with hard limestone	23 - 39	787 - 771	16	68
II	Glen Rose	Massive nodular limestone with occasional claystone partings	39 - 102	771 - 708	63	68
IIIA	Glen Rose	Alternating limestone and claystone	102 - 110	708 - 700	8	68
IIIB	Glen Rose	Massive limestone	110 - 163	700 - 647	53	68
IV	Glen Rose	Alternating limestone and claystone with sand lenses	163 - 200	647 - 610	37	68
Upper	Twin Mountains	Soft-to-hard sandstone with interbedded medium-to-hard claystone	200 - 315	610 - 495	115	68

TABLE 2.5.4-5E | 68

(Sheet 2) | 68

LITHO-STRATIGRAPHY OF SUBSURFACE MATERIALS | 68

TO A DEPTH OF 500 FT | 68

Geologic Characterization						
Geologic Units			Superpositional Boundaries			
Zone	Formation	Prevalent Lithologics	Depth Range (Ft)	Elevation Range	Thickness	
Middle	Twin Mountains	Hard claystone	315 - 352	495 - 458	37	| 68
Lower	Twin Mountains	Alternating hard sandstone and claystone	352 - 444	458 - 366	92	| 68
---	Mineral Wells	Massive and hard claystone	444 - 504**	366 - 306**	60**	| 68

** Maximum depth of borings. | 68

TABLE 2.5.4-5F | 68

SUMMARY OF THE RESULTS OF THE POSTCONSTRUCTION CROSSHOLE SURVEY | 68

Formation	Depth (ft)	Elevation (ft)	Comprssional Wave (ft/sec) No. of Test	Comprssional Wave (ft/sec) Mean	Shear Wave (ft/sec) No. of Tests	Shear Wave (ft/sec) Mean	Poisson's Ratio	
								| 68
GLEN ROSE	0 - 194	809 - 615						| 68
- Limestone (upper)	0 - 36	809 - 773	26	8,631	30	3901	0.37	| 68
- Claystone	19 - 36	790 - 773	16	5,766	31	2380	0.40	| 68
	99 - 105	710 - 704						| 68
- Limestone	36 - 99	773 - 710	137	10,802	161	5066	0.36	| 68
	105 - 194	704 - 615						| 68
TWIN MOUNTAINS	194 - 443	615 - 366						| 68
- Sandstone	varies	varies	71	7,495	67	3073	0.40	| 68
- Claystone	varies	varies	38	7,861	38	2884	0.42	| 68
- Limestone	211 - 217	598 - 592	4	10,904	4	5210	0.35	| 68
- Sandstone/Claystone	varies	varies	23	10,452	24	4691	0.37	| 68
MINERAL WELLS	443 - 503+	366 - 306	20	10,162	19	4339	0.39	| 68

TABLE 2.5.4-5G | 68

DYNAMIC ROCK PROPERTIES DETERMINED BY LABORATORY TESTING | 68

OF CORE SAMPLES | 68

Lithology	Formation	Unit Weight (PCF) No. of Tests	Unit Weight (PCF) Mean	Shear Modulus (KSI) and Damping Ratio No. of Tests	Mean Shear Modulus	Mean Damping Ratio	
	Glen Rose	17	137.8	4	69	4.4	68
Claystone	Twin Mountains	11	137.9	2	110	5.0	68
	All	28	137.9	6	83	4.6	68
	Glen Rose	12	154.4	4	727	1.9	68
Limestone	Twin Mountains	3	156.4	1	729	2.3	68
	All	15	154.8	5	727	2.0	68
	Glen Rose	1	144.4	-	-	-	68
Sandstone	Twin Mountains	20	135.0	6	166	4.7	68
	All	21	135.4	6	166	4.7	68
	Total	:64		Total :17			68

NOTE: Samples were obtained from S/L cluster borings S-1, S-1A, L-1. L-2 core samples were not tested. | 68

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TABLE 2.5.4-5H | 68

DYNAMIC FOUNDATION DESIGN PARAMETERS BASES ON PREEXCUVATION | 68
AND POSTCONSTRUCTION GEOPHYSICAL SURVEYS | 68

Zone Description			Seismic Velocities (Ft/Sec)					
Depth (ft)	Elevation (ft)	Thickness (ft)	Compressional V_p	Shear Vs	Unit Weight (PCF)	Shear Modulus** (KSI)	Poisson's Ratio	
0*	810	23***	9400	4500	150	650	0.35	68
23	787	16	5600	2700	145	230	0.35	68
39	771	161	11400	5500	155	1000	0.35	68
200	610	160	6400	3100	135	280	0.35	68
360	450		9400	4500	150	650	0.35	68

* Plant Grade | 68

** Shear Modulus = Mass Density $(Vs)^2$ | 68

*** Some boring logs indicate substantial claystone strata within this zone. Such strata are treated as subzones and properties of the zone extending from elevation 787 to elevation 771 are applied. | 68

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TABLE 2.5.4-6

SUMMARY OF STATIC BEARING CAPACITY ANALYSIS | 68
OF CATEGORY I STRUCTURES | 68

Structure	Founding Elevation Ft.	Maximum Pressure Ksf	Ultimate Bearing Capacity Ksf	Factor of Safety	
Reactor Containments	795.5/769.2	12.2	60.0	4.9	68
Safeguard Structures	805/767.5/ 781.75	5.7	60.0	10.5	68
Fuel Building	805.3/ 795.2/780.5	10.0	60.0	6.0	68
Auxiliary Building	785/773	6.9	60.0	8.7	68
Service Water Intake Structure	749	12.1	60.0	5.0	68
Condensate Storage Tank	805.5	5.1	60.0	11.7	68
Refueling Water Storage Tank	805.5	5.1	60.0	11.7	68
Reactor Makeup Water Storage Tank	806.6	4.7	60.0	12.7	68

TABLE 2.5.4-6A

SUMMARY OF DYNAMIC BEARING CAPACITY ANALYSIS | 68
OF CATEGORY I STRUCTURES | 68

Structure	Founding Elevation Ft.	Maximum Pressure Ksf	Ultimate Bearing Capacity Ksf	Factor of Safety	
Reactor Containments	795.5/769.2	24.9	60.0	2.4	68
Safeguard Structures	805/767.5/ 781.75	9.6	60.0	6.3	68
Fuel Building	805.3/ 795.2/780.5	13.9	60.0	4.3	68
Auxiliary Building	785/773	13.4	60.0	4.5	68
Service Water Intake Structure	749	18.1	60.0	3.3	68
Condensate Storage Tank	805.5	11.9	60.0	5.0	68
Refueling Water Storage Tank	805.5	11.9	60.0	5.0	68
Reactor Makeup Water Storage Tank	806.6	11.0	60.0	5.4	68

TABLE 2.5.4-7

SUMMARY OF SETTLEMENT ANALYSES
FOR CATEGORY I STRUCTURES | 68

Building	Maximum Settlement (in.) Center	Maximum Settlement (in.) Edge	Differential Settlement (in.)	
Containment Structures (2)	0.26	0.16	0.10	68
Safeguard Buildings (2)	0.12	0.04	0.08	68
Fuel Building	0.18	0.09	0.09	68
Auxiliary Building	0.18	0.09	0.09	68
Service Water Intake Structure	0.08	0.04	0.04	68
Condensate Storage Tank	0.04	0.02	0.02	68
Refueling Water Storage Tank	0.04	0.02	0.02	68
Reactor Makeup Water Storage Tank	0.02	0.01	0.01	68

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TABLE 2.5.4-11

CYLIC SHEAR STRENGTH CRITERIA

| 68

Normal Effective Stress (Kips/ft^2)	Cyclic Shear Stress to 0.05 Strain a /2 (Kips/ft^2)		Cyclic Stress Ratio to 0.05 Strain a/2 o		
	5 cycles	10 cycles	5 cycles	10 cycles	| 68
0	0	0	0	0	
2	1.20	1.05	0.600	0.525	
4	2.10	1.85	0.525	0.463	
6	2.90	2.60	0.483	0.433	| 68
8	3.65	3.25	0.456	0.406	| 68

This table was developed based on Figure 2.5.6-48.

See Figure 2.5A-17 for Filter "A" cyclic shear strength criteria. | 68

TABLE 2.5.4-12

CYCLIC TRIAXIAL TEST PROGRAM 68

SUMMARY OF DENSITIES 68

Material	Specific Gravity	d min (pcf)	d max (pcf)	Density Criteria	Required d test (pcf)	Average d test (pcf)	
Filter A (Sample #1)	2.65	96.8(1)	127.8(1)	80% relative density	120.1	120.6	68
Fine Aggregate (Sample #2)	2.62	95.6(1)	121.6(1)	80% relative density	115.3	115.7	68
Glen Rose Crushed Stone (Sample #3)	2.51		123.6(2)	95% of standard Proctor maximum density	117.4	118.6	68
Crowder Quarry Crushed Stone (Sample #4)	2.60		134.6(2)	95% of standard Proctor maximum density	127.9	128.6	68

1) Relative density test ASTM D2049

2) Maximum density AASHTO T99, Method D

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Table 2.5.4-13

SUMMARY OF CYCLIC STRENGTH VALUES

0.05 Double Amplitude Axial Strain Failure Criteria

Soil Type	Confining Pressure psf	Density Required pcf	Density Tested pcf	Max Shear Stress Required psf	Max Shear Stress Tested psf	Double Amplitude Axial Strain Developed in/in 5 cycles	10 cycles	Dynamic Criteria Achieved 0.05 D.A. Axial Strain 5 cycles	10 cycles	68
Sample #1										
Filter "A"	2000	120.1	120.6	1200	1260	0.0222	0.0953	Yes	No	
	4000		120.5	2100	2120	0.0167	0.0657	Yes	No	
	6000		120.6	2900	2940	0.0271	0.1138	Yes	No	
	8000		120.6	3650	3680	0.0294	0.105	Yes	No	
Sample #2										
C-33, Fine	2000	115.3	114.6	1200	1220	0.0131	0.0225	Yes	Yes	
Aggregate	4000		115.7	2100	2140	0.0121	0.0199	Yes	Yes	
	6000		116.1	2900	2970	0.0108	0.0173	Yes	Yes	
	8000		116.4	3650	3700	0.0131	0.0216	Yes	Yes	
Sample #3										
Glen Rose	2000	117.4	118.1	1200	1230	0.0084	0.0127	Yes	Yes	
Crushed	4000		118.6	2100	2160	0.0117	0.0237	Yes	Yes	
Stone	6000		118.7	2900	2980	0.0182	>.20	Yes	No	68
	8000		119.1	3650	3740	0.0270	>.20	Yes	No	68
Sample #4										
Crowder	2000	127.9	127.9	1200	1220	0.0050	0.0062	Yes	Yes	
Quarry	4000		127.9	2100	2140	0.0050	0.0068	Yes	Yes	
Crushed	6000		129.7	2900	2960	0.0060	0.0081	Yes	Yes	
Stone	8000		129.0	3650	3720	0.0070	0.0084	Yes	Yes	

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Table 2.5.4-13A

SUMMARY OF ADDITIONAL CYCLIC SHEAR STRENGTH TEST | 68

SOIL TYPE	CONFINING PRESSURE (psf)	DENSITY TESTED (psf)	STRESS RATIO	DOUBLE AMPLITUDE AXIAL STRAIN DEVELOPED (IN/IN)		DYNAMIC CRITERIA ACHIEVED 0.05 D. A. AXIAL STRAIN		
				5 CYCLES	10 CYCLES	5 CYCLES	10 CYCLES	68
Sample #1	2000	119.8	0.51	.013	.073	Yes	No	68
Filter "A"	2000	119.3	0.36	.004	.012	Yes	Yes	68
	4000	120.1	0.46	.016	.083	Yes	No	68
	6000	120.0	0.43	.031	.127	Yes	No	68
	8000	120.8	0.41	.062	.211	No	No	68
Sample #3	6000	118.3	0.43	.017	.141	Yes	No	68
Glen Rose	8000	117.8	0.40	.045	>.20	Yes	No	68
Crushed								68
Store								68

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2.5A Construction Records and Design Verification of SSI Dam

2.5A.1 Design Verification at Fifty percent (50%) Construction Completion

In order to verify that the SSI Dam was constructed according to design criteria, two borings were drilled through the impervious core when the embankment was approximately 50% complete. The location of the borings were as follows:

Boring No.	Station	Ground Surface Elevation
SSI-1	6+00	756.4
SSI-2	10+00	756.3

Continuous samples of the impervious core were taken utilizing a Shelby-tube sampler. A detailed log of each boring is shown on the attached Figures 2.5A-1 & -2.

The following laboratory tests were conducted on samples obtained from the impervious core: unit dry weight determinations, Atterberg limit determinations and hand vane shear tests. The results of these tests are summarized on Table 2.5A-1, -2 for Borings SSI-1 & 2, respectively. The results of the Atterberg limit tests confirmed that the impervious core met the liquid limit specification requirement. The hand vane shear tests were conducted as a relative indicator of the uniformity of the layer compaction. For each specimen tested, a vane shear reading was made in both the top and bottom of the sample. Differences in the readings are interpreted to represent the relative variation in the compaction of the embankment. Only slight variations were noted in the vane shear readings.

A series of hydrometer and sieve analyses were conducted on representative samples and are shown on Figure 2.5A-3. All the tested specimens are classified as CL material.

68 The strength parameters of the impervious core were evaluated by a series of unconfined compression tests as well as three (3) multi-stage triaxial compression tests. Summarized on Table 2.5A-3 are the results of the unconfined compression tests. Figure 2.5A-4 is a typical stress-strain curve developed for one of the unconfined compression tests. The results of the triaxial tests are summarized on Figure 2.5A-5. The results confirmed that the material actually possessed a strength higher than assumed during design.

In summary, the results of these tests confirmed that all design criteria and specification requirements had been met at 50 percent completion of the embankment.

2.5A.2 Design Verification at One Hundred Percent (100%) Construction Complete

68 Upon completion of the SSI dam, a core boring was drilled through the entire embankment at centerline station 9+75 to obtain undisturbed Shelby-tube samples throughout the entire depth of the impervious core section. A detailed "Log of Boring" is attached as Figure 2.5A-6.

Utilizing all the samples obtained of the impervious core, a series of laboratory tests, including Atterberg Limits, unit dry weight determinations, and hand vane shear tests was conducted. The results are summarized on Table 2.5A-4. The results of the Atterberg limit tests confirmed that the impervious core met the liquid limit specification requirement. The results of the vane shear readings show the material to be relatively uniform in compaction.

To evaluate the shear strength of the impervious core, a series of consolidated undrained triaxial tests were conducted with the results plotted on Figure 2.5A-7. This is a plot of the Mohr Stress circles developed from the individual tests with the design criteria failure envelope drawn showing its relationship to the test results. As can be seen on this Figure, the results of the CU triaxial tests fit very closely to the adopted design strengths.

On Figures 2.5A-8 through 10 are results of consolidation tests to the impervious core.

The results of the testing shows that the impervious core met all specification requirements and design criteria.

2.5A.3 Field Density Test Distribution

To further show the adequacy of the completed embankment, a ploy showing the distribution of density control tests has been developed and is shown on Figures 2.5A-11, Sheets 1 through 15. These plots show the distribution of the density test conducted for the impervious core, filter zone and rock fill material at 5-foot intervals through the embankment. In addition, the test results have been shown in histogram plots. These histogram plots are shown as Figure 2.5A-12 through 15. In summary, the results showed the following: | 68

Data Base	No. of Tests	Mean	Standard Deviation	Design Criteria
% Compaction of Impervious Core	309	102%	2.3	>95%
Moisture Content Variation, W_o-W_f	309	-0.59%	1.1	-3%+1%
Percent Relative Density-Filter	585	96.7%	9.3	>80%
Rock Fill Density	75	125.0pcf	9.8	115pcf

Histograms of the grain size distributions for the rock fill, filter A and filter B are presented in Figures 2.5A-19 and 2.5A-20. | 68

2.5A.4 Dynamic Stability Re-evaluation

During construction a series of cyclic triaxial tests were conducted on filter "A" material to verify whether or not his material met the adopted design criteria as discussed in Section 2.5.6.4.3.4. According to the test results presented in Section 2.5.4.7, filter "A" material failed to meet the adopted design criteria. Based on the cyclic triaxial test results for the filter "A" material, Figure 2.5A-16 was developed which relates the cyclic strength of the material based on a failure criteria for both 5% and 10% of double amplitude strain. From this figure, new cyclic design criteria for 10 cycles of loading and a failure criteria of 5% double amplitude strain was developed for the filter material and is shown on Figure 2.5A-17. Utilizing this new design criteria, a new seismic stability cross-section of the SSI Dam was developed. Figure 2.5A-18 relates shear stress induced by the artificial ground motion, $_d$, to the cyclic shear stress required to cause 5% double amplitude strain, $_f$. As was the cause in Section 2.5.6.5.2.6, the ratio $_d/_f$ represents a local safety factor against the development of 5% strain in 10 cycles of loading. Figure 2.5A-18 is the plot showing the calculated local safety factors at the nodal points of the finite element grid. Several points had values less than one (1) for the rock shell at points on the outer slope surfaces near the crest of the dam. These values result from the presence of little or no normal stress. An interpretation of these results would be that, during the occurrence of the SSE, an occassional rock particulate on the outer slope may be displaced enough to tumble or roll some short distance down the slope.

Near the top of the filter zone, two nodal points have safety factors calculates as 0.94 and 0.97. This is interpreted as meaning that, during the occurrence of the SSE, the upper zone of the filter material will experience 5% double amplitude strains. This will not result in the liquefaction of the filter material and will not impare the function of the dam.

TABLE 3.2-1

QUALITY STANDARDS

Components	Safety Class 1 (Note 1)	Safety Class 2 (Notes 2 and 5)	Safety Class 3 (Notes 2 and 5)	NNS Class (Notes 3 and 4)	
					| 56 | 56
Pressure Vessels	ASME B&PV Code, Section III, Nuclear Power Plant Components, Class 1, Components NB-3300	ASME B&PV Code, Section III, Nuclear Power Plant Components, Class 2, Components NC-3300, NC-3200	ASME B&PV Code, Section III, Nuclear Power Plant Components, Class 3, Components ND-3300 and ND-3800.	ASME B&PV Code, Section VIII, Division 1	| 56 | 56 | 56 | 4 | 4 | 68
Atmospheric Storage Tanks	N/A	N/A	ASME B&PV Code, Section III, Nuclear Power Plant Components, Class 3, Components Article ND-3800	ASME B&PV Code, Section VIII Division 1	| 4 | 56 | 56 | 21
Steel-lined Concrete Storage Tanks	N/A	ACI-318	ACI-318	N/A	| 21 | 21 | 21
Supports	ASME B&PV Code Section III, Nuclear Power Plant Components Class 1 Components Article NF	ASME B&PV Code Section III, Nuclear Power Plant Components Class 2, Article NF	ASME B&PV CODE Section III, Nuclear Power Plant Components Class 3, Article NF, ANSI B31.1	ANSI B31.1	| 56 | 4 | 4 | 4 | 56

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42 3.3.2.2 Tornado Protection Design Features

42 Venting from the building interior compartments to the exterior is provided by roll-up door F-4E, tornado pressure relief dampers, and tornado pressure relief blowout panels.

42 1. Roll-up door F-4E

The Fuel Building is primarily vented by an opening covered by
68 roll-up door F-4E. This door is specified to partially blowout when the differential pressure across the door reaches 1.0 psi.

55 2. Tornado Pressure Relief Dampers in Exterior Walls

55 All exterior tornado dampers are specified to remain closed during winds with speeds up to 119 mph. Tornado dampers in
42 exterior walls are specified to open in either direction when the differential pressure across the damper reaches 0.25 psi in either direction. Concrete missile shields, as shown on Figure 3.3-1, protect the dampers and the building interior from tornado generated missiles.

55

42 3. Tornado Pressure Relief Blowout Panels

42 Airtight blowout panels are used in the venting of the Auxiliary Building, the Control Room and the Safeguards Building. All blowout panels are specified to remain airtight during winds with
52 speeds up to 119 mph. Blowout panels are specified to open when the differential pressure across the panel reaches 0.25 psi.
68 Where a missile can impact a blowout panel, a Seismic Category I building is protected by reinforced concrete missile resisting walls and roof so arranged as to stop a missile as shown in Figure 3.3-2 and Figure 3.3-3. Where a missile is stopped by a concrete barrier and could then enter by gravity into a Category I building protective grating is provided as shown in Figure 3.3-3.

The Diesel Generator Building is primarily vented through the Diesel 42
Generator air intakes. Venting between interior compartments that 55
are separate fire areas is provided by fire rated architectural door
openings, tornado pressure relief dampers (in series with fire
dampers), and fire rated HVAC air transfer grilles.

1. Fire Rated Architectural Door Openings 42

Roll-up door E-3A is specified to blowout of the E&C building 68
into the turbine building. The door is specified to blowout
when the differential pressure across the door reaches 1.0 psi.

The cable spreading rooms are vented by specially modified 55
security doors E-23, E-24 and E-25; each of these doors is
equipped with a pneumatically operated pressure sensitive door
opener. The door opener is designed to open the door at a
maximum time of 1 second after a differential pressure of 1.0 psi 42
is reached. Each door is equipped with reserve air and
electrical systems. Each of these doors are fire-rated and
designed to open only during tornadic loading.

Several standard hollow metal doors are held open with fusible 42
link arm-holders designed to close the door during a fire.
Wire-mesh doors are also provided for radiation protection
(access control) as required. Where other considerations, such
as HVAC integrity, require that a hollow metal door be closed,
and venting is still required the door is modified in such a way
as to allow it to release during a tornado. These doors are
also restrained during release to prevent destruction of the
door. Fire rated doors are also provided in the same doorway
and are also held open with fusible link arm-holders designed to
close the door during a fire.

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2. R. G. Gido, G. J. E. Willcutt, Jr., J. L. Lunsford, and J. S. Gilbert, COMPARE-MOD 1 Code Addendum, NUREG/CR-1185, Addendum 1 (LA-7199-MS, Addendum), June 1980, and Attachment September, 1980. | 68

3. K. C. Mehata, et al. The Kalamazoo Tornado of May 13, 1980. Committee on Natural Disasters, Commission on Sociotechnial Systems. National Research Council, National Academy Press 1981. | 42

4. Amercian Society for Testing and Materials. Standard Test Method for Structural Performance of Exterior Windows, Curtain-Walls, and Doors by Uniform Static Air Pressure Difference, ANSI/ASTM E 330-79. | 42

5. American Society of Civil Engineers, Nuclear Power Plant Tornado Design Considerations Journal of the Power Division, Proc. ASCE, Vol. 97, No. PO2. | 42

TABLE 3.3-1

(Sheet 1 of 4)

NORMAL DOOR POSITIONS FOR TORNADO VENTING

DOOR NO.	POSITION	DOOR NO.	POSITION	
Fuel Building				
F-4A	Open	F-12	Open	55
F-4B	Open	F-12x	Closed	55
F-4E	Closed	F-13	Open	55
F-5	Open	F-13x	Closed	55
F-9	Open	F-15	Closed	55
		F-4FX	Open	66
Turbine Building				
T-40	Closed			
Electrical & Control Building				
E-1	Open	E-20B	Open	68
E-1C	Closed	E-22A	Open	55
E-1D	Open	E-22B	Open	55
E-1DX	Closed	E-22X	Open	55
E-1E	Open	E-23	Closed	55
E-1EX	Closed	E-28	Open	55
E-1F	Closed	E-29	Closed	55
		E-33	Open	66
E-3B	Open	E-34	Closed	55
E-3C	Open	E-35	Open	55
E-4	Open	E-40A	Closed	55

TABLE 3.3-1
(Sheet 2)

NORMAL DOOR POSITIONS FOR TORNADO VENTING

DOOR NO.	POSITION	DOOR NO.	POSITION	
E-5	Open	E-40B	Closed	55
E-6	Open	E-41	Closed	55
E-7	Open	E-41A	Open	55
E-8	Open	E-41C	Open	55
E-9	Open	E-41H	Open	55
E-10	Closed	E-43	Open	55
E-14	Open	E-45	Closed	55
E-15	Open	E-45A	Closed	55
E-16	Open	E-45B	Closed	55
E-17	Open	E-45C	Closed	55
E-18	Open	E-45D	Closed	55
E-19	Closed	E-46	Open	55
E-20A	Open			

Auxiliary Building

A-10B	Closed	A-29	Closed	
A-10E	Open	A-30	Closed	
A-10F	Open	A-32	Closed	
A-14A	Closed	A-32A	Closed	
A-16	Closed	A-32B	Closed	
A-17	Closed	A-32C	Closed	
A-18	Closed	A-32D	Closed	
		A-36	Open	66
		A-37	Open	66
A-22	Closed	A-38	Open	
A-23	Closed	A-39	Open	
A-23A	Closed	A-40E	Open	
A-24	Closed	A-44	Open	
A-28	Closed	A-46	Open	

TABLE 3.3-1
(Sheet 3)

NORMAL DOOR POSITIONS FOR TORNADO VENTING

Safeguards Buildings - Units 1 & 2

Note Door Numbers and Positions are generic for both units | 68

DOOR NO.	POSITION	DOOR NO.	POSITION	
S-1	Open			66
S-1X	Closed			66
				68
S-10X	Closed	S-33X	Closed	55
S-11	Open			68
S-11X	Closed			68
S-12X	Closed			68
S-13	Closed			66
S-14	Closed	S-35	Closed	
S-15	Closed	S-35B	Closed	
S-16	Closed	S-35C	Closed	
S-17	Closed	S-37	Open	68
S-18	Closed	S-37X	Closed	68
S-19	Closed	S-38A	Closed	
S-19A	Closed	S-38B	Closed	
S-20	Closed	S-38C	Closed	
S-20A	Closed	S-38D	Closed	
S-22	Open	S-38E	Closed	
		S-38F	Closed	66
S-24	Closed	S-38G	Closed	
		S-40	Closed	66
S-26	Closed	S-40A	Closed	

TABLE 3.3-1

(Sheet 4)

NORMAL DOOR POSITIONS FOR TORNADO VENTING

DOOR NO.	POSITION	DOOR NO.	POSITION	
S-28	Closed	S-40B	Closed	
S-28A	Open	S-40C	Closed	
		S-43	Open	| 66
S-29C	Closed	S-44	Open	
S-29D	Closed	S-45	Open	
S-29E	Closed	S-46	Open	
				| 68

3.6B.1.2 Description

Essential systems are defined as those systems that are needed to shut down the reactor and mitigate the consequences of the pipe break for a given postulated piping break.

3.6B.1.2.1 Protection Criteria

Depending upon the type and location of the postulated pipe break, certain safety equipment may not be classified as essential for the particular event. Some safety equipment will be essential for almost all cases. This category includes service water to the ultimate heat sink and the pressurizer level instrumentation. The containment integrity and leak tightness will be maintained for any LOCA break. | 40
The effects of a postulated piping failure, including environmental conditions resulting from the escape of contained fluids, will not preclude habitability of the control room or access to surrounding areas important to the safe control of reactor operations needed to cope with the consequences of the piping failure. Accordingly, protection from the effects of pipe rupture will be provided for only that safety-related equipment considered as essential on a case-by-case basis.

The systems or portions of systems and equipment for which protection against postulated pipe failures is required are identified below. However, in general, protection from pipe failure need not be provided if any of the following conditions exists:

a. The piping is physically separated (or isolated) from structures, systems, or components important to safety by protective barriers, or is restrained from whipping.

68 b. Following a single break, the unrestrained pipe movement of either end of the ruptured pipe about a plastic hinge formed at the location determined by calculation cannot impact any structure, system, or component important to safety.

c. The internal energy level associated with the whipping pipe can be demonstrated to be insufficient to impair the safety function of any structure, system, or component to an unacceptable level.

The following systems or portions of these systems are required to mitigate the consequences of a postulated pipe failure:

1. Auxiliary Feedwater System

2. Chemical and Volume Control System

3. Feedwater System

4. Main Steam System

5. Reactor Coolant System

6. Residual Heat Removal System

7. Safety Injection System

8. Safety Chilled Water System

9. Hydrogen and Nitrogen System

10. Containment Spray System

11. Diesel Generator System

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15 b) The primary plus secondary stress intensity range derived on an elastically calculated basis under loadings associated with the OBE and normal and upset plant conditions exceeds 2.4 S_m but is less than 3.0 S_m, and the cumulative usage factor is less than 0.1 <u>or</u>

15 c) The primary plus secondary stress intensity range derived on an elastically calculated basis under loadings associated with the OBE and normal and upset plant conditions exceeds 3.0 S_m but the stress ranges computed by Equations (12) and (13) of Subparagraph NB-3653 of ASME III are less than 2.4 S_m and the cumulative usage factor is less than 0.1.

15 Where intermediate break locations are not required, based upon the preceding criteria, two postulated pipe break locations will be selected on the basis of the highest Equation (10) stress. However, only one intermediate break need be postulated in sections of straight pipe where there are no fittings, valves, or welded attachments.

B. ASME Section III, Code Class 2 and 3 Piping

Q112.2

ASME B&PV Code, Section III, Class 2 and Class 3 piping breaks are postulated to occur at terminal ends and intermediate
68 locations in each piping run or branch run. Breaks at intermediate locations are selected by either of the following criteria:

Q112.2

68 a. At each location where the stresses associated with normal and upset plant conditions and an OBE event, calculated by the sum of Equations (9) and (10) of paragraph NC-3652 of the ASME B&PV Code, Section III, exceed
68 $0.8(1.2\ S_h+S_A)$.

Q112.2

b. Certain pipe runs contain Class 2 piping extensions to 2
Class 1 lines up to the first anchor point beyond the Class
1/Class 2 boundary. For these pipe runs a break is 68
postulated at the Class 2 terminal end. Breaks are not
postulated in the Class 2 portion if the stresses in the 2
Class 1 portion, calculated using Equation (10) of NB-3653,
are above 2.4 S_m and the sum of the stresses in the Class
2 portion calculated using Equations (9) and (10) of NC-
3652, are below 0.8 (1.2 S_h + S_A).

Q112.3

C. Non-Nuclear Piping

Breaks are postulated to occur in non-nuclear piping systems at the locations as specified for ASME Section III, Class 2 and 3 piping in accordance with Section B criteria above when a seismic stress analysis is performed.

Breaks in non-nuclear piping systems are postulated at terminal ends and at intermediate points, such as fittings (elbows, tees, reducers, etc.), welded attachments and valves in each run or branch run where a stress analysis is not performed.

D. Fluid System Piping Between Containment Isolation Valves

Q112.3

There are no ASME Section III, Class I piping penetrations. The 2
piping between containment Isolation valves is ASME B&PV Code,
Section III, Class 2 piping.

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Q112.3

68 Breaks are not postulated in the break exclusion areas of high energy fluid system piping from the inside containment process-pipe-to-penetration flued head weld to and including the outside containment isolation valve moment restraint forging as shown in figures 3.6B-15, 16, 17, 18, 23, 25, 26, 27, 28, 29, 30, 42, 82, 85, 87 and 88. Moment restraints are provided to protect the break exclusion areas from a postulated piping failure beyond these portions of piping. A typical moment restraint is shown in Figure 3.6B-5.

68 The following stress limits of Branch Technical Positions APCSB 3-1 [3] and MEB 3-1 [2] for ASME B&PV Code Section III Class 2 piping are met:

Q112.3

2 1. The maximum stress ranges as calculated by the sum of Equations (9) and (10), paragraph NC-3652 of the ASME B&PV Code, Section III, considering normal and upset plant conditions (i.e., sustained loads, occasional loads, and thermal expansion), and an OBE event do not exceed
2 $0.8\ (1.2S_h + S_A)$.

Q112.2

68 2. The maximum stress as calculated by Equation (9), paragraph NC-3652 of the ASME B&PV Code, Section III, under the load combination of internal pressure, dead weight and a postulated piping failure of fluid system piping beyond these portions of piping do not exceed $1.8S_h$.

Welded attachments, for pipe supports or other purposes, to these portions of piping are avoided except where detailed stress analyses or tests are performed to demonstrate compliance with the limits defined in D.1 and D.2 above. In addition, the number of circumferential and longitudinal piping welds and branch connections is minimized. Guard pipes are not used.

The length of these portions of piping is reduced to the minimum length practical. The design of pipe anchors or restraints (e.g., connections to Containment penetrations and pipe whip restraints) does not require welding directly to the outer surface of the piping (e.g., flued-integrally forged pipe fittings are used). If welded attachments are required, such 40
welds are 100-percent volumetrically examinable in service, and a detailed stress analysis shall be performed to demonstrate compliance with the limits defined in D.1 and D.2 above.

3.6B.2.1.3 Type of Breaks Postulated in Fluid System Piping Other Than the RCS Main Loop

Circumferential and longitudinal breaks are postulated in piping systems at locations previously discussed. The following are definitions and criteria used to determine the type of break to be considered:

A. Circumferential Pipe Breaks

Circumferential breaks are defined as a full cross section area break, with at least one diameter lateral displacement of the ruptured pipe. Circumferential breaks are postulated in piping with nominal size greater than 1", unless the break separation is physically limited by piping restraints, by structural members, or by piping stiffness as demonstrated by inelastic limit analysis. Circumferential breaks are perpendicular to the pipe axis, and the break area is equivalent to the internal cross-sectional area of the ruptured pipe. Dynamic forces at the break location are based on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. These forces separate the piping axially causing the ruptured

3.6B.2.2.2 High-Energy Piping Other Than RCS Main Loop

The time dependent function representing the thrust forces caused by the jet flow from a postulated pipe break or crack includes the combined effects of the thrust impulse resulting from the sudden pressure drop at the initial moment of pipe rupture, the thrust transient resulting from wave propagation and reflection, and the blowdown thrust resulting from buildup of the discharge flow rate which may reach steady state if there is a fluid energy reservoir having sufficient capacity to develop a steady jet for a significant interval. Alternatively, in a simplified method, the jet thrust force is represented by a steady state function. This function, representing the force, would have a magnitude not less than:

$$F_{ss} = K_t PA$$ | 68

where: | 31

F_{ss} = Steady State Thrust Force (lb_f) | 31

P = system pressure prior to pipe break (lb_f/in^2) | 31

A = pipe break area (in^2) | 31

K_t = steady state thrust coefficient | 68

The steady state thrust coefficient K_t is dependent on the fluid state and the frictional loss terms. The value of steady state thrust coefficient and the time to reach steady state flow conditions are calculated from references [15], [16] and [22]. | 68

The rigorous time dependent blowdown forces resulting from a postulated pipe rupture are determined using the RELAP-5 computer code [6]. RELAP-5 is a thermal/hydraulic program commonly used in | 68

68 the nuclear industry to evaluate the behavior of water cooled reactor
systems during postulated accidents such as pipe ruptures. The
program is acceptable (see Reference [7]) as a means of determining
68 the hydraulic forcing function at the pipe break. CALPLOTF [20], Plot
Processor Program to Relap Program, is used to develop the break force
time - history plots.

68 The RELAP-5 program solves the transient energy, momentum, and fluid
state equations to determine the system flow, pressure, and
thermodynamic conditions. The break force is computed using the
one-dimensional momentum equation and the appropriate density,
internal energy, and pressure values. The rupture load is the
summation of the pressure, momentum, and change in momentum terms at
the time interval in question.

68 RELAP-5 has the capability of solving the fluid state equation for
subcooled water, flashing water, two-phase steam/water mixtures, and
superheated steam. The ASME steam tables [9] have been incorporated
into RELAP-5 so that the fluid state properties are accurately
determined. Moody's critical flow model [10] is used to establish
the maximum flow rate (choked flow), and an inertial mode is used to
compute the nonchoked flow. The program uses the lower of these two
values at the designated flow. A bubble rise model can be used to
describe the depressurization of a large, essentially stagnant volume
68 as in the case of a steam generator after a pipe ruptures. RELAP-5
has a provision for modeling components such as valves, check valves,
pumps, heat exchangers, and reactors along with the associated piping.
Transients can be initiated by the control card added to the
program which is used to describe leaks (pipe breaks), valves opening
and closing, check valve pressure drop-flow-characteristics, pump
coastdowns, and so forth.

The flow system is described as a series of volumes connected by flow
68 paths or junctions. RELAP-5 requires input data that completely

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describe the thermodynamic conditions and physical data of the system being analyzed. Pressure, temperature, and flow conditions along with physical dimensions, flow areas, friction characteristics, and path inertias must all be specified as initial conditions. The break area can be deduced by an analytically or experimentally derived discharge coefficient. However, in lieu of such data it is conservatively assumed that the discharge coefficient is 1.0 for both longitudinal and circumferential breaks. In a similar manner, the break area is assumed to open within one millisecond (0.001 second). | 68

The piping dynamic responses resulting from a postulated pipe rupture are determined using the PIPERUP [13], SHPLAST 2267 [24] or ABAQUS [21] computer codes. The program is an adaptation of the finite element method to the requirements of pipe rupture analyses. It performs a dynamic, nonlinear, elastic-plastic analysis of piping systems subjected to time-history forcing functions. These forces result from fluid jet thrust at the location of a postulated longitudinal or circumferential rupture of high energy piping. | 68

The piping is mathematically modeled in the PIPERUP, SHPLAST 2267 or ABAQUS program as an assembly of weightless structural members connecting discrete nodal points. A typical pipe whip mathematical model is shown in Figure 3.6B-6. Weight of the system, including distributed weight of the piping and concentrated weights (e.g., valves), is lumped at selected mass points (lumped parameter analysis model). Nodal points are placed in such a manner as to isolate particular types of piping elements such as straight runs of pipe, valves, elbows, etc. for which force-deformation characteristics may be determined. Nodal points are also placed at all discontinuities such as piping restraints, branch lines, and changes in cross-section. Piping restraints are modeled in PIPERUP with an initial gap and bilinear stiffness curve, or, in SHPLAST 2267 and ABAQUS with multilinear stiffness curve. A typical piping stress-strain curve is shown in Figure 3.6B-7. The first stiffness represents linear elastic behavior and the second stiffness models linear strain hardening | 68 behavior. PIPERUP utilizes a direct step-by-step integration method to determine the time history response of the ruptured piping system. A typical restraint impact curve is shown in Figure 3.6B-8. An incremental procedure is used to account for the nonlinear deformation and elastic-plastic effect of the pipe and restraints. The incremental equation of motion is evaluated by the Newmark's method [14].

3.6B.2.3 Dynamic Analysis Methods to Verify Integrity and Operability

3.6B.2.3.1 Reactor Coolant System Main Loop

61 The leak-before-break technology has been applied to CPSES Units 1 and 2 to exclude from the design basis the dynamic effects of postulated ruptures in the RCS main loop piping. This applies, in particular, to jet impingement loads on components and supports.

61 Jet loads from large branch nozzle breaks are addressed in Section 3.6B.2.3.2.

3.6B.2.3.2 High-Energy Piping Other than the RCS Main Loop

Pipe beaks are postulated in high-energy piping in accordance with the criteria in Section 3.6B.2.1.2. The analyses for determining the dynamic effects of pipe break are as follows:

B. Pipe Whip Dynamic Analysis Criteria

An analysis of the pipe run or branch is performed for each longitudinal and/or circumferential postulated rupture at the break locations determined in accordance with the criteria of Section 3.6B.2.1.2. The loading condition of a pipe run or branch prior to postulated rupture, in terms of internal pressure, temperature and stress state, is assumed to be the condition associated with the normal plant operating condition.

For a circumferential rupture, pipe whip dynamic analyses are only performed for that end (or ends) of the pipe or branch that is connected to a contained fluid energy reservoir having sufficient capacity to develop a jet stream. Dynamic analytical methods used for calculating the piping and piping/restraint system response to the jet thrust developed after a postulated rupture adequately account for the effects of the following:

a. Translational masses (and rotational masses for major components) and stiffness properties of the piping system, restraint system, major components and support walls.

b. Transient forcing function(s) acting on the piping system and jet thrusts on affected structures.

c. Elastic and inelastic deformation of piping and/or restraint.

A 10 percent increase of minimum specified design yield strength (S_y) is used to account for strain rate effects in inelastic nonlinear analyses.

3.6B.2.3.3 Pipe Whip Restraint Design Criteria

A. Design Bases

Pipe whip restraints function primarily as a load carrying member for the low probability occurrence of a pipe break. The restraints are designed for one time use only and function to control the movement of the ruptured pipe. The design basis for the pipe break event is that of a faulted condition. The restraints and the structure which supports them are analyzed accordingly.

B. Functional Requirements

68

High-energy pipe whip restraints are designed to ensure that the pipe whip will be eliminated or minimized. On the other hand, the restraints are designed to permit the predicted thermal and seismic movements of the pipes.

C. Design Parameters

After the pipe restraint locations are identified, the following design parameters are determined:

1. Jet thrust force

2. Pipe seismic displacements

3. Pipe thermal displacements

4. Pipe insulation thickness

5. Minimum allowable tolerance between restraint and pipe insulation.

6. Maximum allowable pipe movement.

The jet thrust force and maximum allowable pipe movement are used in the analysis process. Tolerance, insulation, and seismic and thermal movements are used in determining the minimum gap between the restraint and pipe surfaces.

The pipe whip restraints used for the CPSES project are U-bar, crushable pipe, crushable pad (honeycomb), and elastic hard restraints as shown in Figures 3.6B-1 through 3.6B-4.

D. Analysis and Design

The maximum allowable design limits for the restraints are as follows:

| Q112.5 68

a. The permanent strain in the metallic ductile materials is limited to [illegible] percent of the minimum ultimate uniform strain (strain corresponding to the maximum stress point on the appropriate engineering stress-strain curve) based on restraint material tests and/or ASME Code [4].

b. The design limit for crushable pipe restraints longer than three outside pipe diameters is 50 percent of energy absorbing capacity (crushed to 70 percent of inside diameter of pipe).

Other than for the Emergency Core Cooling System lines, which must circulate cooling water to the vessel, the engineered safety features are located outside of the steam generator compartment walls. The Emergency Core Cooling System lines which penetrate the steam generator compartment walls are routed around and outside the walls to penetrate the walls in the vicinity of the loop to which they are attached.

3.6B.2.5.2 High-Energy Piping Other Than RCS Main Loop

In this section, a summary is presented giving the results of the detailed stress analysis and, describing methods of protection employed to protect essential equipment against the effects of pipe breaks for the high energy systems outlined in Section 3.6B.1.2.1.

1. Main Steam System

   A. General Description

      The main steam piping inside containment is carbon steel ASME SA-155, Grade KCF 70 material designed in accordance with the ASME Code, Section III, Class 2 criteria. The main steam system inside containment consists of four 32 inch OD (1.25 inch minimum wall thickness) lines running from each steam generator to the containment penetrations.

The main steam piping outside containment from the containment penetrations to the main steam isolation valve moment restraints is the same as the main steam piping inside containment. The piping from the MSIV moment restraints to the high pressure turbine is carbon steel ASME SA-155 Grade KC 70 material designed in accordance with ANSI B31.1 as non-nuclear class piping. These lines are 34 inch OD (1.25 inch minimum wall thickness). The piping connected to the main steam drip pots are of carbon steel material with portions designed in accordance with ANSI B31.1 and ASME Code, Section III, Class 2 criteria. The Class 2 portion of the system consists of ASME SA-333 Grade 6, two inch schedule 80 pipe and the non-nuclear portion of the system consists of ASME SA-106 Grade B, two inch schedule 80 pipe. The location and configuration of the main steam lines with respect to structures, equipment, and other piping are shown on Figures 1.2-8, 1.2-14 and 1.2-25. The criteria described above and as follows is applicable for both Units 1 and 2.

B. Pipe Whip Analysis

Isometrics of the main steam lines inside Containment indicating the location of the highest stress node points, postulated breakpoints, and restraints are provided in Figures 3.6B-11 through 3.6B-14. The systems and equipment necessary to mitigate the consequences of a main steam line
68 break are described in Section 3.6B.1. Breakpoints were postulated at the terminal ends of the piping run and at intermediate locations in accordance with the criteria outlined in Section 3.6B.2.1. The steam generator nozzles and the flued heads at the containment penetrations are
68 considered terminal ends. Intermediate breaks are postulated as shown. A circumferential break is postulated

to occur at any one of these points. Restraints are provided on each line to prevent impact on essential components. The pipe whip restraints are designed to prevent plastic hinge formation and thereby preclude adverse pipe whip effects.

Isometrics of the main steam lines outside containment indicating the location of postulated breakpoints and restraints are provided in Figures 3.6B-15 through 3.6B-18. Since these lines consist of non-nuclear piping, pipe breaks are postulated at each fitting, valve, or welded attachment. Since these lines are greater than 4 inches in diameter, circumferential or longitudinal breaks are postulated. Pipe whip restraints are provided as necessary that are designed to prevent plastic hinge formation and thereby preclude adverse pipe whip effects. A portion of the main steam lines between the containment penetrations and isolation valves, are designed in accordance with the criteria in Section 3.6B.2.1.2. Therefore, circumferential 40
pipe breaks are not postulated in the regions between the penetrations and the moment restraints after the isolation valves. However, a one square foot crack is postulated and evaluated for environmental effects, in accordance with the criteria in BTP ASB 3-1. As shown on Figure 3.6B-25, pipe breaks are postulated in the main steam blowdown lines outside containment. Since these lines consist of non-nuclear piping, break points are postulated at terminal ends and at each intermediate fitting, valve or welded attachment with the exception of the four break exclusion areas as shown.

Review of the piping layout and plant arrangement showed that breaks in portions of the main steam lines could adversely impact the safeguards, switchgear, electrical and controls and turbine buildings. Accordingly, these lines are restrained as necessary to protect the structures. The pipe whip restraints are designed to prevent plastic hinge formation and thereby preclude adverse pipe whip effects on the structures. Since there are no essential systems or components in the turbine building, protection is primarily provided to protect the structures noted above from a main steam pipe break.

C. Jet Impingement Analysis

40 The jet impingement analysis for this system is performed to determine the effects of jet impingement loading on essential components (as defined in Section 3.6B.1.2.1), associated supports and building structures.

40 In those cases where the analysis shows that the component or structure is not capable of withstanding the load then protection is required. Protection consists of either relocating the target, or installing jet shield or barriers to protect the target.

40 D. Environmental Analysis

40 The safety-related systems required to mitigate the consequences of a main steam line break inside Containment are designed to perform their safety function under the environmental conditions resulting from a LOCA or MSLB as discussed in Section 3.11.

40 Any one of the main steam lines is postulated to develop a 1 ft^2 crack in the penetration area outside Containment. The postulation of this crack is in accordance with Branch Technical Position ASB 3-1.

The penetration area is isolated from the rest of the 40
safeguards building by the seismic gap seal between the safeguards building and the containment wall, fast closing isolation dampers in the interconnecting HVAC ducts and the pressure resisting watertight doors. These features preclude this crack from producing any detrimental environmental effects on the rest of the plant.

2. Feedwater System

A. General Description

The main feedwater lines inside containment are carbon steel ASME SA-333, Grade 6 material designed in accordance with the ASME Code, Section III, Class 2 criteria. Each of the lines consists of an 18 inch schedule 80 seamless pipe running from the Containment penetration to each steam generator. The main feedwater piping outside Containment from the containment penetration to the feedwater containment isolation valve is the same as the feedwater piping inside Containment. The piping from the feedwater containment isolation valve to the feedwater control valve moment restraint is also the same as the feedwater piping inside containment, except that these lines are 18 inch schedule 140 seamless pipe. The main feedwater lines from the feedwater control valve moment restraint connect to the main feedwater header combining into one main feedwater line which originates from the feedwater heater in the turbine building. The feedwater lines to the main feedwater header and the feedwater control valve by-pass lines are carbon steel ASME SA-106, Grade B material designed in accordance with ANSI B31.1 as non-nuclear class piping. These lines consist of 18 inch schedule 140 and eight inch schedule 120 pipe. The main feedwater header and feedwater line in the turbine building are carbon steel ASME SA-155 Grade KC 60 material designed in accordance with ANSI B31.1 as non-nuclear class piping. These lines are 30 inch OD (2.125 inch minimum wall thickness) pipe. The location and configuration of the feedwater lines with respect to structures, equipment, and other piping are shown in Figures 1.2-8, 1.2-13 and 1.2-25.

The criteria described above and as follows is applicable for both Units 1 and 2.

B. Pipe Whip Analysis

Isometrics of the main feedwater lines inside Containment indicating the location of the highest stress node points, postulated breakpoints, and restraints are provided in Figures 3.6B-19 through 3.6B-22. The systems and equipment necessary to mitigate the consequences of a main feedwater
68 line break are described in Section 3.6B.1. Breakpoints were postulated at the terminal ends of the piping run and at intermediate locations in accordance with the criteria outlined in Section 3.6B.2.1. The flued heads at the containment penetrations and the steam generator nozzles
68 are considered terminal ends. Intermediate breaks are postulated as shown. A circumferential break is postulated to occur at any one of these points. Pipe whip restraints are provided as required to prevent impact on essential components. The pipe whip restraints are designed to prevent plastic hinge formation and thereby preclude adverse pipe whip effects.

Isometrics of the main feedwater lines outside containment, indicating the location of postulated breakpoints and restraints, are provided in Figures 3.6B-23 and 3.6B-24.

High energy flooding is evaluated in the same manner as 40
moderate energy flooding per Section 3.6B.2.5.3.

3. Auxiliary Feedwater System 14

A. General Description 14

The Auxiliary Feedwater System lines are carbon steel, ASME 14
SA 106, Grade B material designed in accordance with ASME Code Section III, Class 2 or 3 criteria as applicable. The suction lines of the auxiliary feedwater pumps consist of 10, 8 and 6 inch category 152 piping which is schedule 40, with design pressure of 150 psig. These lines run from the condensate storage tank and service water piping to the suction of each auxiliary feedwater pump. The discharge lines of the auxiliary feedwater pumps consist of 6, 4 and 3 inch category 2002 (category 2003, directly after the isolation valve) piping which is schedule 160 up to 3 inches, and schedule 120 for sizes above 3 inches. The design pressure is 1800 psig. The piping that connects to the main feedwater piping after the last auxiliary feedwater isolation valves is category 1303 which is schedule 80, with a design pressure of 1200 psig to match the main feedwater piping. The discharge lines of the auxiliary feedwater pumps connect to the main feedwater lines. The location and configuration of the auxiliary feedwater lines with respect to structures, equipment and other piping are shown in Figure 1.2-10. The criteria described above and as follows are applicable for both Units 1 and 2.

14 B. Pipe Whip Analysis

14 Isometrics of the Auxiliary Feedwater System lines indicating the location of the highest stress node points, postulated breakpoints and restraints are provided in Figures 3.6B-26 through 3.6B-33. The systems and equipment necessary to mitigate the consequences of a Auxiliary Feedwater System line break are described in
68 Section 3.6B.1. Breakpoints were postulated at the terminal ends of the piping run and at intermediate locations in accordance with the criteria outlined in Section 3.6B.2.1. Intermediate breaks are postulated as shown. A circumferential break is postulated to occur at
14 each of these points. Pipe whip restraints are provided as required to prevent impact on essential components. The pipe whip restraints are designed to prevent plastic hinge formation and thereby preclude adverse pipe whip effects.

40 C. Jet Impingement Analysis

40 The jet impingement analysis for this system is performed to determine the effects of jet impingement loading on essential components (as defined in Section 3.6B.1.2.1), associated supports and building structures.

40 In those cases where the analysis shows that the component or structure is not capable of withstanding the load then protection is required. Protection consists of either relocating the target, or installing jet shield or barriers to protect the target.

5. Steam Generator Blowdown Cleanup System

A. General Description

The Steam Generator Blowdown Cleanup System Piping is comprised of non-nuclear safety related, carbon steel ASME SA-106 Grade B material, or non-nuclear safety related stainless steel ASTM A-312 TP 304 material.

The blowdown lines from each steam generator up to the isolation valve moment restraints are 3" or 4" safety class 2, Schedule 80, category 1303 piping with a design pressure of 1200 psig. The piping connecting the 8-inch header of the steam generator blowdown heat exchanger to the pressure reducing valve PV-5180 is schedule 80, category 1302, Class 5, with a design pressure of 1200 psig. The portion of the blowdown piping on the Switchgear Building roof is category 1302G, and not seismically supported. The piping from valve PV-5180 up to valve SB-170 is 8-inch, category 302, Class 5, schedule 40, with a design pressure of 450 psig. The piping from valve SB-170 to the filters and demineralizers is 6-inch, category 301, Class 5, schedule 40S with a design pressure of 370 psig. The piping that connects the discharge of relief valve SB020 to the condenser is 4 and 8 inch, category 302, Class 5, Schedule 40, with a design pressure of 450 psig. The piping downstream of the filters is moderate energy piping. The location of the steam generator blowdown cleanup system with respect to structures and equipment are shown in Figures 1.2-31 and 1.2-35.

B. Pipe Whip Analysis

Isometrics of the Steam Generator Blowdown Cleanup System Piping indicating the locations of the highest stresses postulated breakpoints and restraints are provided in Figures 3.6B-38 through 3.6B-47. The systems and equipment necessary to mitigate the consequences of a Steam Generator Blowdown Cleanup System line break are described in Section
3.6B.1. Break locations were postulated at the terminal 68
ends of the piping run and at intermediate locations in accordance with the criteria outlined in Section 3.6B.2.1 except for category 1302G piping, where breaks are postulated at terminal ends and at each intermediate fitting, valve or welded attachment. A circumferential break is postulated to occur at each of these points. Pipe whip restraints are provided as required to prevent impact on essential components. The pipe whip restraints are designed to prevent plastic hinge formation and thereby preclude adverse pipe whip effects.

C. Jet Impingement Analysis

The jet impingement analysis for this system is performed 40
to determine the effects of jet impingement loading on essential components (as defined in Section 3.6B.1.2.1), associated supports and building structures.

In those cases where the analysis shows that the component 40
or structure is not capable of withstanding the load then protection is required. Protection consists of either relocating the target, or installing jet shield or barriers to protect the target.

REFERENCES

1. NRC Regulatory Guide 1.46, Protection Against Pipe Whip Inside Containment, U.S. Nuclear Regulatory Commission, 5/73.

2. Acceptance Criteria of NRC Standard Review Plan, Section 3.6.1, Plant Design For Protection Against Postulated Piping Failures in Fluid Systems Outside Containment and Section 3.6.2, Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping. Also Branch Technical Position MEB 3-1, Postulated Breaks and Leakage Locations in Fluid System Piping Outside Containment.

3. Branch Technical Position APCSB 3-1, Protection Against Postulated Piping Failures in Fluid Systems Outside Containment.

4. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, 1974 with Addenda.

5. WCAP-8082-P-A (Proprietary), January 1975, and WCAP 8172 A (Nonproprietary), January 1975, Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop.

6. NUREG/CR-1826, RELAP-5/MOD 1, Rev 2 Sept. 1981. | 68

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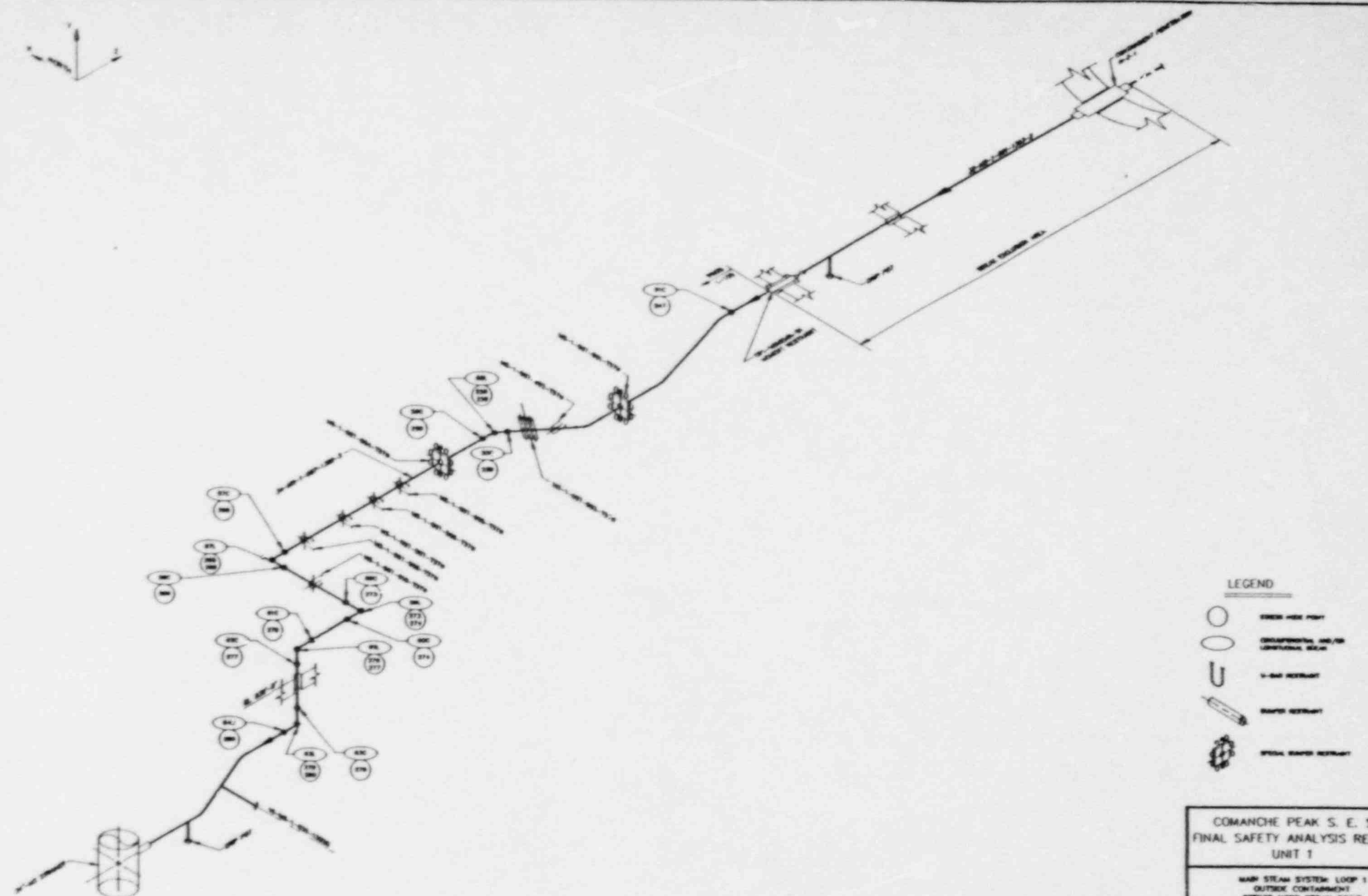
22. Design Basis for Protection of Light Water Nuclear Power Plants Against Effects of Pipe Rupture, American National Standard ANSI/ANS 58.2 (Working draft 08/02/84). | 68

23. Federal register 12502 Vol. 51, No. 70, April 11, 1986 "Modification of General Design Criterion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures". | 68

24. EBASCO Services Inc., "SHPLAST 2267, Dynamic Pipe Whip Analysis" version 0 Rev. 27, June 1983. | 68



COMANCHE PEAK S. E. S.
FINAL SAFETY ANALYSIS REPORT
UNIT 1

MAIN STEAM SYSTEM: LOOP 1
OUTSIDE CONTAINMENT
STRESS NODE BREAK POINT
AND RESTRAINT LOCATIONS

FIGURE 3.6B-15 PROOF 1-100

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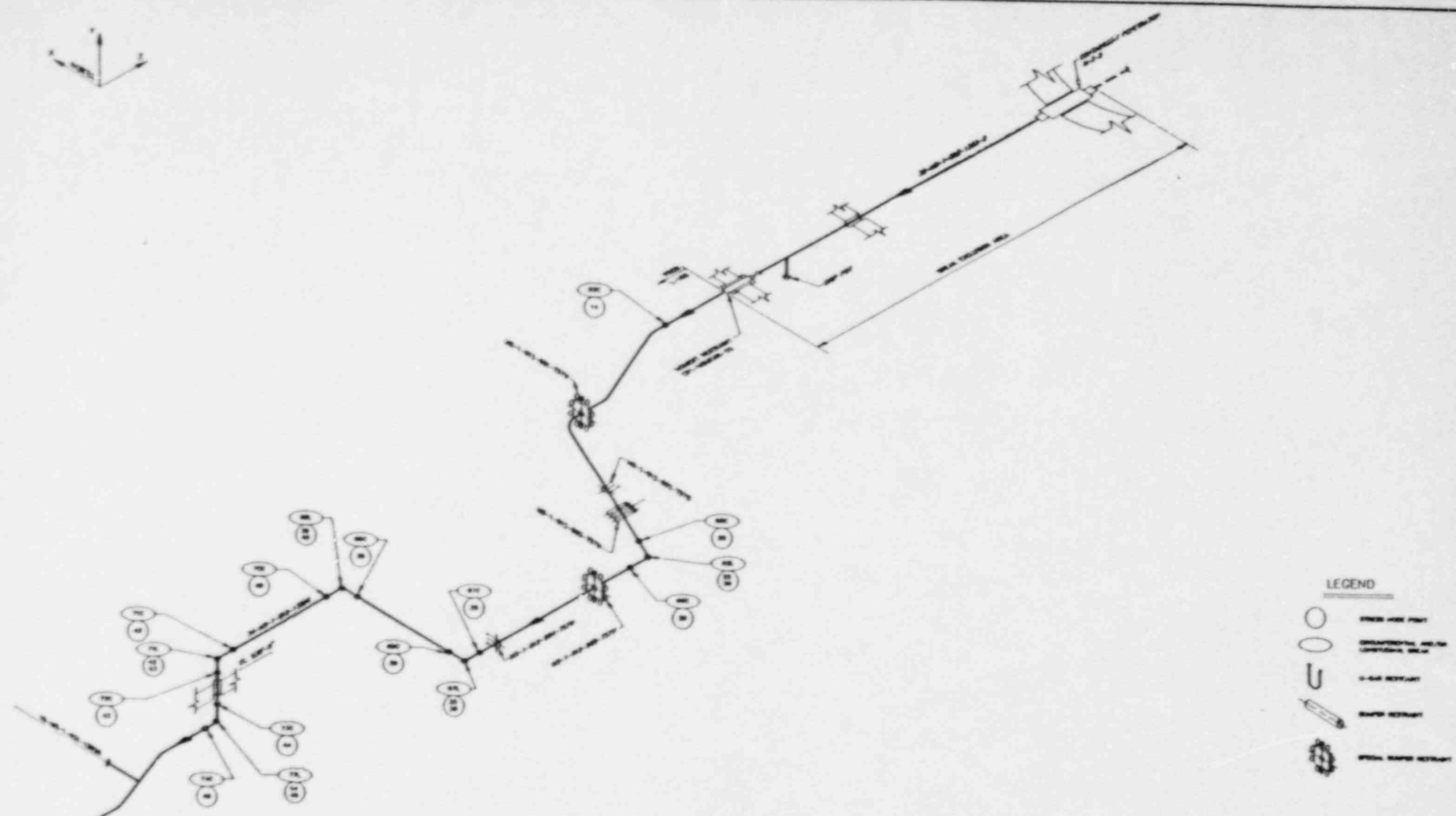

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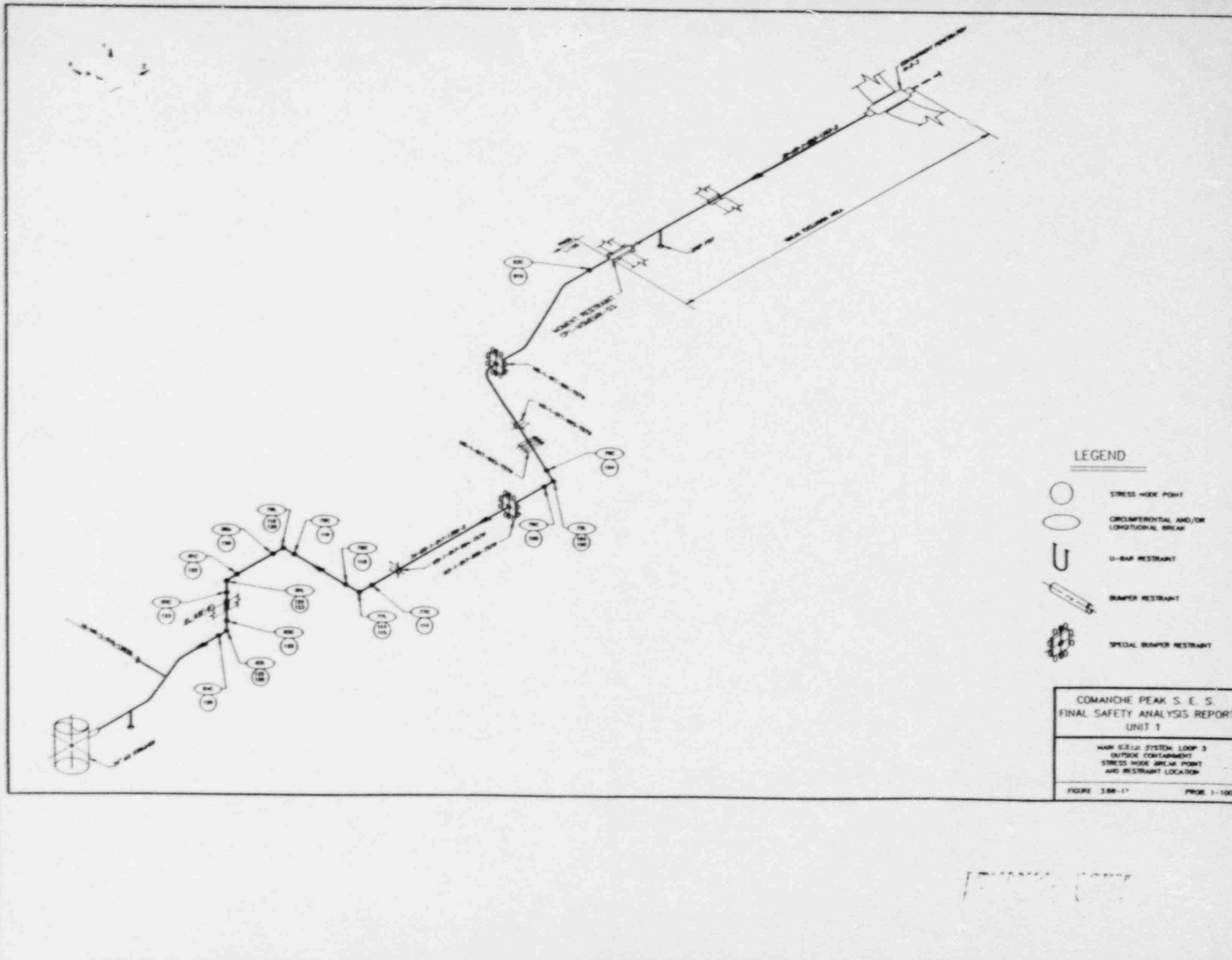
FINAL SAFETY ANALYSIS REPORT
UNIT 1

MAIN STEAM SYSTEM LOOP 2
OUTSIDE CONTAINMENT
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-16 PROB. 1-100



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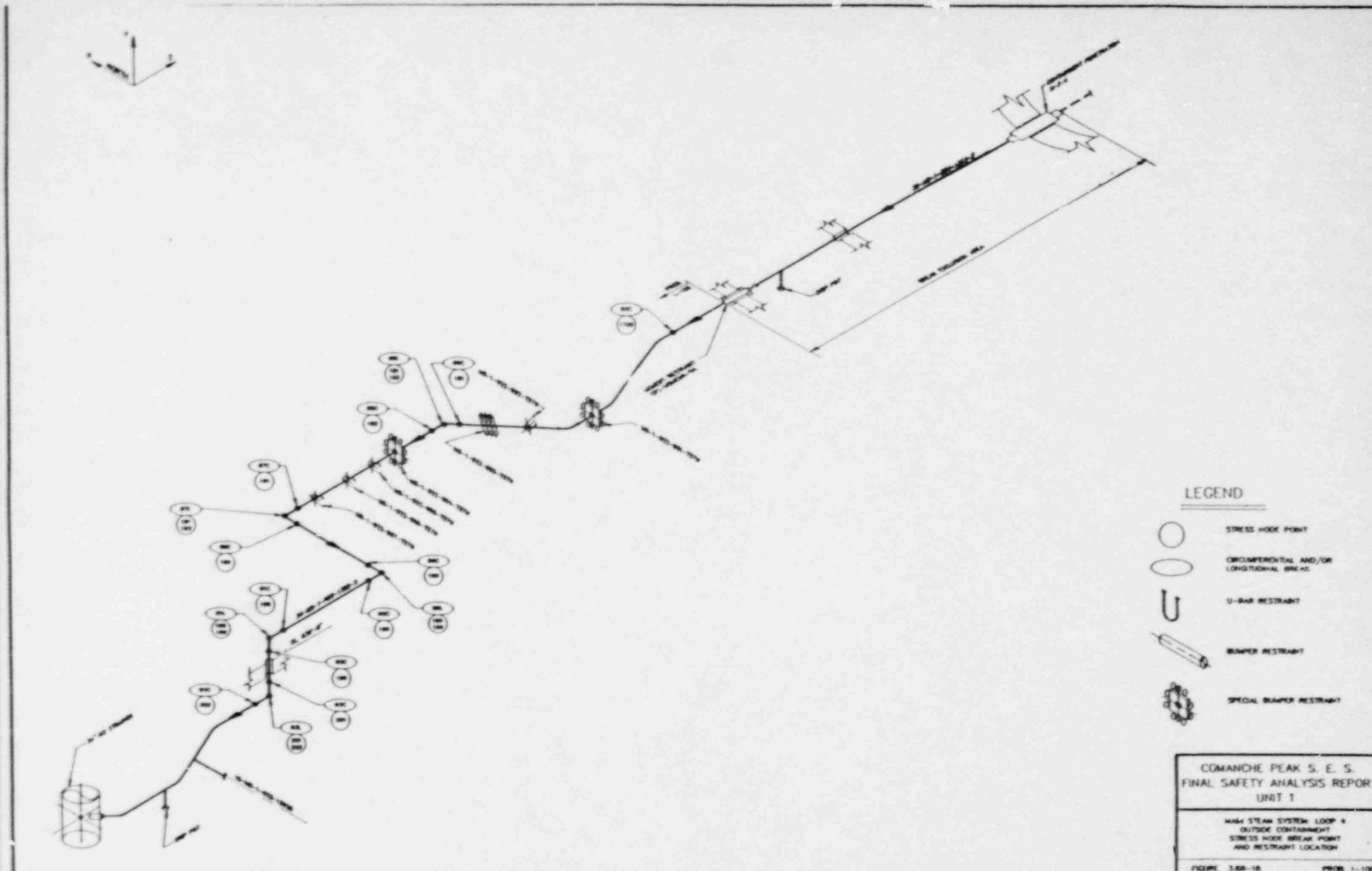
FINAL SAFETY ANALYSIS REPORT
UNIT 1

MAIN STEAM SYSTEM LOOP 3
OUTSIDE CONTAINMENT
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-17 PROB. 1-100



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FINAL SAFETY ANALYSIS REPORT
UNIT 1

MAIN STEAM SYSTEM: LOOP 4
OUTSIDE CONTAINMENT
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-18 PROB. 1-100

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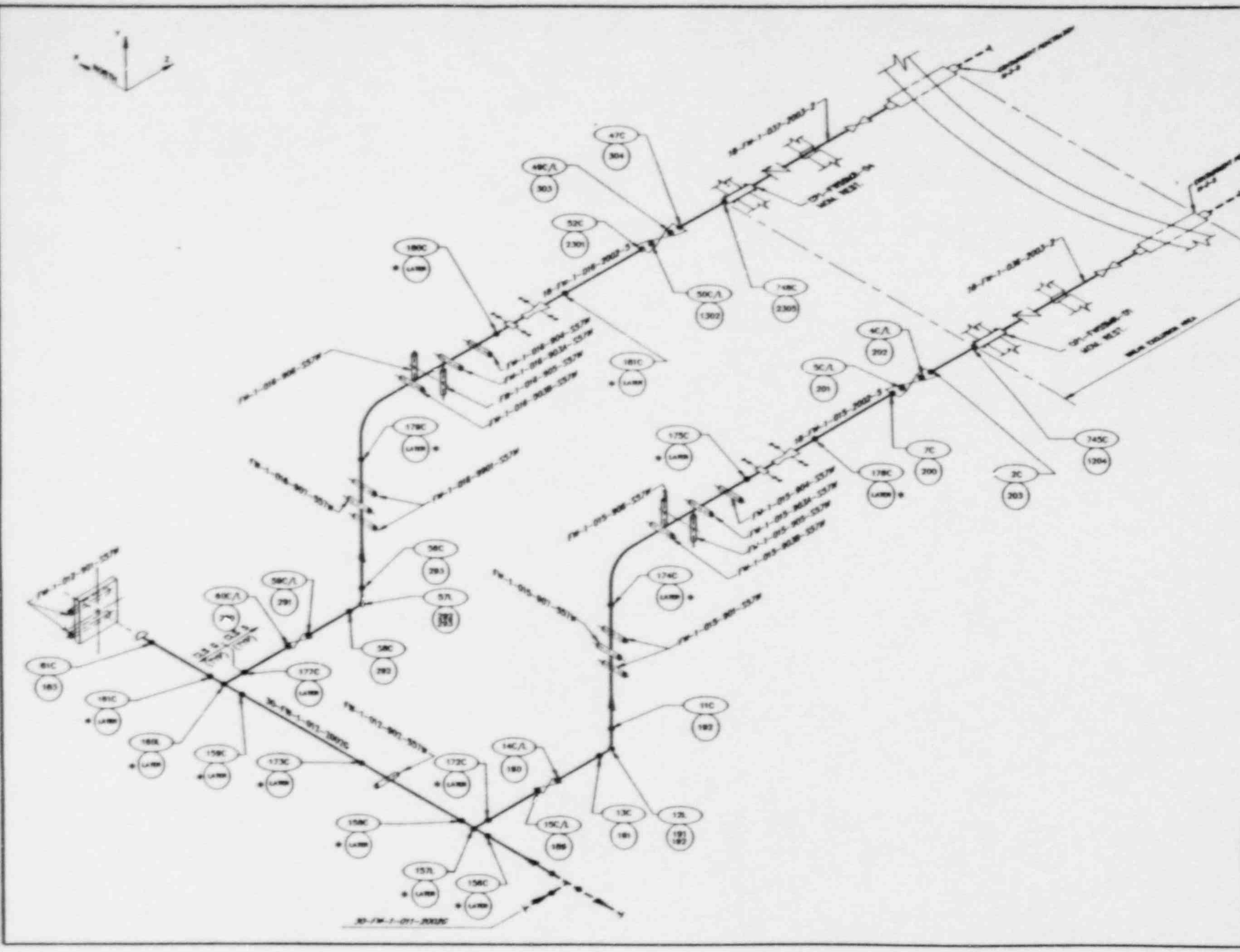

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FINAL SAFETY ANALYSIS REPORT
UNIT 1

FEEDWATER SYSTEM
OUTSIDE CONTAINMENT
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.5B-23-1 PROB. 1-104

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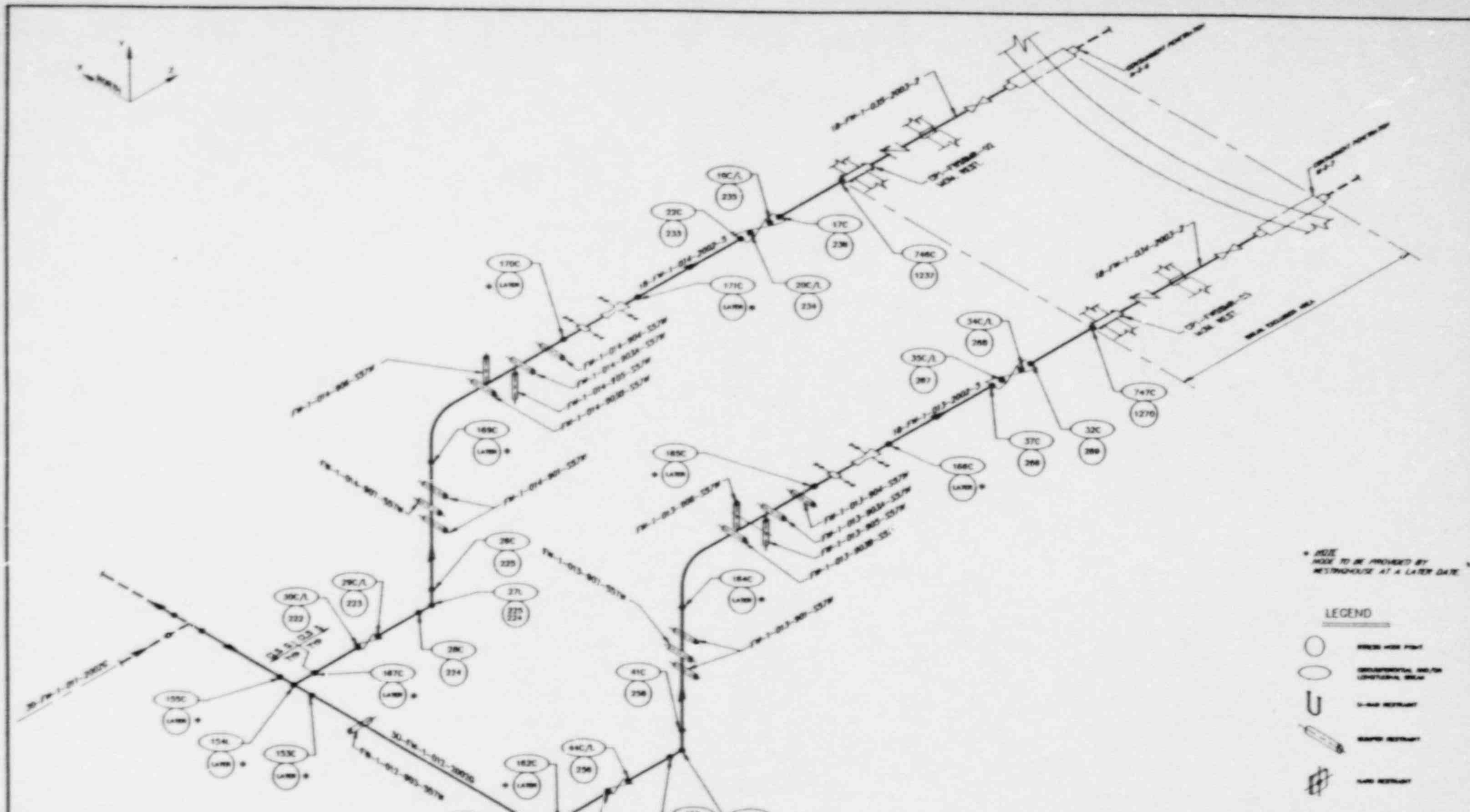

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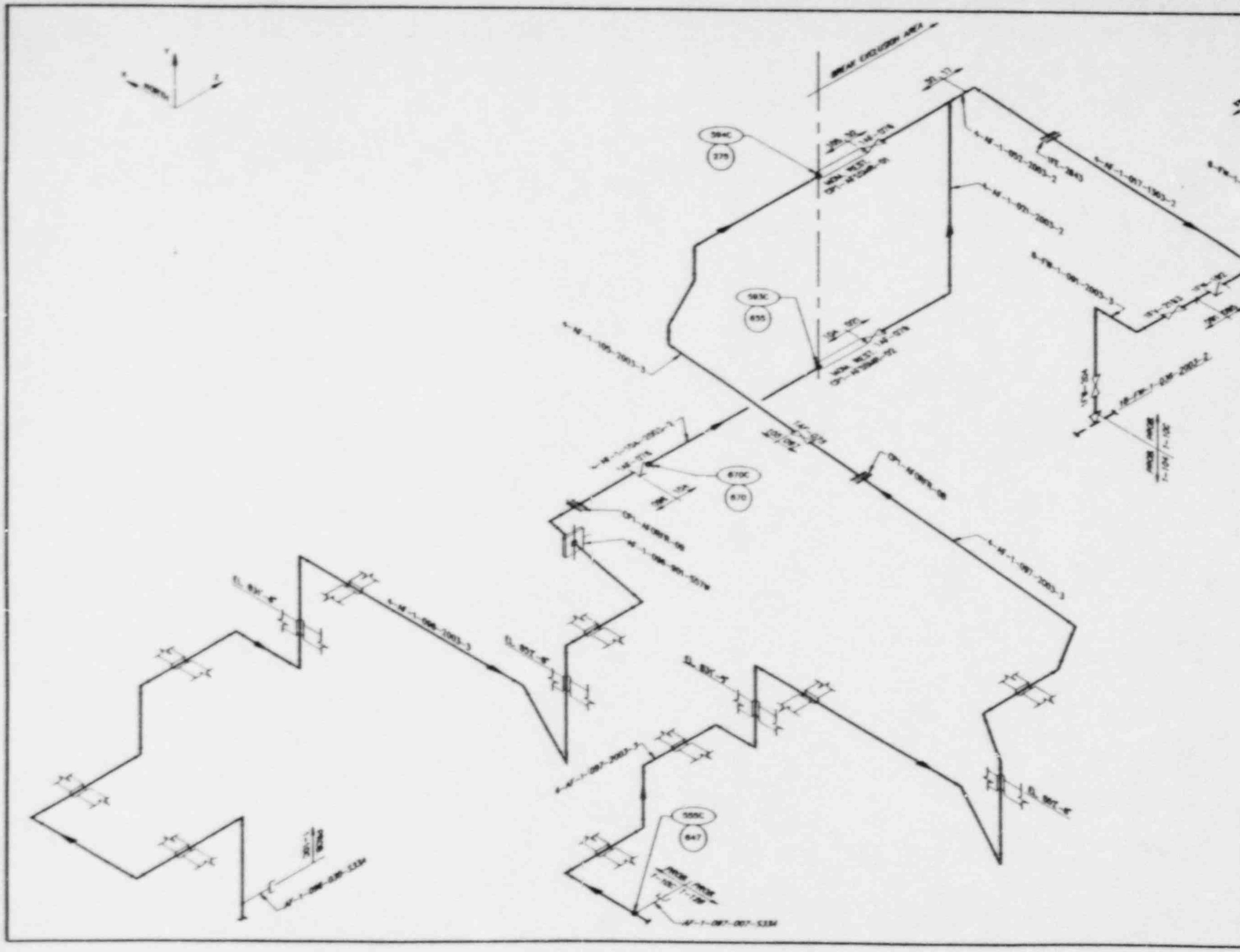
FINAL SAFETY ANALYSIS REPORT
UNIT 1

FEEDWATER SYSTEM
OUTSIDE CONTAINMENT
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-25-2 PROB 1-104



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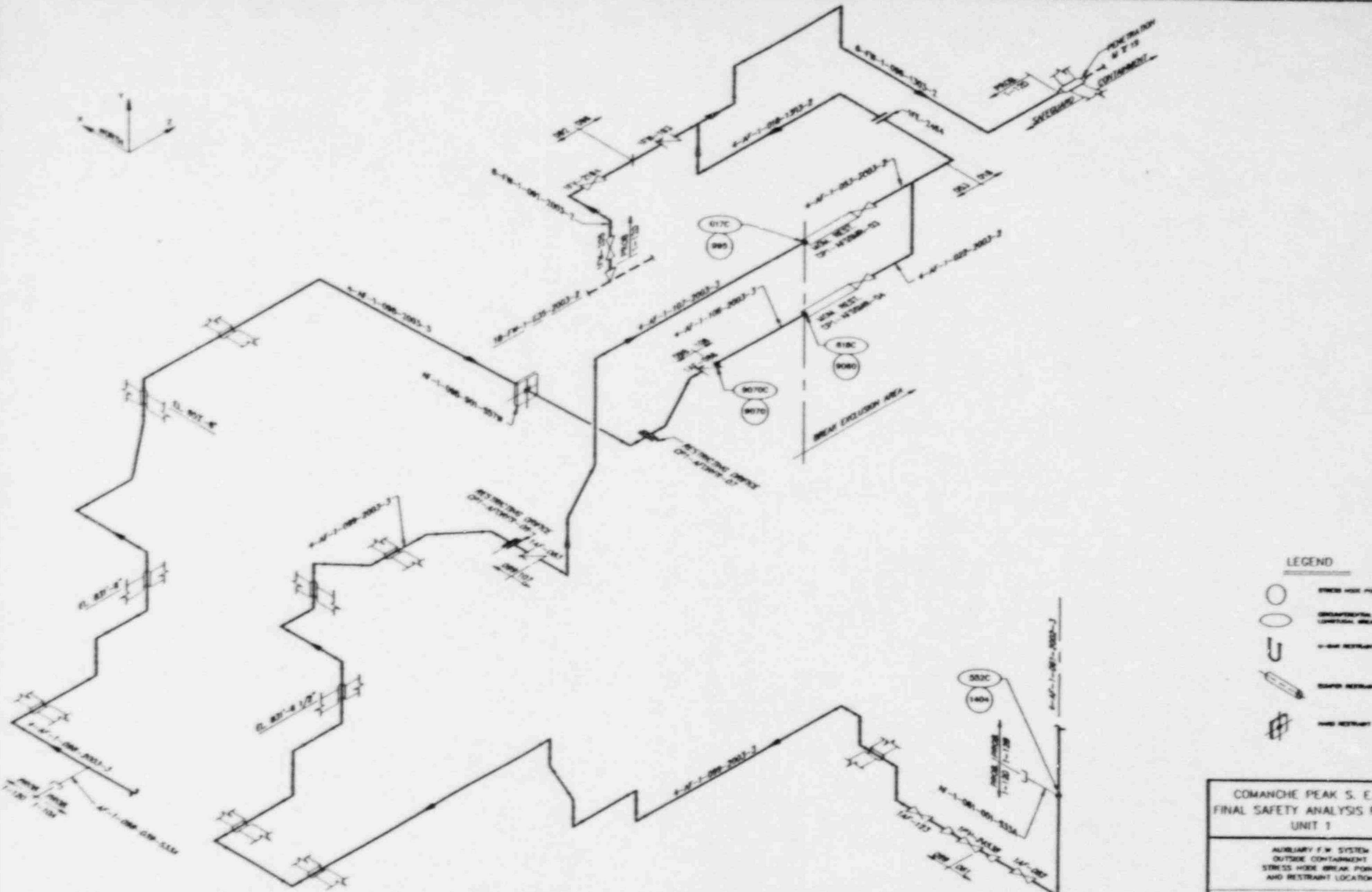
FINAL SAFETY ANALYSIS REPORT
UNIT 1

AUXILIARY F.W. SYSTEM LOOP 1
OUTSIDE CONTAINMENT
STRESS MODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-27 PROB. 1-10C



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UNIT 1

AUXILIARY F.W. SYSTEM
OUTSIDE CONTAINMENT
STRESS MODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-28 PROB 1-120

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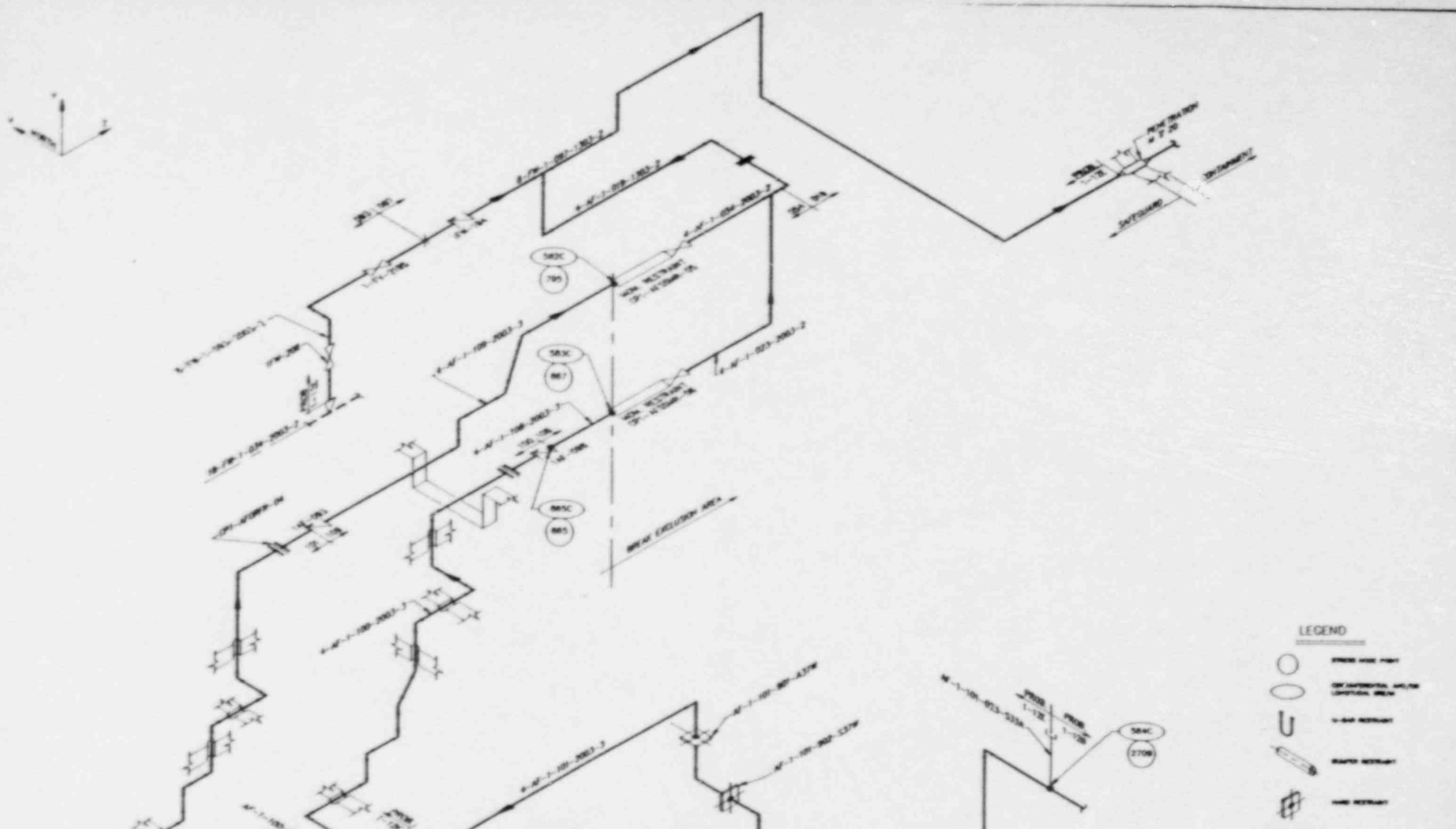

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FINAL SAFETY ANALYSIS REPORT
UNIT 1

AUXILIARY F.W. SYSTEM LOOP 3
OUTSIDE CONTAINMENT
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-29

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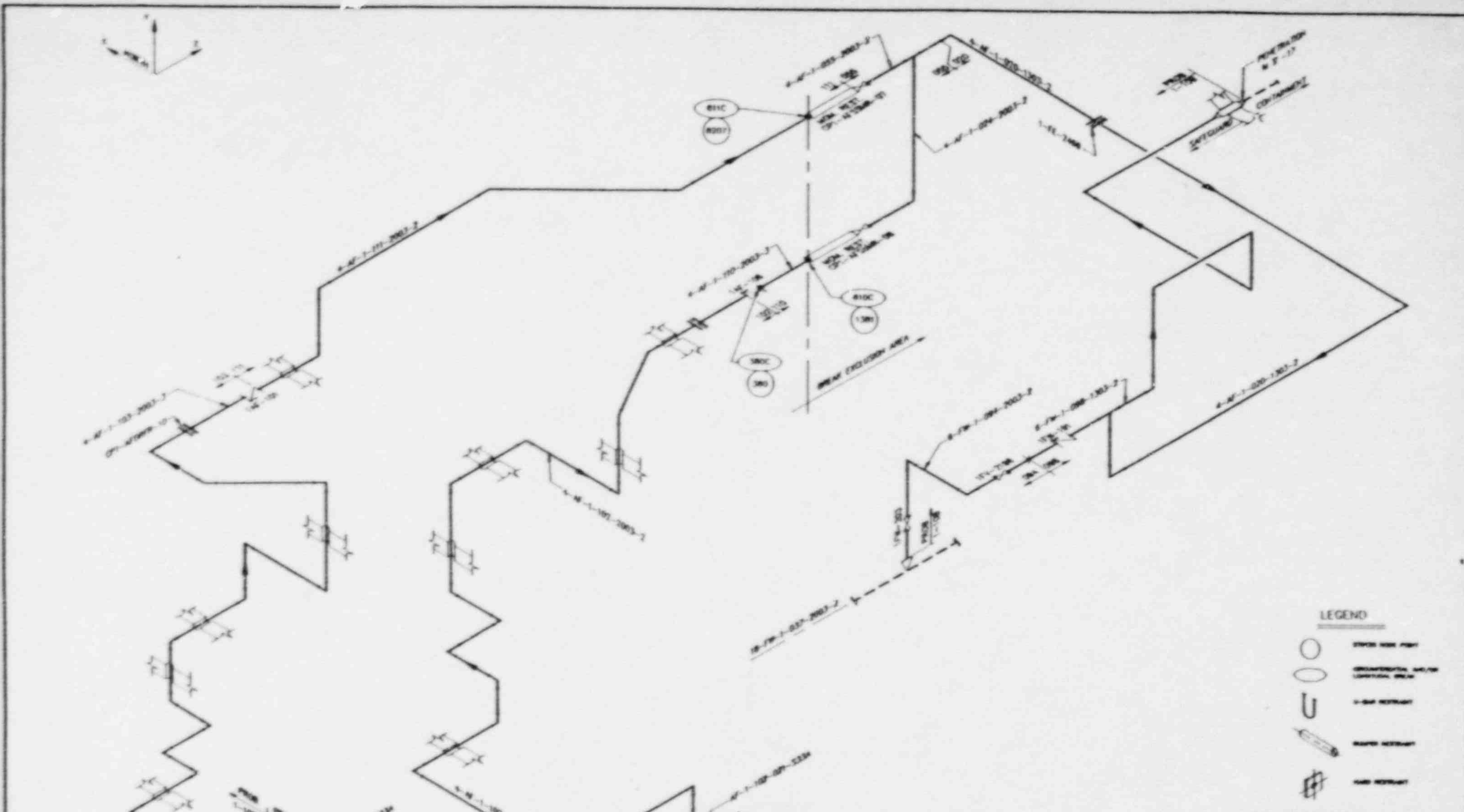

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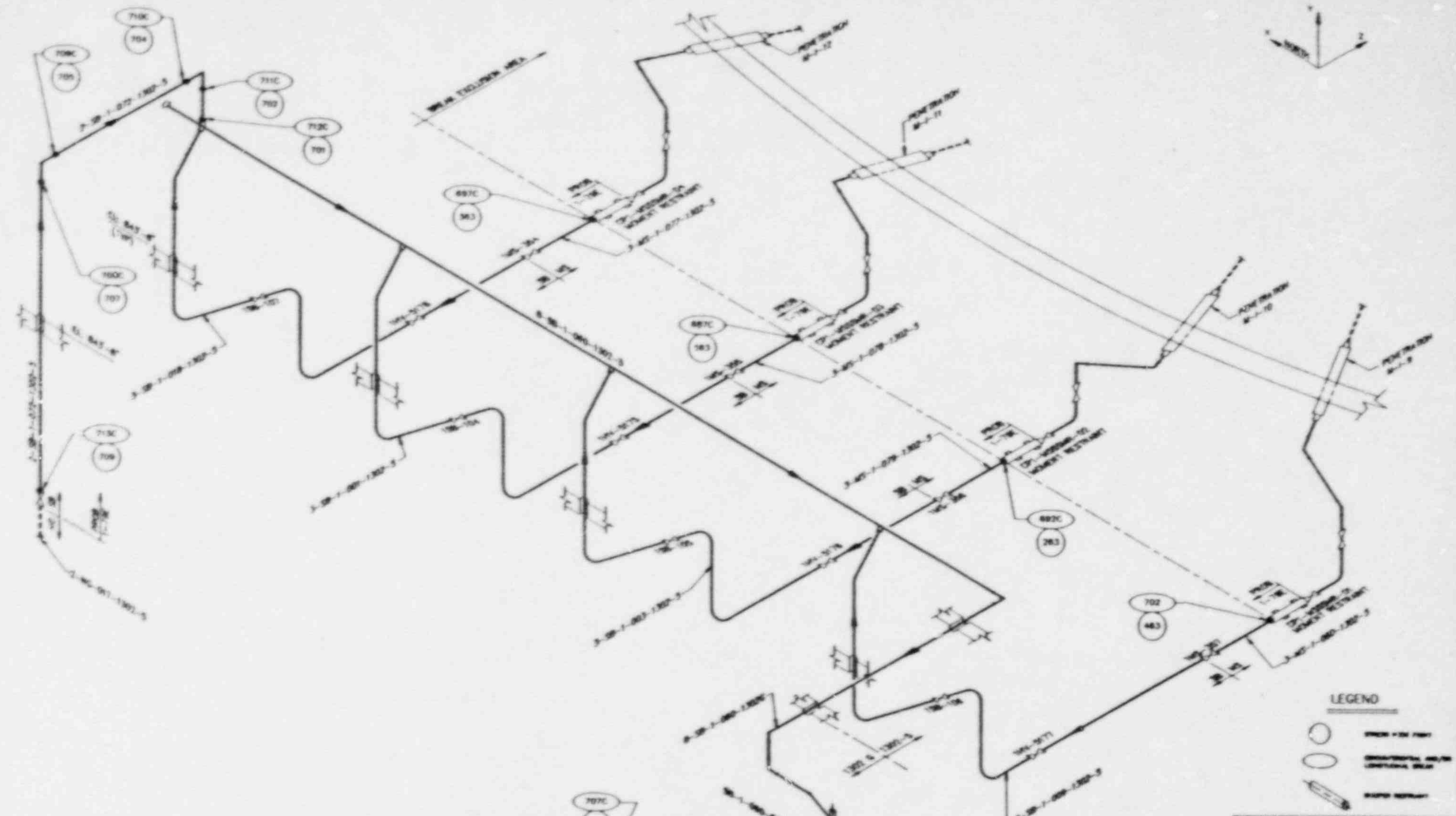
FINAL SAFETY ANALYSIS REPORT
UNIT 1

AUXILIARY F.W. SYSTEM
OUTSIDE CONTAINMENT
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-30 PROB 1-108



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FINAL SAFETY ANALYSIS REPORT
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STEAM GENERATOR BLOWDOWN SYSTEM
OUTSIDE CONTAINMENT
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-42 PROB 1-79E

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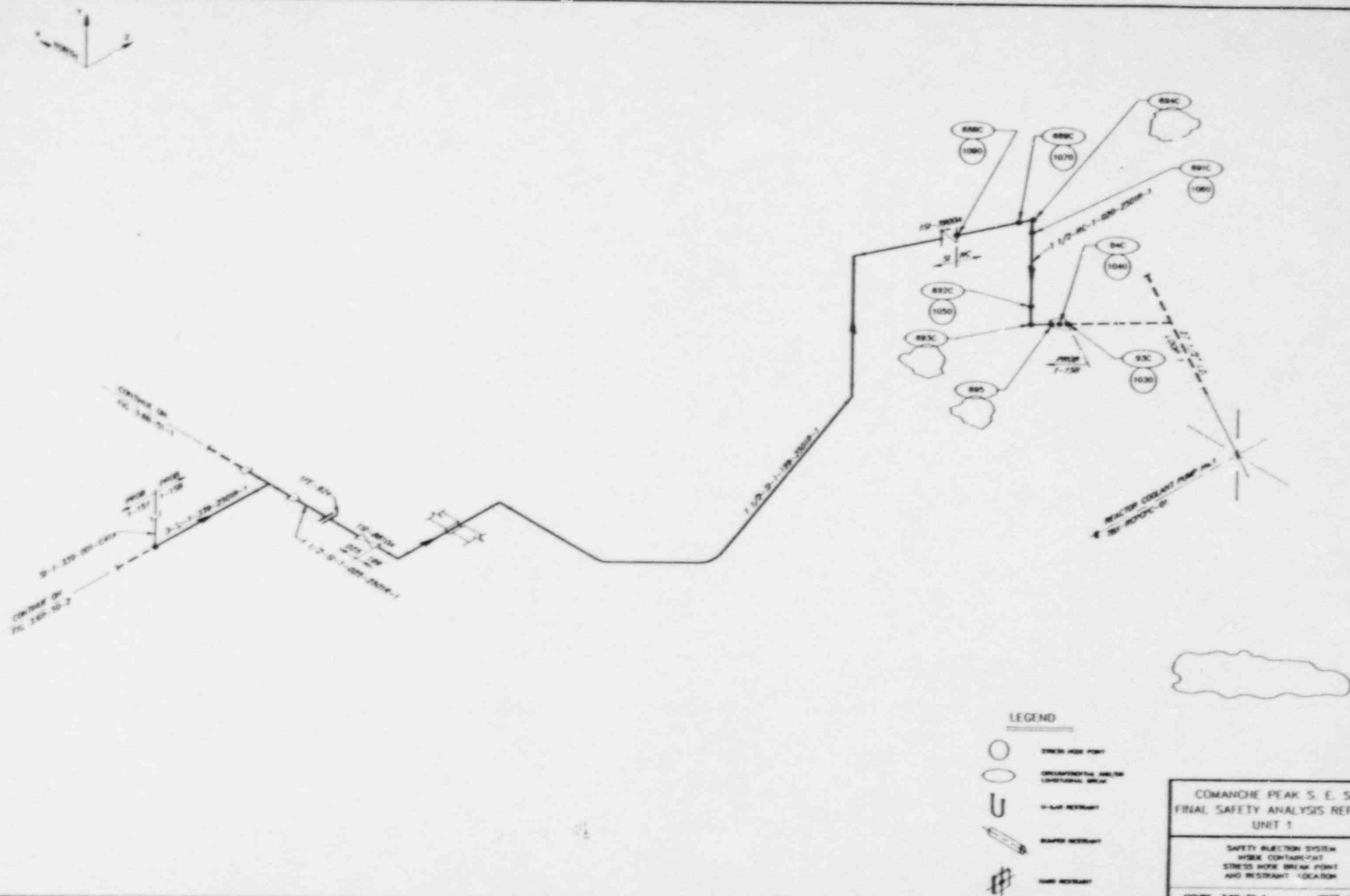

COMANCHE PEAK S. E. S.
FINAL SAFETY ANALYSIS REPORT
UNIT 1

SAFETY INJECTION SYSTEM
INSIDE CONTAINMENT
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-50-1 PROB. 1-156

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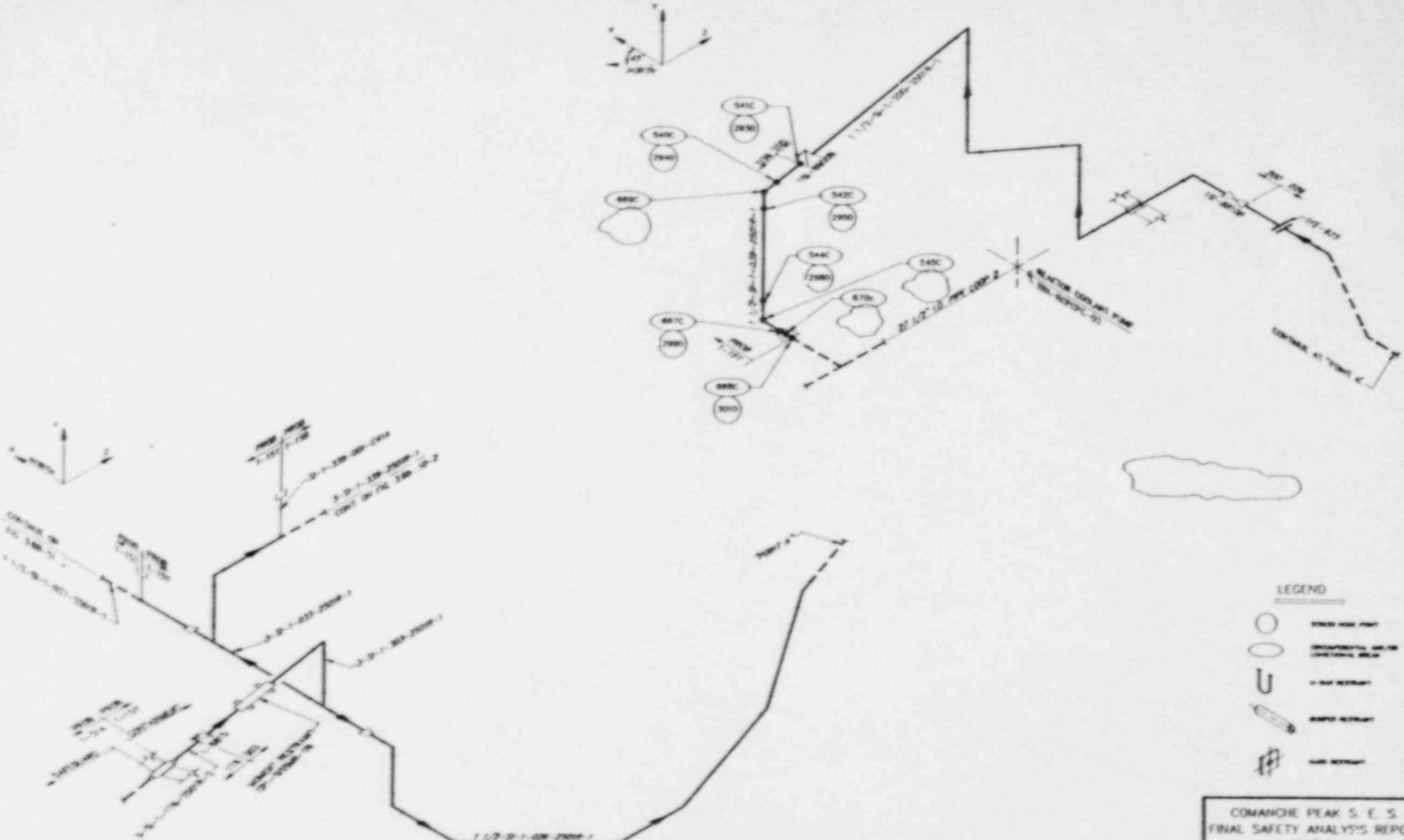

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FINAL SAFETY ANALYSIS REPORT
UNIT 1

SAFETY INJECTION SYSTEM
INSIDE CONTAINMENT
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-50-2 PROB. 1-107

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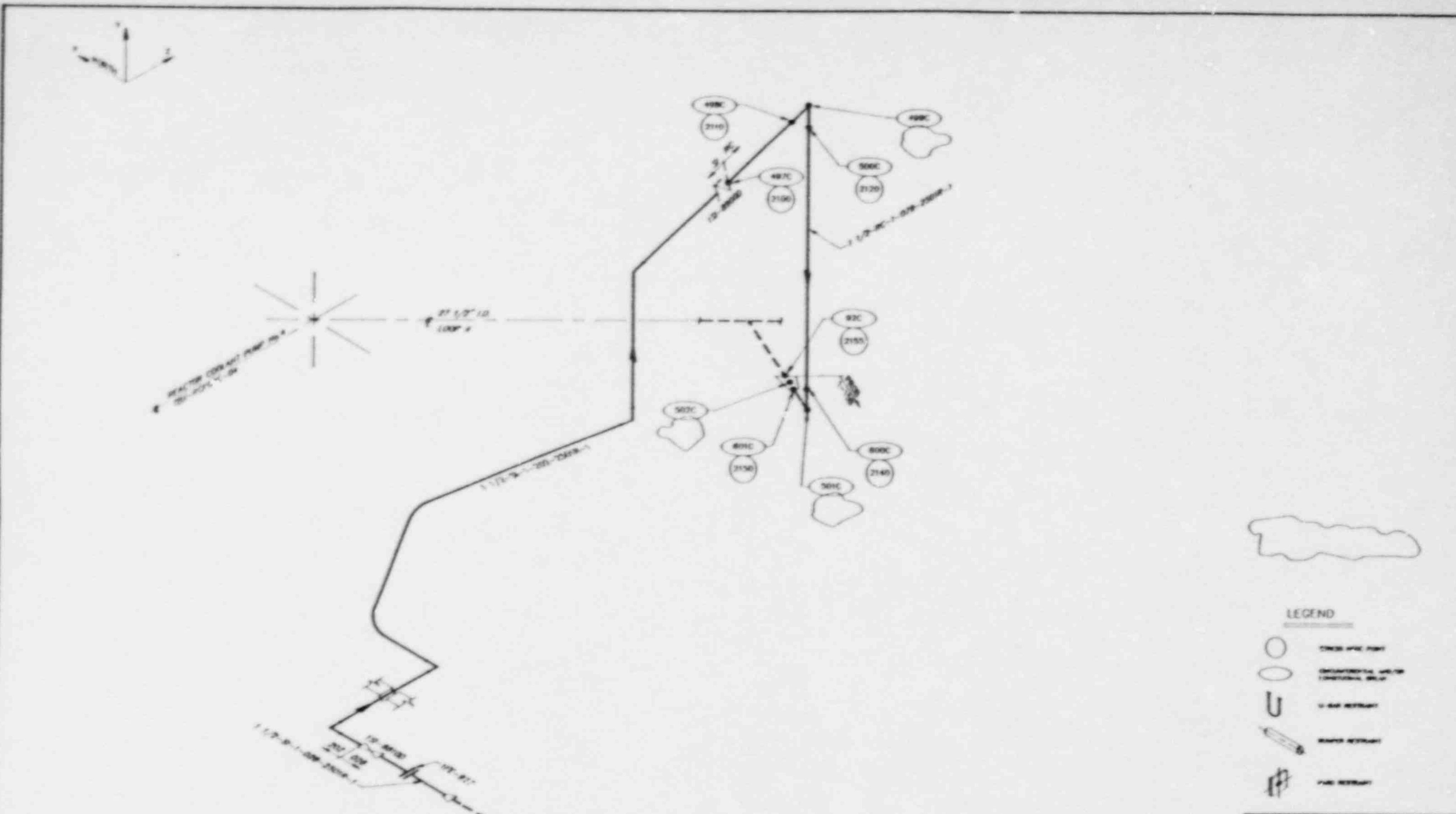

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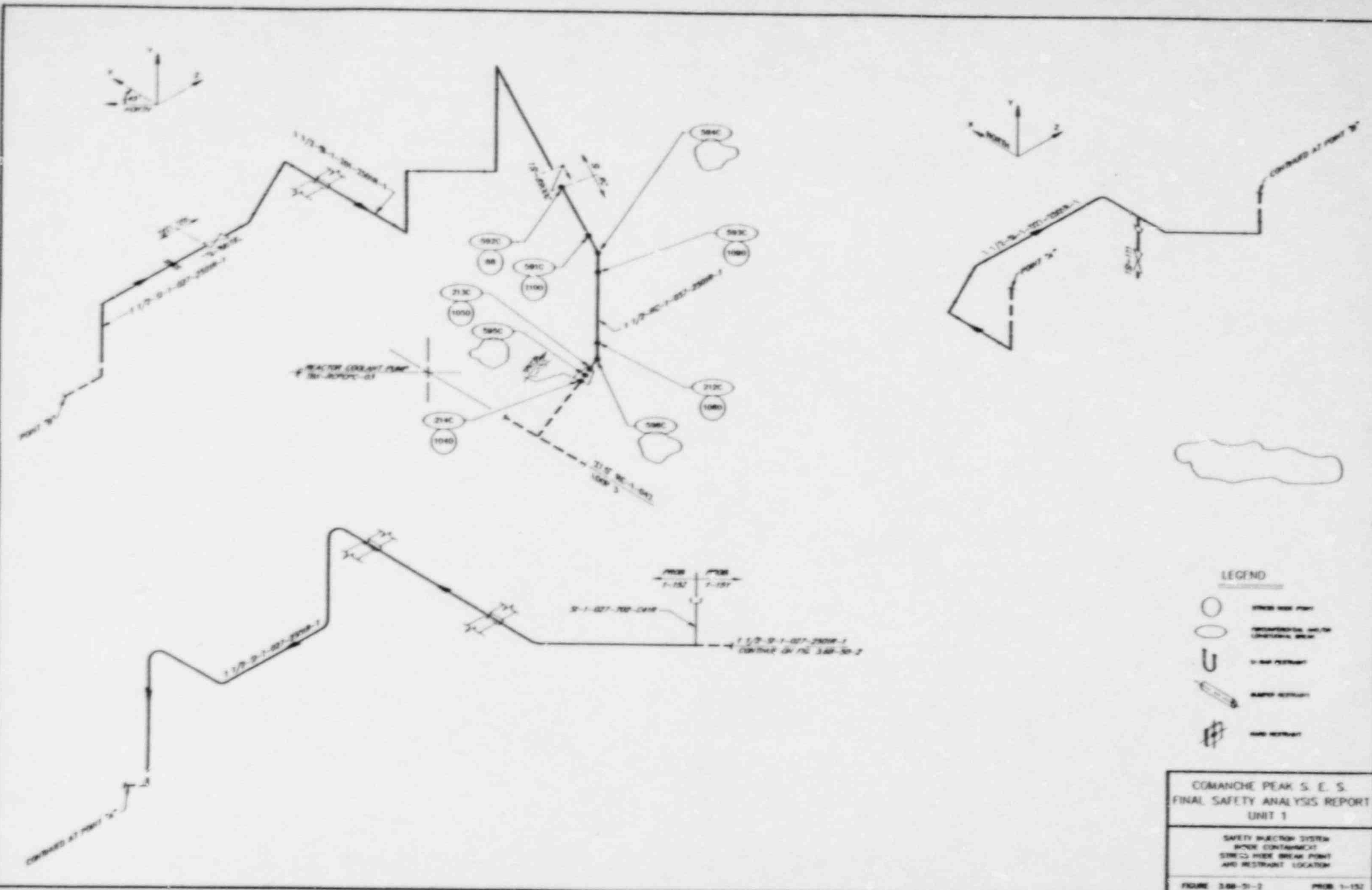
FINAL SAFETY ANALYSIS REPORT
UNIT 1

SAFETY INJECTION SYSTEM
INSIDE CONTAINMENT
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-51-1 PROB. 1-150


LEGEND
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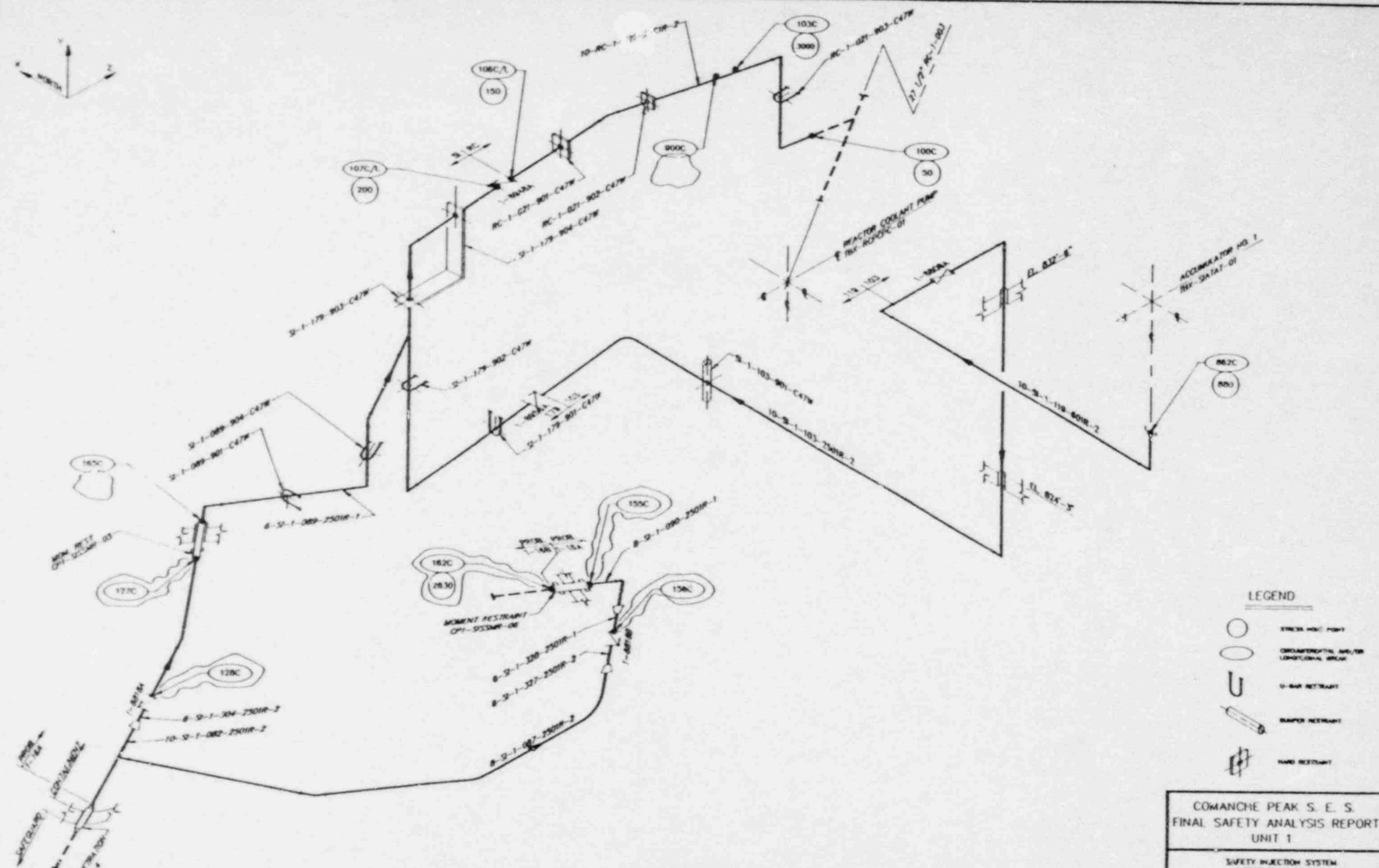
FINAL SAFETY ANALYSIS REPORT
UNIT 1
SAFETY INJECTION SYSTEM
INSIDE CONTAINMENT
AND RESTRAINT LOCATION
584C
592C
591C
593C
213C
595C
212C
214C
596C
1050
1040
1080
1090
1100
REACTOR COOLANT PUMP



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UNIT 1

SAFETY INJECTION SYSTEM
INSIDE CONTAINMENT
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-52 PROB 1-16A

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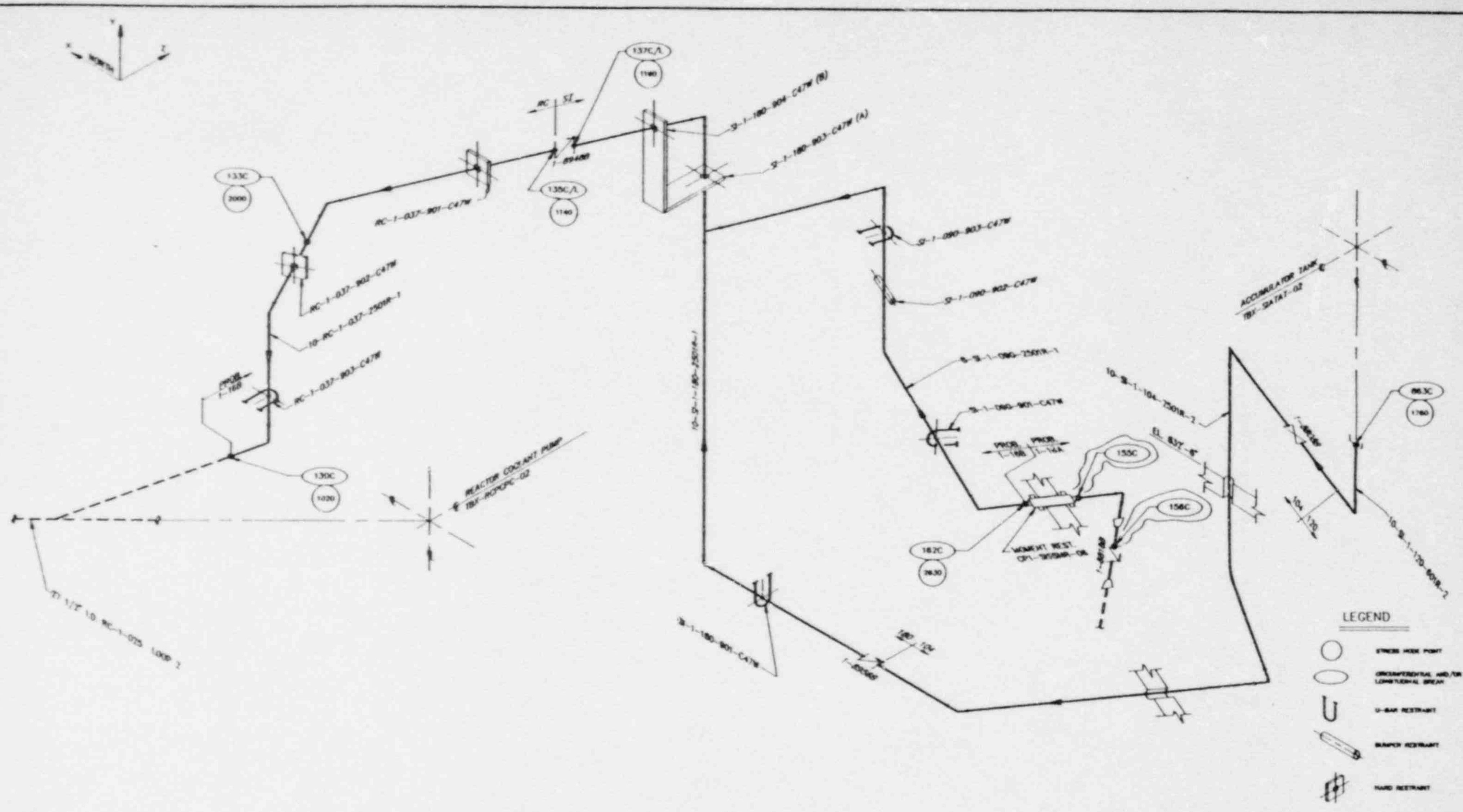

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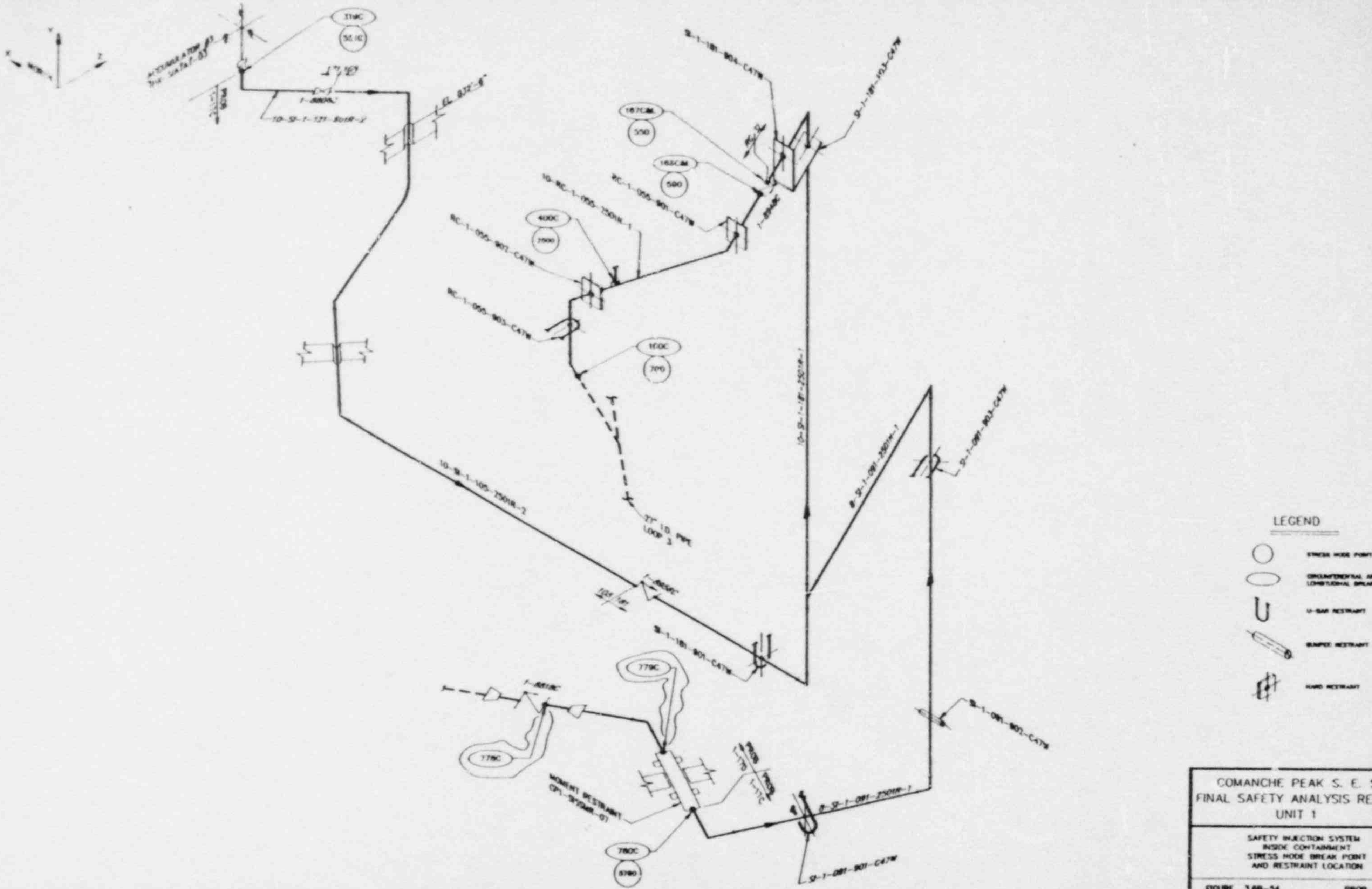
FINAL SAFETY ANALYSIS REPORT
UNIT 1

SAFETY INJECTION SYSTEM
INSIDE CONTAINMENT
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-53 PROB. 1-16B



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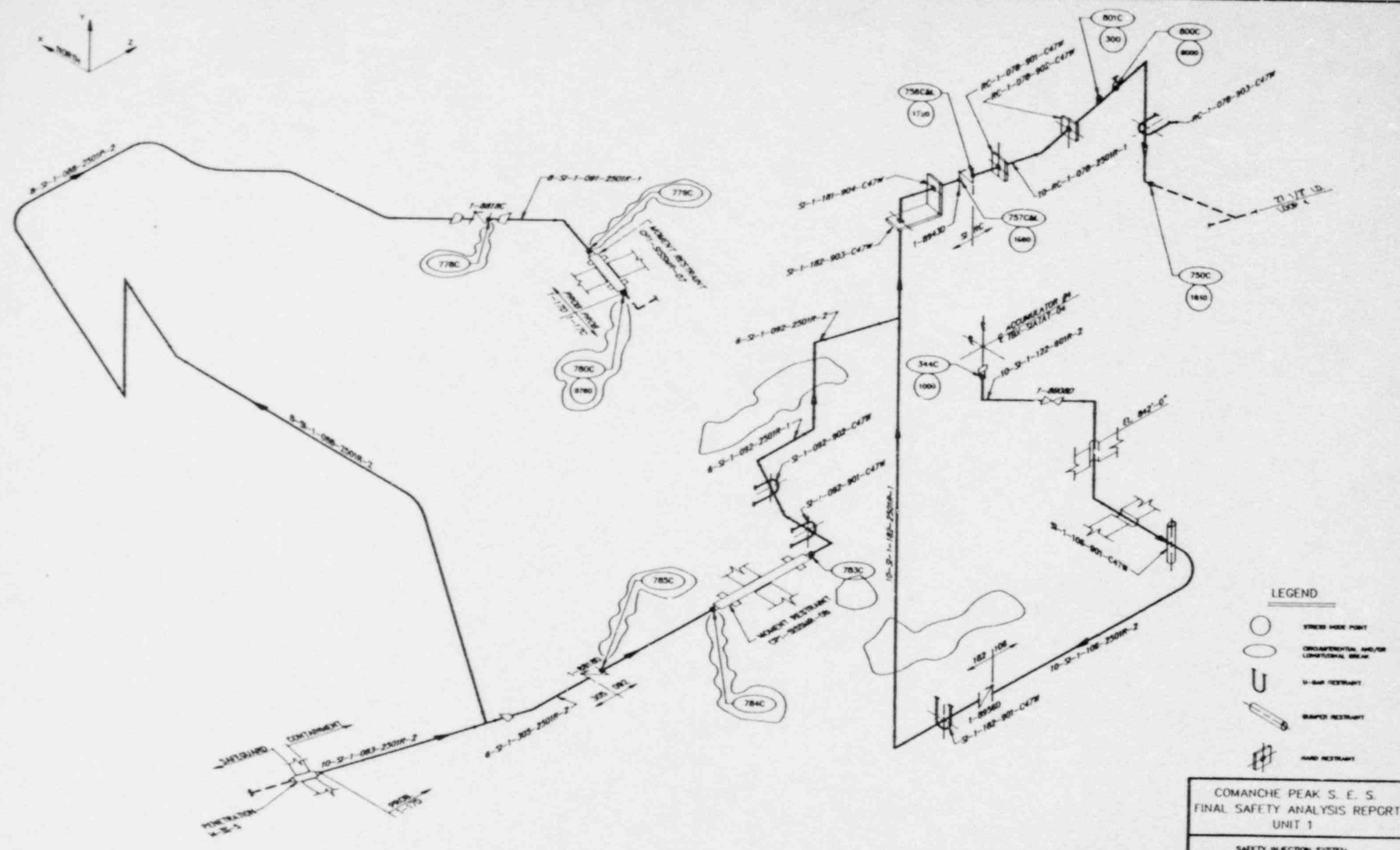
UNIT 1

SAFETY INJECTION SYSTEM
INSIDE CONTAINMENT
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-54 PROB. 1-17C



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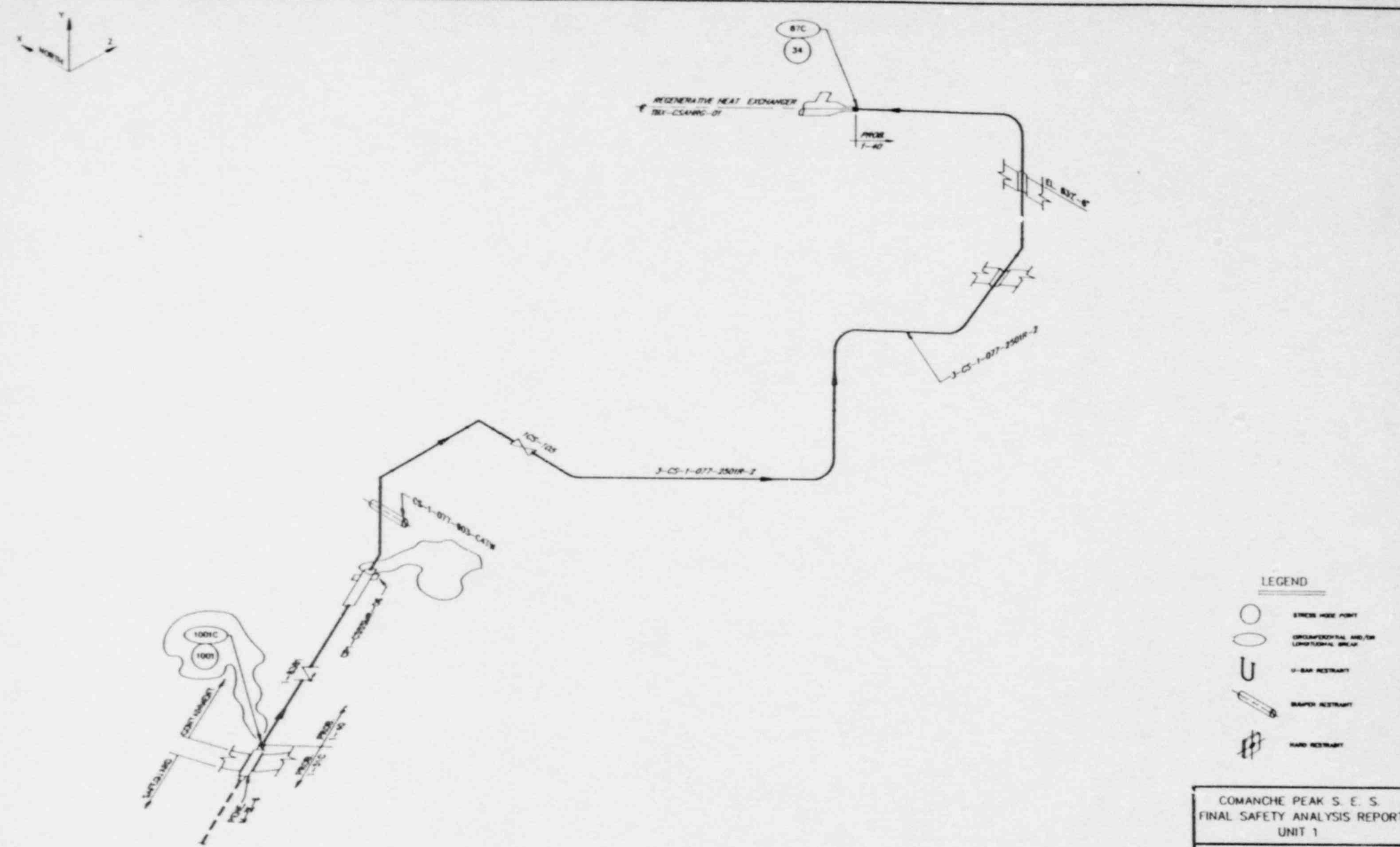
FINAL SAFETY ANALYSIS REPORT
UNIT 1

SAFETY INJECTION SYSTEM
INSIDE CONTAINMENT
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-55 PROB. 1-17D



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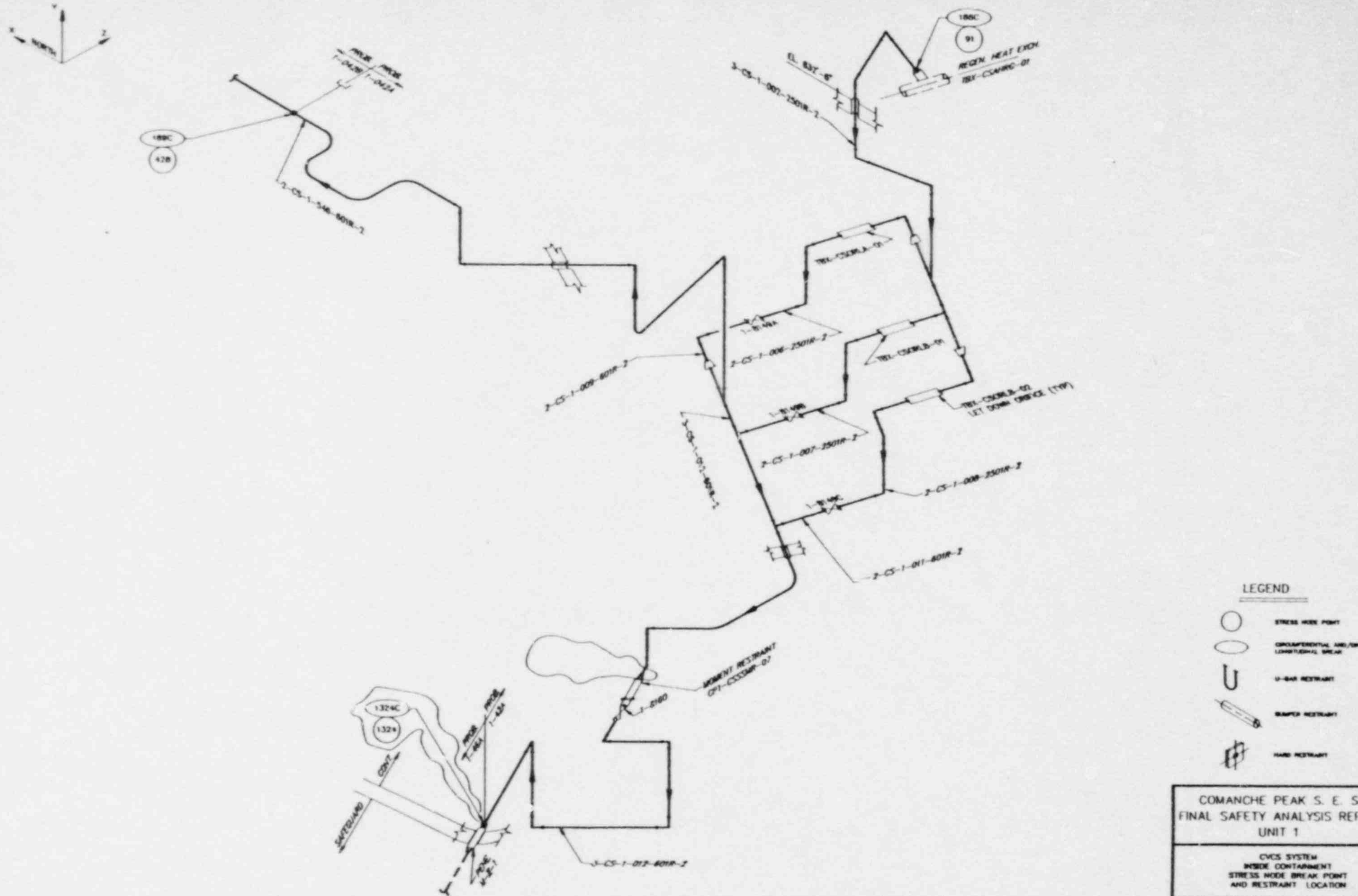
UNIT 1

CVCS SYSTEM
INSIDE CONTAINMENT
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-70 PROB. 1-40



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FINAL SAFETY ANALYSIS REPORT
UNIT 1

CVCS SYSTEM
INSIDE CONTAINMENT
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-72 PROB. 1-42A

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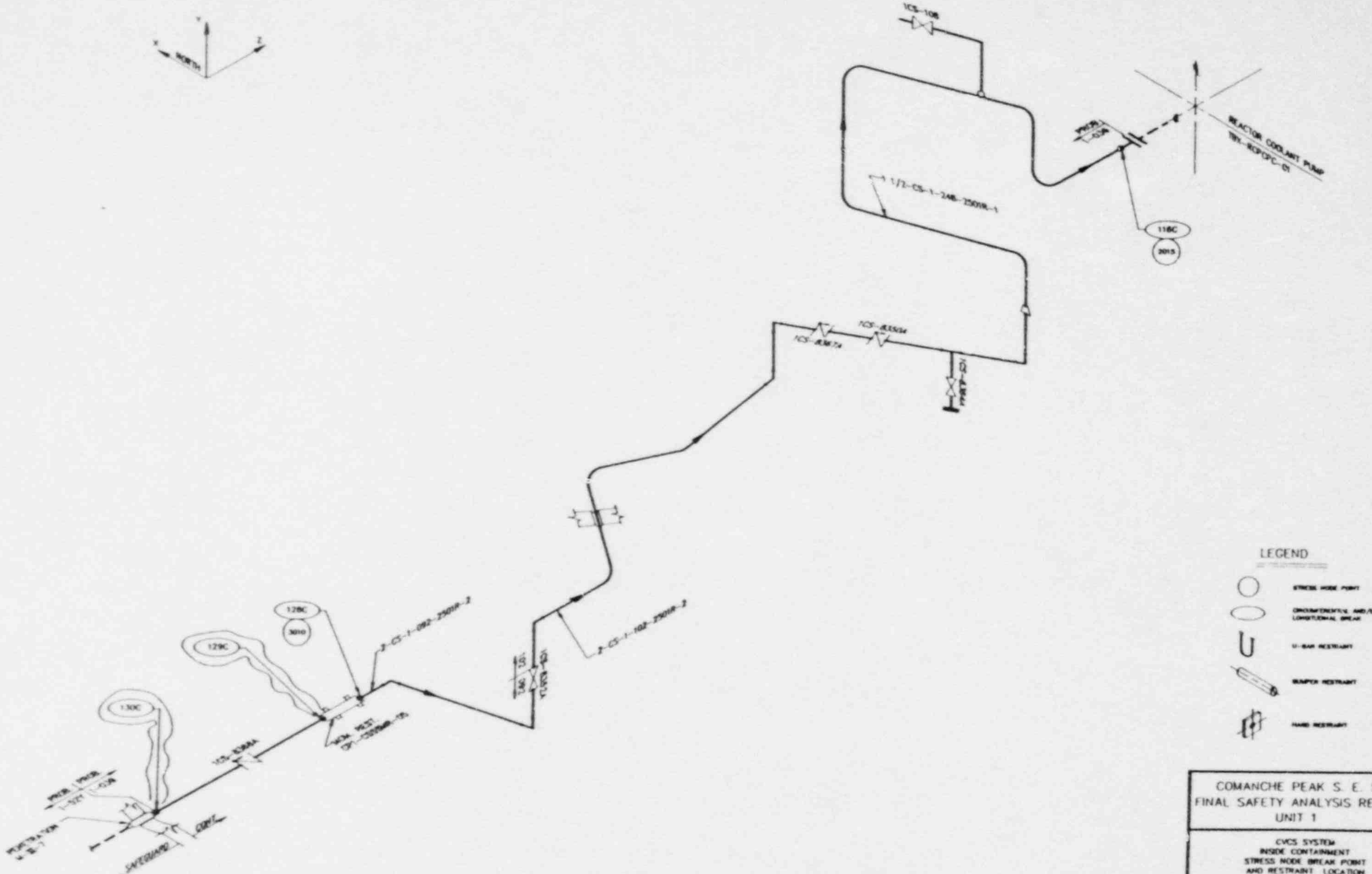

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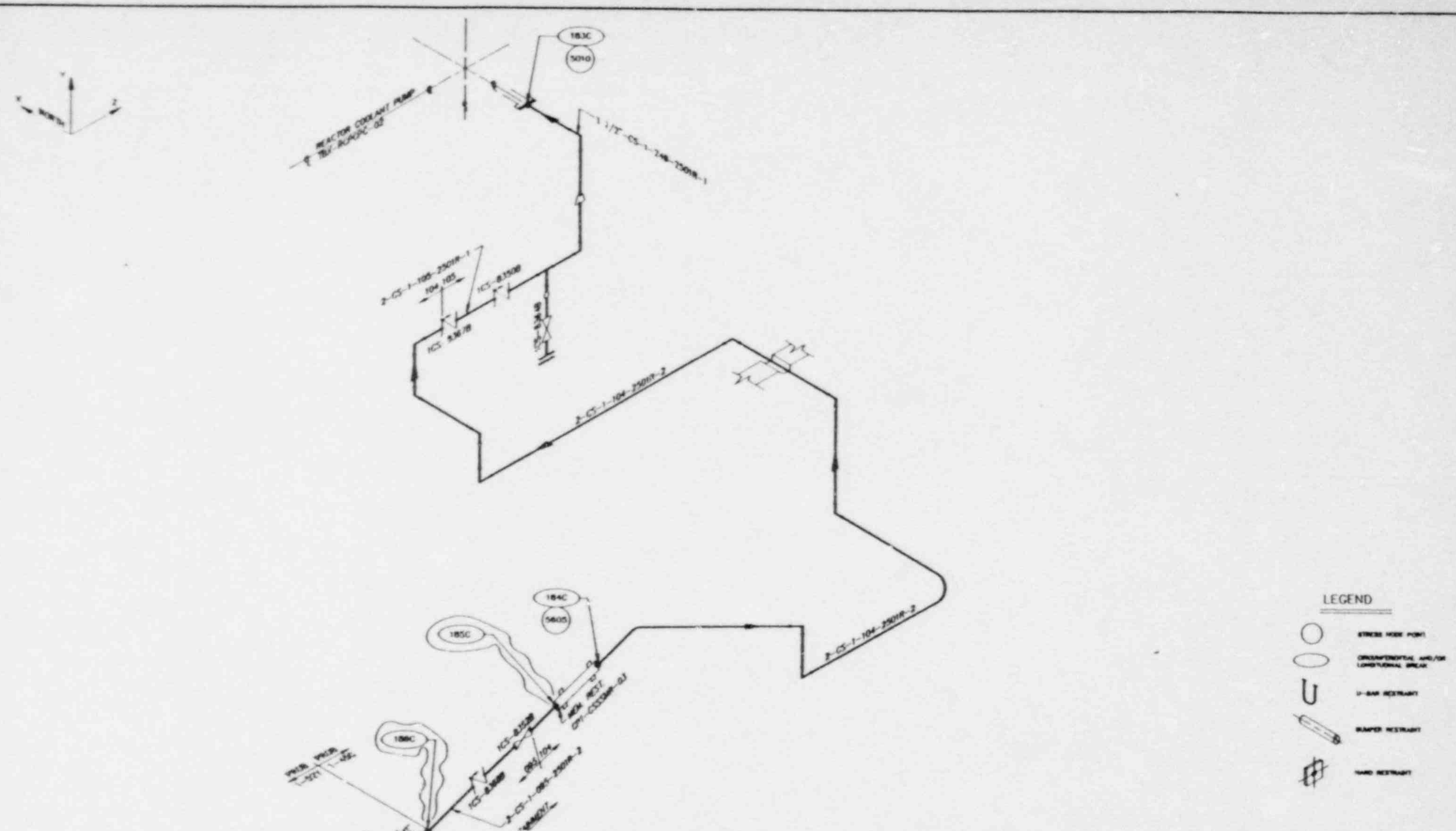
UNIT 1

CVCS SYSTEM
INSIDE CONTAINMENT
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

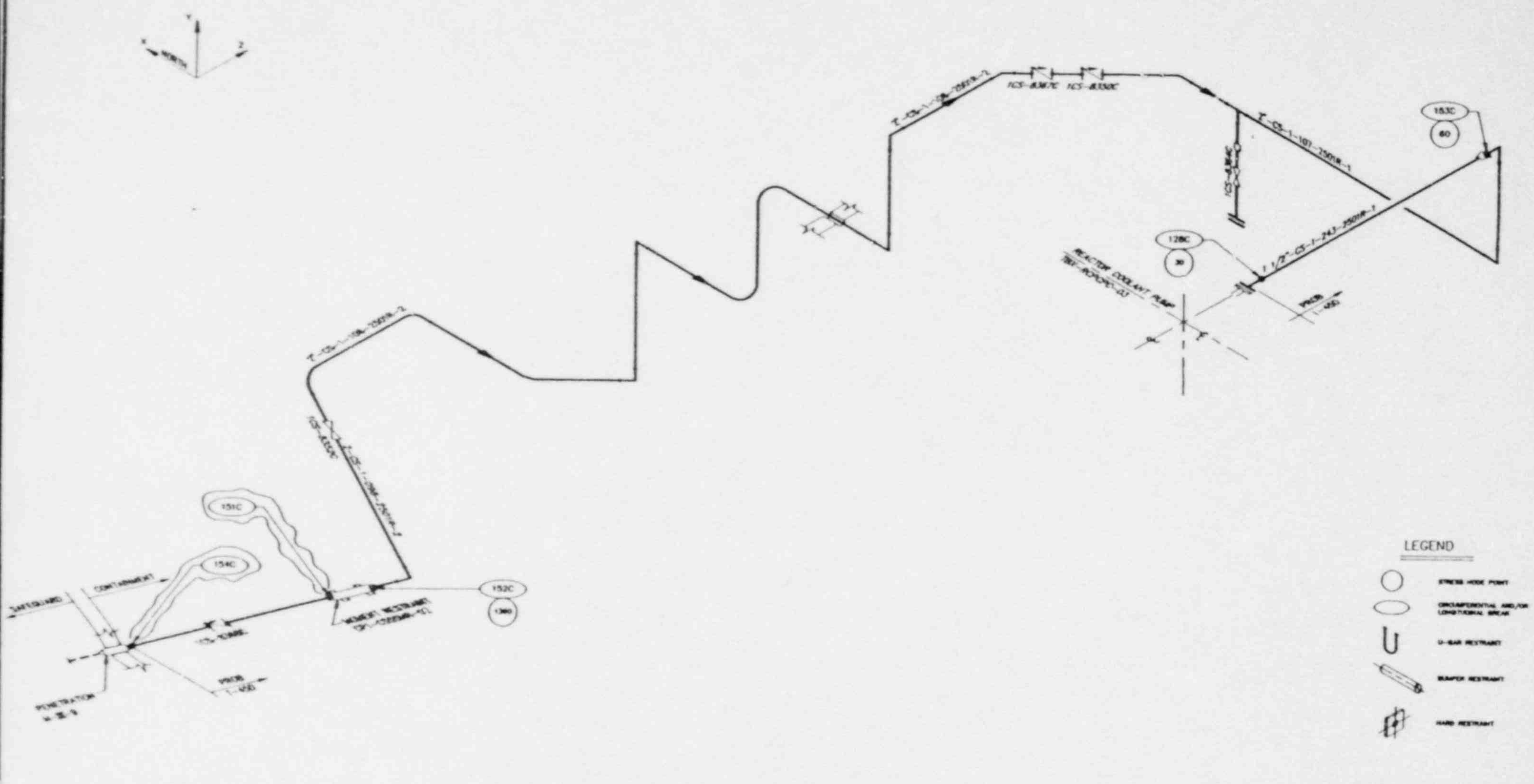
FIGURE 3.6B-78 PROB. 1-38



COMANCHE PEAK S. E. S.
FINAL SAFETY ANALYSIS REPORT
UNIT 1

CVCS SYSTEM INSIDE CONTAINMENT
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-79 PROB. 1-45C



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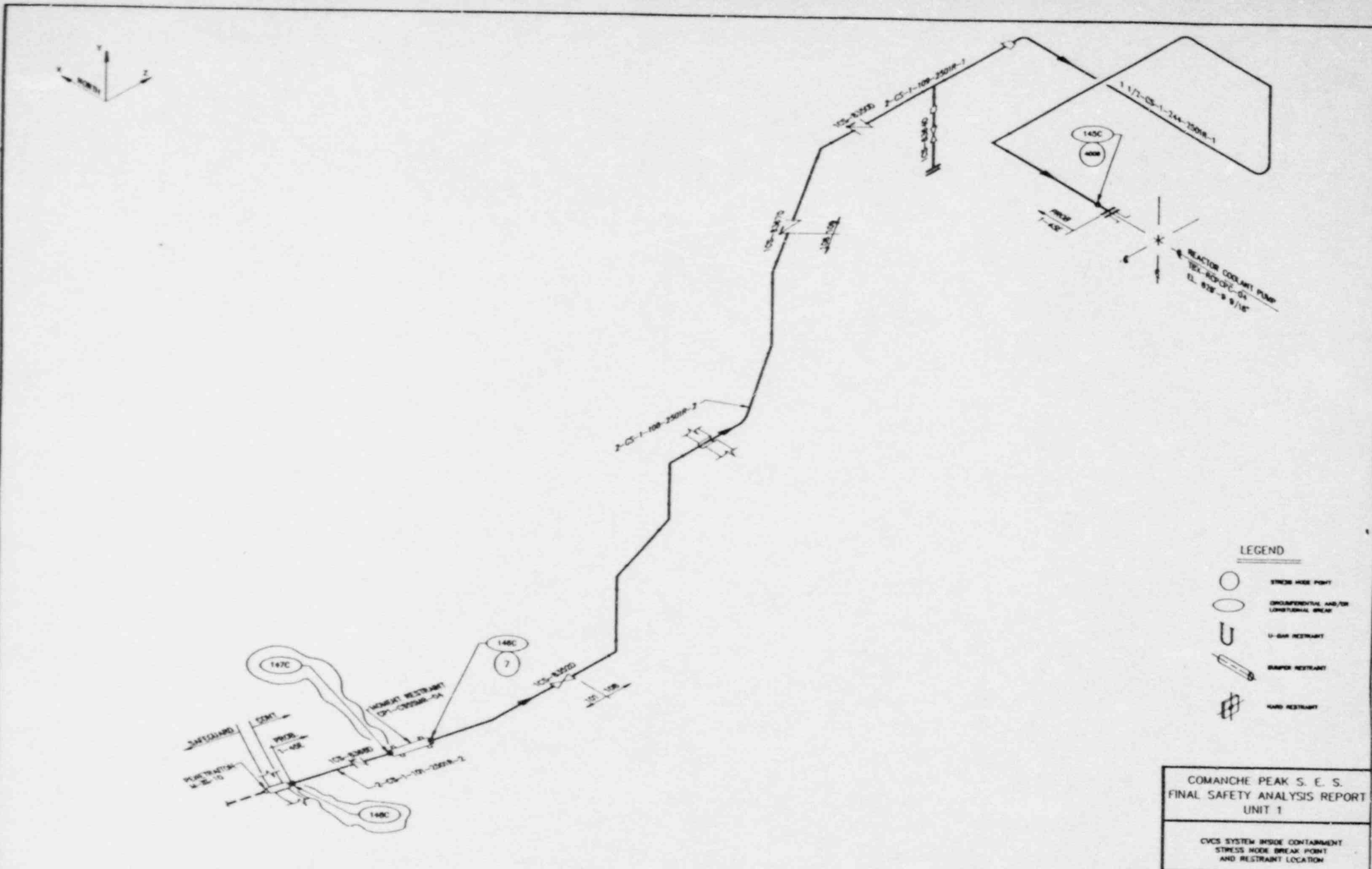
FINAL SAFETY ANALYSIS REPORT
UNIT 1

CVCS SYSTEM INSIDE CONTAINMENT
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-80 PROB. 1-450



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FINAL SAFETY ANALYSIS REPORT
UNIT 1

CVCS SYSTEM INSIDE CONTAINMENT
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-81 PROB. 1-45E

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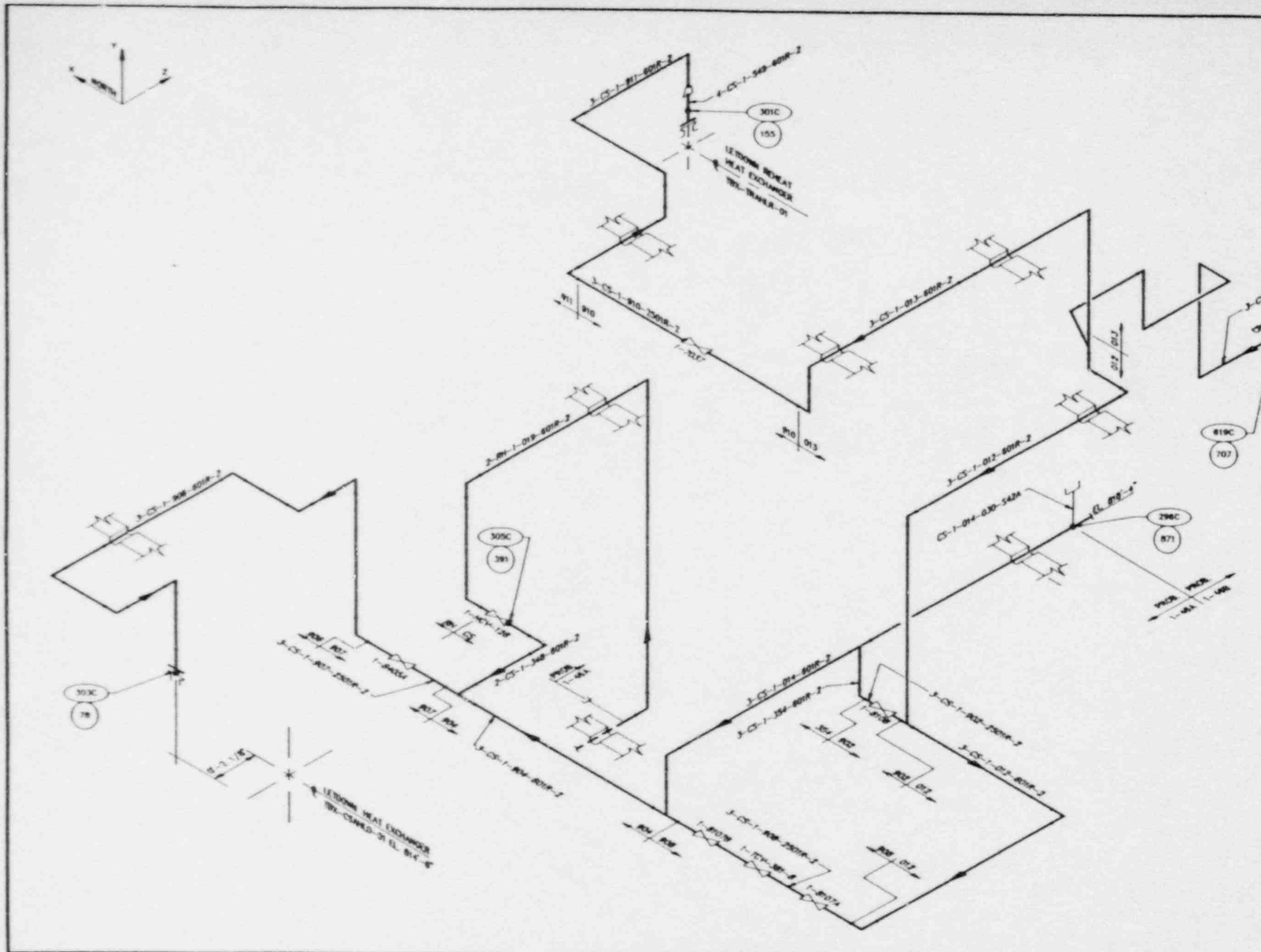

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FINAL SAFETY ANALYSIS REPORT
UNIT 1

CVCS SYSTEM OUTSIDE CONTAINMENT
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-82-1 PROB. 1-46A

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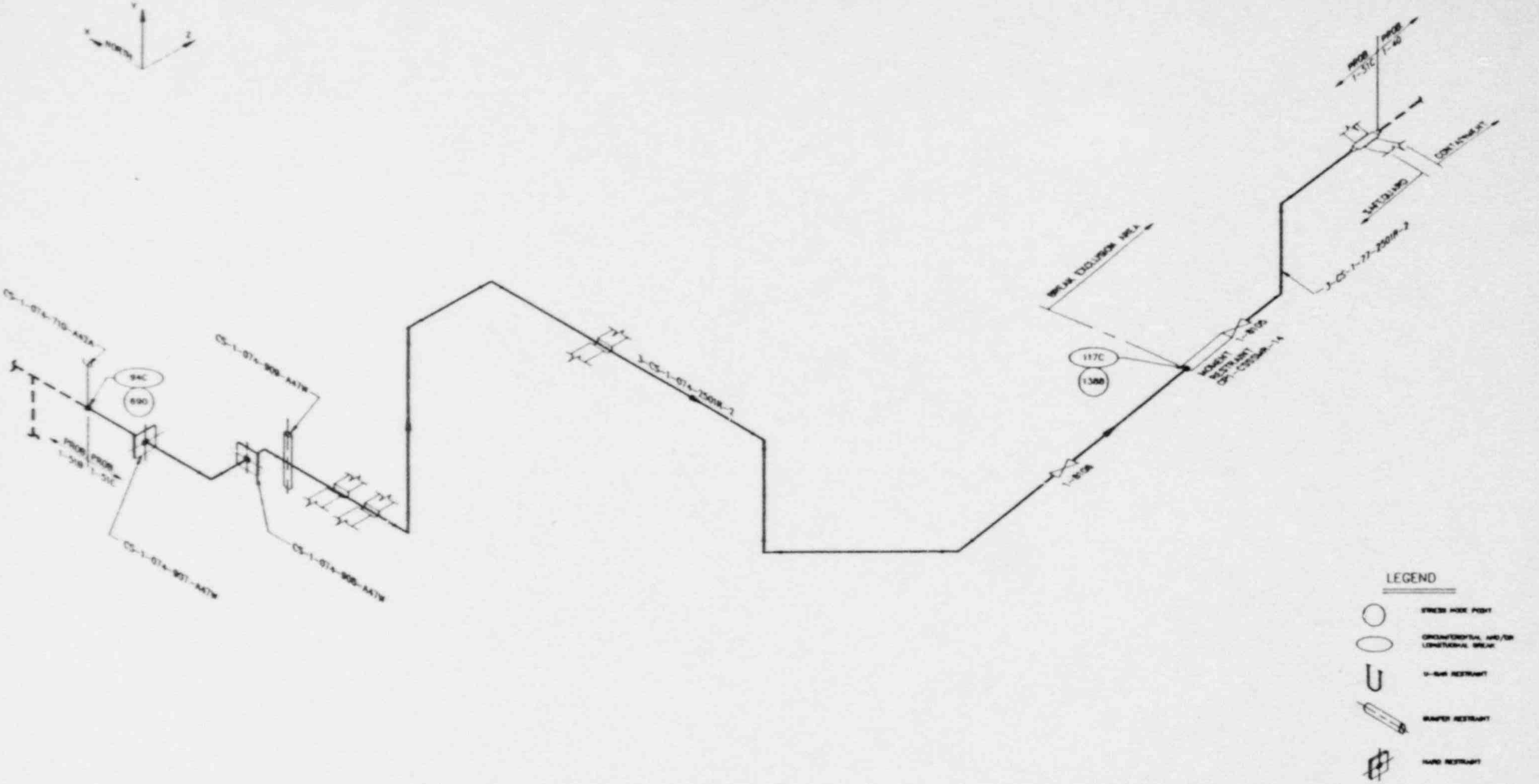

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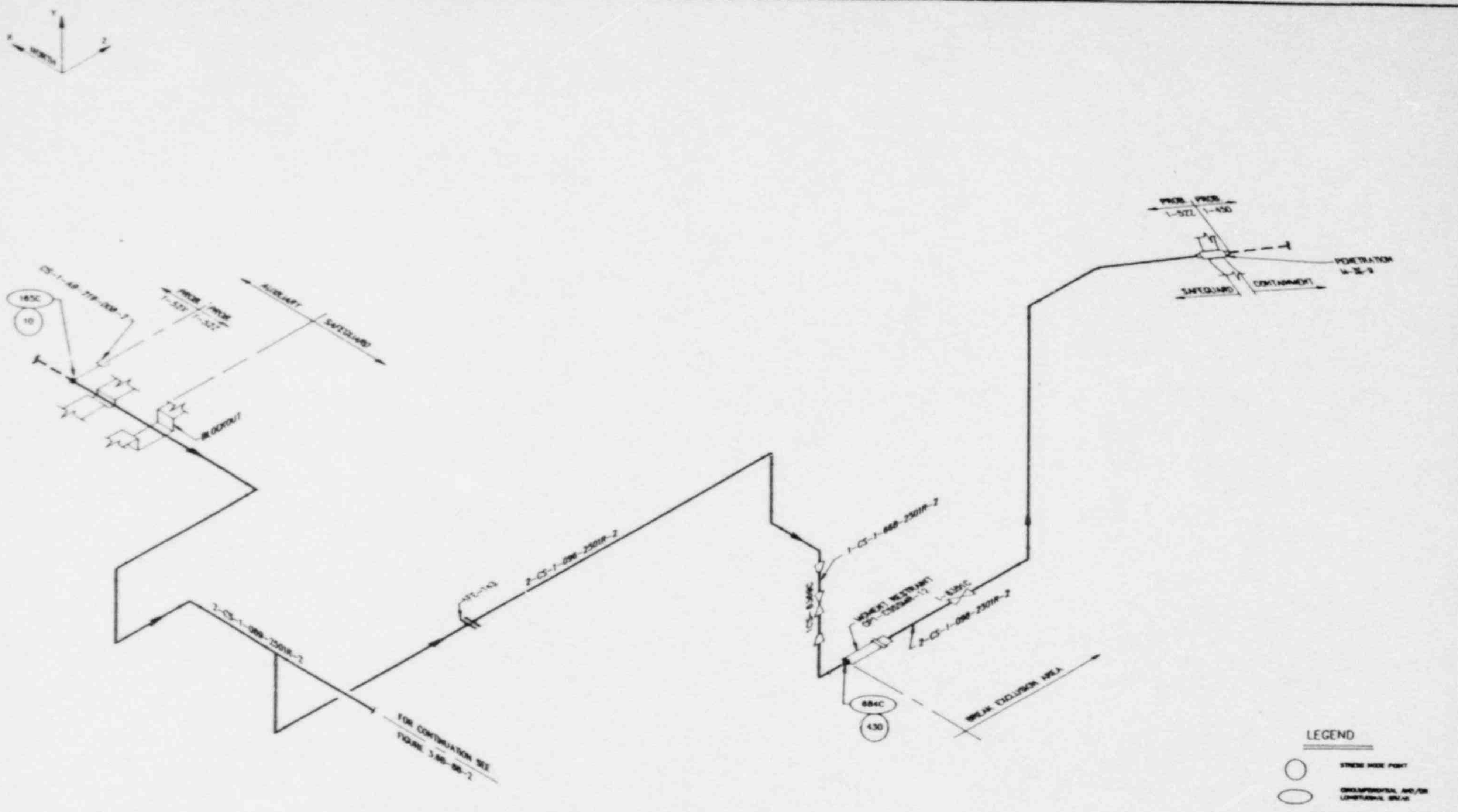
FINAL SAFETY ANALYSIS REPORT
UNIT 1

CVCS SYSTEM
AUX. BUILDING
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-85 PROB. 1-51C



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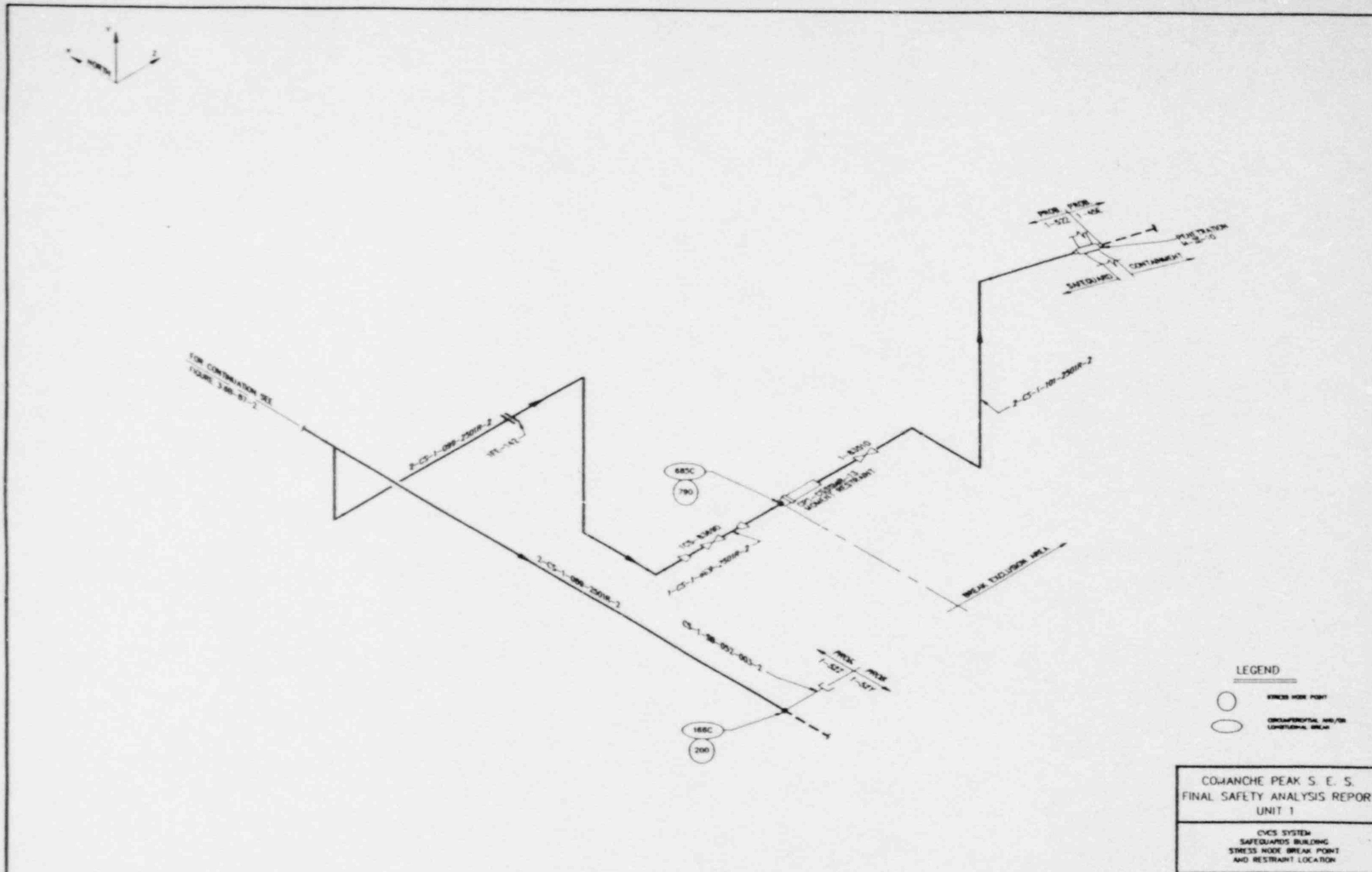
FINAL SAFETY ANALYSIS REPORT
UNIT 1

CVCS SYSTEM
SAFEGUARDS BUILDING
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-87-2 PROB. 1-522



COMANCHE PEAK S. E. S.
FINAL SAFETY ANALYSIS REPORT
UNIT 1

CVCS SYSTEM
SAFEGUARDS BUILDING
STRESS NODE BREAK POINT
AND RESTRAINT LOCATION

FIGURE 3.6B-88-2 PROB 1-572

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The results of these two steps, the dynamic inertia analysis and the static differential motion analysis, are combined absolutely with due consideration for the ASME classification of the stresses.

3.7N.3.10 Use of Constant Vertical Static Factors

Constant vertical load factors are not used as the vertical floor response load for the seismic design of safety-related components and equipment within Westinghouse's scope of responsibility.

3.7N.3.11 Torsional Effects of Eccentric Masses

The effect of eccentric masses, such as valves and valve operators, is considered in the seismic piping analyses. These eccentric masses are modeled in the system analysis and the torsional effects caused by them are evaluated and included in the total system response. The total response must meet the limits of the criteria applicable to the safety class of the piping.

3.7N.3.12 Buried Seismic Category I Piping Systems and Tunnels

Refer to Section 3.7B.3.12.

3.7N.3.13 Interaction of Other Piping with Seismic Category I Piping

Refer to Section 3.7B.3.13.

3.7N.3.14 Seismic Analyses for Reactor Internals

The seismic analysis of the reactor internals is conducted in accordance with the guidelines specified in Regulatory Guide 1.92. The seismic analysis determines the response of the reactor internals to Operational Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) vertical and horizontal seismic shock components. The horizontal and 20

20 vertical seismic analysis use the modal response spectrum method and the WECAN general purpose finite element program to determine the internals response. The method used to obtain the combined response of the modal spectral responses is square-root-of-the-sum-of-the squares (SRSS).

20 The effect of closely spaced modes is considered using the Ten Percent Method (Regulatory Guide 1.92, Paragraph 1.2.2); however, the effect has been shown to be insignificant. The maximum, or total, seismic response value of the reactor internals is obtained by taking the SRSS of the maximum values of the co-directional responses due to the three components of earthquake motion. In general, this combination is made in the Stress Analysis section of the particular structural component.

68 The horizontal and vertical seismic models contain 118 and 23 active dynamic degrees of freedom, respectively. Results from the modal analysis of the horizontal and vertical systems indicates, in general, 12 and 3 modes present with frequencies less than 33Hz.

20 In developing the seismic model of the reactor vessel and internals, a systematic approach was used to ensure that basic fundamental frequencies, i.e., both component and system frequencies, are described and inherent in the mathematical models. The approach used to verify the mathematical modeling of reactor vessel and internals was to compare and require that the system frequencies and mode shapes, from the mathematical models, be in agreement with plant test and scale model test data.

20 In determining the seismic response of the reactor system, due to the excitation of unidirectional shock spectrum, only those modes contributing to the first 80-90 percent of total system mass were considered in the solution.

20 Hydrodynamic mass effects, for horizontal and vertical directions, was included in the reactor vessel-internals system models. The numerical values for the various hydrodynamic mass effects within the reactor system were based on scale model and plant tests and applicable analytical expressions; e.g., Fritz, Fritz & Kiss, etc.

3.7B SEISMIC DESIGN

3.7B.1 SEISMIC INPUT

3.7B.1.1 Design Response Spectra

Design response spectra for both horizontal and vertical ground motion for the SSE are shown in Figures 3.7B-1 and 3.7B-7, respectively. Response spectra for 2, 5, 7, 10, and 15 percent of critical damping are provided for both the horizontal and vertical motions and are scaled to the maximum ground accelerations of 0.12g and 0.08g selected for the SSE. For the OBE, a scaling factor of 0.5 is applied to the SSE design spectra.

The response spectra are constructed on the basis of the 68
recommendations of Newmark, Blume, and Kapur [14] and conform to the requirements of NRC Regulatory Guide 1.60, Revision 1, with the exception of the 33 Hz to 50 Hz frequency range. In that range, the vertical response spectrum of NRC Regulatory Guide 1.60, Revision 1, differs from the vertical response spectrum of Reference [14]. The effects of this deviation on the results of the analyses of structures and systems are negligible because they only affect the modes which have low amplification. Similarly, the method recommended in Reference [14] for the construction of vertical response spectra leads to a slight deviation from NRC Regulatory Guide 1.60, Revision 1 recommendations for accelerations corresponding to 3.5 Hz. The magnitudes of these differences are negligible.

The response spectra indicate the estimated response of a structure subject to significant nearby earthquake ground motion. The spectra are presented over a range of frequencies corresponding to natural frequencies of structural elements, and they represent the maximum amplitude of motion in structural elements for typical degrees of

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structural damping. Because the design response spectra have been developed from a large number of real records, following the procedures recommended by Newmark, the effect of strong motion duration and distance of focal depth are included [29].

There are, of course, general associations between duration of strong motion and the size of an event. Longer durations of strong motion are expected with greater-sized earthquakes. Higher frequency accelerations are attenuated with greater distance from the epicenter of the earthquake. These conditions are inherent in the strong motion records which are the source of Newmark's work. In no case are the amplification factors less than one.

3.7B.1.2 Design Time History

68 | An artificial time history has been produced for each of the horizontal and vertical motions resulting from the SSE. These artificial records are based on the design response spectra requirements presented in Section 3.7B.1.1.

As an alternative to a site-dependent analysis, these artificial time history records are suitable for use as base excitations for the dynamic structural analysis.

The mathematical procedures used to generate these artificial time history records can be briefly summarized as follows:

1. The spectral characteristics of the selected site SSE design response spectra are extracted to construct a frequency response function with proper phase factor modification.

2. A fast Fourier transform of the frequency response function is performed to obtain a filter impulse response function.

3. The filter impulse response function is then integrated with a set of pseudorandom numbers to obtain an artificial time history record.

4. A comparison is made between the response spectrum derived from the artificial time history and the site SSE design response spectrum. Any unacceptable deviations are corrected by adding a series of sinusoidal impulses with proper amplitude and phase angles until the desirable fit is achieved.

Q130.6
Q130.7

5. The artificial time history records meet the minimum acceptance criteria given by Table 3.7.1-1 in Section 3.7.1 of the Standard Review Plan. 68

68

The response spectra derived from the horizontal artificial time history record and the selected site SSE design response spectra are presented in Figures 3.7B-2 through 3.7B-6, for each of the five structural damping values. The corresponding artificial time history is presented in Figure 3.7B-14. The response spectra from the vertical artificial time history record and the SSE design response spectra are presented on Figures 3.7B-8 through 3.7B-12, and the corresponding artificial time history is presented on Figure 3.7B-19. 68

Time history durations of approximately 10 sec have been found necessary to allow the modifications of the time histories to match response spectra values at periods of three to four sec. A 10 sec record allows two to three cycles for modification by sinusoidal impulses. A record length of 10.24 sec is obtained because the fast Fourier transform used for this purpose operates on sets of numbers which are as powers to time; i.e., 1024 is equal to two raised to the tenth power.

The artificial time history records are generated at 0.01 sec equal time intervals with a time duration of 10.24 sec. They are in the digitized form of 1024 acceleration values.

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3.7B.1.3 Critical Damping Values

The specific percentages of critical damping values used for Category
I structures, systems, and components are based on the materials,
stress levels, and type of connections of the particular structure or
component. They are determined in accordance with the recommendations
61 of NRC Regulatory Guide 1.61 and Reference [14]. For piping systems
analyzed by the response spectrum method, ASME Code Case N-411 damping
values may also be used in lieu of the damping values in Regulatory
66 Guide 1.61. The ASME Code Case N-411 damping values are also used in
an analysis of the primary system loop to determine seismic loads
transmitted to the steam generator upper and lower lateral
restraints. However, qualification analyses of the steam generator
and other primary loop components incorporate damping values as
defined in Section 3.7N.

Structure and component damping values used in the response spectrum
and time history analyses are given in Table 3.7B-1. Damping factors
associated with foundation springs are discussed in Section 3.7B.2.4.
66 Damping values used for qualifying Westinghouse equipment are shown in
Section 3.7N.

3.7B.1.4 Supporting Media for Seismic Category I Structures

All seismic Category I structures are founded on the firm, unweathered Glen Rose Limestone which constitutes the principal bedrock formation on the site.

Below the Glen Rose unit lies the Twin Mountains Formation, which forms a gradational contact with the Glen Rose unit and is composed principally of sandstone, limestone, and clay stone. The portion of the Glen Rose unit which provides the founding material for the Category I structures consists of argillaceous limestone with lenses and zones of calcereous clay stone. Approximately 150 to 160 ft of

this formation is present beneath the lowermost foundation. The upper portion of the Glen Rose unit consists of weathered rock and a soil cover of a few feet. Prevailing rock characteristics are presented | 66
in Table 3.7B-3.

The soil cover and the upper 40 ft (approximately) of the Glen Rose Limestone are totally removed by foundation excavation. Thus, all of the moderately-to-severely weathered rock present at the site is removed.

With the exception of the Service Water Intake Structure, no structural backfill is used under or against Category I structures.

More detailed description of the site geology, the subsurface | 68
conditions, and the engineering properties of site materials are included in Section 2.5.

Foundation elevations, depths of embedment, total structural heights, and foundation plan dimensions for the Category I structures are presented in Table 3.7B-4.

3.7B.2 SEISMIC SYSTEM ANALYSIS

3.7B.2.1 Seismic Analysis Methods

Methods of seismic analysis used for seismic Category I structures, systems, and components, as well as applicable stress and deformation criteria, and mathematical models, are described in this section.

Seismic analysis of seismic Category I structures, systems, and components is performed by the use of the response spectrum or the time history concept of analysis, or both [28], [30], [35]. The use of the response spectrum concept provides a convenient procedure for seismic analysis. Spectrum analysis uses the natural frequencies, mode shapes, and appropriate modal dampings as a fraction of critical damping, and is an approximate method for determining the seismic response of linear elastic multidegree-of-freedom systems with lumped masses and elastic properties in discrete parts.

In a time history analysis, there are two basic ways of using the time history for linear elastic systems, namely, by a modal analysis time history, which uses the same free vibration characteristics and damping factors as the spectrum analysis, or by solving a system of coupled differential equations of motion by direct numerical integration. In the latter case, the numerical integration using a suitable technique must be performed simultaneously for all of the coupled equations. This procedure is cumbersome, requiring a large amount of computations, and is susceptible to computational difficulties. For example, it is difficult to know how small the time intervals should be to avoid mathematical instability. Furthermore, there is no really satisfactory way to determine all of the damping coefficients in these coupled differential equations of motion. Because of these difficulties, the modal method of analysis is used. Only in the case of nonlinear behavior when structures, systems, and components cannot be regarded as linear elastic, such as springs with nonlinear restoring-force functions and nonlinear elastic properties of materials, is the method of direct numerical integration of coupled differential equations of motion used.

Where the aforementioned methods do not provide reliable results, or where analysis appears impractical, dynamic testing of equipment is performed to ensure functional integrity.

The methods used for seismic analysis of particular seismic Category I structures, systems, and components are summarized in Table 3.7B-2.

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It should be noted that the modal analysis time history method is used to generate responses at selected locations, such as the ones required for the development of instructure response spectra. Responses at selected locations resulting from both response spectrum concept and time history are compared. Static loads resulting from a dynamic analysis are used in the design of some structural components such as foundation mats, floors, and shear walls [34].

3.7B.2.1.1 Idealization of Seismic Category I Structures, Systems, and Components

A most important part of seismic analysis is devising a mathematical model that satisfactorily represents the dynamic behavior of a seismic Category I structure, system, or component. The modeling technique used results in mathematical models composed of a network of lumped masses and elastic properties in discrete parts. Normally, characteristic points or nodes are selected so that they coincide with concentrations of mass, e.g., at floors, changes of cross sectional area, or at locations which are important for stiffness. The characteristic points for lumping of the masses of an axisymmetric shell-type structure are selected at the centroids of horizontal cross-sections through individual components of the structure. These centroids lie on the vertical centerline of the structure. Each mass has six degrees of freedom, namely, three translations in the three principal orthogonal directions and three rotations about the three principal orthogonal axes. In general, responses associated with all of these degrees of freedom can be coupled and excited by each direction of seismic motion. Bending and shearing effects are considered in determining the discrete rigidities between the lumped masses.

For all seismic Category I structures except the Service Water Intake | 68
Structure and Seismic Category I Tanks, finite element techniques that |
simulate floor slabs and shear wall assemblies are used to generate |

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68 the reduced stiffness matrix associated with the number of dynamic degrees of freedom required for the dynamic analysis. The mathematical model for which this reduced matrix is generated consists of lumped masses, viscous dashpots, and elastic properties in discrete
68 parts. For the Service Water Intake Structure and the Seismic Category I Tanks stiffness properties are calculated for the structural elements between lumped mass elevations using standard structural techniques. The mathematical models representing the seismic Category I structures and the method chosen for the selection of the number of masses are described in Subsection 3.7B.2.1.6.

For ease of computation, the mathematical model is reduced to contain as few dynamic degrees of freedom as feasible so that it can be analyzed successfully by means of algorithms adopted for today's high-speed digital computers.

Foundation structure interaction is represented by decoupled springs, dashpots, and effective masses generally associated with the six degrees of freedom in a global orthogonal system. The methods used to determine the foundation parameters related to torsion, rocking, and translation are described in Subsection 3.7B.2.4.

3.7B.2.1.2 Analytical Approach

In order to analyze the response of a linear elastic lumped mass system, the natural frequencies and corresponding mode shapes are first determined. This determination is accomplished by extracting the eigenvalues and associated eigenvectors from a homogeneous system of equations which result from undamped free vibration and are comprised of stiffness or flexibility and mass matrices developed from the mathematical model. These free vibration characteristics are calculated by using any one of the suitable algorithms coded into the computer programs, such as the diagonalization method originated by Jacobi, Householder's tridiagonalization method combined with the

Sturm sequence method, and methods such as those used in computer programs presented in Section 3.7B(A). After establishing the free vibration characteristics, such as natural frequencies and associated mode shapes, the next step consists of response computations obtained by using the response spectrum approach or time history analysis or both [28], [30], [31], [35], [38].

1. Response Spectrum Analysis

   The response spectrum analysis is performed using various computer programs consisting of different subroutines developed by TU Electric or it's engineering services contractor as described in Section 3.7B(A).

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   The analysis of the structures founded on bedrock uses spectral values from the free-field horizontal and vertical ground response spectra developed for this site. Spectral values associated with modal dampings and natural frequencies are obtained for each mode. Then the maximum absolute accelerations, inertia forces, shears, moments, and relative displacements are obtained in each mode. The maximum modal responses of all the modes are combined by the square root of the sum of the squares (SRSS), by absolute sum, and by combinations thereof, as discussed in Subsection 3.7B.2.7. | 2

   A separate analysis is made on the model representing the structure founded on bedrock for each of the three orthogonal principal directions of input ground excitations.

   Vertical and two horizontal ground excitations are assumed to act simultaneously. Hence, the combined effects of earthquakes on structures, components, or elements are computed by taking the SRSS of the particular effects at any particular point, caused by each of the three components of earthquake motion (two horizontal motions at right angles and one vertical motion).

In the case of shell structures when shell theory is used, maximum stress resultants (membrane shears, moments, and forces), as well as unit stresses and displacements, are obtained. This is accomplished by applying distributed inertia forces and using a suitable computer program.

The total overturning moment at the base of a structure is obtained. The maximum dynamic foundation pressure is evaluated to ensure that it is within permissible limits.

The analysis is performed for both the SSE and OBE unless it is apparent that one of these controls the design.

2. Time History Analysis and Instructure Response Spectra

After the mathematical models of structures are analyzed for their characteristics of free vibration, the time history responses at selected mass points are obtained using the artificial time history ground motion [30], [31], [38]. Derivation of the appropriate time history ground motion is discussed in Subsection 3.7B.1.2. Once the time history response of a selected mass point is generated, the next step is to subject a single-degree-of-freedom system, with the natural frequency range of interest and various damping ratios, to this time history motion. A spectrum response curve is obtained by plotting the maximum acceleration response as ordinates and the corresponding natural periods of the single-degree-of-freedom
68 system as abscissa. The enveloping technique used for the construction of instructure response spectra consists of enveloping the maximum peaks. Since the frequencies of the structures can only be computed approximately because of the linear and nonlinear deformability, the energy dissipation, variation in elastic properties of both structure and foundation,

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and the idealization of structure with lumped masses and elastic properties in discrete parts, parametric studies are performed in order to take into account these effects for the construction of instructure response spectra. These effects result in the shifting of the resonance peaks of the instructure response spectra. The peaks are widened by at least 10 percent of the resonance frequencies to account for these effects. The widening exceeds 10 percent if the parametric studies indicate that such widening is necessary to achieve conservative results. These spectra are generated on the basis of an artificial time history ground motion with maximum horizontal ground accelerations normalized to 12 and 6 percent of gravity for SSE and OBE, respectively, and maximum vertical ground accelerations normalized to 8 and 4 percent of gravity for SSE and OBE, respectively. Parametric studies and spectra widening are discussed in Subsection 3.7B.2.9. | 68

The preceding analyses are accomplished by using suitable computer programs as presented in Section 3.7B(A) and in accordance with References [30], [31], [36], and [38].

3.7B.2.1.3 Testing and Analysis for Equipment

Seismic Category I equipment, equipment supports, and components are designed to ensure functional operability during and after an earthquake of magnitude up to and including the SSE (refer to Section 3.2 for the list of seismic Category I mechanical and electrical equipment). The capability of all seismic Category I electrical and mechanical equipment and equipment supports to satisfy this requirement is verified by testing or analysis, or both.

The seismic qualification of seismic Category I electrical equipment and equipment supports is in accordance with IEEE 344-1975. Seismic Category I mechanical equipment and equipment supports are qualified in accordance with the ASME B&PV Code, Section III, Division 1, and the applicable NRC regulatory guides.

66 When dynamic analysis is employed, the equipment specification furnishes the vendor with appropriate instructure response spectra for both SSE and OBE with instructions on their use. In the case where a supporting structure is required to be modeled as an integral part of the equipment, the vendor is furnished with the mathematical model representing the structure consisting of lumped masses and elastic properties in discrete parts. The lumped masses, flexibility or stiffness characteristics, or both, can be presented in matrix form. Any additional information such as the floor motion time history is furnished only upon request.

Where applicable, the response spectrum technique is employed in the dynamic analysis of seismic Category I equipment and components. The time history modal analysis of seismic Category I structures, as previously explained, generates time histories at various support elevations and instructure response spectra for use in analysis of systems and equipment.

In general, at each level of the supporting structure where vital items are located, two horizontal and one vertical response spectra corresponding to coupled translational motions of the supporting structure in the three orthogonal principal directions and the coupled rotational motions of the structure about the three axes for each direction of ground excitation are developed. The instructure response spectra are smoothed so that the response curve is an upper
68 bound envelope of all the acceleration points. Parametric studies and spectra widening are discussed in Subsection 3.7B.2.9.

Generally, multimass presentation of seismic Category I equipment and components, except piping, is used in accordance with the lumped parameter modeling techniques and normal mode theory, as described in the references listed in Section 3.7B. Piping systems are analyzed as described in Subsection 3.7B.3.8.

Simplified analytical models such as one or two degree-of-freedom models are used where they provide a suitable representation of the systems.

Equipment whose lowest dominant natural frequency is 33 Hz or higher is considered rigid. In this case the acceleration of the equipment is assumed to be the same as the zero period acceleration of the appropriate response spectrum

Where dynamic testing is used to ensure functional integrity, test performance data and results reflect the following:

1. Performance data for equipment which, under the specified conditions, has been subjected to equal or greater dynamic loads than those to be experienced under the specified seismic conditions

2. Test data from previously tested comparable equipment which, under similar conditions, has been subjected to equal or greater dynamic loads than those specified

Mechanical and electrical equipment and components which are required to maintain functional integrity and operability during and after a postulated seismic event are described in Subsection 3.9B.2 and Section 3.10B. Actual testing of such equipment and components to demonstrate seismic adequacy is performed by the suppliers using a shake table, test bed, or other appropriate device.

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1. Seismic testing for equipment operability conforms to the following:

   a. A test required to confirm the functional operability of seismic Category I electrical and mechanical equipment and instrumentation during and after an earthquake of magnitude up to and including the SSE is performed. Analysis without testing may be performed only if structural integrity alone can ensure the design intended function. When a complete seismic testing is impracticable, a combination of test and analysis is performed.

   b. The characteristics of the required input motion are specified by one of the following:

      1) Response spectrum

      2) Power spectral density function

      3) Time history

      Such characteristics, as derived from the structures or systems seismic analysis, are representative of the input motion at the equipment mounting locations.

   c. Where practicable, equipment which is required to function during and/or after an earthquake is tested in the operational condition. Operability is verified during and/or after the testing.

   d. The actual input motion is characterized in the same manner as the required input motion and the conservatism in amplitude and frequency content is demonstrated. The frequency spectrum covers the range from 1 through 33 Hz.

e. Seismic excitations generally have a broad frequency content. Therefore, random vibration input motion is generally used. Single frequency input, such as sine beats, can be applicable provided one of the following conditions is met:

1) The characteristics of the required input motion indicate that the motion is dominated by one frequency, i.e., by structural filtering effects.

2) The anticipated response of the equipment is adequately represented by one mode.

3) The input has sufficient intensity and duration to excite all modes to the required magnitude so that the testing response spectra envelope the corresponding response spectra of the individual modes.

f. To account for the simultaneous occurrence of random earthquake motions, multiaxis testing is performed. Test input motions are applied simultaneously in the vertical and each of the horizontal principal directions. Single-axis testing is accepted in accordance with the provisions of IEEE 344-1975, i.e., where justified by demonstrating that earthquake motions at the mounting locations are amplified in one direction, or that the equipment is constrained to respond in one direction, or that coupling between the responses is negligible. In the case of single-frequency input, the time phasing of the inputs in the vertical and horizontal directions is such that a purely rectilinear resultant input is avoided.

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The acceptable alternate to this procedure is to test the equipment with vertical and horizontal inputs in-phase, and then repeat the test with inputs 180 degrees out-of-phase. 5

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5 | In addition, it is required that the test be repeated with the equipment rotated 90 degrees horizontally.

g. The fixture design meets the following requirements:

1. Simulates the actual service mounting

2. Causes no dynamic coupling to the test item

h. The in situ application of vibratory devices to superimpose the seismic vibratory loadings on the complex active device to provide data for operability testing is acceptable when application is justifiable.

i. The test program may be based upon selectively testing a representative number of mechanical components according to type, load level, size, and similar items on a prototype basis.

j. Where a random vibration input motion is used, the amplitudes of the excitations are controlled to provide a test response spectrum which meets or exceeds the applicable required response spectra.

2. Seismic design adequacy of supports conforms to the following:

a. Analyses or tests are performed for all supports of electrical and mechanical equipment and instrumentation to ensure their structural capability to withstand seismic excitation.

In accordance with the intent of IEEE 344-1975, if supports are similar and justified as such, or if based on engineering investigation, the worst case support is chosen from a group of supports to be qualified and is justified, only one of the similar supports or the worst case support requires a complete dynamic seismic analysis or full scale test, or a combination of both.

Justification of this procedure is based upon comparison analysis or past experience that the supports to be qualified are similar or that the worst case has been chosen. Upon such justification and dynamic analysis or full scale testing, or a combination of both, of the similar or worst case support, the group of supports being investigated is accepted as seismically qualified.

b. The analytical results include the following:

1) The required input motions to the mounted equipment are obtained and characterized in the manner stated in Item 1.b of this subsection.

2) The combined stresses of the support structures are within the limits of the ASME B&PV Code, Section III, Division 1, Subsection NF, Component Support Structures, or other comparable stress limits.

c. Supports are tested with equipment installed. If the equipment is inoperative during the support test, the response at the equipment mounting locations is monitored and characterized in the manner stated in Item 1.b. In such a case, equipment is tested separately and the actual input to the equipment is made more conservative in amplitude and frequency content than the monitored response.

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d. The requirements of Items 1.b, 1.d, 1.e, 1.f, and 1.g are applicable when tests are conducted on the equipment supports.

Equipment, when under test, is mounted in a manner that simulates the intended service mounting in its operating condition.

When sinusoidal beat input motion is applicable as per Items 1.e.1), 1.e.2), and 1.e.3), testing is normally performed in two phases as follows:

1. Phase 1 consists of a low-amplitude continuous sweep frequency search, with a sinusoidal steady-state input of at least 0.2g over a minimum frequency range of 1.0 to 33 Hz in order to determine potential resonant regions.

2. Phase 2 consists of sinusoidal beat testing at resonant frequencies determined in Phase 1 with amplitudes correlated to the maximum acceleration of the equipment support. In addition to sinusoidal beat testing at the known resonant frequencies, the test is also performed at a number of preselected frequencies in the 1.0 to 33 Hz band. This ensures that any shifts in resonant frequencies caused by nonlinear effects are adequately evaluated.

   A sinusoidal beat consists of 10 cycles at the frequency being tested with amplitudes of each cycle varying as a sine function from zero to a maximum at the fifth cycle and then decreasing to zero at the tenth cycle. The maximum sine beat amplitude is made to correspond to the maximum zero period SSE acceleration of the structure at the appropriate floor.

   A test at any frequency consists of five beats with a pause between beats of approximately two sec or more, as required to ensure no superposition of motion.

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Sinusoidal steady-state testing at resonant frequencies rather than sinusoidal beat testing is performed at the option of the equipment supplier.

When sinusoidal beat testing or sinusoidal steady-state testing is used, the equipment is subjected to the input motion for at least the entire duration of an earthquake of 30 sec.

All tests are performed independently for each of the two horizontal directions and the vertical direction. Three-axes simultaneous testing is also required. Satisfactory alternatives as recommended in Section 6.6.6 of IEEE 344-1975, Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations, are allowed where three-axis testing is impractical.

Equipment when under test is required to demonstrate a capability to perform its intended function, and sufficient monitoring equipment is used to evaluate performance during test.

Other test procedures proposed by equipment suppliers equivalent to the methods described previously are reviewed by TU Electric or its engineering services contractor, and used for seismic qualification of equipment when formally approved by TU Electric or its engineering services contractor. 66

The equipment supplier is required to document his previous experience in analysis or testing of seismic Category I equipment similar in nature to that being provided.

The equipment supplier is required to furnish documentation to demonstrate that the equipment meets the seismic design criteria established in the specification in both performance and structural integrity.

If proof of performance is obtained by analytical means, the equipment supplier is required to present documentation in a step-by-step form which is readily auditable by persons skilled in such analyses.

If proof of performance is obtained by testing, the test data include detailed information on the followin

1. Equipment identification

2. Test facility (including location)

3. Test equipment

4. Test method

5. Test data (includi g unsuccessful test of components and subsequent remedial measures)

6. Results and conclusions

7. Attestation

3.7B.2.1.4 Differential Seismic Movement of Interconnected Components

The seismic analysis of seismic Category I subsystems and equipment subjected to differential support motion is performed in three parts using lumped mass mathematical models as follows:

1. Modal response spectrum analysis is performed for all three principal orthogonal directions of support motion for each direction of ground excitation using appropriate instructure response spectra constructed on the basis of superimposing the spectra for all support points and enveloping them as stated in

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Subsection 3.7B.2.5. The analysis of these subsystems or components follows the same considerations as those described in Subsection 3.7B.2.1 for seismic Category I structures. The vertical analysis is combined with both horizontals, according to the statement in Subsection 3.7B.2.1.2, to produce basic dynamic loading conditions.

2. The same multimass lumped parameter model is subjected to a stress analysis due to differential displacements of the support points. The displacements used are consistent with the directions of structural excitation being considered in the spectrum analysis. This results in basic differential displacement loading conditions.

3. The results obtained from the spectrum analysis and differential displacement analysis are then combined directly. The effects of these loading conditions on the components and the supporting structures are determined.

3.7B.2.1.5 Stress and Deformation Criteria

The maximum horizontal ground accelerations are 6 and 12 percent of gravity for OBE and SSE, respectively. The maximum vertical ground accelerations are equal to two-thirds of the horizontal. Horizontal and vertical ground motions are assumed to act simultaneously. Horizontal ground response spectra for the SSE are shown on Figure 3.7B-1.

Primary steady-state stresses including the effects of the normal operating loads plus the OBE loads are maintained well within the elastic limit of the material affected.

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For systems and equipment, self-limiting secondary stresses may exceed allowable primary stress to the extent permitted by the appropriate codes. For the OBE, the equipment function is performed without permanent deformation.

Primary steady-state stresses, including the effects of the normal operating loads plus the SSE loads, are limited to prevent loss of
66 function of the equipment. For the purpose of calculation, the no-loss-of-function stresses are limited to 90 percent of the yield strength of the material, except when valid plastic analysis demonstrates structural integrity or when the stress limits are specifically controlled by an applicable code or standard as committed to by CPSES for example, see Section 3.9B. Local, self-limiting, secondary stresses may exceed yield stress levels to the extent set forth in the appropriate design standards and codes.

Deformations resulting from the combined influence of normal operating loads and the loads from the SSE are investigated to verify that they do not impair the functional performance required for a safe and orderly shutdown of the plant.

For fatigue analysis required by some codes, the number of expected earthquakes, the duration of strong motion vibration, and the number of cycles the equipment or system is exposed to are evaluated for the OBE, and specified as 600 load cycles.

Because the earthquake transients are part of mechanical loading conditions, the origin of their determination is separate and distinct from those transients which result from fluid pressure and temperature. Therefore, the superposition of the earthquake cycles on the fluid pressure and thermal transients is not considered.

3.7B.2.1.6 Descriptions of Mathematical Models

The mathematical models used for the dynamic analysis of the major structures are described as follows:

1. Model for Containment and Internal Structures

   The mathematical model for the Containment and Internal Structures consists of lumped masses, elastic properties, and viscous dashpots in discrete parts. The number of masses and associated degrees of freedom is kept to a minimum to reduce unnecessary complexity.

   Plan and elevation views of the Containment and Internal Structures are shown on Figures 3.7B-23 and 3.7B-24. The corresponding mathematical models are shown on Figure 3.7B-34. The coordinates of the lumped masses are listed in Table 3.7B-9. The values of the masses and mass moments of inertia associated with the lumped masses of the structures and foundation are presented in Table 3.7B-14. The degrees of freedom assigned to the lumped masses of the structures are identified in Table 3.7B-19.

   The dome and the cylinder of the Containment Structure are simulated by five masses. The Internal Structure, made up of concrete walls, columns, slabs, and beams, is represented by three masses lumped at floor levels. The mat is simulated by a single mass. Each mass is assumed to have six dynamic degrees of freedom, namely, three translations in the three orthogonal principal directions and three rotations about these three axes.

   Two parametric analyses are performed for the Containment Structure where the full thickness (uncracked) and the half thickness (cracked) of the cylindrical wall are considered to be effective.

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Foundation-structure interaction, such as the foundation spring constants, damping ratios associated with each foundation spring, and effective masses of the foundation material, is based on the theory of a circular base resting on elastic half space as described in Subsection 3.7B.2.4 [1], [2], [23], [32], [37]. These values are determined for all six degrees of freedom of the global orthogonal system. The effect of the embedment is also evaluated. Estimated values, upper bound values, and lower bound values of foundation spring constants used in the parametric analyses described in Subsection 3.7B.2.4 are presented in Table 3.7B-24.

The stiffness or the flexibility matrices of the Containment Structure and Internal Structure are generated using suitable computer programs based on finite element techniques. The Containment Structure wall and dome and Internal Structure walls, columns, and floors are modeled with finite elements. The stiffness matrix which corresponds to the finite element model is then reduced to the number of dynamic degrees of freedom required for the dynamic analysis [3], [23], [30], [38].

Past experience shows that the mathematical model described previously conservatively predicts the dynamic behavior of the actual structure subjected to seismic disturbances.

68 2. Models for the Safeguards Building, Electrical and Auxiliary Buildings, Fuel Building, Service Water Intake Structure and Category I Tanks.

68 The mathematical models for these structures are comprised of lumped masses, elastic properties, and dashpots in discrete parts.

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The locations of the mass points are chosen at floor levels and points considered of critical interest, such as equipment support levels. Because the structures cannot be considered symmetrical and the torsional modes of vibration can be excited by ground motions, each mass is assumed to have six degrees of freedom, namely, three translations in the orthogonal principal directions and three rotations about the three principal axes which account for the torsional and rocking modes of vibration.

Plan and elevation views of the Safeguards Building, Electrical and Auxiliary Buildings, Fuel Building, and Service Water Intake Structure are shown on Figures 3.7B-25 through 3.7B-33. The corresponding mathematical models are shown on Figures 3.7B-35 through 3.7B-38. The coordinates of the lumped masses are listed in Tables 3.7B-10 through 3.7B-13. The values of the masses and mass moments of inertia associated with the lumped masses of the structure and foundation are presented in Tables 3.7B-15 through 3.7B-18. The degrees of freedom assigned to the lumped masses of the structures are identified in Tables 3.7B-20 through 3.7B-23.

Foundation spring constants associated with three orthogonal 68
principal translations and two rocking motions about the two
horizontal orthogonal axes are determined on the basis of a
rectangular or circular base resting on an elastic half space [1],
[2], [23], [37]. Torsional foundation spring constants, damping
ratios, and effective masses and rotary inertias for foundation
below the vibrating mat associated with the foundation springs
are determined on the basis of the equivalent radius for the
rectangular base of dimensions 2c by 2d using the theory of the
elastic half space for a circular footing according to Subsection
3.7B.2.4 [1], [2], [23], [32], [37]. The effects of the
embedment of the structures are evaluated and taken into
consideration in the analysis. Values of foundation spring 68
constants used in the parametric analyses described in Subsection
3.7B.2.4 are presented in Tables 3.7B-25 through 3.7B-29.

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68 The stiffness matrices of the buildings are generated using suitable computer programs based on finite element techniques or on stiffness properties calculated for the structural elements between lumped mass elevations. For unsymmetric structures the stiffness matrices include the effects of torsional rigidities of shear wall assemblies between floors. The stiffness matrices obtained for finite element models are reduced to conform to the number of degrees-of-freedom of the dynamic models which are used in the dynamic analysis [3], [23], [30], [38].

3.7B.2.2 Natural Frequencies and Response Loads

68 The natural frequencies and modal participation factors for all modes resulting from the parametric analyses of representative seismic Category I structures are presented in Tables 3.7B-30 through 3.7B-45. Response loads for these structures obtained by the square root of the sum of the squares method (SRSS) are summarized in Tables 3.7B-46 through 3.7B-50 in the form of modal accelerations.

For comparison, envelope values of time history analysis results for the Electrical and Auxiliary Buildings and for the Fuel Building are presented in Tables 3.7B-51 and 3.7B-52.

Seismic loads used for the design of seismic Category I structures are obtained by multiplying the response accelerations with the appropriate masses.

Response spectra at all floors are developed for all seismic Category I structures as indicated in Subsection 3.7B.2.5.

3.7B.2.3 Procedure Used for Modeling

The structures and their contents possess mass which contributes to the inertia loading of the structure. The complexity of the spatial distribution makes it necessary to concentrate the mass at characteristic points or nodes. These points are selected so that they coincide with concentrations of mass, e.g., at the floors, or with locations which are important for stiffness. In some instances, the nodes are selected at intermediate points of structures and equipment that can be regarded as being of uniform construction. This discretization into characteristic points permits a more accurate prediction of the dynamic behavior of actual structures and equipment.

At each node, the structure or system is given six degrees of freedom (three translation components and three rotation components).

No simplifications aimed at reducing the total number of degrees-of-freedom considered in the analysis are made. All six degrees-of-freedom of each node are treated as generalized displacements for all seismic Category I structures.

The idealization of the mass is performed on the basis of relative displacements. If the horizontal cross-section of the structural component, for example, does not deform significantly, and the contents undergo essentially the same displacement as the structure, all mass in a given place can be represented by a point mass placed at the centroid.

It is not feasible to formulate a mathematical model which would include, in addition to the primary structure, all of the equipment, piping systems, and other lightweight structures. These subsystems are therefore uncoupled from the primary structures and are analyzed by the response spectrum approach procedure. In order to use the spectrum analysis for secondary systems, floor response spectra are developed as described in Subsection 3.7B.2.5.

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The criteria employed for system/subsystem decoupling are consistent with the provisions of USNRC Standard Review Plan, Subsection 3.7.2. | 2

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2 June 1975. They are based on the mass ratio, R_m of the supported subsystem mass to the corresponding support mass, and the frequency ratio, R_f of the supported subsystem fundamental frequency to the corresponding supporting system dominant frequency such that:

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2 1. If $R_m < 0.01$, decoupling can be done for any R_f

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2 2. If $0.01 < R_m < 0.1$, decoupling can be done if $R_f < 0.8$ or $R_f > 1.25$

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2 3. If $R_m > 0.1$, an approximate model of the subsystem should be included in the primary system model.

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2 where:

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2
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$$R_m = \frac{\text{Total mass of supported subsystem}}{\text{Mass of supporting structure}}$$

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2
2

$$R_f = \frac{\text{Fundamental frequency of the supported system}}{\text{Frequency of the dominant support motion}}$$

The floor response spectra are generated using the mathematical models which consist of the lumped masses computed from tributary structure dead loads, a portion of live loads, and fixed equipment loads. In some cases, the uncoupled mathematical models, with lumped masses representing the equipment, include the effective masses and flexibility of the supporting structure.

3.7B.2.4 Soil-Structure Interaction

The mathematical model for performing the dynamic analysis of seismic Category I structures supported on the ground is comprised of lumped masses and elastic properties in discrete parts. Because these structures are founded on sound bedrock (Glen Rose Limestone) with

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shear wave velocities of 5500 to 6000 ft/sec, the foundation-structure interaction is evaluated using the conventional elastic half-space theory in accordance with References [1], [2], [23], [32], and [37]. The justification for the use of this theory is based on the fact that sound bedrock is much closer to being a truly elastic material than any other common foundation material. Using the half-space theory, foundation spring constants with associated effective masses of the rock and damping ratios caused by radiation damping are determined.

Principles of radiation damping apply more correctly for this case than for foundations on other materials. The radiation damping is associated with the energy which is carried away from the immediate vicinity of the foundation by stress waves. In addition to the radiation damping, the internal damping is also determined. The internal damping within the foundation material arises primarily because of a nonlinear effect known as hysteresis; i.e., the stress-strain relationship during unloading is not the same as that during loading. The combined effects of radiation and internal damping are evaluated in accordance with References [32] and [37]. However, damping values used in the analysis are limited to a maximum of 10 percent associated with horizontal and vertical translational foundation springs and five percent for rocking and torsional foundation springs.

Because of the possible variations in measuring the properties of the foundation material, parametric studies are performed on the basis of the best estimate as well as the upper and lower bound estimates of the foundation spring constants with the exception of the Fuel Building and Service Water Intake Structure where only lower bound and upper bound values are used. The range of foundation spring constants considered in the parametric studies accounts for the variations in the properties of the foundation material encountered at the site, as well as the anticipated variations in measuring these properties. These variations are considered to be 10 percent in measuring the mass density and 15 percent in measuring the shear-wave velocity. 52

3.7B.2.4.1 Evaluation of Soil-Structure Interaction Parameters

The foundation-structure interaction in the mathematical models represented by foundation spring constants, associated damping factors, and effective foundation material masses and rotary inertias (in general, corresponding to all six degrees-of-freedom) is evaluated from the theory of a rigid base resting on an elastic half space in accordance with References [1], [2], [23], [32], and [37]. The solution of this problem was originally given by Boussinesq [8].

Based on the preceding theory, the foundation spring constants are obtained from Tables 3.7B-5 and 3.7B-6 in conjunction with Figure 3.7B-39 for rigid circular and rectangular bases resting on the surface of an elastic half-space. An approximate evaluation is made by using an equivalent circular base having the same second moment about its centroidal axis for the contact area. On this basis, the equivalent radius for the rectangular base of dimensions 2c by 2d is determined by the following equation:

$$r_o = \sqrt[4]{\frac{8cd(c^2+d^2)}{3\pi}} \qquad (3.7B\ 1)$$

The determination of radiation damping (see Table 3.7B-7), occasionally called geometric damping, caused by foundation structure interaction is also based on vibrations of a rigid circular base on the elastic half-space.

The internal damping caused by energy loss during stress reversals is evaluated on an empirical basis.

Combined effects of radiation and internal damping are taken into account by direct addition of the two values of damping. For rotational motions, the radiation damping is low, and the internal damping can be a significant part of the total damping. However, for translational motions, radiation damping is much greater than internal damping.

There are practical reasons why the damping shown by the theory may not be realistic in actual mass foundation systems. Therefore, damping values that are used in the dynamic analysis are limited to the maximum values set forth in Subsection 3.7B.2.4.

The effective masses and rotary inertias of the foundation material are estimated on the basis of the theory for the elastic half-space.

Effective mass and mass moment of inertia for foundation material below a vibrating footing are estimated from the following formulas:

$$M_{eff} = \alpha \cdot \rho \cdot r_o^3 \qquad (3.7B\text{-}2)$$

and

$$I_{eff} = \beta \cdot \rho \cdot r_o^5 \qquad (3.7B\text{-}3)$$

where

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M_{eff} denotes the effective mass of the foundation material.

I_{eff} is the effective rotary inertia of the foundation material.

ρ is the mass density of the foundation material.

α and β are the coefficients given by theory and shown in Table 3.7B-8 [2].

For rectangular footings of dimensions 2c by 2d, the equivalent radii are obtained for torsion from Equation 3.7B-1 and for translation and rocking from the following equations:

For translation:

$$r_o = \sqrt{\frac{4cd}{\pi}} \qquad (3.7B-4)$$

For rocking:

$$r_o = \sqrt[4]{\frac{16cd^3}{3\pi}} \qquad (3.7B-5)$$

68 The values of effective foundation masses and mass moments of inertia determined for representative seismic Category I structures are presented in Tables 3.7B-14 through 3.7B-18. They are calculated
52 based upon the assumption that the centers of gravity of the effective soil masses are located on the top of the soil at the centroid of the soil mat contact areas. These values are then added to the lumped mass points representing the building foundations.

Foundation embedment depths, including a description of the foundation medium and its properties, are given in Subsection 3.7B.1.4.

The effect of embedment on foundation rigidities is also determined approximately from the theory of elastic half-space. For example, the torsional rigidity and the rigidity in horizontal translation are determined on the basis of the assumption that the vertical contact area, engaging the elastic half-space, is an imaginary area of rectangular shape whose vertical dimension is twice that of the actual contact area of embedment. The rigidities obtained by the foregoing procedure are divided in half to account for the actual vertical contact area of embedment.

The equations used for evaluating translational and rotational spring constants for embedment are presented in Subsection 3.7B.2.4.2.

The additional nomenclature used in Tables 3.7B-5, 3.7B-6, 3.7B-7, and 3.7B-8 is as follows:

E and G are the modulus of elasticity and the shear modulus of the foundation material, respectively. Nu is the Poisson's ratio of the foundation material.

The shear modulus is obtained from the field-measured, shear-wave velocity C and mass density of the foundation material from the equation

$$G = C_s^2 \cdot \rho \qquad (3.7B\text{-}6)$$

The methods of determining the shear-wave velocity are discussed in Section 2.5.4.4. Strains developed by these methods are in the order of 10^{-5} to 10^{-3} percent. Based on the shear modulus of the competent rock on which the structure is founded (8×10^5 lb/in^2), the estimated strains caused by a seismic event are of the same order of magnitude. In this range of strains, the shear modulus of rock is relatively independent of the strain levels.

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52 The analysis of all CPSES seismic Category I structures is based upon a rigid mat approach, and the spring constants of the foundation material are determined by means of the formulas listed in Table 3.7B-5 and Table 3.7B-6 and by means of the coefficients in Figure 3.7B-39.

3.7B.2.4.2 Effect of Embedment on Foundation Rigidities

The effect of the embedment on foundation rigidities is based on the conservative assumption that there is no friction between the vertical contact surfaces of the structures and the foundation medium. Therefore, the effect of the embedment on vertical translational rigidity and torsional rigidity, in the case of circular mats and axisymmetric shells of revolution structures, is neglected. The effect of the embedment on rocking rigidities, about the two horizontal axes located at the base, is normally small and is also neglected, unless a significant contribution to the total rocking rigidities due to this cause is expected to play an important role in the analysis and design.

The effect of the embedment on foundation torsional rigidity due to the two opposite, vertical contact surfaces of a structure having a rectangular shape is obtained, according to References [23] and [33], from the following formula:

$$k_\theta = \frac{G}{1-\nu} \beta\psi 4hd^2 \qquad (3.7B\text{-}13)$$

where

h = the depth of the embedment

2d = the length of the contact surface

G = the rigidity modulus

ν = the Poisson's ratio

$\beta\psi$ = the value given in Figure 3.7B-39 for various values of d/h.

For example, for h = 20 ft and d = 100 ft, d/h = 5; this corresponds to $\beta\psi$ which equals 0.9 in Figure 3.7B-39. Then, the foundation torsional rigidity constant is:

$$k_e = \frac{G}{1-\nu} 0.9 \times 4 \times 20 \times 100^2 = 720{,}000 \frac{G}{1-\nu}$$

This torsional rigidity is added to the torsional rigidity associated with the other two opposite, vertical surfaces and the torsional rigidity associated with the horizontal contact surface obtained in accordance with Subsection 3.7B.2.4.

The effect of the embedment on horizontal translational rigidity is given by the following formula:

$$k_x = \frac{G}{1-\nu} \beta_z\sqrt{hd} \qquad (3.7B\text{-}14)$$

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where β_z is the value given in Figure 3.7B-39 for various values of d/h.

Taking the same example as for torsional rigidity, the following horizontal translational rigidity constant is obtained for the value of $\beta_z = 2.5$ resulting from Figure 3.7B-39:

$$k_\psi = \frac{G}{1-\nu} 2 \cdot 5 \sqrt{20 \times 100} = 112 \frac{G}{1-\nu}$$

This horizontal translational rigidity is added to the horizontal translational rigidity corresponding to the horizontal contact surface obtained as presented in Subsection 3.7B.2.4.

A matter of practical interest is that the rigidities in torsion and horizontal translation due to the effects of embedment, obtained on the basis of the elastic half-space theory, are essentially in agreement with the rigidities obtained using References [39] and [40] when the sloping planes of the effective foundation zone are assumed to slope at an angle of 30 degrees. It should be noted that these values are also consistent for practical purposes with the values obtained from Reference [7].

Where the effect of embedment on rocking rigidities plays an important role in the analysis and design of seismic Category I structures, equipment, systems, and components, the rocking rigidity constants for embedment are based on the conservative assumption that the side pressure distribution is uniform due to horizontal load. In this case, the horizontal translational spring constant k_x is obtained from the formula presented in Table 3.7B-6. Then, the rocking rigidity constant due to two opposite vertical contact surfaces of the embedment is obtained using Equation 3.7B-9 of Subsection 3.7B.2.4 as follows:

$$k_\psi = \frac{1}{12} \frac{G}{1-\nu} \beta_z h^2 \sqrt{2hd} \qquad (3.7B\text{-}15)$$

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where β_z is a value given in Figure 3.7B-39 for various values of 2d/h.

For example, the rocking rigidity constant corresponding to a length of embedment 2d = 200 ft and a depth of embedment h = 20 ft (β_z = 2.85) is:

$$k_{\psi} = \frac{1}{12} \frac{G}{1-\nu} 2 \cdot 85 \times 400\sqrt{4,000} = 6000 \frac{G}{1-\nu}$$

It should be noted that this value can be insignificant compared to the rocking rigidity constant associated with the horizontal contact surface of the base with dimensions 2c by 2d. For example, assume that the width of the base mat is 2c = 100 ft (c/d = 0.5); then, the rocking rigidity constant corresponding to the horizontal contact surface is:

$$k_{\psi b} = \frac{G}{1-\nu} 0 \cdot 42 \times 8 \times 100 \times 50^2 = 840,000 \frac{G}{1-\nu}$$

This value is much larger than the value of rocking rigidity constant obtained for the effect of embedment. Therefore, in this case, the effect of the embedment on rocking rigidity is neglected.

In reality, the rocking rigidity constant for embedment is higher in value than the one obtained here. Perhaps a more realistic value can be obtained by assuming that the vertical contact surface of the embedment with the depth h has a mirror image surface with the depth of 2h. Then half of the value for rocking rigidity constant based on the elastic half space theory seems to be more appropriate when the ratio of the actual depth of embedment to the length of embedment is less than unity. For example, using this approach, the following value for rocking rigidity constant for embedment is obtained:

$$k_{\psi} = \frac{G}{1-\nu} 0 \cdot 4 \times 8 \times 100 \times 20^2 \times \frac{1}{2} = 64,000 \frac{G}{1-\nu}$$

Incidentally, this value and the values obtained for the ratios of the depth to the length of embedment less than one are in close agreement with the values obtained on the basis of the approach to the problem for cohesive soils as presented in References [39] and [40]. These values also compare well for practical purposes with the ones obtained using formulation presented in Reference [7].

68 For the dynamic analysis of seismic Category I structures which have relatively shallow depths of embedment (such as the Safeguards, Electrical and Auxiliary, and Fuel buildings), the effect of embedment on rotational foundation rigidities (torsion and rocking) is negligible. The Service Water Intake Structure, which has a greater depth of embedment, is analyzed by including the effects of embedment in both translational and rotational foundation rigidities on the basis of the pressure distribution for a perfectly rigid base on an elastic half-space.

3.7B.2.5 Development of Floor Response Spectra

The methods of seismic analysis are covered in Subsection 3.7B.2.1. The response spectrum method for the development of instructure response spectra is not used.

Instructure response spectra at selected locations of interest are developed on the basis of computed responses to an artificial time history input of ground motion. The time history of the simulated earthquake ground motion is developed to be compatible to the given ground response spectra. Having established the time history of the ground motion, the lumped mass mathematical models of seismic Category I structures are analyzed and time histories at desired masses lumped at floor levels or any other location of interest are generated. Once the time history of the floor motion is obtained, the next step consists of subjecting a single degree-of-freedom system with the natural frequency range of interest and various damping ratios to the floor time history motion. The maximum acceleration responses obtained are then plotted as ordinates and the corresponding natural periods of the single oscillators are plotted as abscissa. The envelope of maximum peaks is used for the construction of instructure response spectra.

In constructing instructure response spectra, uncertainties inherent to the analysis, such as the material properties of the foundation material and the structures, damping values, soil structure interaction, approximations in the modeling techniques, and computation of structure natural frequencies, are accounted for by parametric variations incorporated into the analysis and by broadening of the peaks of the resulting envelope response spectra as described in Subsection 3.7B.2.9.

The procedure of parametric variations consists of evaluating and using in the dynamic analysis lower bound, best estimate, and upper bound values for the foundation spring constants in the case of all seismic Category I structures with the exception of the Fuel Building and the Service Water Intake Structure where only lower bound and upper bound values are used. In addition, the analysis of the Containment Building is performed for each set of foundation spring constants by considering a cracked and an uncracked Containment wall.

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Responses including translational and rotational effects are 68
calculated for each structure at each center of mass, and transferred to the points farthest from the center of mass. Responses in each direction from each of the three directions of earthquake input are calculated separately, and resulting response spectra are combined by the square root of the sum of the squares (SRSS) method to calculate the total response.

44 Typical refined response spectra at critical locations for floor elevation 852.5 ft. of the Safeguards Building and corresponding to 2-percent equipment damping and SSE intensity are shown on Figure 3.7B-50A. Curves Ax, Ay, and Az represent the spectra in the X, Y, and Z directions for the combined effect of the three simultaneous earthquakes. The coupling effects of the nonsymmetric structure are included.

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For certain special subsystems such as the RCL subsystem, response spectra at the exact locations of the subsystems considered (e.g., at the steam generator support or the reactor nozzle) are developed as follows: Floor time histories for the three translational and three rotational degrees-of-freedom and for each earthquake excitation (SSE and OBE) are derived at the nodes corresponding to the floors which contain the selected locations. Response spectra are developed at these nodes by subjecting a single-degree-of-freedom system with the natural frequency range of interest and various damping ratios to the floor time history motions obtained. The response spectra at the selected points are then developed by rigid body transformations.

Figures 3.7B-51, 3.7B-52, and 3.7B-53 represent the response spectra of translational accelerations in three orthogonal directions at the location of the outermost support of the steam generator for two percent equipment damping and for SSE excitations in X, Y, and Z directions, respectively.

3.7B.2.6 Three Components of Earthquake Motion

The three orthogonal components of the design earthquake motion are assumed to act simultaneously. The combined responses (shears, moments, deflections, and so forth) of structures, components, and elements to the simultaneous application of the two horizontal and one vertical ground excitations are obtained by means of the SRSS method because it is considered unlikely that the peak values of the responses from ground excitations in different directions can coincide. This procedure is in conformance with the recommendations of NRC Regulatory Guide 1.92.

3.7B.2.7 Combination of Modal Responses

When the response spectrum concept of analysis is used, only the maximum modal responses are known and the phasing of modes cannot be determined as in the time history analysis. Therefore, the total response at a point in the multi-degree-of-freedom system can only be approximated. The maximum modal responses are normally combined by SRSS, by absolute sum, or by combinations thereof.

The method of combining maximum modal responses is not straightforward. When frequencies of the modes are closely spaced (differences of 10 percent in frequency of the lower mode), the absolute sum procedure of combining the responses in these modes is used. 66

When the absolute sum procedure for combining some of the modal responses is used, the total maximum response is obtained by treating the responses resulting from the absolute sum as pseudomodal responses and combining them with all other modal responses in an SRSS manner. This procedure conforms to the recommendations of NRC Regulatory Guide 1.92. When additional conservatism is desired, the total maximum response is obtained by adding the values of the responses resulting from the absolute sum to the SRSS value of the rest of the modal responses.

3.7B.2.8 Interaction of Non-Category I Structures with Seismic Category I Structures

A number of structures such as the Turbine Building, the Switchgear Buildings, the Circulating Water Intake and Discharge Structures, the Maintenance Building, and the Administration Building are designated as non-Category I.

The only non-Category I structures which are adjacent to any seismic Category I structure are the Turbine Building and the Switchgear Buildings. These structures do not share a common mat with the adjacent seismic Category I structure, and all structures are founded on firm rock. Therefore, there is no possible interaction of non-Category I structures with seismic Category I structures resulting from seismic motion. Sufficient space is provided between the Turbine and Switchgear Buildings and the adjacent seismic Category I structure so as to prevent contact because of deformations occurring in the structures during a seismic event.

The possibility of structural failure during a seismic event is considered for the Turbine Building. Structural failure in the direction of the adjacent seismic Category I structure is prevented by anchoring the Turbine Building to the turbine generator pedestals. The Switchgear Buildings are design to withstand a seismic event equal to the SSE.

54 Non-Category I equipment and components located in seismic Category I buildings are investigated by analysis or testing, or both, to ensure that under the prescribed earthquake loading, structural integrity is maintained, or the non-Category I equipment and components do not adversely affect the integrity or operability, or both, of any

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designated seismic Category I structure, equipment, or component to | 54
the extent that these seismic Category I items cannot perform their
required functions.

3.7B.2.9 Effects of Parameter Variations on Floor Response Spectra

The instructure response spectra are developed using the time history of the instructure motion resulting from the ground motion time history. The instructure response is evaluated by performing time history modal analysis on a lumped-mass mathematical model which simulates the structure and foundation-structure interaction. Because of the uncertainties associated with the energy dissipation, the variation in elastic properties of both structure and foundation, the idealization of structure with lumped masses and elastic properties in discrete parts, and the frequency content and amplitude modulation of simulated ground motion, the free vibration characteristics and the response of structures can only be approximately computed. Parametric studies and conservative assumptions are made to take into account these uncertainties in construction of instructure response spectra.

The amplifications at resonance peaks of the instructure response spectra are generally not sensitive to frequency-shifting because the eigenvalues and eigenvectors of the structure do not change appreciably with small to moderate changes of mass or flexibility and the amplification region of the free field is very wide at dominant structure modes.

Damping factors play an important role in determining amplification of the structure; the smaller the damping, the greater the amplification. Therefore, the damping values are determined conservatively as discussed and presented in Subsections 3.7B.1.3, 3.7B.2.4, and
3.7B.2.15. The conservatism of the amplification is also reflected in | 68
the chosen artificial ground motion.

68 The enveloping technique used for the construction of instructure response spectra consists of enveloping the maximum peaks. Since the frequencies of the structures can only be computed approximately because of the linear and nonlinear deformability, the energy dissipation, variation in elastic properties of both structure and foundation, and the idealization of structure with lumped masses and elastic properties in discrete parts, parametric studies are performed in order to take into account these effects for the construction of instructure response spectra. These effects result in the shifting of the resonance peaks of the instructure response spectra. The peaks are widened by at least 10 percent of the resonance frequencies to account for these effects. The widening exceeds 10 percent if the parametric studies indicate that such widening is necessary to achieve conservative results. These spectra are generated on the basis of an artificial time history ground motion with maximum horizontal ground accelerations normalized to 12 and 6 percent of gravity for SSE and OBE, respectively, and maximum vertical ground accelerations normalized to 8 and 4 percent of gravity for SSE and OBE, respectively.

68 The preceding analysis are accomplished by using suitable computer programs as presented in Section 3.7B(A) and in accordance with References [30], [31], [36], and [38].

3.7B.2.10 Use of Constant Vertical Static Factors

Constant static factors such as vertical response loads for the seismic design of seismic Category I structures, systems, and components are not used. Instead, multimass dynamic analysis for both horizontal and vertical directions of excitation is performed as described in Subsection 3.7B.2.1.

3.7B.2.11 Method Used to Account for Torsional Effects

The methods of seismic analysis of structures are presented in Subsection 3.7B.2.1. The torsional effects on structures depend on the geometric configuration of the structure, the elastic properties of the material of the structure, and the foundation structure interaction.

Torsional effects in unsymmetric structures are induced primarily because the centroids of masses which simulate floor slabs and portions of the shear walls between the adjacent floors do not coincide with the shear centers of the shear wall assemblies.

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A finite element computer program is used for the seismic Category I | 68
structures except the Service Water Intake Structure and seismic |
Category I Tanks. The structure walls and floors are modeled with |
finite elements. The stiffness matrix corresponding to the finite
element model is reduced to the number of dynamic degrees-of-freedom
required for the dynamic analysis in accordance with References [3],
[23], and [38]. For the Service Water Intake Structure and the | 68
seismic Category I tanks, stiffness properties are calculated for the |
structural elements between lumped mass elevations using standard |
structural techniques. Static factors are not used nor are any other |
approximate methods used in lieu of a combined vertical, horizontal,
and torsional system dynamic analysis to account for torsional
accelerations in the seismic design of seismic Category I structures.

For seismic Category I structures, the mathematical model is normally composed of masses lumped at floor levels. Each mass is assumed to have six degrees-of-freedom; namely, three translations in the three orthogonal directions and three rotations about these axes which account for the rotational modes of vibration. Torsional rigidities of shear wall assemblies between the floors are determined and taken into account in the analysis. Torsional spring constants of the foundation, associated damping ratios, and effective inertia of the foundation are determined on the basis presented in Subsection 3.7B.2.4.

3.7B.2.12 Comparison of Responses

The results of the response spectrum analysis are used for the design of all seismic Category I structures. The procedure for combining modal responses is presented in Subsection 3.7B.2.7.

The time history method of analysis is used to generate time histories of the instructure support motion at selected critical locations of interest. The maximum responses attained from the time history analysis are compared with the ones resulting from the spectrum analysis. This comparison is used to check both analyses and to verify the conservatism of the procedure for combining modal responses in the spectrum analysis. Responses at selected points in the seismic Category I structures resulting from both response spectrum concept and the time history analysis are listed in Tables 3.7B-46 through 3.7B-52.

3.7B.2.13 Methods for Seismic Analysis of Dams

The Safe Shutdown Impoundment (SSI) Dam impounds water which serves as the ultimate heat sink for the power plant.

The SSI Dam is analyzed using a two-dimensional finite element model in which the dam is assumed to behave elastically and the elastic continuum is modeled as an assemblage of discrete elements connected at a set of common nodal points [25]. The elastic properties are selected based on predictions of the strain distribution in the dam during an earthquake.

Hydro-dynamic forces are included, using one-dimensional elements to model the water [26]. The effect of dam-foundation interaction is studied by extending two-dimensional elements into the foundation. Vertical as well as horizontal ground motion is considered.

The finite element representation yields the stiffness and mass matrices for the structure. The resulting eigenvalue problem for undamped free vibration is solved using conventional eigenvalue techniques [27] to give the natural frequencies and mode shapes for the dam. The mode shapes and spectral response curves are used in a conventional modal superposition method of analysis in which damping coefficients are introduced into the various modes.

It should be noted that the main dam is not considered as a seismic Category I structure. Therefore, no seismic analysis of the main dam is performed.

3.7B.2.14 Determination of Seismic Category I Structure Overturning Moments

A description of the dynamic methods and procedures used to determine seismic Category I structure overturning moments is provided in this subsection. A description of the procedures used to account for foundation reactions and vertical earthquake effects is also included.

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All dynamic analyses performed for seismic Category I structures are based on the assumption that the structures can be simulated by mathematical models corresponding to linear elastic lumped systems.

66 | A coefficient of stability C_{st} and an eccentricity e shown on Figure 3.7B-40, which govern the evaluation of the dynamic contribution to the foundation pressure beneath the base, are determined.

66 | The degree of stability is dependent on the coefficient of stability C_{st}, which is defined as the ratio of the resisting moment to the overturning moment taken about the extreme point A of the foundation (see Figure 3.7B-40) as expressed in the following equation:

$$C_{st} = \frac{Wa}{M_o} \qquad (3.7B\text{-}22)$$

where

W = the weight of the entire structure

a = the horizontal distance from the center of gravity of the entire structure to the point of rotation designated as A on Figure 3.7B-40

M_o = the total overturning moment about point A induced by seismic disturbances.

The overturning moment in each significant structure mode is given as:

$$M_i = \sum_{n=1}^{N} M_{i,n} + \sum_{n=1}^{N} H_{i,n} \overline{Y}_n + \sum_{n=1}^{N} V_{i,n} \overline{X}_n \qquad (3.7B\text{-}23)$$

where

N = the total number of nodes in the dynamic model of the structure

$M_{i,n}$ = the modal inertia moment at each node

$H_{i,n}$ = the horizontal modal inertia force at each node

$V_{i,n}$ = the vertical modal inertia force at each node

$\overline{X}_n$ = the horizontal distances from the lines of action of the vertical inertia forces to the point of rotation A

$\overline{Y}_n$ = the vertical distances from the lines of action of the horizontal inertia forces to the point of rotation A

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The total overturning moment M_o in the stability Equation 3.7B-22 is obtained from the combined effect of all dominant structure modes. The modal overturning moments for each direction of ground excitation are combined, as discussed in Subsection 3.7B.2.7. The combined overturning moments resulting from the vertical and the two horizontal ground motions are calculated by the SRSS combination. In addition, the overturning moment resulting from the vertical ground motion is combined with the larger of the two overturning moments resulting from the two horizontal ground motions by absolute sum combination. Both results are compared and the stability of the structures is checked based upon the most critical of the two cases. For a conservative evaluation of stability against overturning, the vertical seismic forces acting on the seismic Category I systems are considered to act in an upward direction and thus combined with the lateral forces. | 2

An alternate method for calculating the total overturning moment Mo is as follows: | 68

68 Horizontal and vertical inertia forces are obtained by multiplying the building mass at each story level by the corresponding building acceleration. The building acceleration used considers the three directions of earthquake motion, (i.e., the square root of the sum of the squares of the maximum representative accelerations of the codirectional accelerations caused by each of the three components of earthquake motion). The overturning moment due to horizontal inertia forces is determined by multiplying the individual horizontal inertia force by the vertical distance from it's center of mass to the point of rotation. Similarly, the overturning moment due to vertical inertia is found by multiplying the individual vertical inertia force by the horizontal distance from its center of mass to the point of rotation. The total overturning for the structure, in a particular direction, is found by summing the individual overturning moments due to the horizontal inertia force in the corresponding direction and the overturning moment due to the vertical inertia force.

The criteria used to select an acceptable structural configuration are as follows:

1. The stability coefficient C_{st} is not less than 1.10 for the SSE.

2. Maximum static and dynamic foundation pressures remain within the allowable ultimate capacity.

To calculate dynamic foundation pressures, the resultant reaction R and the eccentricity e shown on Figure 3.7B-40 are first determined. The resultant reaction is a function of the weight of the entire structure and the total vertical seismic load. The total vertical seismic load is determined by the combined effects of all dominant structure modes induced by horizontal and vertical seismic excitations; i.e., the resultant modal vertical earthquake loads for each direction of ground motion and the simultaneous effect of three earthquakes are combined as discussed in Subsection 3.7B.2.7.

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TABLE 3.7B-1

DAMPING VALUES

Item, Equipment, or Structure	Damping, Percent Critical[1] OBE	SSE	
Equipment	2	3	68
			68
Welded steel structures	2	4	
Bolted steel structures	4	7	
Reinforced concrete structures	4	7	
Piping [2]	variable		68

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1 NRC Regulatory Guide 1.61

2 Damping used in piping analysis is as specified in Code Case N-411 of the ASME Boiler and Pressure Vessel Code. | 68

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TABLE 3.7B-2

(Sheet 1 of 2)

METHODS OF SEISMIC ANALYSIS USED FOR SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS

Structure System, or Component	Equivalent Static Load	Response Sepctrum Analysis	Modal Analysis Time History	Remarks	
1. Structures					
Containment and internal structures	X	X	X		
Safeguards Building	X	X	X		
Electrical and Auxiliary Building	X	X	X		
Fuel Building	X	X	X		
Condensate storage tank	X	X	X		68
RWST	X	X	X		68
Reactor makeup Water storage tank	X	X	X		68
Service Water Intake Structure	X	X	X		

TABLE 3.7B-2

(Sheet 2)

METHODS OF SEISMIC ANALYSIS USED FOR SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS

Structure System, or Component	Equivalent Static Load	Response Sepctrum Analysis	Modal Analysis Time History	Remarks
2. Mechanical Components and Systems				
Westinghouse equipment				See Section 3.7N
Other equipment	X	X		
3. Electrical Components and Systems				
Westinghouse equipment				See Section 3.7N
Other equipment	X	X		

TABLE 3.7B-13

SERVICE WATER INTAKE STRUCTURE
MASS POINT NODAL COORDINATES

Mass Point	X (ft)	Y (ft)	Z (ft)	
1	29.65	0	54.86	| 68
2	21.91	20	60.81	| 68
25	21.92	41	55.56	| 68
3	19.96	62	53.36	| 68
				| 68

NOTES:
a. The origin of coordinates is at elevation 776 ft 0 in. | 68
b. For orientation of coordinate axes, see Figure 3.7B-32
c. For structure dynamic model showing the mass point numbers, see Figure 3.7B-38.

TABLE 3.7B-18

SERVICE WATER INTAKE STRUCTURE

MASS POINT MASS DATA

Mass Point	Mass (kip-sec^2/ft) Mx	My	Mz	Mass Moment of Inertia (kip-sec^2/ft) Ix	Iy	Iz	
1	736	1446	736	1.964×10^6	1.308×10^6	5.79×10^5	| 68
2	280	280	280	3.59×10^5	4.17×10^5	1.08×10^5	| 68
25	98	98	98	1.57×10^5	1.83×10^5	$.335 \times 10^5$	| 68
3	89	89	89	0.99×10^5	1.21×10^5	$.25 \times 10^5$	| 68

NOTES:

a. Mass 1 includes the equivalent soil mass. | 68

b. The origin of coordinates is at elevation 776 ft, 0 in. For orientation of coordinate axes, see Figure 3.7B-32. | 68

c. For structure dynamic model showing the mass point numbers, see Figure 3.7B-38.

TABLE 3.7B-23

SERVICE WATER INTAKE STRUCTURE DEGREES OF FREEDOM

Mass Point	Degree of Freedom	Direction of Motion		
1	1	Translation	X	
1	2	Translation	Y	
1	3	Translation	Z	
1	4	Rotation	θ_x	
1	5	Rotation	θ_y	
1	6	Rotation	θ_z	
2	7	Translation	X	
2	8	Translation	Y	
2	9	Translation	Z	
2	10	Rotation	θ_x	
2	11	Rotation	θ_y	
2	12	Rotation	θ_z	
25	13	Translation	X	| 68
25	14	Translation	Y	| 68
25	15	Translation	Z	| 68
25	16	Rotation	θ_x	| 68
25	17	Rotation	θ_y	| 68
25	18	Rotation	θ_z	| 68
3	19	Translation	X	| 68
3	20	Translation	Y	| 68
3	21	Translation	Z	| 68
3	22	Rotation	θ_x	| 68
3	23	Rotation	θ_y	| 68
3	24	Rotation	θ_z	| 68

NOTES:

a. The origin of coordinates is at elevation 776 ft 0 in. | 68

b. For orientation of coordinate axes, see Figure 3.7B-32.

c. For structure dynamic model showing the mass point numbers, see Figure 3.7B-38.

TABLE 3.7B-29

SERVICE WATER INTAKE STRUCTURE

FOUNDATION SPRING CONSTANTS

Direction	Upper Bound	Lower Bound	
Translation along X axis (kip/ft)	5.21×10^{7}	2.60×10^{7}	68
Translation along Y axis (kip/ft)	4.14×10^{7}	2.07×10^{7}	68
Translation along Z axis (kip/ft)	5.21×10^{7}	2.60×10^{7}	68
Rotation along X axis (kip-ft/rad)	1.28×10^{11}	0.64×10^{11}	68
Rotation along Y axis (kip-ft/rad)	1.94×10^{11}	0.97×10^{11}	68
Rotation along Z axis (kip-ft/rad)	7.72×10^{10}	3.86×10^{10}	68

NOTES:

a. The origin of coordinates is at elevation 776 ft 0 in. | 68

b. For orientation of coordinate axes, see Figure 3.7B-32.

c. Spring constants include embedment effects. | 68

TABLE 3.7B-44

SERVICE WATER INTAKE STRUCTURE
MODAL FREQUENCIES AND PARTICIPATION FACTORS
LOWER BOUND, SSE

Mode	Frequency (Hz)	Participation Factors X-Earthquake	Participation Factors Y-Earthquake	Participation Factors Z-Earthquake	
1	11.313	-21.293	-1.791	.213	68
2	14.626	-.5070001	3.043	-25.66	68
3	16.241	-1.043	47.581	1.991	68
4	17.707	-3.707	.337	2.373	68
5	23.975	-28.005	-.264	.01	68
6	24.767	-.155	.998	-22.391	68
7	29.819	-.363	.288	13.161	68
8	33.756	-3.922	-2.451	-.89	68
9	34.974	-9.597	.324	-1.413	68
10	35.444	-3.913	-1.371	1.243	68
11	41.936	-1.891	.861	8.776	68
12	43.327	-3.692	2.796	-2.289	68
13	46.842	-4.94	-.888	.158	68
14	49.277	-1.391	-.141	-3.584	68
15	52.073	-1.658	-.116	.915	68
16	54.557	-.759	.269	-.205	68
17	57.317	-1.173	.076	.382	68
18	59.172	-.04	-.063	-.677	68

TABLE 3.7B-45

SERVICE WATER INTAKE STRUCTURE

MODAL FREQUENCIES AND PARTICIPATION FACTORS

UPPER BOUND, SSE

Mode	Frequency (Hz)	Participation Factors X-Earthquake	Participation Factors Y-Earthquake	Participation Factors Z-Earthquake	
1	11.808	-19.204	-.995	.186	| 68
2	16.07	-.466	1.042	-23.067	| 68
3	18.161	-2.899	-.288	2.952	| 68
4	22.329	-.714	47.003	1.004	| 68
5	27.753	-23.194	-.41	-.318	| 68
6	31.033	-.3	-1.656	21.343	| 68
7	35.528	-3.489	.61	-5.757	| 68
8	35.606	-4.602	2.56	6.371	| 68
9	39.008	-10.54	1.926	-1.433	| 68
10	41.556	-8.903	-7.544	-.517	| 68
11	47.207	-10.72	3.002	6.139	| 68
12	47.285	-3.796	1.4	-14.066	| 68
13	51.626	-12.732	-.8	.785	| 68
14	52.885	-1.152	-.052	-11.36	| 68
15	55.16	-.838	-.038	.066	| 68
16	59.357	-.354	.197	2.053	| 68
17	61.053	-3.046	.465	-.466	| 68
18	66.59	-1.022	.073	.668	| 68

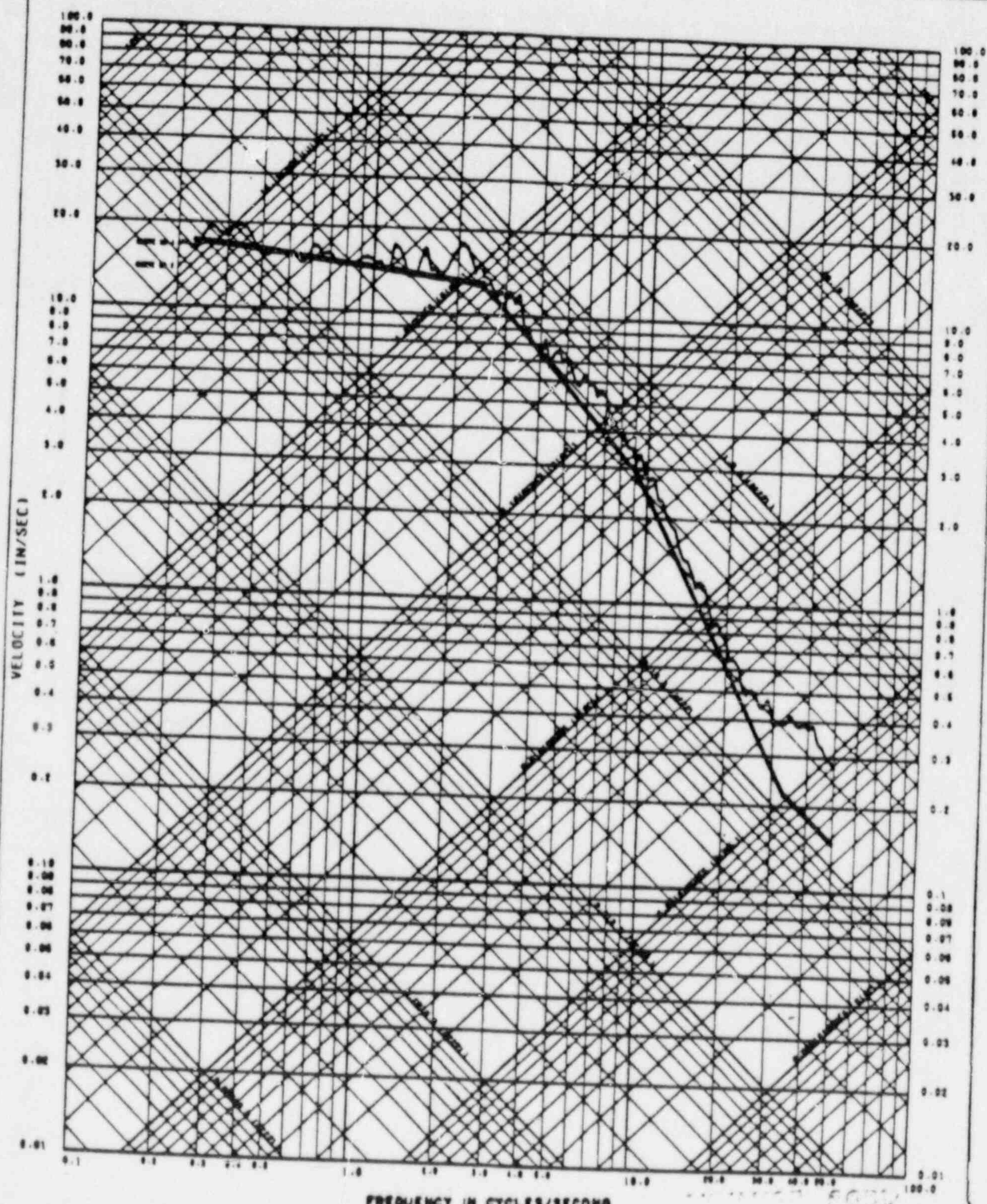
TABLE 3.7B-50

SERVICE WATER INTAKE STRUCTURE
SRSS ACCELERATIONS FOR SSE
FROM RESPONSE SPECTRUM ANALYSIS
UPPER BOUND SOIL CASE | 68

Degree of Freedom[1]	X-Earthquake[2]	Y-Earthquake[2]	Z-Earthquake[2]	
1	2.19	0.41	0.75	68
2	0.52	4.16	0.32	68
3	0.75	0.25	2.01	68
4	0.0136	0.0050	0.0317	68
5	0.0105	0.0019	0.0049	68
6	0.0285	0.0127	0.0187	68
7	4.74	0.33	0.85	68
8	0.32	5.18	0.36	68
9	0.68	0.41	4.26	68
10	0.0127	0.0095	0.0447	68
11	0.0577	0.0120	0.0237	68
12	0.0588	0.0282	0.0340	68
13	7.98	1.14	2.28	68
14	2.09	6.58	1.25	68
15	1.08	0.60	7.29	68
16	0.0302	0.0128	0.0969	68
17	0.0370	0.0033	0.0316	68
18	0.1745	0.0455	0.0748	68
19	12.00	1.03	1.20	68
20	3.05	7.40	2.04	68
21	1.03	0.71	10.25	68
22	0.0440	0.0165	0.1271	68
23	0.0776	0.0126	0.0472	68
24	0.2329	0.0571	0.0973	68

[1] Degrees of freedom are identified in Table 3.7B-23 | 68

[2] Tabulated accelerations are in units of ft/sec./sec. for | 68
transitional degrees of freedom and radians/sec./sec. for | 68
rotational degrees of freedom. | 68



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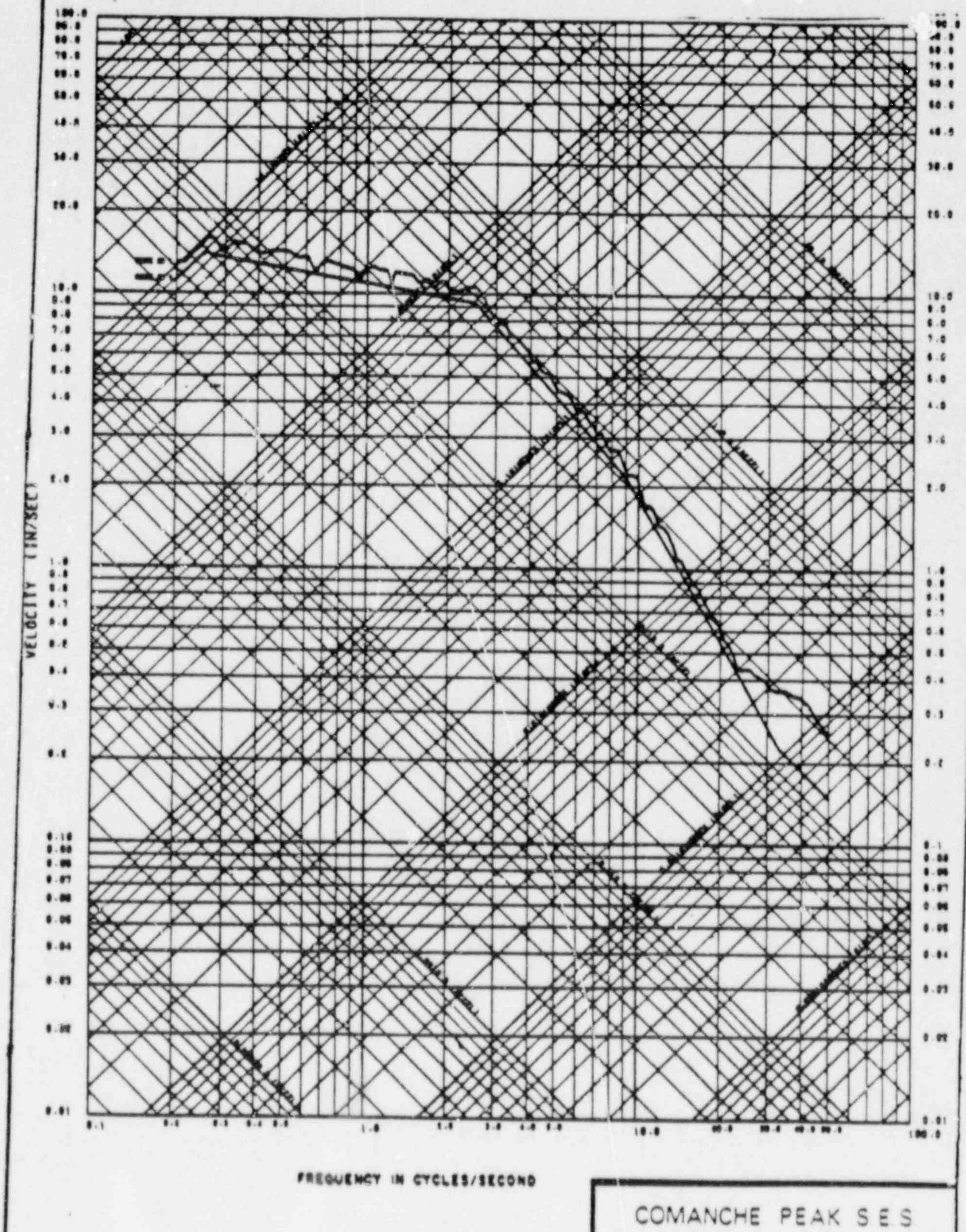
UNITS 1 and 2

HORIZONTAL RESPONSE SPECTRA
SAFE SHUTDOWN EARTHQUAKE
2 PERCENT DAMPING

FIGURE 3.7B-2



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HORIZONTAL RESPONSE SPECTRA
SAFE SHUTDOWN EARTHQUAKE
5 PERCENT DAMPING

FIGURE 3.7B-3

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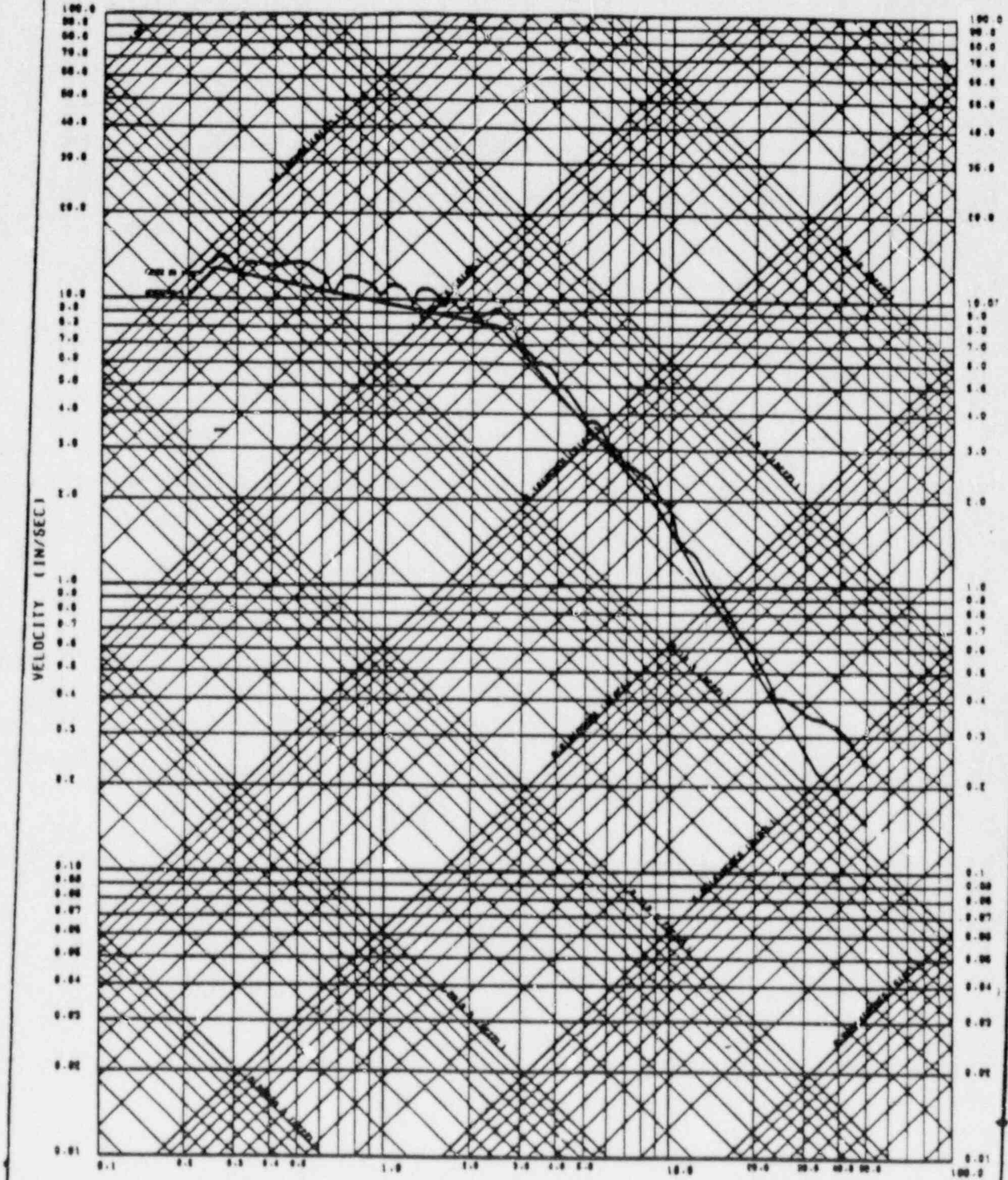

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HORIZONTAL RESPONSE SPECTRA
SAFE SHUTDOWN EARTHQUAKE
7 PERCENT DAMPING

FIGURE 3.7B-4

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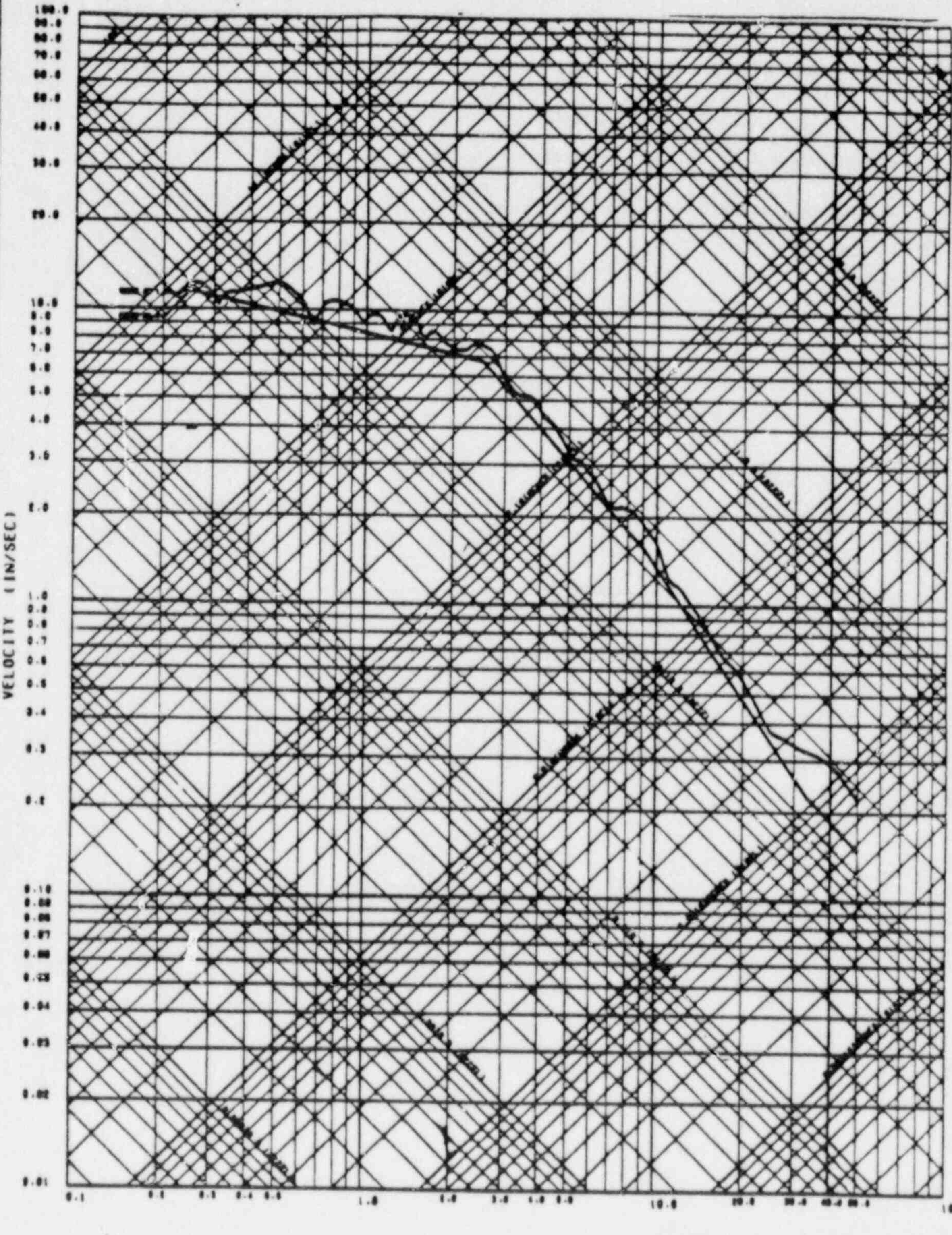

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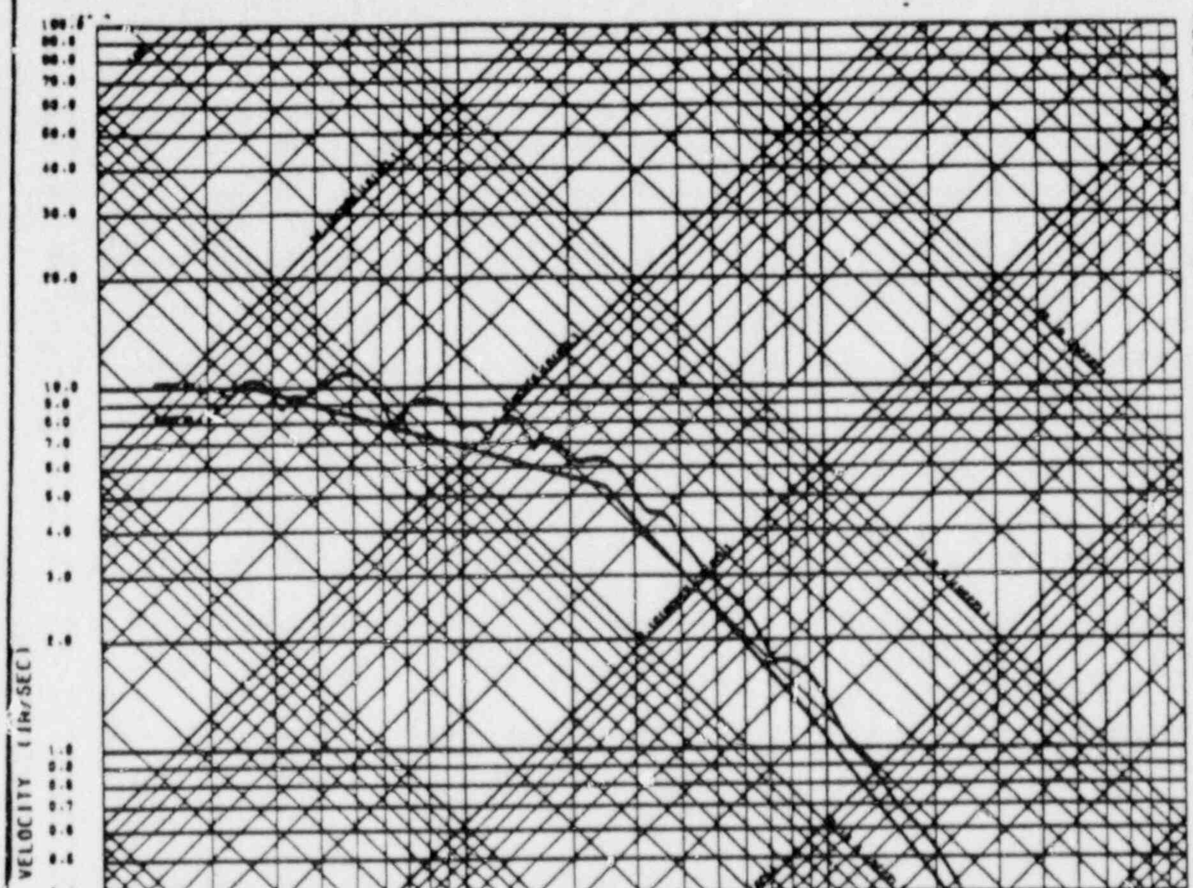
HORIZONTAL RESPONSE SPECTRA
SAFE SHUTDOWN EARTHQUAKE
10 PERCENT DAMPING

FIGURE 3.7B-5



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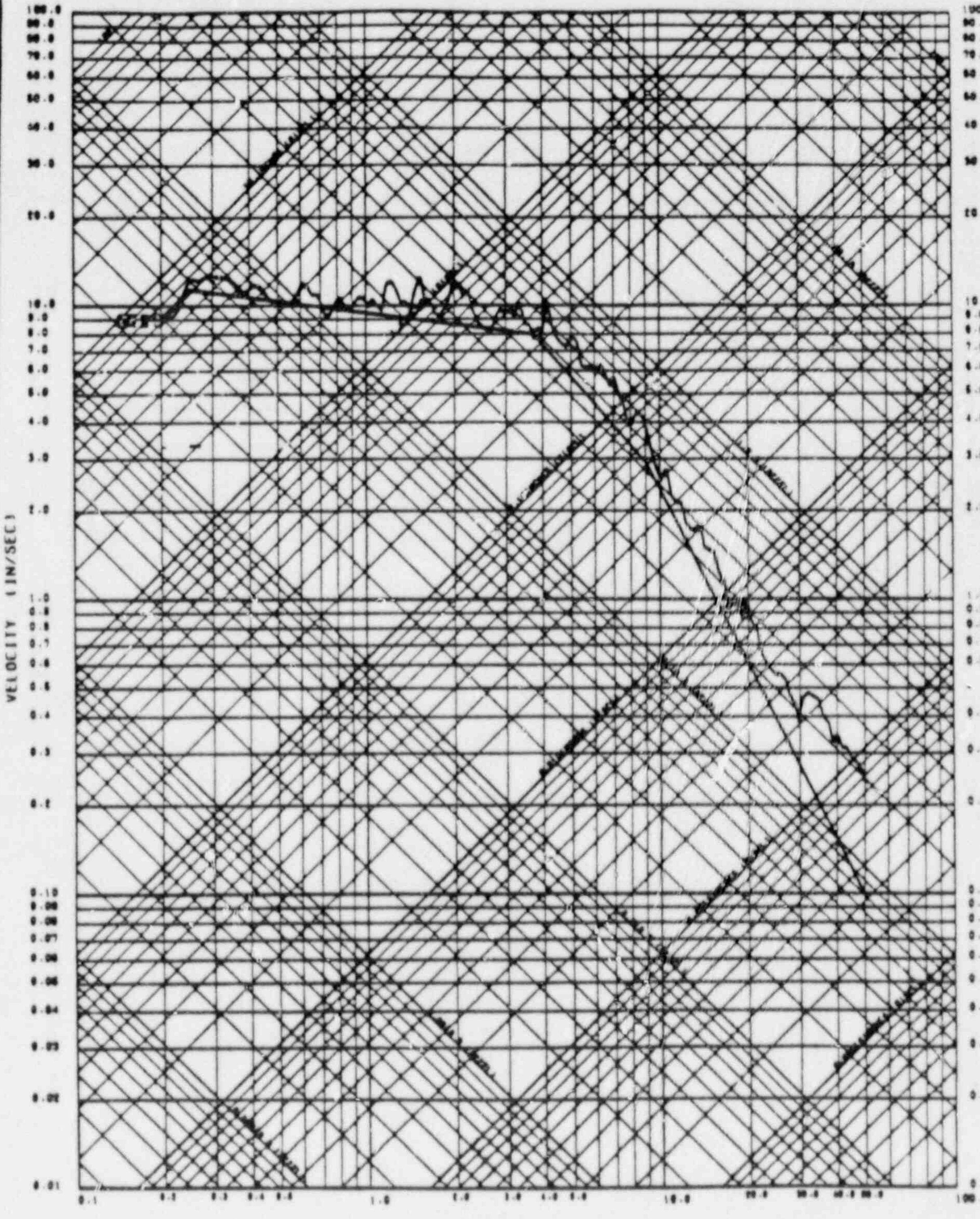
HORIZONTAL RESPONSE SPECTRA
SAFE SHUTDOWN EARTHQUAKE
15 PERCENT DAMPING

FIGURE 3.7B-6



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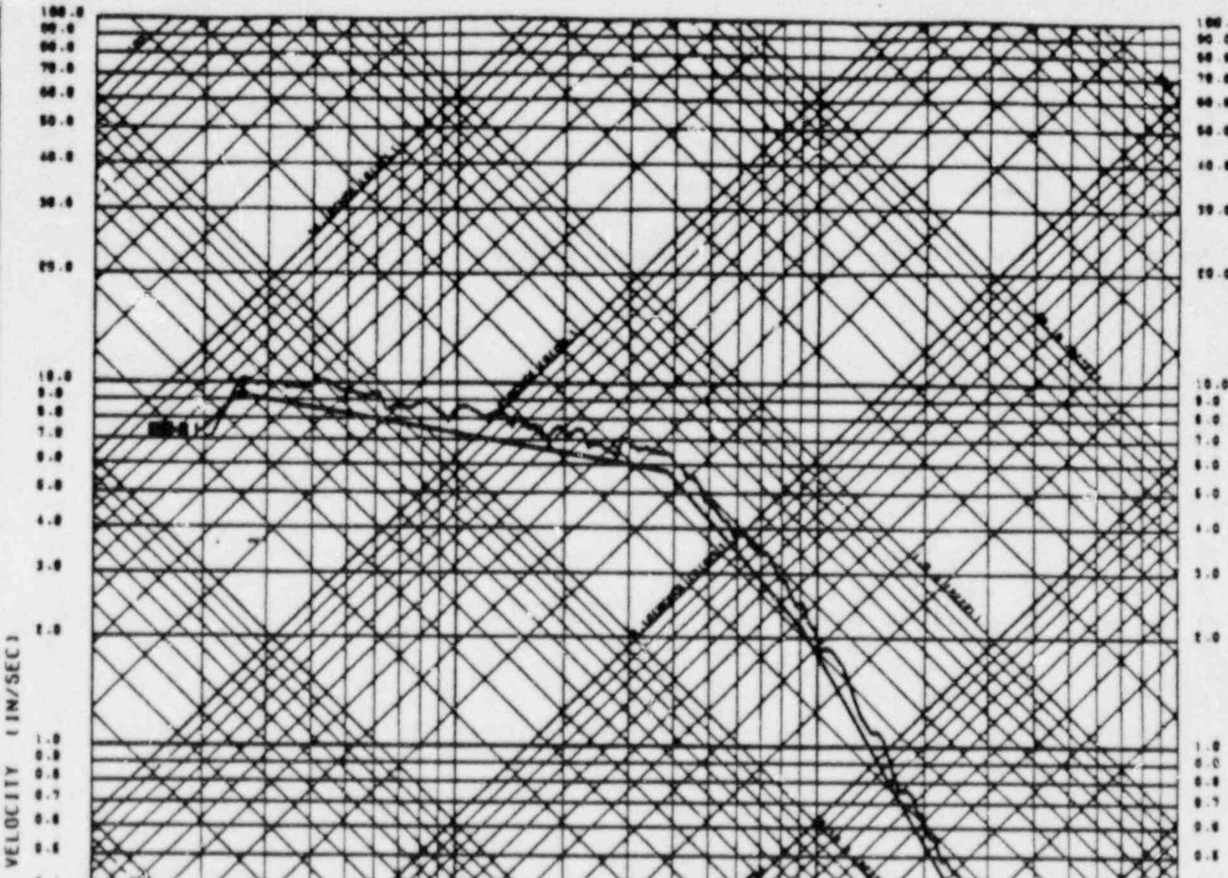
VERTICAL RESPONSE SPECTRA
SAFE SHUTDOWN EARTHQUAKE
2 PERCENT DAMPING

FIGURE 3.7B-8



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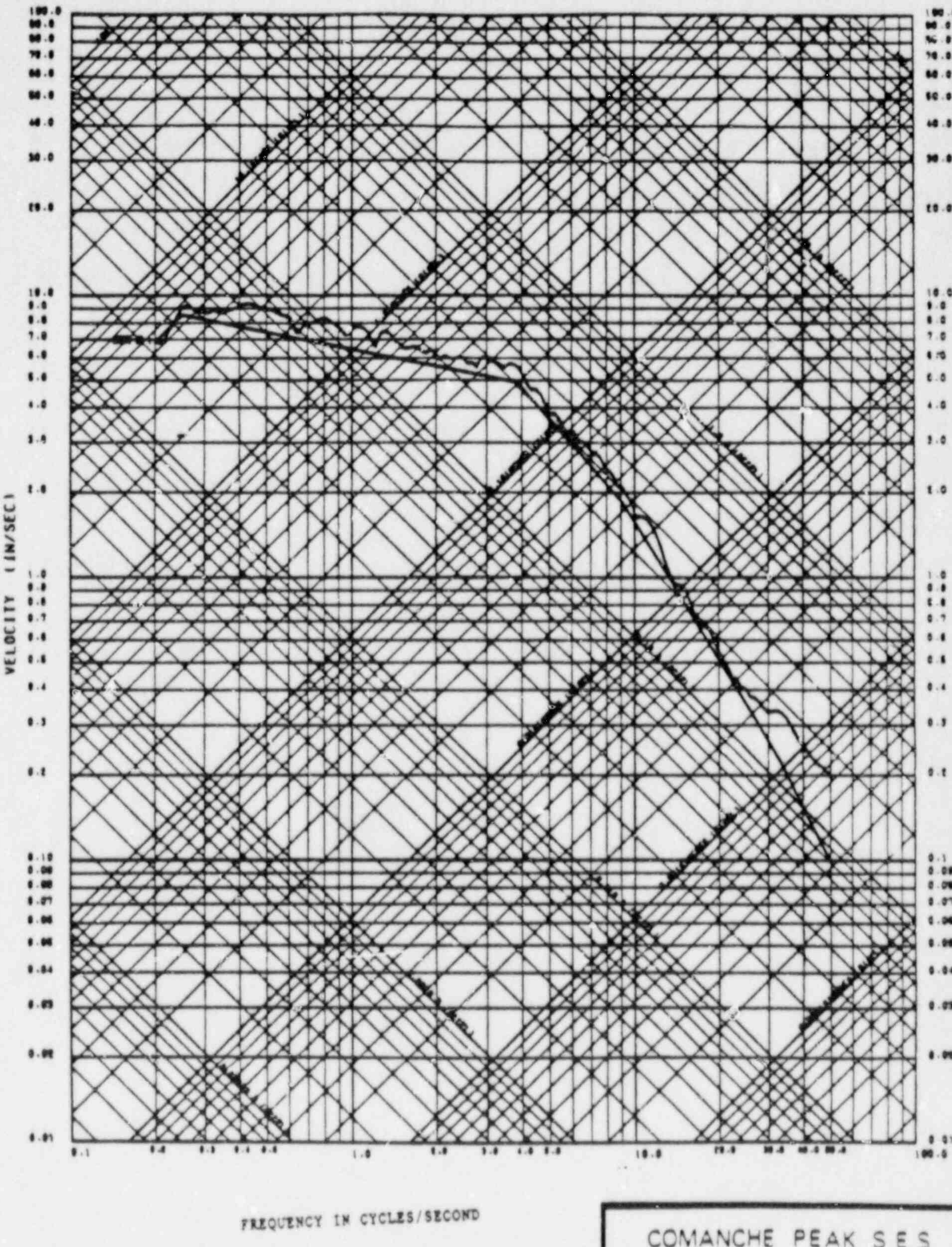
UNITS 1 and 2

VERTICAL RESPONSE SPECTRA
SAFE SHUTDOWN EARTHQUAKE
5 PERCENT DAMPING

FIGURE 3.7B-9



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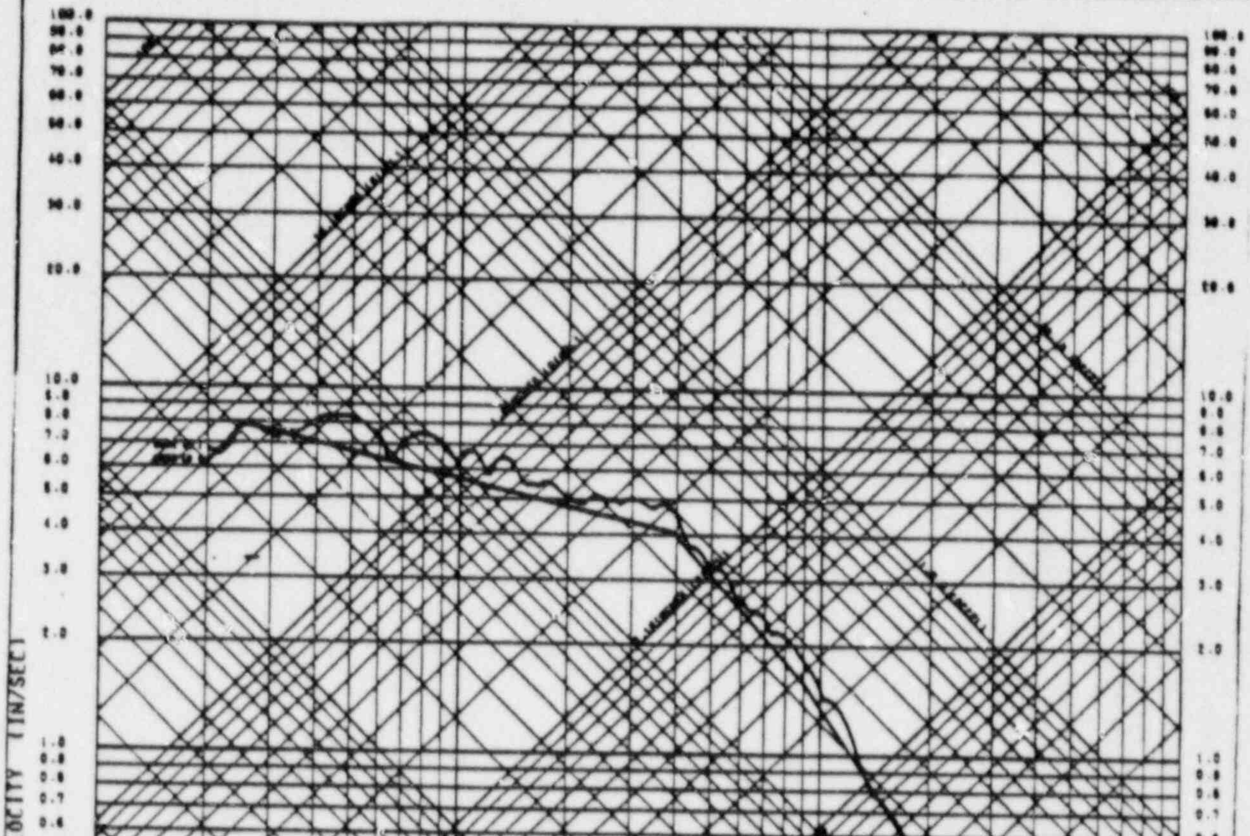
VERTICAL RESPONSE SPECTRA
SAFE SHUTDOWN EARTHQUAKE
7 PERCENT DAMPING

FIGURE 3.7B-10



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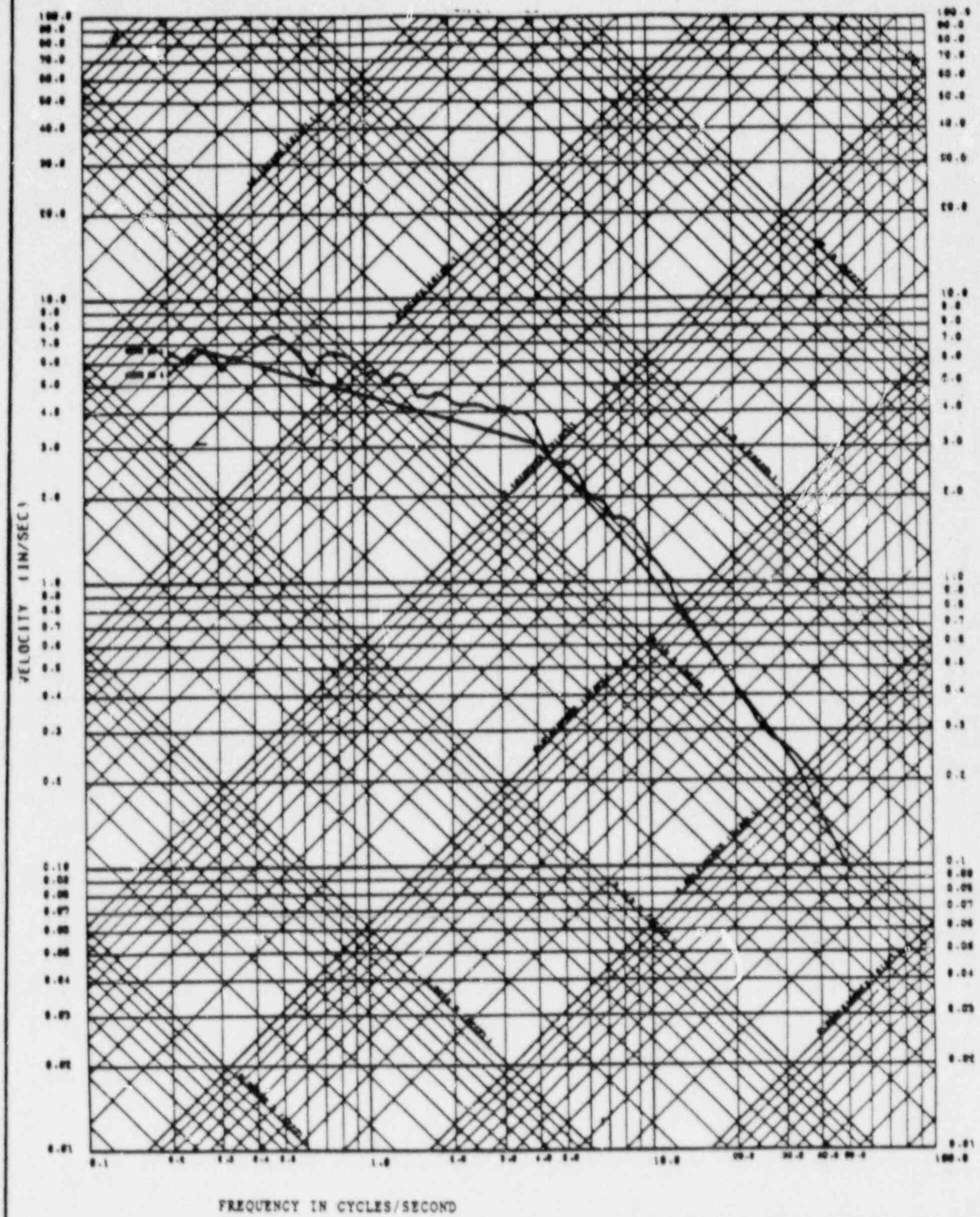
VERTICAL RESPONSE SPECTRA
SAFE SHUTDOWN EARTHQUAKE
10 PERCENT DAMPING

FIGURE 3.7B-11



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VERTICAL RESPONSE SPECTRA
SAFE SHUTDOWN EARTHQUAKE
15 PERCENT DAMPING

FIGURE 3.7B-12

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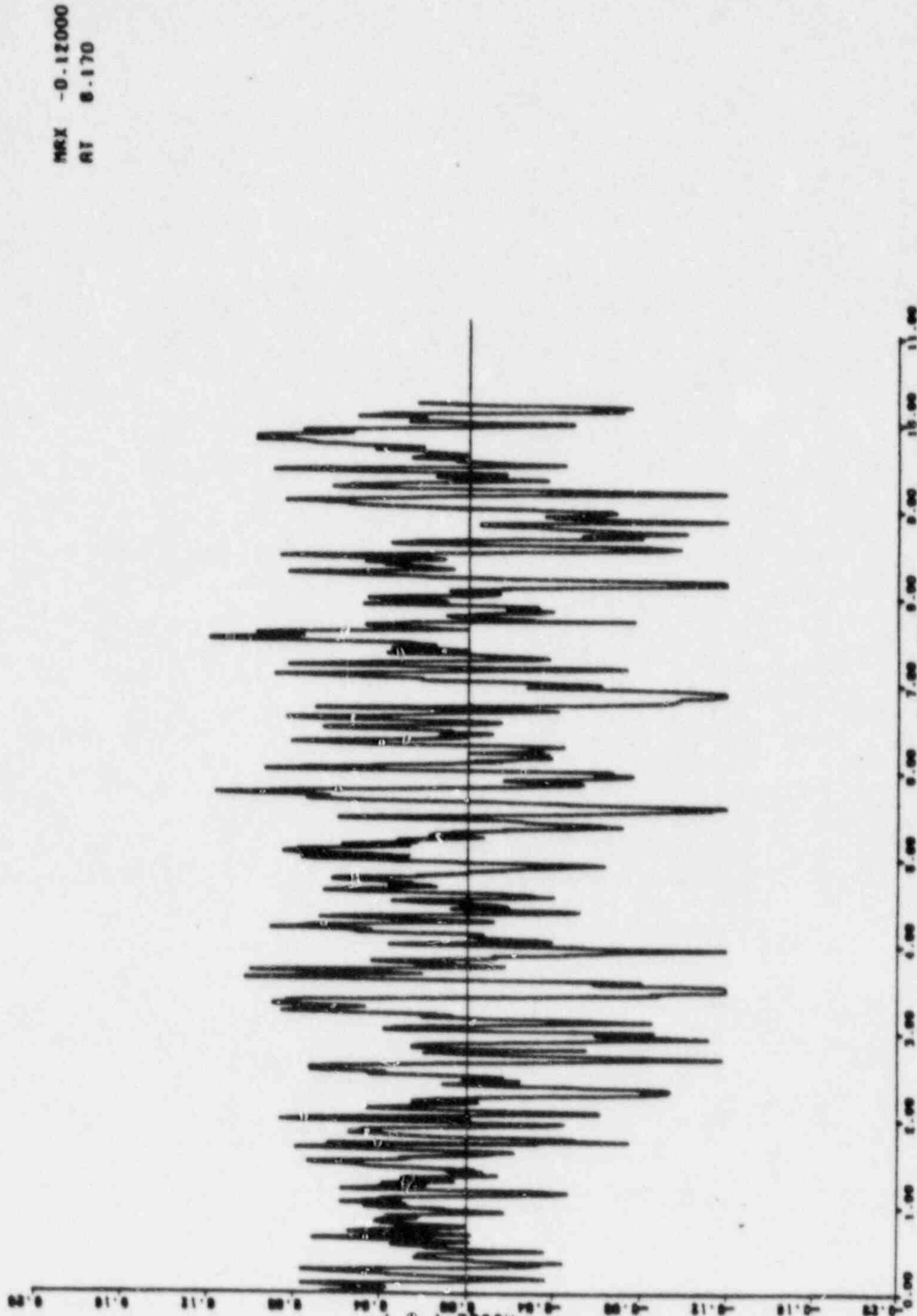
FIGURE 3.7B-13



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HORIZONTAL ARTIFICIAL
ACCELERATION TIME HISTORY

FIGURE 3.7B-14

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FIGURE 3.7B-13

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FIGURE 3.7B-16

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FIGURE 3.7B-17

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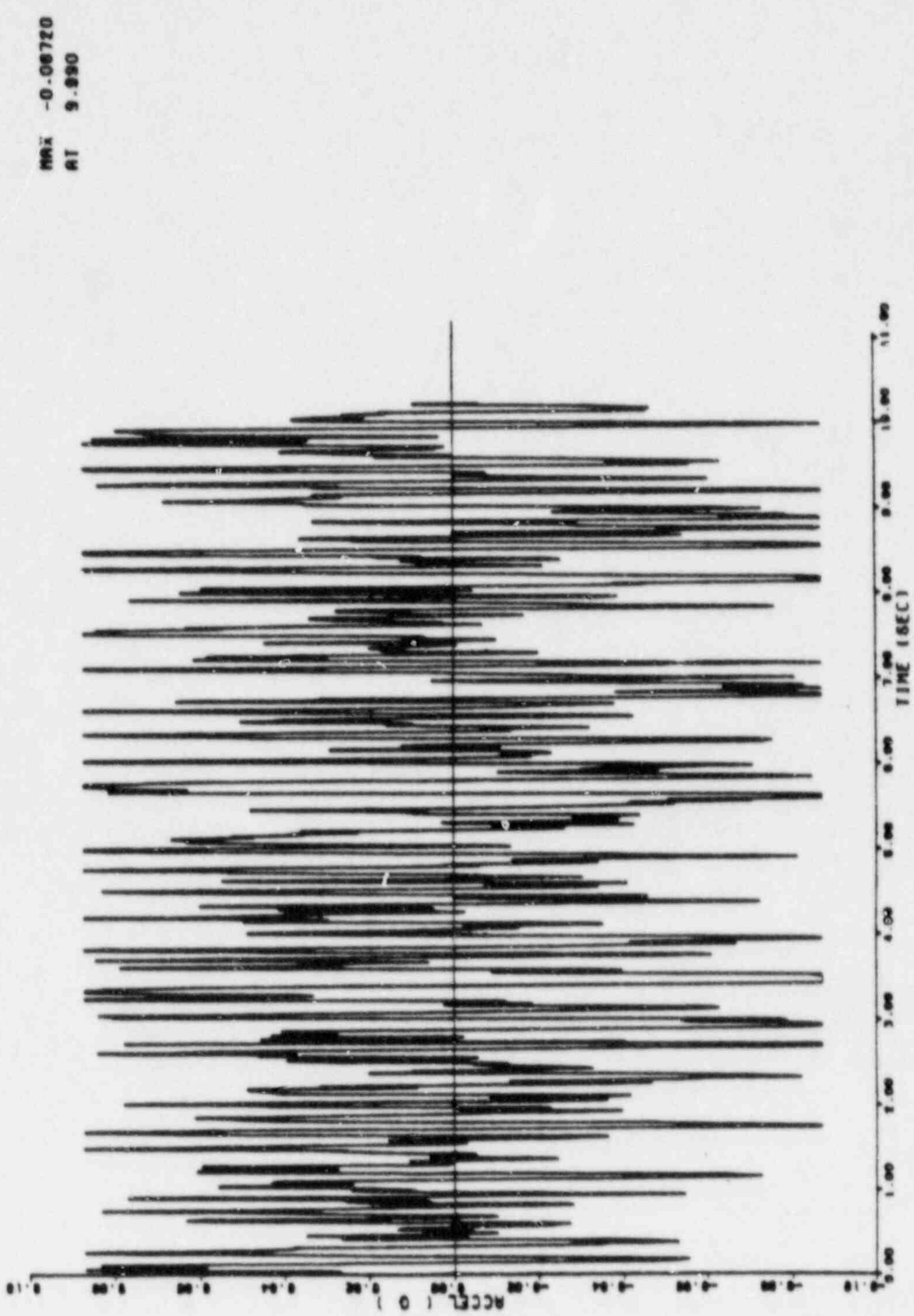
FIGURE 3.7B-18



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VERTICAL ARTIFICIAL
ACCELERATION TIME HISTORY

FIGURE 3.7B-19

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FIGURE 3.7B-20

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FIGURE 3.7B-21

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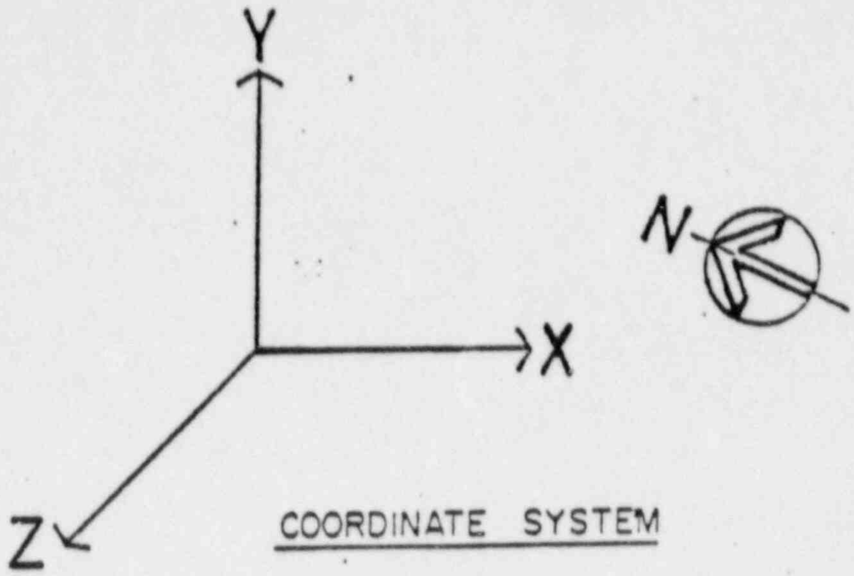
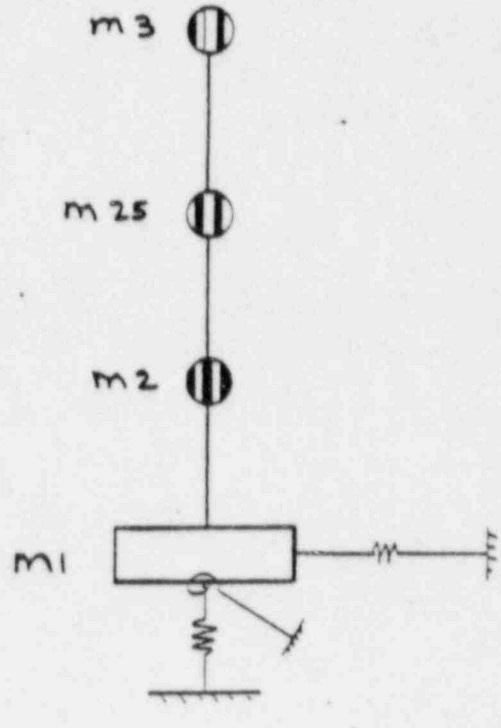
FIGURE 3.7B-22



COORDINATE SYSTEM



DYNAMIC MODEL

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COMANCHE PEAK S.E.S. FINAL SAFETY ANALYSIS REPORT UNITS 1 and 2
DYNAMIC MODEL SERVICE WATER INTAKE STRUCTURE
FIGURE 3.7B-38

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UNITS 1 and 2

FIGURE 1.7B-11

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UNITS 1 and 2

FIGURE 3.7B-42

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FIGURE 1.7B-41

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UNITS 1 and 2

FIGURE 3.7B-44

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UNITS 1 and 2

FIGURE 3.7B-45

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FIGURE 1.7B-46

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UNITS 1 and 2

FIGURE 3.7B-47

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UNITS 1 and 2

FIGURE 3.7B-48

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FIGURE 1.7B-49

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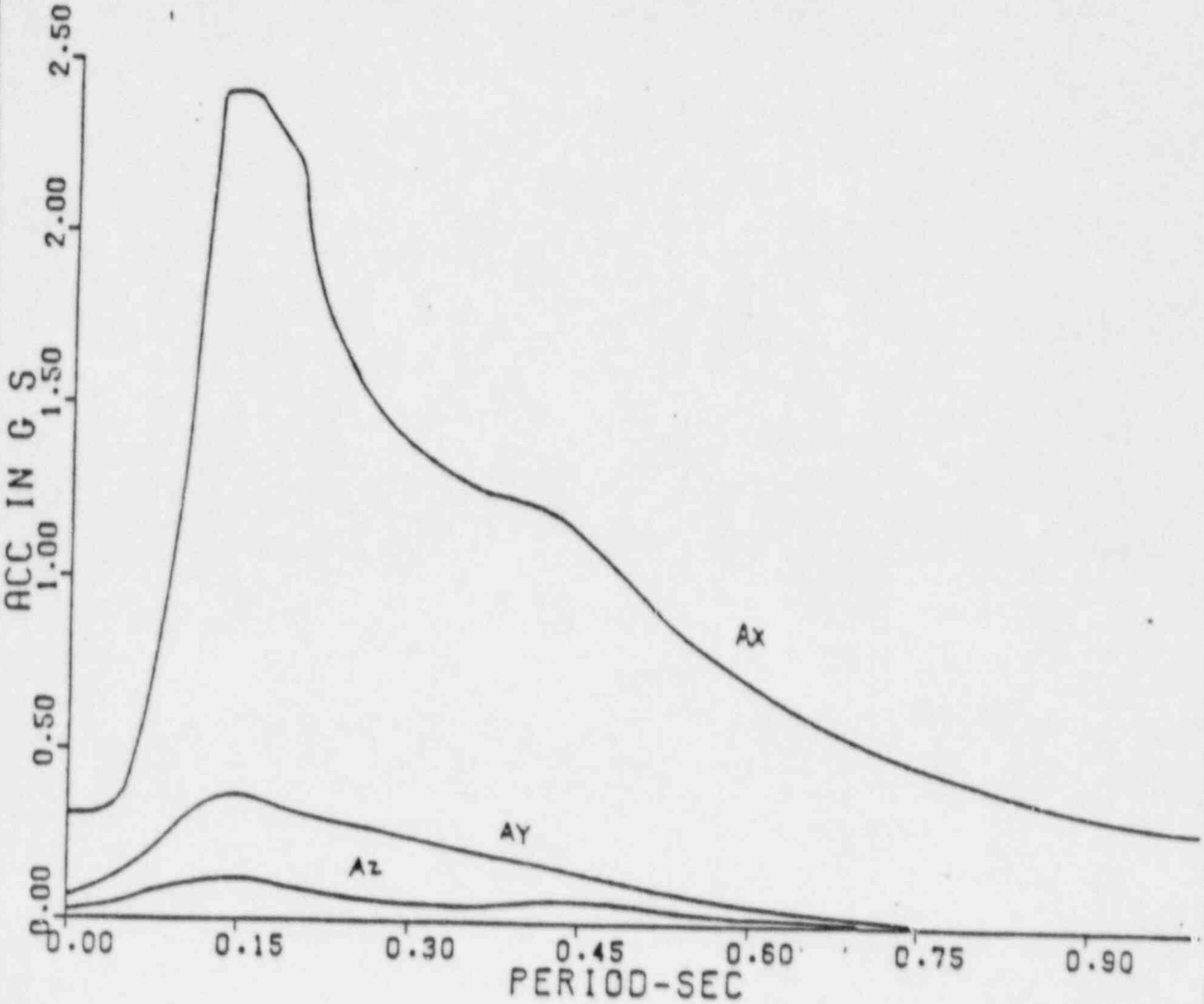
UNITS 1 and 2

FIGURE 3.7B-50

INSTRUCTURE RESPONSE SPECTRA FOR:

STEAM GENERATOR SUPPORT
SSE EARTHQUAKE DIRECTION: X
EQUIPMENT DAMPING: 0.02
RESPONSE SPECTRA DUE TO NODAL TRANSLATION:
AX: N-S AY: VERT. AZ: E-W



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INSTRUCTURE RESPONSE SPECTRA
STEAM GENERATOR SUPPORT
X-EARTHQUAKE
SSE, 2 PERCENT DAMPING

FIGURE 3.7B-51

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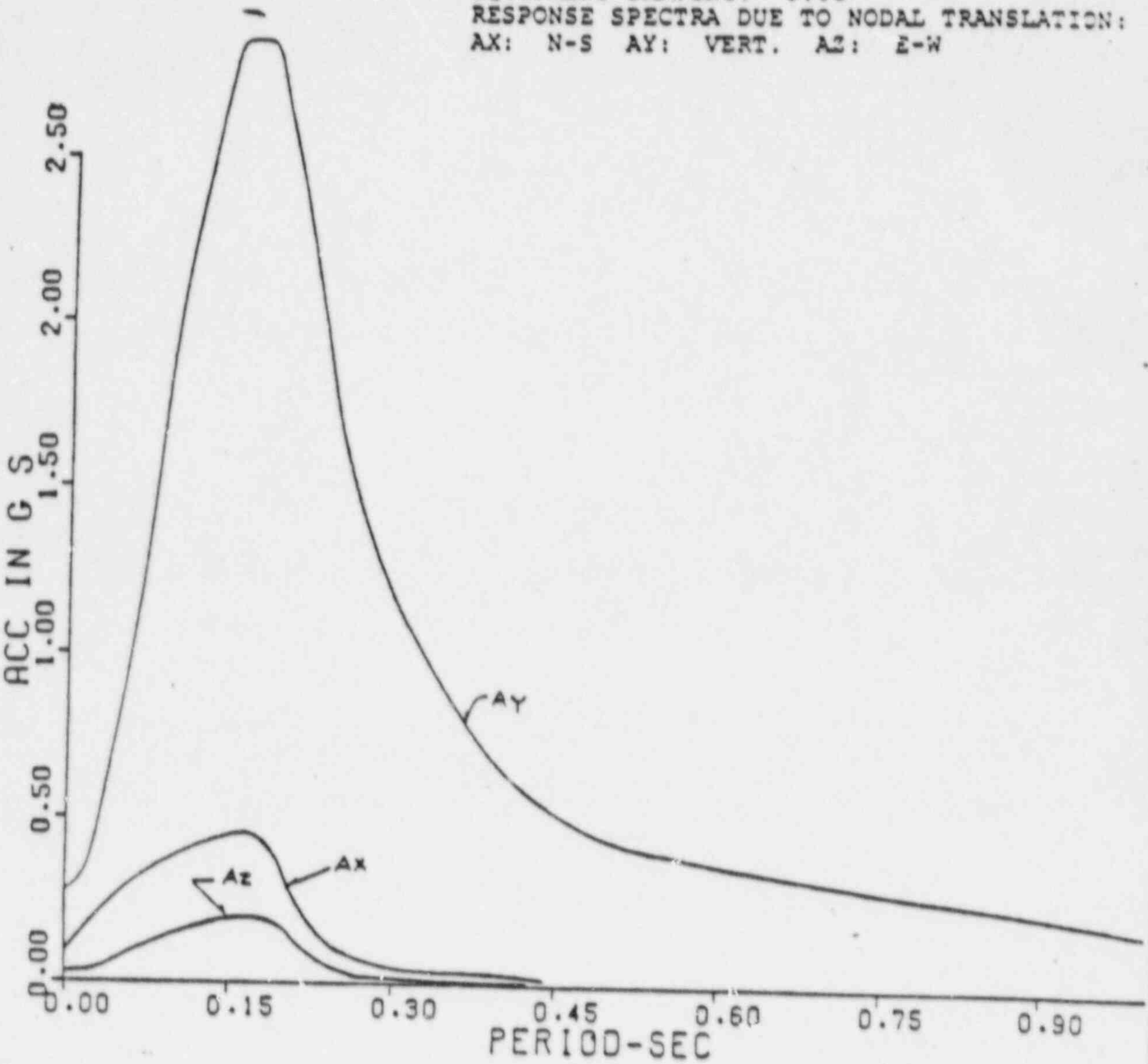
INSTRUCTURE RESPONSE SPECTRA FOR:

STEAM.GENERATOR SUPPORT
SSE EARTHQUAKE DIRECTION: Y
EQUIPMENT DAMPING: 0.02
RESPONSE SPECTRA DUE TO NODAL TRANSLATION:
AX: N-S AY: VERT. AZ: E-W



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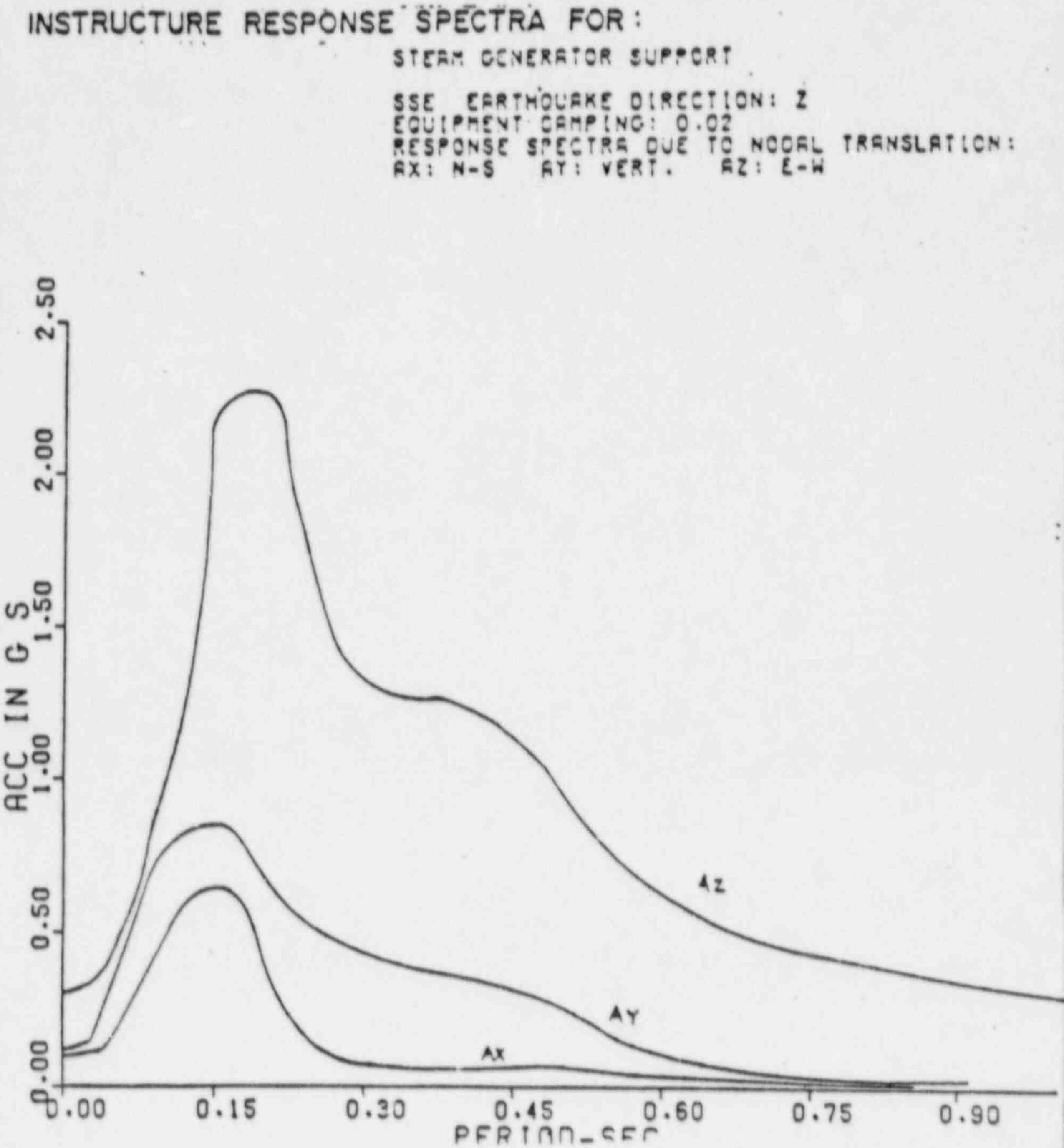
INSTRUCTURE RESPONSE SPECTRA
STEAM GENERATOR SUPPORT
Y-EARTHQUAKE
SSE, 2 PERCENT DAMPING

FIGURE 3.7B-52



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INSTRUCTURE RESPONSE SPECTRA
STEAM GENERATOR SUPPORT
Z-EARTHQUAKE
SSE, 2 PERCENT DAMPING

FIGURE 3.7B-53

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3.8 DESIGN OF CATEGORY I STRUCTURES

3.8.1 CONCRETE CONTAINMENT

3.8.1.1 Description of the Containment

3.8.1.1.1 General Description

The Reactor Containment structure is a fully continuous, steel-lined, reinforced concrete structure. It consists of a vertical cylinder and a hemispherical dome and is supported on an essentially flat foundation mat with a reactor cavity pit projection. The Containment superstructure is independent of the adjacent interior and exterior structures. Sufficient space is provided between the Containment and the adjacent structures to prevent contact under all combinations of loadings.

3.8.1.1.2 Dimensions of Containment

The dimensions of the Containment are as follows:

1. Inside diameter (ID): 135 ft 0 in.

2. Height of cylinder (top of foundation mat to dome spring line): 195 ft 0 in.

3. Inside radius of hemispherical dome: 67 ft 6 in.

4. Thickness of cylindrical walls: 4 ft 6 in.

5. Thickness of dome: 2 ft 6 in.

6. Foundation mat thickness: 12 ft 0 in.

7. Top of the foundation mat: approximately 4 ft 6 in. below grade

3.8.1.1.3 Containment Function

The Containment structure is designed to serve the following functions:

1. Provide vapor containment and limit the leakage from the Containment following an accident within the Containment

2. Isolate the RCS from postulated extreme environmental conditions, including tornadoes and external missiles

3. Biological shielding

3.8.1.1.4 Arrangement of Main Reinforcing Steel

The principal reinforcement used in the mat, walls, and dome are No. 18 bars, made continuous at splices by the use of Cadweld mechanical connectors produced by the Erico Corporation.

The reinforcing steel pattern in the cylinder wall consists of vertical bars at each face, horizontal hoop bars at each face, and 45-degree diagonal bars, in each direction, near the outside face.

The foundation mat is reinforced with top and bottom layers of bars placed as shown on Figure 3.8-12.

The dome reinforcement consists of top and bottom meridional layers of rebars, extending from the vertical bars of the cylindrical wall and top and bottom layers of circumferential hoop bars, as shown on Figure 3.8-11.

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The meridional reinforcement terminated in the dome is anchored by the use of a positive mechanical anchor, such as a bearing plate cadwelded to the end of the bar, and satisfies the other anchorage requirements in accordance with CC-3531.1.2 of the ASME-ACI 359 document.

At penetration openings, reinforcing steel is generally curved around the openings where practical, and supplemental bars are provided around the opening as required. At large major penetrations such as the personnel lock and the equipment hatch some of the wall reinforcement is terminated at the opening by cadwelding steel plates on the end of the bar. Additional reinforcing is provided around these openings to carry stress concentrations and redistributions at these discontinuities. For details, see Figures 3.8-13 and 3.8-14.

3.8.1.1.5 Steel Liner

The entire inside face of the Containment (mat, walls, and dome) is lined with a continuous welded steel liner plate, attached with anchors to the reinforced concrete, to ensure a high degree of leaktightness. The thickness of the liner in the wall is 3/8 in. and in the dome is 1/2 in.; a 1/4 in. thick plate is used on top of the foundation mat and covered with a layer of concrete. Local thickened liner plate sections are provided at penetrations, at major pipe and duct support attachments and at crane and rotating platform girder brackets, and at the bottom of the cylindrical wall's steel liner. Overlay plates and/or structural shapes may be used on the interior side of the liner for support of minor pipes and ducts, conduits, cable trays, and equipment. | 68

Leak-chase channels are provided at liner seams which, after construction, are inaccessible for other means of leaktightness examination. For typical liner details, see Figures 3.8-5 and 3.8-6.

Q022.7 Q022.19 3.8.1.1.6 Containment Penetrations and Attachments

5 Access to the Containment structure is provided by a personnel
airlock, an emergency airlock, and an equipment hatch. Containment
airlocks will be tested once every three days following multiple
12 openings. A constant pressure of P_a will be used to pressurize the
volume between the airlock seals.

1. Personnel Airlock

Q022.7
68 The personnel airlock is an approximately 9 ft diameter double-door
assembly. Each door is hinged and gasketed, with leakage test taps
aligned to the annulus between the gasket sealing surfaces. Both
doors are interlocked so that if one door is open, the other cannot
68 be activated. Both doors are also furnished with automatic as well
as manual pressure equalizing connections and equalizing valves which
can be operated by persons leaving or entering the personnel hatch.
2 Plan and elevation drawings of the personnel airlock are shown in
Figures 3.8-20 and 3.8-21. The configuration of the airlock seal
test provisions is shown in Figure 3.8-21. The personnel airlock has
provisions for test pressurization at a pressure of P_a of the space
between the two grooves at both ends of the airlock as well as
provisions for pressurization at a pressure of P_a of the volume
between the airlock doors. The doors are designed to maintain their
functional capability during testing with no additional requirements
for blocking beyond normal locking procedure.

2. Emergency Airlock

The emergency airlock is an approximately 5-ft 9-in. diameter double-
68 door assembly, with 2-ft 6-in. diameter doors. Both doors of the

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emergency airlock are furnished with manually operated pressure- 68
equalizing connection and valves. The reactor building to airlock door (interior) requires installation of strongbacks for the performance of the overall leakage check. Other operating features are similar to those of the personnel airlock described previously.

3. Equipment Hatch

The equipment hatch is a 16 -ft 0-in. ID single closure penetration. 68
The bolted hatch cover is mounted on the inside of the Containment, and is double-gasketed with a leakage test tap between the gaskets. The hatch cover is provided with a hoist for handling. For details of the airlocks and equipment hatch, see Figure 3.8-9.

4. Pipe Penetrations

Other smaller penetrations through the Containment include the main steam and feedwater lines, hot and cold pipes, the fuel transfer tube, and electrical conductors. All penetration sleeves are welded to the liner and anchored into the reinforced concrete Containment wall. For typical details, see Figures 3.8-7 and 3.8-8.

5. Fuel Transfer Tube Penetration

A fuel transfer tube penetration is provided for fuel transfer between the refueling canal in the Containment structure and the spent fuel pools in the Fuel Building. The penetration consists of a 20 in. stainless steel pipe inside a carbon steel sleeve. The inner pipe acts as the transfer tube; the outer tube is welded to the Containment liner. Bellows expansion joints are provided to permit differential 68
movements.

68 6. Electrical Penetrations

68 Header plate type penetrations are used for electrtical conductors passing through the Containment. The penetration header plate with double O-ring gaskets is bolted to a weld neck flange which is welded to a steel penetration sleeve. The steel penetration sleeves are welded to the Containment vessel liner.

68 7. Liner Attachments

68 Major pipe and duct supports, and crane and Containment access rotating platform girder support brackets are welded to a thickened section of the liner plate, and anchored into the reinforced concrete Containment wall, as shown on Figure 3.8-6. Overlay plates and/or structural shapes may be used from the interior side of the liner, for support of minor pipes and ducts, conduits, cable trays and equipment.

3.8.1.1.7 Drawings

For various Containment structure details described in Subsection 3.8.1.1, see Figures 3.8-1 through 3.8-15.

3.8.1.2 <u>Applicable Codes, Standards, and Specifications</u>

3.8.1.2.1 Basic Code

The basic code used for the materials, design, fabrication, construction, examination, testing, and surveillance of the Containment are the appropriate portions of the Proposed Standard Code for Concrete Reactor Vessels and Containments (April 1973), which was issued for trial use and comments. This basic code was developed by the joint ACI-ASME Technical Committee on Concrete Pressure Components for Nuclear Service, which is made up of ACI Committee 359 and the ASME B&PV Code, Section III, Division 2, Subgroup on Concrete

Components. The specific portions of this document that apply (except where otherwise specifically indicated in this FSAR) are as follows:

1. Subsection CA (General Requirements)

   Article CA-4000 Quality Assurance

2. Subsection CC (Concrete Containment)

   Article CC-1000 Introduction

   Article CC-2000 Materials (except CC-2232.4, CC-2240, and CC-2400, for Preplaced Aggregate Concrete, Grout, and Materials for Prestressing Systems; and except portions of CC-2231.2 for testing for: 68

   coefficient of thermal expansion - CRD-C-39 68
   thermal conductivity - CRD-C-44 68
   creep - ASTM C-512 68
   shrinkage coefficient - ASTM C-157 68
   aggregates for radiation - shielding concrete - ASTM C-637) 68

   Article CC-3000 Design (except CC-3830 for Transitions from Concrete Containment to Steel Containment Vessels and Prestressed Concrete sections)

Article CC-4000	Fabrication, Construction, and Installation (except CC-4230 for Preplaced Aggregate Concrete, and CC-4400 for Fabrication and Installation of Prestressing Systems)
Article CC-5000	Construction, Testing and Examination (except CC-5235 for Preplaced Aggregate Concrete and CC-5400, for Examination of Prestressing Systems)
18 | Article CC-7000	Concrete Containment Structures Protection Against Overpressure
Article CC-9000	Inservice Surveillance (except CC-9230 for Structural Integrity of Prestressed Concrete Containments)
Appendix I	Tables of Prestressing and Liner Materials (except that materials for prestressing do not apply)
Appendix III	Glossary of Terms and Symbols
Appendix VI	Porosity Charts
Appendix IX	Nondestructive Examination Methods
Appendix D	Nonmandatory Preheat Procedures

The procedures used in the design of this facility are consistent with the requirements contained in the applicable portions of the ASME-ACI 359 document as described in the following paragraphs.

The code (hereinafter referred to as the ASME-ACI 359 document) is used as the basic code for this facility because it is by far the most complete, comprehensive document available for the design, construction, inspection and testing of a concrete containment structure. Another advantage is that this document is virtually complete, in itself, without reference to portions of other codes or standards. Reference to other documents such as ASTM specifications 4
or ACI standards are made only where the total document is applicable.

68

The previously listed portions of the ASME-ACI document that apply are all of the applicable requirements of Subsection CC (which comprises the technical requirements for a concrete containment), the applicable appendices, and some aspects of the Quality Assurance Article (CA-4000). The exceptions taken concern items which are not applicable (preplaced aggregate, grout, and prestress systems). Also, in answer to AEC (presently NRC) Question 3.6 of Amendment No. 1, conservative modifications are made regarding some load combinations and allowable stresses. (See Subsections 3.8.1.2.1, 3.8.1.2.5, 3.8.1.3.1, 3.8.1.3.2, 3.8.1.4.5, 3.8.1.5.1, 3.8.1.5.2, and 3.8.3.2.4.)

Requirements for punching shear are in accordance with Section 11.10.3 68
of the 1971 edition of the ACI 318 Code.

Specific references to the articles in Subsection CA, General Requirements, which are of a legal nature rather than a technical nature (Articles CA-1000, CA-2000, CA-3000, CA-4000, CA-5000, and CA-8000) have been omitted. These articles include requirements for such items as code jurisdiction, effective dates of code edition and addenda, certificates of authorization, responsibility of parties, stamping of containment, inspector's certification, authorized inspection agency, and so forth. These legal requirements are not applicable to the Comanche Peak Steam Electric Station (CPSES) since the Code edition in force for this project is the trial use and comments issue.

3.8.1.2.2 ACI Committee 349

Since the ASME-ACI 359 document used as the basic code is co-sponsored by the ACI, it supersedes the ACI Committee 349 document, Criteria For Reinforced Concrete Nuclear Power Containment Structures; therefore, the ACI 349 criteria, published in the ACI Journal, January 1972, is not used as the basic containment criteria. However, the following sections of the ACI 349 criteria complement the requirements of the ASME-ACI 359 document and are used as a reference in the design of this facility:

1. Section 2.2.1 — concerning strain limitations on self-limiting-type bending moments (used in conjunction with CC-3110(b) of the ASME-ACI 359 document)

68 | 2. Appendix C — method of calculating stresses and strains when a thermal gradient is combined with other loads.

68 | 3. The ASME-ACI 359 document does not provide guidance in determining thermal stresses. Therefore, the guidance provided in ACI 349-76, "Code Requirements for Nuclear Safety Related Concrete Structures", Appendix A, "Thermal Considerations" is used.

3.8.1.2.3 Additional Specifications and Standards

The following is a list of specifications and standards that are referred to in the applicable portions of the ASME-ACI 359 document described in Subsection 3.8.1.2.1 and which are applicable to this facility.

1. Liner, Penetrations, Containment Vessel Metal Components, and Attachments:

   ASME B&PV Code, Section III, Division I, Subsection NE, 1971 through and including the 1973 Summer addenda (for the electrical penetration sleeves, fuel transfer tube penetration sleeve, emergency and personnel air locks, and equipment hatch) and 1974 through and including the Summer 1976 addenda (for process piping penetrations subjected to pressure-induced stresses and unsupported by concrete for load-carrying purposes). | 68

   ASME B&PV Code, Section V, 1974 (for liner radiographic examinations) | 68

   ASME B&PV Code, Section IX, 1971 through and including the Summer 1973 addenda (for welding qualifications) | 68

   ASME B&PV Code, Section II, 1971 through and including the Summer 1973 addenda, Part A (ferrous materials) and Part C (welding rods, electrodes, and filler materials) | 68

   AISC Specification for the Design, Fabrication and Erection of Structural Steel For Buildings, 1969 including Supplement Numbers 1, 2, and 3 hereafter referred to as AISC Specification. | 68

68 Except: when supported by an engineering analysis, connections using A325 or A490 high strength bolts need not be pretensioned to the values required by AISC Specification, Table 1.23.5 (for steel brackets and attachments)

ASTM A 20-72a	Specification for General Requirements for Delivery of Steel Plates for Pressure Vessels
SA 370-74	Specification for Methods and Definitions for Mechanical Testing of Steel Products
ASTM A 578-71b	Specification for Straight-Beam Ultrasonic Examination of Plain and Clad Steel Plates for Special Applications
SA 537-74	Specification for Carbon Manganese Silicon Steel Plates, Heat Treated for Pressure Vessels
SA 333-74	Specification for Seamless and Welded Steel Pipe for Low-Temperature Service
68 SA 182-74	Specification for Forged or Rolled Alloy Steel Pipe Flanges, Forged Fittings, and Valve and Parts for High-Temperature Service
68 SA 350-74	Specification for Forged or Rolled Carbon and Alloy Steel Flanges, Forged Fittings, and Valves and Parts for Low-Temperature Service

SA-320-74	Specification for Alloys Steel Bolting Materials for Low-Temperature Service	68
SA-105-74	Specifications for Forged or Rolled Steel Pipe Flanges, Forged Fittings, and Valves and Parts for High-Temperature Service	68
SA-354-74	Specification for Quenched and Tempered Alloy Steel Bolts and Studs with Suitable Nuts	68
ASTM A 36-74	Specification for Structural Steel (for miscellaneous attachments)	
ASTM A 108-73	Specification for Steel Bars, Carbon, Cold-Finished, Standard Quality	
SA 516-74	Specification for Carbon Steel Plates for Pressure Vessel for Moderate and Lower Temperature Services	

2. Reinforcing Steel

ASTM A 615-72	Specification for Deformed and Plain Billet Steel Bars for Concrete Reinforcement	33
ASME SFA 5.1	Specification for Mild Steel Covered Arc-Welding Electrodes	

3. Concrete

ASTM C 150-74	Specification for Portland Cement

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ASTM C 33-74	Specification for Concrete Aggregates
ASTM C 131-69	Test for Resistance to Abrasion of Small Size Coarse Aggregate by Use of the Los Angeles Machine
ASTM C 142-71	Test for Clay Lumps and Friable Particles in Aggregates
ASTM C 117-69	Test for Materials Finer Than No. 200 Sieve in Mineral Aggregates by Washing
ASTM C 87-69	Test for Effect of Organic Impurities in Fine Aggregate on Strength of Mortar
ASTM C 40-73	Test for Organic Impurities in Sands for Concrete
ASTM C 289-71	Test for Potential Reactivity of Aggregates (Chemical Method)
ASTM C 136-71	Test for Sieve or Screen Analysis of Fine and Coarse Aggregates
ASTM C 88-73	Test for Soundness of Aggregates by Use of Sodium Sulfate or Magnesium Sulfate
ASTM C 127-73	Test for Specific Gravity and Absorption of Coarse Aggregate
ASTM C 295-65	Recommended Practice for Petrographic Examination of Aggregates for Concrete

ASTM D 512-67	Tests for Chloride Ion in Water and Waste Water
ASTM C 151-74a	Test for Autoclave Expansion of Portland Cement
ASTM C 191-74	Test for Time of Setting of Hydraulic Cement by Vicat Needle
ASTM C 260-74	Specification for Air-Entraining Admixtures for Concrete
ASTM C 494-71	Specification for Chemical Admixtures for Concrete
ACI 211.1-74	Recommended Practice for Selecting Proportions for Normal and Heavy-Weight Concrete
ACI 304-73	Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete
ASTM C 143-74	Test for Slump of Portland Cement Concrete
ASTM C 172-71	Sampling Fresh Concrete
ASTM C 192-69	Making and Curing Concrete Test Specimens in the Laboratory
ASTM C 31-69	Making and Curing Concrete Test Specimens in the Field

ASTM C 39-72	Test for Compressive Strength of Cylindrical Concrete Specimens
ASTM C 109-73	Test for Compressive Strength of Hydraulic Cement Mortars (Using 2-in. (50 mm) cube specimens)
ASTM C 231-74	Test for Air Content of Freshly Mixed Concrete by the Pressure Method
ACI 214-65	Recommended Practice for Evaluation of Compression Test Results of Field Concrete
ASTM C 78-64	Test for Flexural Strength of Concrete
ASTM C 496-71	Test for Splitting Tensile Strength of Cylindrical Concrete Specimens
ASTM C 469-65	Test for Static Modulus of Elasticity and Poisson's Ratio of Concrete in Compression
ASTM C 642-69T	Test for Specific Gravity, Absorption, and Voids in Hardened Concrete
ASTM C 94-74	Specification for Ready-Mixed Concrete
ACI 347-68	Recommended Practice for Concrete Formwork
ACI 305-72	Recommended Practice for Hot Weather Concreting

ACI 306-66 — Recommended Practice for Cold Weather Concreting

4. Testing and Surveillance

10 CFR Part 50, Appendix J, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors (2/5/73)

ANSI N45.4-1972 Leakage-Rate Testing of Containment Structures for Nuclear Reactors

3.8.1.2.4 Specifications and Standards Not Referred to in ASME-ACI 359

AWS D12.1-61 — Recommended Practices for Welding Reinforcing Steel, Metal Inserts, and Connections in Reinforced Concrete Construction

3.8.1.2.5 Summary of Principal Plant Specifications

1. The principal plant specifications for the steel-lined, reinforced concrete containment are the fabrication and construction specifications for the following:

   a. Liner (including penetrations and attachments)

   b. Reinforcing steel

   c. Splicing of reinforcing steel

   d. Concrete

2. The applicable portions of the ASME-ACI 359 document, as described in Subsection 3.8.1.2.1, are included in the plant construction specifications in regard to materials, construction techniques, fabrication, welding, examination, testing, and so forth. The following are the principal appropriate portions of the ASME-ACI 359 document which are incorporated into the plant specifications:

    a. Liner Specifications

        CC-2500 Materials for Liners

        CC-2520 Special Materials Testing

        CC-2530 Examination and Repair of Liner Materials

        CC-2540 Marking of Liner Materials

        CC-2600 Welding Materials

        CC-2612 Weld Metal Tests

68 NB-2432 may be used in lieu of CC-2612.2 for chemical analysis of filler metal or weld deposit.

68 CC-2620 Stud Welding Materials

68 Except, 1100 Aluminum may be used as a flux for stud welding without a chemical analysis of each batch as stipulated by CC-2623.2.

        CC-2630 Identification of Welding Materials

        CC-2700 Materials Manufacturers' Quality Assurance Programs

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CC-3840 Design of Welded Construction

CC-4120 Certification of Materials and Fabrication by Component Manufacturer and/or Installer and/or Constructor

CC-4520 Forming, Fitting, and Aligning

Except for the following: 68

When qualifying the procedure for the forming and bending process, the Charpy V-notch impact test temperatures, of the specimens used to establish a transition curve, may be conducted at a minimum of five different temperatures distributed throughout the transition region, in lieu of CC-4521.3.2(e) requirement for conducting tests at each temperature increment of 10°F. 68

When thermal cutting is performed to prepare weld joints or edges, to remove attachments or defective material or any other purpose, preheating may be in accordance with the applicable section of the ASME III B&PV Code in lieu CC-4521.1.1. 68

Engineers may approve on a case by case basis other stud welding equipment in addition to the requirements of CC-4543.5(a). 68

Engineers may review post-weld heat treatment records in lieu of CC-4552.2.2 requirement of review by inspector. 68

CC-4530 Welding Qualifications

CC-4540 Rules Governing Making, Examining, and Repairing Welds

CC-4550 Heat Treatment

68

CC-5520 Required Examination of Welds

68 The criteria for extent and frequency of radiographic examination may be as follows in liew of CC-5521.1.1.(a) (b) (c) (d) (e) and (f):

68 a. For each welder and welding position (flat, vertical, horizontal and overhead), the first 10 feet of weld shall be examined radiographically. If this radiograph meets the acceptance standards that 10 feet of weld shall be accepted.

68 b. For the first three shell rings only, for each welder and welding position (flat, vertical, horizontal and overhead) the first 10 feet of weld shall be performed on a respresentative mock-up. These mock-ups shall simulate as close as practicable the actual conditions that the welder will experience during production. These welds shall be 100 percent radiographically examined. Should a question of interpretation arise as to the acceptance of the weld in accordance with the radiographic acceptance standard, a cross-sectional coupon can be cut from the weld to visually verify or refute the film interpretation.

Further, all production welds on the first three shell rings are to be 100 percent examined by magnetic particle inspection and vacuum box tested, in lieu of radiographic examination. 68

c. If the radiography in "a" above does not meet the acceptance standards, the portions of the 10 foot increment of weld which do not meet the performance requirements shall be repaired and re-radiographed. 68

d. Welders who have satisfactorily welded the first 10 feet of weld as described above shall have one 12-inch-long radiograph made of each subsequent 50 feet of weld or fraction thereof which he produces. If the first radiograph in each 50 foot increment meets the acceptance standards, the welder shall be permitted to continue welding the next 50-foot increment of production weld. A minimum of 2 percent of all liner seam welds shall be examined by radiography. 68

e. If the 12-inch radiograph in the 50-foot-long increment of weld does not meet the acceptance standards, two 12-inch films shall be taken at least 1 ft removed on each side from the original spot within the 50-foot-long increment. If these radiographs meet the radiographic acceptance standards, the 50 feet of weld represented shall be accepted. The defective areas shown in the first radiograph shall be repaired and re-radiographed. 68

68 f. If either of the second radiographs does not meet the acceptance standards, the entire weld test unit is unacceptable. The remaining portion of the 50-foot increment of this weld shall be radiographed. The portions of the 50-foot increment of weld which do not meet the acceptance standards are to be repaired and re-radiographed.

68 The criteria for magnetic-particle (MT or MPE) or liquid-penetrant (PT or LPE) examination in lieu of radiographic examination may be used as follows in lieu of CC-5521.2.1:

68 Where radiographic examination of liner seam welds is not feasible or where the weld is located in areas which will not be accessible after construction, the entire length of weld shall be examined by the magnetic-particle or liquid-penetrant method. Where magnetic-particle or liquid-penetrant inspection discloses welding that does not meet the magnetic-particle or liquid-penetrant acceptance standards, additional testing shall be performed to the same extent as required for radiography in CC-5520. Unacceptable indications of the weld shall be eliminated or repaired by welding as required.

68 In addition to the required radiography of CC-5520, all seam welds at abrupt changes in liner configuration (i.e., cylinder to sphere) shall be examined by the magnetic particle method for their entire length.

CC-5530 Acceptance Standards

The acceptance standards for the liner butt welds 68
examined by radiography may be in accordance with ASME B&PV Code Section VIII, Division 1, paragraph UW-51 and AEC Regulatory Guide 1.19 in lieu of CC-5532 Radiography Acceptance Standards.

Non-full penetration attachment welds to the 68
reactor containment liner may use the following acceptance standards:

The fillet welds of the attachment to the 68
reactor containment liner shall be examined by either the magnetic particle method or liquid penetrant method. The acceptance standards shall be in accordance with the following criteria.

For attachment welds, fillet weld size 68
specified on the drawings are the minimum size required, except as permitted below, along the full length of the weld joint. Additional welding is acceptable provided it does not distort the items being joined together.

A fillet weld in any single continuous weld 68
may be less than the fillet weld dimension by not more than 1/16 inch provided that the undersize portion of the weld does not exceed 10 percent of the length of weld.

68 Liquid Penetrant Acceptance Standard

68 1. Linear indications in which the length is more than three times the width.

68 2. Round indications are indications which are circular or elliptical with the length less than three times the width.

68 3. Only indications with major dimensions greater than 1/16 inch are considered relevant.

68 4. Unless otherwise specified, the follwoing relevant indications are unacceptable.

68 a. Any cracks and linear indications.

68 b. Rounded indications with dimensions greater than 3/16 inch.

68 c. Four or more rounded indications aligned and separated by 1/16 inch or less, edge to edge.

68 d. Ten or more rounded indications in any 6 square inch of surface, the major dimension of this area not exceeding 6 inches, with the area taken in the most unfavorable location relative to the indications being evaluated.

68 Magnetic Particle Acceptance Standards

1. Only indications with major dimensions greater than 1/16 inch are considered relevant. | 68

2. Unless otherwise specified, the following relevant indications are unacceptable: | 68

   a. Any cracks or linear indications. | 68

   b. Rounded indications with dimensions greater than 3/16 inch. | 68

   c. Four or more rounded indications aligned and separated by 1/16 inch or less, edge to edge. | 68

   d. Ten or more rounded indications in any 6 square inch of surface, the major dimension of this area not exceeding 6 inches, with the area taken in the most unfavorable location relative to the indications being evaluated. | 68

CC-5540 Examination of Stud Welds | 68

(For protective coatings on liner and other steel, see Subsection 3.8.1.6.5, Item 2.g.)

b. Reinforcing Steel Specification

CC-2300 Materials for Reinforcing Systems

CC-2320 Material Identification

CC-2330 Special Materials Testing

CC-2700 Materials Manufacturers' Quality Assurance Programs

68 | CC-3430 Concrete Temperatures

CC-3533 Reinforcing Steel Cover and Spacing Requirements

CC-4120 Certification of Materials and Fabrication by Component Manufacturer and/or Installer and/or Constructor

CC-4320 Bending of Reinforcing Bars

CC-4340 Placing Reinforcement

CC-4350 Spacing of Reinforcement

CC-4360 Surface Condition

c. Specification for Mechanical Butt Splicing (Cadwelds) of Reinforcing Steel

CC-2700 Materials Manufacturers' Quality Assurance Programs

CC-4120 Certification of Materials and Fabrication by Component Manufacturer and/or Installer and/or Constructor

CC-4333 Mechanical Butt Splices Using Sleeve and Filler Material

CC-4335 Mechanical Joints

CC-5320 Examination of Sleeve with Filler Metal Connection

d. Concrete Specification

CC-2220 Materials for Concrete

CC-2230 Concrete Mix Design

CC-2250 Marking and Identification of Concrete Materials

CC-2700 Materials Manufacturers' Quality Assurance Programs

CC-4220 Batching, Mixing, and Transporting

CC-4225 Depositing

CC-4240 Curing

CC-4250 Formwork and Construction Joints

CC-4260 Cold and Hot Weather Conditions

CC-4270 Repairs to Concrete

CC-5220 Concrete Materials (testing and examination of)

CC-5230 Concrete (testing and examination of)

3.8.1.2.6 Applicable NRC Regulatory Guides

The following NRC (formerly AEC) Regulatory Guides are applicable to this Containment and are complied with:

NRC Regulatory Guide 1.10	Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures (Revision 1, 1-2-73, of former Safety Guide 10)
NRC Regulatory Guide 1.15	Testing of Reinforcing Bars for Category I Concrete Structures (Revision 1, 12-28-72 of former Safety Guide 15)
NRC Regulatory Guide 1.18	Structural Acceptance Test for Concrete Primary Reactor Containments (Revision 1, 12-28-72 of former Safety Guide 18)
NRC Regulatory Guide 1.19	Nondestructive Examination of Primary Containment Liner Welds (Revision 1, 8-11-72 of former Safety Guide 19)
NRC Regulatory Guide 1.28	Quality Assurance Program Requirements (Design and Construction) (6-7-72 of former Safety Guide 28)
68 NRC Regulatory Guide 1.29	Seismic Design Classification (Revision 2, 2-76 of former Safety Guide 29)
NRC Regulatory Guide 1.55	Concrete Placement in Category I Structures (6-73)

68

3.8.1.3 Loads and Load Combina..ons

3.8.1.3.1 Loads

The following loads are considered in the design of the steel-lined, reinforced concrete Containment structure (essentially in accordance with the ASME-ACI 359 document):

1. D = dead load of the Containment, and all superimposed permanent loads

2. L = live loads, comprising conventional floor and roof live loads, movable equipment loads, cables, and lateral soil pressure

3. Pa = Containment pressure load due to the DBA, at 50 psig

4. T = thermal effects

   a. To = thermal loads during normal operating conditions, including liner expansion and temperature gradients in the wall

      1) Normal operating temperature range inside the Containment is 60°F to 120°F.

      2) Ambient temperature range at the outside face of the Containment wall is 0°F to 110°F.

   b. Ta = added thermal loads (over and above operating thermal loads), exerted by the liner, which may occur during an accident and which correspond to the factored accident pressure (i.e., 1.0 Pa, 1.25 Pa, or 1.5 Pa); the accident temperature causes an almost instantaneous increase in the liner temperature, with little initial effect on the temperature of the relatively thick concrete wall. This sudden increase in liner temperature creates compressive stresses and strains in the liner which thrusts against the reinforced concrete section, having an effect on the reinforcing steel similar to an added internal pressure.

1) The saturation temperature corresponding to an accident pressure of 50 psig is $280^{\circ}F$.

2) The saturation temperature corresponding to an accident pressure of 1.25 x 50 psig is $295^{\circ}F$.

3) The saturation temperature corresponding to an accident pressure of 1.5 x 50 psig is $305^{\circ}F$.

c. Tt = thermal loads during pressure test, including liner expansion and a temperature gradient in the wall; a maximum gradient of $40^{\circ}F$ is assumed during test.

5. Seismic loads representing two magnitudes of earthquake are considered, as follows:

a. E' = SSE

b. E = 1/2 of SSE=OBE

68 The vertical and horizontal earthquake accelerations are assumed to act simultaneously.

The earthquake forces acting on the Containment structure are taken from the results of dynamic analyses, based on seismic input described in Section 3.7.

6. W = design wind load (See Section 3.3.)

7. Wt = tornado load including wind, differential pressure and missiles (See Sections 3.3 and 3.5.) 68

8. Ro = piping loads acting on the Containment during operating conditions

9. Ra = piping loads acting on the Containment, due to increased temperature resulting from the design accident

10. Yr = equivalent static load on structure or penetration generated by the reaction on the broken high-energy pipe during the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load. 68

11. Pv = negative internal pressure during operation; maximum Pv equals 5 psi.

12. Yj = equivalent static jet impingement load on a structure generated by the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load. 68

13. Ym = equivalent static missile impact load on a structure generated by or during the postulated break, as from pipe whipping, and including an appropriate dynamic load factor to account for the dynamic nature of the load. 68

Q130.15 |

1 | 14. Ha = represents the load on the Containment resulting from post-
| LOCA internal flooding.

Q130.16 |

3.8.1.3.2 Load Combinations

The design of the reinforced concrete Containment structure incorporates the service load combination requirements and the factored load combination requirements, as follows (in accordance with the ASME-ACI 359 document):

1. Service Load Combinations

   a. Construction Category

      1.0 D + 1.0 L + 1.0 To

   b. Test Category

      1.0 D + 1.0 L + 1.15 Pa + 1.0 Tt

   c. Normal Category

      1) 1.0 D + 1.0 L + 1.0 To + 1.0 E + 1.0 Ro + 1.0 Pv

      2) 1.0 D + 1.0 L + 1.0 To + 1.0 W + 1.0 Ro + 1.0 Pv

2. Factored Load Combinations

   a. Abnormal Category

      1.0 D + 1.0 L + 1.5 Pa + 1.0 (To + Ta) + 1.0 Ra

   b. Extreme Environmental Category

      1) 1.0 D + 1.0 L + 1.0 To + 1.0 Wt + 1.0 Ro + 1.0 Pv

      2) 1.0 D + 1.0 L + 1.0 To + 1.0 E' + 1.0 Ro + 1.0 Pv

c. Abnormal Severe Environmental Category | Q130.15 | Q130.16

1) $1.0\ D + 1.0\ L + 1.25\ Pa + 1.0\ (To + Ta) + 1.25\ E + 1.0\ Ra + 1.0\ (Yr+Yj+Ym)$

2) $1.0\ D + 1.0\ L + 1.25\ Pa + 1.0\ (To + Ta) + 1.25\ W + 1.0\ Ra + 1.0\ (Yr+Yj+Ym)$

3) $1.0\ D + 1.0\ L + 1.0\ To + 1.0\ E + 1.0\ Ha$ | 1

4) $1.0\ D + 1.0\ L + 1.0\ To + 1.0\ W + 1.0\ Ha$ | 1

d. Abnormal-Extreme Environmental Category

$1.0\ D + 1.0\ L + 1.0\ Pa + 1.0\ (To + Ta) + E' + 1.0\ Ra + 1.0\ (Yr+Yj+Ym)$

3. Variable and Interrelated Loads

For loads which may vary, the values (within the possible range) which produce the most critical combination of loading are used in design. For loads which are interrelated as a function of time, the maximum values of these effects do not necessarily occur simultaneously. Recognition is given to the time increments associated with such conditions.

4. Allowable Stresses

The allowable stresses associated with the service load combinations and the factored load combinations are given in Subsection 3.8.1.5.

5. Load Combinations for Localized Areas

The design load combinations used to examine the effects on localized areas such as penetrations, shell discontinuities, crane girder brackets, and local areas of high thermal gradients are the same load combinations used for the general Containment structure, as previously described.

6. Time-Dependent Loads

Time-dependent loads such as the effects of creep, shrinkage, and other related effects are ignored in the design of the reinforced concrete Containment structure. (See Subsection 3.8.1.4.1, Subsection 2).

7. Explanation of the Use of an Ultimate Strength Approach With a Load Factor of 1.0

Factored load combinations that include extreme environmental effects (SSE or tornado effects) incorporate a load factor of 1.0 using an ultimate strength approach with stresses within the range of general yield. This approach is justified based on the fact that the extreme environmental effects that are considered are of an upper bound conservative magnitude and have an extremely low probability of occurrence. In addition, the maximum SSE is assumed to occur concurrently with the DBA, an extremely unlikely occurrence. Additional margin of safety is provided by the fact that, under these factored load combinations, the average stress in the reinforcing steel is limited to 90 percent of yield, rather than full yield.

3.8.1.4 Design and Analysis Procedures

3.8.1.4.1 General Analysis of Entire Containment Structure

The Containment structure, including the foundation mat, is analyzed and designed for all load combinations as described in Subsection
3.8.1.3. Table 3.8-1 shows critical loading combinations, type of | 5
stress and computed and corresponding allowable stresses at key |
locations in the Containment structure. |

1. Foundation Mat Analysis

   The Containment structure foundation mat is analyzed by a finite | 68
   element method of analysis using the ANSYS computer program. |
   (See Appendix 3.7A for a description of the program.) The program uses the stiffness method of structural analysis and contains the various types of finite elements, i.e., triangular, rectangular, and quadrilateral plate elements representing
   membrane or bending behavior, or both, and beam elements. The | 68
   model used for the foundation mat analysis includes the mat and |
   the Containment cylindrical wall to a height of approximately 76 |
   ft above the mat. This height of Containment wall is sufficient |
   to represent the effect of the Containment wall stiffness on the mat behavior under the various loading conditions. The rock beneath the foundation mat is represented by appropriate linear springs. The rock mat contact area under various loading combinations is discussed in Subsection 3.8.5.4.1, Item 1.

   The input to this program consists of the geometry of the structure, the material properties, the appropriate boundary
   conditions, and the loadings. The boundary conditions at the cut | 68
   Containment wall section (approximately 76 ft above the mat) are |
   |

68 represented by the equivalent load reactions at this point for each type of loading (dead load, pressure, thermal gradient, seismic, and so forth). The loads acting on the mat from the concrete internal structure are represented in the model as equivalent pressures at the interface surface of the mat.

The output of this program contains the displacements, rotations, forces, shears, moments, and stresses throughout the structure.
68 This output is used for the design of the foundation mat. In regard to the Containment wall design, the output from this analysis is used only to check the design of the wall at the junction with the mat. The Containment superstructure design (walls and dome) is based on a supplementary analysis as described in Subsection 3.8.1.4.1, Item 2.

2. Containment Superstructure Analysis

68 The Containment shell is analyzed using the SHELL-1 computer program (See Appendix 3.7A for a description of the program.) This program uses a numerical method which combines the direct integration and the finite difference techniques for solving general shell equations. The model consists of the hemispherical dome and the cylindrical wall down to the top of the mat. The input to this model consists of the geometry of the structure, the
68 material properties, and the loadings. Fixed boundary conditions are used at the junction of the bottom of the wall and the top of the mat. The results of the mat analysis are used to verify the design at the bottom of the wall.

Creep and shrinkage of concrete are important considerations in the analysis and design of a prestressed concrete containment. However, for the nonprestressed reinforced concrete containment being used on this project, the effects of creep and shrinkage are not significant and can be safely ignored. Shrinkage in a reinforced concrete containment results in meridional and radial displacements which are the opposite of the displacements caused by the principal loadings, internal pressure, and temperature. Since it is not additive to these major loads, it can be ignored.
The amount of cracking in a reinforced concrete containment is 68
dependent on the tensile stresses in the shell. For load combinations which do not include internal pressure, the containment is assumed to be completely uncracked. For load combinations which include internal pressures, the analysis uses a variably cracked model, in which the concrete is completely cracked in the membrane regions, with the stiffness of the concrete ignored in both directions and only the properties of the reinforcing steel considered. The stiffness is increased in the vertical direction of the cylinder due to the decrease in the net tensile force when the dead weight of the containment is included. Also the stiffness is adjusted at the discontinuities to account for the increased stiffness when compression and/or high moment exist.

68

The overall Containment shell is analyzed, neglecting the presence of the penetrations. The analysis of the portions of the containment shell where the stress pattern is influenced by the major penetrations (airlocks and equipment hatch) is performed as described in Subsection 3.8.1.4.2.

3. Check of Model Validity and Analysis Results

For methods of checking the validity of the model and the results of the analysis, see Subsection 3.8.3.4.1.

3.8.1.4.2 Analysis at Major Penetrations

The effect of the major penetrations (airlocks and equipment hatch) on
68 the Containment wall is analyzed using a finite element model. The
computer program used ANSYS (See Appendix 3.7A for description.) For
the purpose of this analysis, a segment of the Containment wall
containing the penetration is isolated and analyzed for the same
loading conditions as those for which the entire Containment shell is
analyzed. The boundary of the segment is approximately three times
the penetration diameter from the center of the penetrations, except
at the boundary with the mat which is approximately two times the
penetration diameter from the center of the penetration. The finite
element model of the segment consists of solid finite elements or
plate elements, or both connected at their nodes. The boundary
conditions applied to this model are obtained from the analysis of the
entire Containment shell, as described in Subsection 3.8.1.4.1, Item
2. The model considers the various degrees of cracking as described
in Subsection 3.8.1.4.1, Item 2. The program used in this analysis
considers, in addition to the strains in the plane of the wall,
strains in the orthogonal direction.

The output of this analysis includes the displacements, rotations, forces, shears, and moments which are used for the design of the reinforced concrete ring girder around the penetration.

3.8.1.4.3 General Design Criteria and Procedures

The General Design Criteria (GDC) are in accordance with CC-3000 of the ASME-ACI 359 document, except as otherwise specifically indicated. | 68

1. Concrete Tensile Strength

   Concrete tensile strength is not relied upon to resist flexural and membrane tension.

2. Interaction of Liner Plate and Reinforced Concrete

   The SHELL-1 analysis considers the interaction of the liner plate and the concrete structure under all conditions of loading. However, if the action of the liner results in lower concrete or reinforcement stresses, the presence of the liner is disregarded for that particular case. | 68

   In considering the interaction between the liner, when subjected to the hot accident temperature, and the reinforcing steel within the cracked concrete section, strain compatibility between the liner and the reinforcing steel is considered. In the equations of strain compatibility for the design of the reinforcing steel for this hot liner condition, the effect of the concrete is ignored, since it is assumed to be cracked and incapable of carrying any of the tensile loads.

   The interaction of the liner plate is only used to increase the effective internal pressure during an accident case. It is not used to reduce the internal pressure acting on the concrete. | 68

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3. Evaluation of Effect of Variations in Assumptions and Materials

68 The fact that reinforced concrete is not a homogeneous material is accounted for in the design; stiffness properties are altered where the section is assumed to crack. Properties of materials are known with sufficient accuracy, and assumptions made are sufficiently conservative so that other variations need not be considered.

4. Temperature Effects

The temperature gradient through the containment wall during operation is essentially linear and is a function of the internal operating temperature and the average external ambient temperature. The accident temperature primarily affects the liner, rather than the concrete and reinforcing steel, due to insulating properties of the concrete. By the time the temperature of the concrete within the interior of the concrete shell begins to rise significantly, the internal accident pressure within the Containment has fallen off to a point below the peak values. Therefore, it is not necessary to consider peak accident temperature in the concrete coincident with peak pressures in the Containment. (The thrust caused by the instantaneously hot liner against the reinforced concrete wall is considered simultaneously with the peak pressure.) Also, temperature stresses of the reinforcing steel in the Containment shell caused by the maximum thermal gradient do not significantly influence the capacity of the structure to resist membrane forces. Temperature gradients induce stresses in the structure which are internal in nature, causing tension on one face and compression on the other face; the resultant membrane force is zero. If loading combinations concurrent with these temperature gradient effects cause local stresses in the horizontal and vertical bars of one face to reach the yield point, any further load is transferred to the unyielding elements on the other face of the wall. Because of the self-limiting nature of stresses resulting from a thermal gradient, the reinforcing bars across a horizontal or vertical section have a magnitude of final membrane load resistance essentially equal to that which would be carried if temperature gradient effects were neglected. This design approach is basically in accordance with CC-3110 of the ASME-ACI 359 document. However, for factored load combinations which include a thermal gradient, the maximum strain in the reinforcing steel is limited to approximately 1.5 times the yield strain (in accordance with ACI 349, Section 2.2.1). The total reinforcement across any section for any factored load combination has an average tensile stress not more than 0.9 times the yield stress.

3.8.1.4.4 Design of Reinforced Concrete at Penetrations

1. Major Penetrations (Airlocks and Equipment Hatch)

   The Containment wall at the major penetrations is designed as a ring beam around the openings and is thickened around these penetrations. The results of the analyses performed at these openings (see Subsection 3.8.1.4.2) are used to design the ring beams. The ring beams are designed to resist biaxial bending moments, axial tension, torsion, and biaxial shear, resulting from all the load combinations listed in Subsection 3.8.1.3.2. At the openings, some of the typical wall reinforcement close to the outside of the opening is curved around the openings. The remainder of the typical wall reinforcement is terminated at the opening by cadwelding steel plates on the end of the bars.

Additional reinforcement is provided around the opening, principally in circumferential and radial directions relative to the centerline of the opening, to limit stresses to the allowable values (see Subsection 3.8.1.5). For the typical arrangement of reinforcing steel at these major openings, see Figure 3.8-14.

2. Smaller Penetrations (for Pipe Lines)

Penetration sleeves are anchored into the reinforced concrete Containment wall by means of steel lugs welded to the sleeve. The anchors and local wall reinforcement are designed to resist the torsion, bending, and shear that the pipe is capable of exerting on the penetration, as limited by the full plastic strength of the pipe and an axial load based on the maximum possible pipe jet reaction. All possible combinations of loadings are considered to act simultaneously. The typical wall reinforcement is curved around these penetrations and kept continuous wherever possible. Additional local reinforcement is provided around the opening, as required to resist the loads imposed by the pipe, as described in the preceding paragraphs. This additional reinforcement is provided in vertical, horizontal, and radial directions, relative to the centerline of the opening. For typical arrangement of reinforcing steel at these penetrations, see Figure 3.8-13.

3.8.1.4.5 Analysis and Design for Shear Effects

1. Tangential Shear

Tangential shear is due principally to horizontal seismic motion. The maximum intensity of concrete tangential shear stress, Vu (in psi), is obtained from the results of the analysis in accordance with Subsection 3.8.1.4.1, Item 2, for all load combinations.

The factored load design for tangential shear is in accordance with CC-3411.5 of the ASME-ACI 359 document, except that the maximum allowable tangential shear stress carried by the concrete, Vc, does not exceed 60 psi. In regard to this section of the ASME-ACI 359 document, this Containment complies with the requirements CC-3411.5 (a) through (d), and it does not support principal equipment laterally; therefore, the maximum allowable tangential shear stress which the concrete can be assumed to safely resist is 60 psi.

Where the maximum tangential shear stress, Vu, exceeds the concrete allowable shear, Vc, the excess (Vu-Vc) is resisted by inclined reinforcement placed near the outside of the wall at 45 degrees in each direction. Design of the reinforcing steel for tangential shear is in accordance with CC-3521.1.1 of the ASME-ACI 359 document for factored load design. In calculating the stresses in the reinforcing steel caused by tangential shear, compatibility of strains between the inclined steel and the vertical and horizontal steel is considered in accordance with M.J. Holley's Provision of Required Seismic Resistance, included in Seismic Design for Nuclear Power Plants by the MIT Press. For arrangement of reinforcing steel, see Figure 3.8-10. (Service load combinations are checked in accordance with CC-3420 and CC-3522 of the ASME-ACI 359 document.)

2. Radial Shear

The maximum radial shear occurs at the junction of the bottom of the Containment wall and the top of the foundation mat, under the pressure loading. The values of radial shear are obtained from the results of the analysis described in Subsection 3.8.1.4.1, Item 2, for all load combinations. Design for radial shear is in accordance with CC-3411, CC-3421, and CC-3521 of the ASME-ACI 359 document. Radial shear loads are resisted by radial bars inclined at 45 degrees with the horizontal and extending between the vertical bars near the inside surface and the outside surface of the cylinder wall. Above the mat, where the radial shear is maximum, plate bars 4 in by 1 in are welded to the vertical reinforcing steel. See Figure 3.8 10 for arrangement of radial shear reinforcement.

3.8.1.4.6 Analysis and Design of Liner and Anchorage

The liner for the cylindrical walls is 3/8 in-thick steel plate and for the dome 1/2 in-thick steel plate, each anchored into the concrete with 5/8 in. by 6 3/8 in. long, headed, welded studs, Type H4, as produced by the Nelson Stud Welding Co., or engineer-approved equal. Studs are spaced to satisfy the criteria described in Subsection 3.8.1.5.3. The approximate spacing of the anchor studs in the cylindrical wall and dome is 12 in. each way. The wall and dome liner serve as the inside formwork for placing of concrete. The liner on top of the mat is 1/4-in. thick. This bottom liner is installed after foundation mat construction by welding at seams to structural members that have been embedded in the top of the mat. These embedded structural anchors are approximately 8 to 10 ft apart. The liner on top of the mat is covered with approximately 30 in. of concrete. The vertical wall liner is anchored at the foundation mat; this end anchor is designed to resist the maximum compression and tension to which the liner plate is subjected. See Figure 3.8-5 for liner anchorage details.

The analysis and design of the liner, anchors, and attachments are in accordance with CC-3120, CC-3600, CC-3700, and CC-3800 of the ASME-ACI 359 document.

The liner and anchors are designed to withstand the effects of all load combinations as described in Subsection 3.8.1.3.2, using load factors equal to 1.0.

The stability of the liner is ensured by anchorage to the reinforced concrete. The anchorage system prevents distortions sufficient to impair leaktightness. The liner plate anchorage system is designed to 4
withstand without rupture all in-plane (shear) loads or deformations exerted by the liner plate and also to resist all loads applied normal to the liner surface. The anchors are designed to elastically carry the forces resulting from the various load combinations, or to have sufficient ductility to relieve the forces, or to bring necessary additional anchors into action without rupture of the liner or anchor. The anchorage is designed so that if any one anchor fails, successive failure of adjacent anchors does not occur in the manner of a chain reaction. See ACI 349, Section 2.6.5.5.

In general the maximum load affecting the design of the liner and anchors, is that caused by the maximum temperature rise due to an accident. This temperature increase causes the liner, which is restrained by the reinforced concrete wall, to be stressed in compression. The compressive stress is calculated by equating the strains between the liner and the reinforced wall. The resulting stresses and strains in the liner are less than allowable as stated in Subsection 3.8.1.5.3. The maximum load in an anchor stud is an in-plane shear load which can occur if the plate on one side of an anchor bows inward in a flexural mode, causing a reduction of membrane compression on one side of the anchor. This inward bowing of the plate can be caused by initial construction deformity, variation of liner plate curvature, loss of an anchor, and similar occurrences.

The resulting unbalanced plate stress imparts a shear load and corresponding displacement on the adjacent anchor, and, to a lesser degree, on each successive anchor further away from the bowed-in plate. The shear load versus displacement of the anchor stud is based on test data developed by the Nelson Stud Welding Co. The analysis to determine the load and displacement on each stud is performed by making a series of successive approximations using the test curves for anchor shear load versus displacement. The analysis is based on maintaining equilibrium of loads in the plate and anchors for any free body cut through a section of plate and compatibility between the strain in the plate and displacement of the studs. The resulting maximum loads and displacements in the studs are less than allowable as stated in Subsection 3.8.1.5.3.

The anchor design and analysis consider the effects of the following (in accordance with CC-3800 of the ASME-ACI 359 document):

1. Variation of liner plate curvature: the anchors are designed for possible inward bowing of the plate, as described previously.

2. Variation in liner plate thickness due to rolling tolerances: the range of maximum and minimum plate thickness is assumed in the design of the liner plate and anchors.

3. Variation of liner plate yield strength: the anchors are analyzed assuming the liner remains elastic under all conditions, i.e., the liner strains are converted to stress using Hooke's Law with the modulus of elasticity and Poisson's Ratio below yield. See CC-3630 of the ASME-ACI 359 document.

4. Liner plate seam offset: stresses due to maximum allowable seam offset, as stated in the construction specification, are considered in design.

5. Variation in anchor spacing: maximum range of anchor spacing, as allowed in the construction specification, are considered in design.

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6. Variation in anchor stiffness due to a variation of the concrete modulus: this variation is considered by modifying the load versus displacement test data for the stud anchors and considering in the analysis a range of minimum and maximum possible values of concrete moduli.

7. Local concrete crushing in the anchor zone: such crushing is reflected in the anchor stud test data.

The liner plate is thickened at penetrations in accordance with the requirements of Subsection NE of the ASME B&PV Code, Section III.

3.8.1.4.7 Design of ASME B&PV Code, Section III, Division 1, Class MC Components

For the design of ASME B&PV Code, Section III, Division 1, Class MC Steel Components, such as the airlocks, equipment hatch, and portions of penetration sleeves subject to pressure induced stresses, see Subsection 3.8.2.

3.8.1.5 <u>Structural Acceptance Criteria</u>

3.8.1.5.1 Reinforcing Steel - Allowable Stresses [illegible]ains

1. Based on factored load combinations, as described in Subsection 3.8.1.3.2, Item 2, the allowable average tensile stress in reinforcing steel at any section in the Containment is 90 percent of the yield stress (0.90 fy) in accordance with CC-3412 of the ASME-ACI 359 document. When considering load combinations which include a self-limiting thermal gradient, the maximum allowable strain in the reinforcing steel in one face may reach 1.5 times the yield strain, provided that the average stress in the reinforcing steel across the entire section does not exceed 0.9 fy. For additional discussion on this criterion, see Subsection 3.8.1.4.3, Item 4.

2. Based on service load combinations, as described in Subsection 3.8.1.3.2, Item 1, the allowable tensile stress in reinforcing steel is 50 percent of the yield stress (0.50 fy). However, this value is increased by 33 1/3 percent when considering load combinations which include any one (or more) of the following temporary loads:

   a. Temporary pressure loads during the test condition

   b. Temperature effects

   This criterion is in accordance with CC-3422 of the ASME-ACI 359 document, except that the allowable stress may not be increased 33 1/3 percent for earthquake or wind loads.

3.8.1.5.2 Concrete - Allowable Stresses and Strains

1. Based on factored load combinations, as described in Subsection 3.8.1.3.2, Item 2, the allowable concrete stresses are in accordance with CC-3411 of the ASME-ACI 359 document, as follows:

   a. Membrane compression = 0.60 f'c

   b. Membrane plus bending compression = 0.75 f'c

   c. Local compression at discontinuities and at the inside face due to temperature gradients from accident conditions = 0.90 f'c.

   d. Concrete tensile strength is not relied upon to resist flexural and membrane tension.

e. Radial shear is in accordance with CC-3411.4.1 of the ASME-ACI 359 document. (Where the calculated shear is greater than the allowable concrete shear, steel shear reinforcement is provided in accordance with CC-3521 of the ASME-ACI 359 document.)

f. Tangential shear is in accordance with CC-3411.5.1 of the ASME-ACI 359, as follows, except that the maximum allowable tangential shear stress carried by the concrete, Vc, does not exceed 60 psi.

   1) Provisions (a) and (d) of CC-3411.5.1 are satisfied, and this Containment does not support principal equipment laterally; therefore, the allowable tangential shear stress that can be resisted by the concrete, Vc, equals 60 psi.

   2) Where the maximum tangential shear stress, Vu, exceeds the concrete allowable shear, Vc, inclined reinforcement is provided to resist the excess (Vu - Vc).

   3) Design of the reinforcing steel for tangential shear is in accordance with CC-3521.1.1 of the ASME-ACI 359 document.

g. Requirements for punching shear are in accordance with Section 11.10.3 of the 1971 edition of the ACI 318 Code. | 66

2. Based on service load combinations as described in Subsection 3.8.1.3.2, Item 1, the allowable concrete stresses are in accordance with CC-3421 of the ASME-ACI 359 document, as follows:

a. Membrane compression = 0.30 f'c

b. Membrane plus bending compression = 0.45 f'c

68 Note: Allowable stresses indicated in Subsection 3.8.1.5.2, Items 2.a and 2.b are increased by 33 1/3 percent when considering load combinations which include thermal. (The allowable increase when considering wind or earthquake, as permitted by the ASME-ACI 359 document, is not applied.)

c. Local compression at discontinuities and in the vicinity of liner anchors = 0.60 f'c

d. Concrete tensile strength is not relied upon to resist flexural and membrane tension.

e. Allowable stress in shear is 50 percent of the values indicated for factored loads, except that 67 percent of the factored load stresses are allowed for load combinations which include the pressure loads during the test condition. Additional requirements are in accordance with ASME-ACI 359 document, CC-3421.3 and CC-3522, except that the tangential shear stress carried by the concrete, Vc, does not exceed 40 psi.

3.8.1.5.3 Liners, Anchors, and Attachments - Allowable Stresses and Strains

1. The allowable stresses and strains in the liner plate are in accordance with Table CC-3700 1 of the ASME-ACI 359 document, as follows:

a. Considering Calculated Membrane Stresses and Strains Only

1) Construction category: tensile or compressive stress = 2/3 x yield stress

2) Test category: compressive strain = .002 in./in.; tensile strain = .001 in./in.

3) Normal category: same as test category

4) Severe environmental category: same as test category

5) Extreme environmental category: same as test category

6) Abnormal category: compressive strain = .005 in./in.; tensile strain = .003 in./in.

7) Abnormal-severe environmental category: same as abnormal category

8) Abnormal-extreme environmental category: same as abnormal category

b. Considering Combined Membrane and Bending Stresses and Strains

1) Construction category: tensile and compressive stress = 2/3 x yield stress

2) Test category: compressive strain = .004 in./in.; tensile strain = .002 in./in.

3) Normal category: same as test category

4) Severe environmental category: same as test category

5) Extreme environmental category: same as test category

6) Abnormal category: compressive strain = .014 in./in.; tensile strain = .010 in./in.

7) Abnormal-severe environmental category: same as abnormal category

8) Abnormal-extreme environmental category: same as abnormal category

c. The load categories indicated previously include loads as defined in Subsection 3.8.1.3.2, except that load factors for all load cases are equal to 1.0.

2. The allowable forces and displacements of the liner anchors are in accordance with Table CC-3700-2 of the ASME-ACI 359 document, as follows:

a. Considering Mechanical Loads Only

1) Test category: applied load equals the lesser of 0.67 x yield load or 0.33 x ultimate load

2) Normal category: same as test category

3) Severe environmental category: same as test category

4) Extreme environmental category: same as test category

5) Abnormal category: applied load equals the lesser of 0.9 x yield load or 0.50 x ultimate load

6) Abnormal-severe environmental category: same as abnormal category

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7) Abnormal-extreme environmental category: same as abnormal category

b. Considering Displacement Limited Loads

1) Test category: displacement equals 0.25 x ultimate displacement

2) Normal category: same as test category

3) Severe environmental category: same as test category

4) Extreme environmental category: same as test category

5) Abnormal category: displacement equals 0.50 x ultimate displacement

6) Abnormal-severe environmental category: same as abnormal category

7) Abnormal-extreme environmental category: same as abnormal category

c. The load categories previously indicated include loads as defined in Subsection 3.8.1.3.2, except that load factors for all cases are equal to 1.0. Mechanical loads are defined as those which are not self-limiting or self-relieving with load application. Displacement limited loads are those resulting from constraint of the structure or constraint of adjacent material and are self-limiting or self-relieving. The yield and ultimate capacity of the liner stud anchors are based on the results of tests performed by the Nelson Stud Welding Company.

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3. Allowable stresses and strains in penetration assemblies are in
68 accordance with CC-3740 of the ASME-ACI 359 document. The design allowables for the penetration nozzles are the same as those used for metal containments (ASME B&PV Code, Section III, Division I), as discussed in Subsection 3.8.2. For additional criteria for ASME B&PV Code, Section III, Class MC steel components such as the airlocks and the equipment hatch, see Subsection 3.8.2.

68 4. The design allowables for overlay plates, brackets, and attachments are the same as those given in AISC Specification for resisting mechanical loads in the construction, test, and normal categories. For all other categories indicated in Subsection 3.8.1.3.2, the allowable can be increased by a factor of 1.5.

68 For overlay plates, brackets, and attachments which resist external mechanical loads and are not continuous through the liner plate, the allowable strength of the liner plate in the through-the-thickness direction will be taken as one-half of that in the transverse direction.

5. The liner is investigated for fatigue using the methods and limits established by ASME B&PV Code, Section III, Division I, Subsection NE.

3.8.1.5.4 Effect of Two- and Three-Dimensional Stress Strain Fields on the Behavior of the Structure

1. Liner

The computed stresses and strains in the liner consider the effect of the two dimensional stress-strain field by the use of

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Poisson's Ratio in stress and strain determination. The liner anchor design considers biaxial liner loading by not relying on liner plate yielding to limit the forces applied to the anchor; liner strains are converted to stress and membrane forces assuming the material remains elastic. Because of the use of this conservative design approach, biaxial yield test values are not required.

2. Major Penetrations

The analysis performed at the major penetrations, as described in Subsection 3.8.1.4.2, considers the three dimensional stress-strain field. Allowable stresses are the same as indicated in Subsections 3.8.1.5.1, 3.8.1.5.2, and 3.8.1.5.3.

3. Reinforced Concrete Section

Based on the conservative assumption of fully cracked concrete, in both directions, used in the design of the reinforcing steel, the effect of a two- or three-dimensional stress-strain field is not a consideration in the design of the typical reinforced concrete section in the wall and dome of the Containment. However, the effect of the two-dimensional-stress strain field in the liner, in calculating the hot liner thrust against the reinforced concrete section, is considered.

3.8.1.5.5 Effects of Repeated Reactor Shutdowns and Startup During the Life of the Plant

1. The number of assumed reactor shutdowns and startups during the life of the plant is assumed to be 120 cycles over a period of 40 years.

2. The cycled stresses and strains in the reinforced concrete sections caused by reactor shutdowns and startups are minor compared to the stresses caused by the critical design loading based on the abnormal (accident pressure and temperature) and extreme environmental (SSE) conditions. Therefore, the cycled stresses and strains caused by reactor shutdowns and startups do not degrade the margin of safety in the reinforced concrete.

3. The effect of cycled stresses and strains in the liner is considered by performing a fatigue analysis, in accordance with Subsection 3.8.1.5.3, Item 5, which includes the reactor shutdown and startup cycles.

3.8.1.5.6 Connections and Joints

The connections and joints of the various elements of the Containment, such as the crane girder bracket, liner end anchorage at bottom of wall, and similar elements, are designed using the same appropriate stress and strain allowables described elsewhere in Subsection 3.8.1.5.

3.8.1.5.7 Conditions at End of Service Life of Structure

Since a conventionally reinforced concrete containment is essentially a passive structure, as compared to a prestressed structure which relies on active prestress forces to meet its design function requirements, the margins of safety against all loading conditions are essentially the same throughout the life of this structure.

3.8.1.6 <u>Materials, Quality Control, and Special Construction Techniques</u>

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3.8.1.6.1 Concrete

1. Materials

a. Cement

Cement is in conformance with the requirements of ASTM C-150-74, Specification for Portland Cement, Type II.

b. Aggregates

Aggregates are in conformance with the requirements of ASTM C 33-74, Specification for Concrete Aggregates, with the following additional requirements:

1) Gradations 357 or 467 are not furnished as one graded aggregate, but are obtained by combining at least two separate gradation sizes.

2) The potential reactivity of the aggregate is established by the methods described in the Appendix to ASTM C 33-74.

3) Aggregate shapes and sizes are in accordance with CC-2222.1.1 of the ASME-ACI 359 document.

c. Mixing Water

Mixing water is clean and free from injurious amounts of oils, acids, alkalis, salts, and organic materials, or other substances which could be deleterious to concrete or steel. Quality control tests are in accordance with the requirements of CC-2223 of the ASME-ACI 359 document.

d. Admixtures

1) Air-entraining admixtures conform to the requirements of ASTM C 260-74, Specification for Air-Entraining Admixtures for Concrete.

2) Chemical admixtures conform to the requirements of ASTM C 494-71, Specification for Chemical Admixtures for Concrete

2. Concrete Strength

Concrete has a minimum compressive strength of 4000 psi, in 28 days, when tested in accordance with ASTM C 39-72.

3. Other Concrete Properties

The following concrete properties are determined in accordance with the noted ASTM standards:

ASTM C 78-64 for flexural strength

ASTM C 496-71 for splitting tensile strength

ASTM C 469-65 for static modulus of elasticity

ASTM C 642-69T for specific gravity of concrete

4. Selection of Concrete Mix Proportions

Concrete mix proportions are established on the basis of laboratory trial batches, in accordance with the requirements of CC-2232 of the ASME-ACI 359 document. The following industry standards are referred to in this document:

ACI 211.1-74	Recommended Practice for Selecting Proportions for Normal and Heavy Weight Concrete
ACI 214-65	Recommended Practice for Evaluation of Compression Test Results of Field Concrete
ASTM C 39-72	Test for Compressive Strength of Cylindrical Concrete Specimens
ASTM C 192-69	Making and Curing Concrete Test Specimens in the Laboratory

5. Construction of Concrete

Concrete construction, including the stockpiling, batching, mixing, conveying, depositing, consolidation, curing, and construction joint preparation, is in accordance with CC-4200 of the ASME-ACI 359 document. The following industry standards are referred to in this document:

ACI 304-73	Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete
ASTM C 94-74	Specification for Ready-Mixed Concrete
ACI 347-68	Recommended Practice for Concrete Formwork
ACI 305-72	Recommended Practice for Hot Weather Concreting

ACI 306-66	Recommended Practice for Cold Weather Concreting

6. Examination, Testing and Other Quality Control Procedures for Concrete

   a. Quality Assurance

      The QA procedures are in accordance with the requirements, in general, as described throughout the ASME-ACI 359 document.

      The quality control (QC) for concrete begins with the selection and testing of the ingredients of the mix and extends through proportioning, batching, mixing, transporting, placing, and curing.

   b. Testing and Examination of Concrete Ingredients

      1) Cement

         QC testing is in accordance with ASTM C 109-73, Test for Compressive Strength of Hydraulic Cement Mortars. Other QA requirements, including testing frequency, are in accordance with CC-5221 of the ASME-ACI 359 document.

      2) Aggregates

4 QC testing is in accordance with CC-5224 of the ASME-ACI 359 document with the exception that the frequency of tests in accordance with ASTM C289 can be one test for each two month period provided low alkali

cement is used. The following industry standards are called for in testing of aggregates to ascertain conformance to ASTM C 33-74, Specification for Concrete Aggregates: 4

ASTM C 131-69	Test for Resistance to Abrasion of 11 Size Coarse Aggregate by Use of the Los Angeles Machine
ASTM C 142-71	Test for Clay Lumps and Friable Particles in Aggregates
ASTM C 117-69	Test for Materials Finer Than No.200 Sieve in Mineral Aggregates by Washing
ASTM C 87-69	Test for Effect of Organic Impurities in Fine Aggregate on Strength of Mortar
ASTM C 40-73	Test for Organic Impurities in Sands for Concrete
ASTM C 289-71	Test for Potential Reactivity of Aggregates (Chemical Method)
ASTM C 136-71	Test for Sieve or Screen Analysis of Fine and Coarse Aggregates
ASTM C 88-73	Test for Soundness of Aggregates by Use of Sodium Sulfate or Magnesium Sulfate
ASTM C 127-73	Test for Specific Gravity and Absorption of Coarse Aggregate

ASTM C 295-65 Recommended Practice for Petrographic Examination of Aggregates for Concrete

3) Mixing Water

Quality control testing requirements, including testing frequency, are in accordance with CC-5225 of the ASME-ACI 359 document.

c. Testing and Examination of Concrete

Concrete is tested in accordance with CC-5230 of the ASME-ACI 359 document. The following is a summary of the major testing requirements, as stated in this document:

1) Slump Test

Testing is performed in accordance with ASTM C 172-71, Sampling Fresh Concrete, and ASTM C 143-74, Test for Slump of Portland Cement Concrete, CC-5232 of the ASME-ACI 359 document.

2) Air Content

Testing is performed in accordance with ASTM C 231-74, Test for Air Content of Freshly Mixed Concrete by the Pressure Method, CC-5233 of the ASME-ACI 359 document.

3) Mechanical Properties

Testing is performed in accordance with ASTM C 39-72, Test for Compressive Strength of Cylindrical Concrete Specimens, and ASTM C 31-69, Making and Curing Concrete Test Cylindrical Specimens in the Field. The samples

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for strength tests are taken in accordance with ASTM C 172-71, Sampling Fresh Concrete. Testing frequency, and the acceptance criteria, are in accordance with CC-5234.2 of the ASME-ACI 359 document.

d. Certification, Marking, and Identification of Materials

Certified materials test reports are prepared in accordance with the requirements of CC-2130 of the ASME-ACI 359 document. Marking and identification of materials are in accordance with CC-2250 of the ASME-ACI 359 document.

e. Certification and Tests and Examinations

Certification of tests and examinations is provided in accordance with the requirements of CC-4120 of the ASME-ACI 359 document.

3.8.1.6.2 Reinforcing Steel

1. Material Specification

Reinforcing steel conforms to the requirements of ASTM A 615-72 Grade 60. | 33

2. Physical Properties

The specified minimum yield strength is 60,000 psi, and the specified minimum ultimate strength is 90,000 psi. The minimum elongation is 7 percent in 8 inches.

3. Chemical Properties

Some arc-welding is performed on the reinforcing steel, such as the welded shear bars at the Containment wall (See Figure 3.8-10); the reinforcing steel which is welded has a limited chemistry as defined in CC-2333 of the ASME-ACI 359 document.

4. Fabrication and Installation of Reinforcing Steel

Fabrication and installation of reinforcing steel are in accordance with CC-4300 of the ASME-ACI 359 document.

5. Examination and Testing of Reinforcing Steel

Special materials testing is performed in accordance with CC-2330 of the ASME-ACI 359 document. This section requires that one full diameter tensile test bar from each bar size be tested for each 50 tons, or fraction thereof, of reinforcing bars produced from each heat of steel. Bend tests also are performed in accordance with CC-2332 of the ASME-ACI 359 document. This section requires that, for No. 14 and No. 18 bars, bend testing of bars be conducted at the rate of one test bar for each bar size from each heat. NRC Regulatory Guide 1.15 is also complied with.

6. Certification, Marking, and Identification of Materials

Certified materials test reports are furnished in accordance with the requirements of CC-2130 of the ASME-ACI 359 document. Marking and identification of reinforcing steel are in accordance with the requirements of CC-2320 and CC-4122 of the ASME-ACI 359 document.

7. Certification of Tests and Examinations

Certification of tests and examinations is provided in accordance with CC-4120 of the ASME-ACI 359 document.

3.8.1.6.3 Mechanical Butt Splices (Cadwelds)

No. 14 and No. 18 reinforcing bars are spliced by use of Cadweld connections, as described in CC-4333 of the ASME-ACI 359 document. Such splices develop the tensile limits shown in Table CC-4330 of that document. Crew qualification is in accordance with CC-5321. Nondestructive testing is in accordance with CC-5322. Tensile testing and test frequency conform to CC-5323 and CC-5324. The procedure for substandard test results conforms to CC-5325. Certified mill test reports are furnished in accordance with CC-2131. Marking and identification conform to CC-4122. Tests and examinations are certified in accordance with CC-4120. The requirements of the NRC Regulatory Guide 1.10 are also complied with.

3.8.1.6.4 Welding to Reinforcing Bars

Limited welding to reinforcing bars, such as required for welding shear bars to vertical reinforcing at the base of the Containment wall, conforms to the requirements of CC-4334 of the ASME-ACI 359 document.

3.8.1.6.5 Liner and Attachments

1. Materials

   a. Liner, including thickened plates at penetrations and crane girder brackets, is in accordance with SA 537-74 Class 2.

b. Penetration sleeves are in accordance with SA 333-74 Grade 6 (seamless), SA 537-74 Class 2, or SA 516-74 Grade 70.

c. Stud anchors are in accordance with ASTM A 108-73.

d. Embedded steel members in mat are in accordance with ASTM A 36-74.

68 e. Penetration caps are in accordance with SA 105-74, SA 350-74 Grade LF1 or LF2, SA 516-74 Grade 60, 65, or 70 or SA 333-74 Grade 6.

68 f. Forgings, including penetration forgings, are in accordance with SA 350-74 Grade LF1 or LF2 or SA 182-74 Type F316.

68 g. Penetration anchorage studs and lugs are in accordance with 108-73 Grades 1015, 1016, or 1018 or ASTM A36-74 or SA 516-74 Grades 60, 65, or 70 or SA 537-74 Class 2.

68 h. Welding materials are in accordance with CC-2600 of the ASME-ACI 359 document, with the exceptions as specified in Section 3.8.1.2.5.2.a.

2. Special Materials Testing and Examination

a. Notch Toughness Testing

Notch toughness testing is performed on liner materials in accordance with the requirements of CC-2520 of the ASME-ACI 359 document.

b. Ultrasonic Testing

Liner plate materials which must transmit orthogonal loads in the through-the-thickness direction, such as the thickened liner plate at the crane girder brackets, are examined by the straight-beam ultrasonic method in accordance with ASTM A 578-71b. This examination, the acceptance standards, and related procedures are in accordance with CC-2530 of the ASME-ACI 359 document.

Liner plate in the area of overlay plates and/or structural 68
attachments are not ultrasonic tested. The liner plate
design, discussed in Subsection 3.8.1.5.3.4, will consider
through-the-thickness properties.

c. Fabrication, Installation, and Welding of Liner

The following is a list of the major portions of CC-4500 of 68
the ASME-ACI 359 document, covering the requirements for
fabrication, installation, and welding of the liner, with the
exceptions as specified in Section 3.8.1.2.5.2.a.

CC-4520	Forming, Fitting, and Aligning Exemptions and clarifications are discussed in Subsection 3.8.1.2.5(2)(a). 68
CC-4522	Forming Tolerances
CC-4530	Welding Qualifications
CC-4540	Rules Governing Making, Examining, and Repairing Welds Clarifications are described in 3.8.1.2.5.2.a 68

CC-4550 Heat Treatment

68 Clarifications are described in 3.8.1.2.5.2.a

d. Examination of Liners and Attachments

68 Examination of liners and attachments are in accordance with CC-5500 of the ASME-ACI 359 document as described in 3.8.1.2.5.2.a. Leak chase systems are installed at inaccessible welds. NRC Regulatory Guide 1.19 is also complied with.

e. Certification, Marking, and Identification of Materials

Certified materials test reports are prepared in accordance with the requirements of CC-2130 of the ASME-ACI 359 document. Marking and identification of liner materials are in accordance with the requirements of CC-2541 of the ASME-ACI 359 document.

f. Certification of Tests and Examinations

Certification of tests and examinations are provided in accordance with the requirements of CC-4120 of the ASME-ACI 359 document.

g. Protective Coatings

Suitable protective coatings are applied to the interior surfaces of the Containment liner. The primary criteria for the selection of the protective coatings are:

1) Capability of withstanding the DBA conditions of pressure, temperature, and radiation

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2) Capability of being easily decontaminated

3) Capability of not reacting chemically with spray solutions

4) Capability of preventing the formation of gaseous or solid waste products

5) Capability of resisting the environmental radiation for the life of the plant

The coatings are required to pass qualification tests as specified in ANSI N101.2-1972, Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities.

3.8.1.6.6 Personnel Airlock, Equipment Hatch, and Emergency Airlock

1. Materials

   a. Plates are in accordance with SA 537-74 Class 2 or SA 516-74 Grade 70.

   b. Forgings are in accordance with SA 350-74.

   c. Bolts are in accordance with SA 320-74.

2. Code Requirements

   The personnel airlock, equipment hatch, and emergency airlock satisfy all the requirements (materials, fabricating, welding, examination, testing, and other requirements) of the ASME B&PV Code, Section III, Division 1, Subsection NE, Class MC components. (See Subsection 3.8.2.) The personnel and emergency airlocks meet the requirements to obtain an N-stamp.

3.8.1.6.7 Quality Assurance Program

The documentation and maintenance of a QA program in the construction of the Containment are in accordance with CC-2700 of the ASME-ACI 359 document and in accordance with Chapter 17 of this FSAR. NRC Regulatory Guide No. 1.28 is also complied with.

3.8.1.7 Testing and Inservice Inspection Requirements

3.8.1.7.1 Structural Acceptance Test of Containment

The completed Containment structure is subjected to an acceptance test by which the internal pressure is increased from atmospheric pressure to a value of 1.15 times the Containment design pressure (1.15 x 50 psig = 57.5 psig) in five or more approximately equal pressure increments. The Containment is depressurized in the same number of
68 increments. The purpose of the structural acceptance test is to demonstrate that the structure responds satisfactorily to the required internal pressure loads by making measurements of deflections and deformations under load to provide correlation with the theoretically predicted response.

Since this Containment is not a prototype, the measurements are limited to gross deformations and crack mapping. Strain measurements are not taken. The prototype containment on which strain measurements are correlated with deflection measurements is Consolidated Edison's Indian Point No. 2, NRC Docket No. 50247-47.

9

10 To the extent feasible, the test is conducted during a period of stable ambient temperature, atmospheric pressure, and humidity. Inside and outside of the Containment, atmospheric temperature, pressure, and humidity are monitored continuously during each test. The test will not be conducted under extreme weather conditions such as snow, heavy rain, or strong wind.

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Deflection measurements are recorded at atmospheric pressure and at
each level of the pressurization and depressurization cycle. At each
level, the pressure is held constant for at least one hour before the
deflections are recorded. Radial displacements of the containment 18
cylinder are measured at five approximately equally spaced elevations
between the base slab and dome springline and at the dome springline.
These measurements are made along four azimuths spaced approximately
equally around the containment. Radial deflections of the
containment wall adjacent to the largest opening (the equipment hatch)
are measured at 12 points as described in paragraph CC-6232(b) of the
ASME B&PV Code, Section III, Division 2 (1980 Edition). The increase
in diameter of the opening is measured in two mutually perpendicular
directions. Vertical displacements of the cylinder at the top
relative to the base are measured along four azimuths as described
above. Vertical deflections of the containment dome at the apex and
at two other equally spaced intermediate points, between a point near
the apex and the springline, are measured along one azimuth. A
listing of the numerical values, in the form of acceptance criteria
for the measurements that are taken during the structural acceptance
test, are given on Figure 3.8-19.

Crack patterns that exceed 0.01 inches in width before, during or 18
after the test are mapped at locations described in paragraph CC-6233
of the ASME B&PV Code, Section III, Division 2 (1980 Edition).

When structural acceptance test results are available, a report will 18
be furnished separately. The analysis of data and preparation of
report will be in accordance with paragraph CC-6260 of the ASME B&PV
Code, Section III, Division 2 (1980 Edition).

All aspects of the structural integrity test, including the acceptance 18
criteria are in accordance with paragraph CC-6000 of the ASME B&PV
Code, Section III, Division 2, 1980 Edition with Summer 1980 Addenda
(except that paragraph CC-6212.1, CC-6234 and CC-6236 do not apply

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18 since the CPSES Containment is not a prototype containment).

3.8.1.7.2 Initial Leakage Rate Tests

Containment leakage testing is in accordance with all the requirements of 10 CFR Part 50, Appendix J, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors (2/5/73).

68

68 A preoperational Type A integrated leakage-rate test is performed at the calculated peak Containment internal pressure and also at a reduced pressure of not less than 50% of the peak pressure. Type B tests of components and Type C tests of Containment isolation valves
68 are performed in accordance with 10 CFR Part 50, Appendix J. For calculated peak containment internal pressure, see Section 6.2.1.

68 The maximum allowable leakage-rate (La as defined in 10 CFR Part 50, Appendix J), related to the maximum Containment leakage under design basis pressurization accident conditions, is 0.10 percent of the weight of contained air, at the calculated peak Containment internal pressure per 24 hour period.

For a discussion of the test objectives and the acceptance criteria, see the Technical Specifications for Containment Tests.

Q022.22

54 Test methods are in accordance with ANSI N45.4-1972, Leakage Rate Testing of Containment Structures for Nuclear Reactors (03/26/73), with the exceptions of isolated penetrations and the use of the mass-plot method per ANSI/ANS 56.8-1981 as stated in Table 14.2-2 (Sheet 59 of 60).

3.8.1.7.3 Inservice Inspection Program

The inservice inspection program conforms to the requirements of 10 CFR Part 50, Appendix J.

3.8.2 STEEL CONTAINMENT

This section, as outlined in the NRC Regulatory Guide 1.70, Rev. 2., Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, regarding a steel containment, is not applicable to the Comanche Peak Containment structure because a steel-lined reinforced concrete containment is being used, as described in Subsection 3.8.1. Certain steel components in the Containment system are classified in 4
accordance with ASME B&PV Code, Section III, as Class MC components, as described in Subsection 3.8.1. These are the personnel airlock, the equipment hatch, the emergency airlock, and other penetrations subject to pressure-induced stresses.

This section addresses itself to the requirements of these ASME B&PV Code, Section III, Class MC steel components.

3.8.2.1 Description of the Containment

For the description of the components (airlocks, equipment hatch, and other penetrations), see Subsection 3.8.1.1.6. The Containment itself is a steel-lined reinforced concrete containment.

3.8.2.2 Applicable Codes, Standards, and Specifications

3.8.2.2.1 Basic Code

The basic code for these steel components is the ASME B&PV Code, Section III, Division 1, Subsection NE, for Class MC components. This code is applicable for all the requirements of the components,

including materials, design, fabrication, examination, testing, and so
68 forth. The 1971 edition of the code, through and including the summer 1973 addenda, is used for the electrical penetration sleeves, fuel transfer tube penetration sleeve, emergency and personnel airlocks, and equipment hatch. The 1974 edition through and including the Summer 1976 Addenda is used for the process piping penetrations.

3.8.2.2.2 Other Applicable Codes, Specifications, and Standards

68 See Section 3.8.1.2.3 and 3.8.2.6.6 of this FSAR.

68 3.8.2.2.3 Applicable NRC Regulatory Guide

68 NRC Regulatory Guide 1.57	Design Limits and loading Combinations for Metal Primary Reactor Containment System Components (6-73) (Applicable only to appropriate containment components constructed in accordance with Subsection NE of the ASME B&PV Code, Section III, Class MC.)

3.8.2.3 Loads and Load Combinations

The applicable loads stated in Subsection 3.8.1.3 are considered in the design of the ASME B&PV Code, Section III, Class MC steel components. For the various load combinations used in the design of Class M components and the related allowable stresses and strains, see Subsection 3.8.2.5.

3.8.2.4 Design and Analysis Procedures

The design and analysis of the Class MC components are in accordance with all the requirements of Subsection NE of the ASME B&PV Code, Section III, including the applicable portions of Appendix A. Analysis procedures also use published formulas such as Roark's Formulas for Stress and Strain and Timoshenko's Theory of Plates and Shells.

3.8.2.5 Structural Acceptance Criteria

3.8.2.5.1 General Criteria

The design is such that all the stress and strain limits, as defined in NE-3000 of the ASME B&PV Code, Section III, are satisfied for pressure loads in combination with all mechanical loads and thermal loads, as discussed in Subsection 3.8.2.5.2.

3.8.2.5.2 Design Load Combination Stress Limits

1. The ASME B&PV Code, Section III, design criteria for Class MC components are based on establishing stress and strain limits which vary depending on the following factors:

   a. Types of Stress

      As defined by ASME B & PV Code Section III: | 68
      P_m - General Primary Membrane Stress | 68
      P_l - Local Primary Membrane Stress | 68
      P_b - Primary Bending Stress | 68
      P_e - Secondary Expansion Stress | 68
      Q - Secondary Membrane plus Bending Stress | 68

68 F - Peak Stress

68 S_y - Yield Strength

68 S_u - Tensile Strength

68 S_m - Design Stress Intensity Value

68 b. Types of Loads

68 The loads considered in the load combinations in 3.8.2.5.2.2 are as defined in Section 3.8.1.3.1 and as follows:

68 P_e - Design External Pressure

68 T_e - Thermal loads under thermal conditions during event causing external pressure

68 R_e - Pipe reaction under thermal conditions during event causing external pressure

68 2. Load Combinations and Acceptance Criteria

	LOAD COMBINATION	ACCEPTANCE CRITERIA
68	$D+L+1.15\ P_a+T_t$	$P_m < 0.9\ S_y$
68		P_l or $P_l + P_b \leq 1.25\ S_y$
68		$P_l+P_b+P_e+Q \leq 3\ S_m$
68		$P_l+P_b+P_e+Q+F$ Fatigue Analysis
68	$D+L+T_o+R_o$	$P_m \leq S_m$
68	$D+L+T_o+R_o+E$	P_l or $P_l+Pb \leq 1.5\ S_m$
68		$P_l+P_b+P_e+Q \leq 3\ S_m$
68		$P_l+P_b+P+Q+F$ Fatigue Analysis
68	$D+L+T_a+R_a+P_a+E$	$P_m \leq S_m$
68	$D+L+T_e+R_e+P_e+E$	P_l or $P_l + P_b \leq 1.5\ S_m$

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LOAD COMBINATION	ACCEPTANCE CRITERIA	| 68
$D+L+T_a+R_a+P_a+E'$	$P_m \leq$ Larger of 1.2 S_m or S_y	| 68
$D+L+T_e+R_e+P_e+E'$	P_l or $P_l + P_b <$ Larger of	| 68
	1.8 S_m or 1.5 S_y	| 68
$D+L+T_a+R_a+P_a+Y_r+Y_j+Y_m+E'$	$P_m \leq$ 0.85 times smaller	| 68
	of 0.7 S_u	| 68
	or $(S_y + \frac{S_u - S_y}{3})$	| 68
		| 68
	P_l or $P_l+P_b \leq (1.5)(0.85)$	| 68
	times smaller of 0.7 S_u or	| 68
	$(S_y + \frac{S_u - S_y}{3})$	| 68
		| 68

3. Compressive Stresses

In areas of compressive stress, buckling criteria are considered in accordance with the applicable sections of NE-3000 of the ASME B&PV Code, Section III.

4. Fatigue Analysis | 68

The requirements for an analysis for cyclic operation is investigated in accordance with NE-3131 (d) and the referenced portions therein.

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

The Containment is a steel-lined reinforced concrete structure. The airlocks, equipment hatch, and other penetrations are fabricated from the following materials in accordance with Section III of ASME-ACI 359, Proposed Standard Code for Concrete Reactor Vessels and Containments (April 1973):

1. Plate is in accordance with ASME SA-537 Class 2.

2. Penetrations are in accordance with ASME SA-333 Grade 6, SA-537 Class 2, or SA-516 Grade 70.

12 Penetration caps are in accordance with ASME SA-105, SA-350 Grade LF1 or LF2, SA-516 Grade 60,65 or 70 or SA-333 Grade 6.

11 3. Forgings including penetration forgings are in accordance with ASME SA-350 Grade LF1 or LF2 or SA-182 Type F316.

4. Bolting is in accordance with ASME SA-320.

The materials meet all the requirements in the ASME B&PV Code, Section III, Division 1, Subsection NE, for Class MC components, including the Charpy impact test requirements.

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

3.8.2.7 Testing and Inservice Inspection Requirements

Testing of Class MC components is in accordance with 10 CFR Part 50, Appendix J. Overall Containment testing is in accordance with Subsection 3.8.1.7.

3.8.3 CONCRETE AND STEEL INTERNAL STRUCTURES OF STEEL OR CONCRETE CONTAINMENTS

3.8.3.1 Description of the Internal Structures

3.8.3.1.1 General Description

The Containment internal structures are primarily of reinforced concrete and consist of the following major elements:

1. Primary Shield Wall (Reactor Cavity)

   The primary shield wall, a heavily reinforced concrete cylinder, is situated at the approximate center of the Containment vessel, and extends up from the interior base slab to surround the reactor vessel. This reactor cavity structure provides support for the reactor vessel. The vessel supports consist of support pads and shoes which are mounted on support members within the concrete cavity structure. During normal operation, the primary shield wall provides biological shielding for maintenance inspection. Under seismic loading, this structure serves to provide seismic shear resistance and stiffens the Containment internal structure.

2. Primary Loop Compartment Walls (Steam Generator Compartment)

   The compartments are formed by the secondary shielding walls on the exterior and by the reactor and refueling canal walls on the interior. These walls extend from the interior base slab up to

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the operating floor. The compartment houses the steam generator, reactor coolant pumps, and the RCLs. The compartment walls provide radiation shielding, isolation of the RCS, and lateral restraint for the steam generator, pump, and pressurizer.

3. Operating Floor

68 The operating floor is supported by the primary loop compartment
walls and concrete columns adjacent to the containment shell which
extend down to the interior base slab. The operating floor
provides a working and access floor during refueling, maintenance,
68 and repair operations. Vent areas are provided where required.

4. Refueling Cavity

The refueling cavity provides shielded access for transport of spent fuel and new fuel between the reactor vessel and the fuel transfer penetration. It also provides shielding storage space for the reactor vessel internals during refueling or maintenance. The cavity is lined with stainless steel.

5. Interior Base Slab

68 The interior base slab is placed on top of the foundation mat liner plate. This slab provides lateral and flexural restraint at the base of the primary loop compartment walls and the primary shield wall. The slab ties the primary loop compartment walls to the primary shield walls and provides a diaphragm for seismic shear distribution at the bottom of the internal structure. It also protects the foundation mat liner from any missiles generated in the primary loop compartments and from the effects of accident temperatures.

6. Missile Shields

The primary loop compartment walls and the operating deck provide missile protection for the RCS from potential missiles outside the primary loop compartment. Conversely, the walls and interior base slab provide missile protection to the Containment and to the safeguard and auxiliary systems located outside the primary loop compartment from postulated missiles originating inside the primary loop compartments. The missile-shielding function of the primary loop compartment walls is supplemented by a control rod drive missile shield positioned over the reactor vessel head, which is designed to contain any postulated ejected control rods from the reactor vessel. This missile shield is removed during refueling operations.

7. Intermediate Floors

Intermediate floors are provided at several elevations, including a principal floor at elevation 860 ft for miscellaneous equipment supports, access, maintenance, and similar items.

8. Removable Slabs and Walls

Removable slabs and walls are provided, where required, for maintenance access, e.g., the missile shield over the reactor and in the walls around the regenerative heat exchanger and reactor coolant drain pumps.

3.8.3.1.2 Polar Crane

A polar crane is provided inside the Containment, on a circular steel runway girder which is supported by brackets from the cylindrical Containment wall. The polar crane will remain stable and not become derailed when subjected to the specified load combinations (See FSAR Sections 3.8.3.3.3 and 9.1.4.3.2). | 68

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68 The primary load path from the crane wheels to the runway girders is maintained within allowable stress limits. Under certain combinations of load and crane position, some degree of local plastic deformation is assumed to achieve this load path.

3.8.3.1.3 Supports for Reactor Pressure Vessel, Steam Generator, Reactor Coolant Pump, Pressurizer, and Loop Piping

Descriptions of the supports for the reactor pressure vessel, steam generator, reactor coolant pump, pressurizer, and loop piping are presented in Section 5.4.14.

3.8.3.1.4 Drawings

For various details of internal structures, see Figure 3.8-15.

3.8.3.2 <u>Applicable Codes, Standards, and Specifications</u>

3.8.3.2.1 Basic Codes

Except where specifically noted otherwise, the basic codes used for the materials, design, construction, and so forth of the Containment internal structures are the applicable sections of the following:

1. Reinforced concrete is in accordance with ACI 318-71 Building Code Requirements for Reinforced Concrete.

68 2. Structural steel is in accordance with AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings (1969) including Supplement Numbers 1, 2, and 3 hereafter referred to as AISC Specification. Except: when supported by an engineering analysis, connections using A325 or A490 high strength bolts need not be pretensioned to the values required by AISC Specification, Table 1.23.5.

Visual inspection of structural welds will be in accordance with 55
Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants (VWAC) Revision 2 dated May 7, 1985. (SEE Appendix 3.8A)

The acceptance criteria contained in VWAC Revision 2 are applicable to 55
all structural steel welds at CPSES other than ASME-class structural welds.

3.8.3.2.2 Supplementary Codes

Where specifically referred to in this section on internal structures, 68
certain portions of the ASME-ACI 359 document apply, as described in Subsection 3.8.1.2.1.

ACI 349-76, Appendix A, is used to determine stresses resulting from 68
thermal gradients.

3.8.3.2.3 Additional Specifications and Standards

The following is a list of additional specifications and standards which are applicable to the internal structures:

1. Concrete

   (Same as Subsection 3.8.1.2.3, Item 3.)

2. Steel

ASTM A 36-74	Specification for Structural Steel
ASTM A 307-74	Specification for Carbon Steel Externally and Internally Threaded Standard Fasteners

	ASTM A 325-74	Specification for High-Strength Bolts for Structural Steel Joints, Including Suitable Nuts and Plain Hardened Washers
	ASTM A 240-74a	Specification for Heat Resisting Chromium and Chromium Nickel Stainless Steel Plate, Sheet, and Strip for Fusion Welded Unfired Pressure Vessels
38	ASTM A 370-74	Methods and Definitions for Mechanical Testing of Steel Products
	ASTM A 578-71b	Specification for Straight-Beam Ultrasonic Examination of Plain and Clad Steel Plates for Special Applications
	ASTM A 6-74	Specification for General Requirements for Delivery of Rolled Steel Plates, Shapes, Sheet Piling, and Bars for Structural Use
68	RCRBSJ - 1974	Specification for Structural Joints Using ASTM A325 or A490 Bolts, AISC 1974 Edition. RCRBSJ - Research Council on Riveted and Bolted Structural Joints.

3. Reinforcing Steel

33	ASTM A 615-72	Specification for Deformed and Plain Billet Steel Bars for Concrete Reinforcement (Grade 60)

4. Polar Crane

Crane Manufacturer's Association of America, Inc. (CMAA) Specification No. 70

3.8.3.2.4 Summary of Principal Plant Specifications

The principal plant specifications for the internal structures are primarily those involving reinforced concrete construction. The applicable portions of the ASME-ACI 359 document, as described in Subsection 3.8.1.2.1, are included in the plant construction specifications in regard to materials, construction techniques, examination, quality control, and so forth. The following principal portions of the ASME-ACI 359 document are incorporated into the plant specifications:

1. Reinforcing Steel Specification Same as Subsection 3.8.1.2.5, Item 2.b.

2. Concrete Specifications
   Same as Subsection 3.8.1.2.5, Item 2.d.

3. Cadweld Splices

   Generally, large-size bars requiring Cadweld splices are not used in the internal concrete structures. Where such splices are used, the requirements indicated in Subsection 3.8.1.2.5, Item 2.c are complied with.

4. Structural Steel

   The plant specification for structural steel includes the requirements in AISC Specification. | 08

3.8.3.2.5 Applicable NRC Regulatory Guides

The following NRC Regulatory Guides are applicable to the internal structures and are complied with:

NRC Regulatory Guide 1.15	Testing of Reinforcing Bars for Category I Concrete Structures (Revision 1, 12-28-72 of former Safety Guide 15)
NRC Regulatory Guide 1.28	Quality Assurance Program Requirements (Design and Construction) (6-7-72 of former Safety Guide 28)

3.8.3.3 <u>Loads and Load Combinations</u>

3.8.3.3.1 Loads

68 The loads and load combinations for supports which are supplied by Westinghouse are provided in Section 3.9N.1.4. The following loads are considered in the design of the internal structures of the Containment:

1. Normal Loads

   Normal loads are those loads which are encountered during normal plant operation and shutdown. They include the following:

   a. D = dead loads, including any permanent equipment loads, and their related moments and forces

68 b. L = live loads, including any movable equipment loads and other loads which vary in intensity and occurrence such as soil and hydrostatic pressures, pressure differences caused by variation in heating, cooling, and their related moments and forces

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c. To = thermal effects and loads during normal operating or shutdown conditions based on the most critical transient or steady-state condition

d. Ro = pipe reactions during normal operating or shutdown conditions based on the most critical transient or steady-state condition

2. Severe Environmental Loads

Severe environmental loads are those loads that could be encountered infrequently during the plant life. This category includes the following:

Feqo = loads generated by 1/2 the SSE
= OBE

3. Extreme Environmental Loads

Extreme environmental loads are those loads which are credible but highly improbable. They include the following:

Feqs = loads generated by the SSE

4. Abnormal Loads

Abnormal loads are loads generated by a postulated high energy pipe break accident within the Containment or compartment thereof. This category includes the following:

a. Pa = pressure equivalent static load within or across a compartment generated by the postulated break, including an appropriate dynamic factor to account for the dynamic nature of the load

b. Ta = thermal loads under thermal conditions generated by the postulated break, including To

c. Ra = pipe reactions under thermal conditions generated by the postulated break, including Ro

d. Yr = equivalent static load on the structure generated by the reaction on the broken high energy pipe during the postulated break, including an appropriate dynamic factor to account for the dynamic nature of the load

e. Yj = jet impingement equivalent static load on the structure generated by the postulated break, including an appropriate dynamic factor to account for the dynamic nature of the load

f. Ym = missile impact equivalent static load on the structure generated by or during the postulated break, such as pipe whipping, including an appropriate dynamic factor to account for the dynamic nature of the load

In determining an appropriate equivalent static load for Yr, Yj, and Ym, elastoplastic behavior is assumed with appropriate ductility ratios as long as excessive deflections do not result in loss of function. For concrete structures, the ductility ratios are described in Section 3.5.3.2.

5. Other Definitions

68 a. For structural steel, S is the required section strength based on the elastic design methods and the allowable stresses defined in AISC Specification.

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A 33-percent increase in allowable stresses for concrete and steel because of seismic or wind loadings is not permitted.

b. For concrete structures, U is the section strength required to resist design loads and is based on methods described in ACI 318-71.

c. For structural steel, Y is the section strength required to resist design loads and is based on plastic design methods described in Part 2 of AISC Specification. | 68

3.8.3.3.2 Load Combinations and Acceptance Criteria for Internal Concrete Structures of the Containment

1. Load Combinations for Service Load Conditions

| Q130.18

a. U = 1.4 D + 1.7 L | 1

b. U = 1.4 D + 1.7 L + 1.9 Feqo | 1

| Q130.18

If thermal stresses due to To and Ro are present the following combinations also apply: | 1

| Q130.18

c. U = .75 (1.4 D + 1.7 L + 1.7 To + 1.7 Ro) | 1

d. U = .75 (1.4 D + 1.7 L + 1.9 Feqo + 1.7 To + 1.7 Ro) | 1

| Q130.18

L is considered for its full value or its complete absence. | 1

2. Load Combinations for Factored Load Conditions

For conditions that represent extreme environmental, abnormal, abnormal/severe environmental, and abnormal/extreme environmental conditions, respectively, the following load combinations are satisfied:

a. $U = D + L + To + Ro + Feqs$

b. $U = D + L + Ta + Ra + 1.5\ Pa$

c. $U = D + L + Ta + Ra + 1.25\ Pa + 1.0\ (Yr + Yj + Ym) + 1.25\ Feqo$

d. $U = D + L + Ta + Ra + 1.0\ Pa + 1.0\ (Yr + Yj + Ym) + 1.0\ Feqs$

In combinations b, c, and d, the maximum values of Pa, Ta, Ra, Yj, Yr, and Ym, including an appropriate dynamic factor, are used unless a
68 time history analysis is performed to justify otherwise. Combinations c and d and the corresponding structural acceptance criteria shall be first satisfied without Yr, Yj, and Ym. When considering these loads, local section strength capacities may be exceeded under these concentrated loads, provided there will be no loss of function of any safety-related system.

L is considered for its full value or its complete absence.

3.8.3.3.3 Load Combinations and Acceptance Criteria for Internal Steel Structures of the Containment

1. Load Combinations for Service Load Conditions

68 Either the elastic working stress design methods of Part 1 of AISC Specification or the plastic design methods of Part 2 of AISC Specification are used.

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a. When the elastic working stress design methods are used, the following apply:

1) $S = D + L$

2) $S = D + L + Feqo$

When thermal stresses caused by To and Ro are present the following combinations are satisfied: | 68

3) $1.5\ S = D + L + To + Ro$

4) $1.5\ S = D + L + To + Ro + Feqo$

L is considered for its full value or its complete absence.

b. When plastic design methods are used, the following apply: | Q130.18

1) $Y = 1.7D + 1.7L$ | 68

2) $Y = 1.7D + 1.7L + 1.7\ Feqo$ | 68

When thermal stresses caused by To and Ro are present, the following combinations are satisfied: | 68

3) $Y = 1.3\ D + 1.3\ L + 1.3\ To + 1.3\ Ro$ | 68

4) $Y = 1.3\ D + 1.3\ L + 1.3\ To + 1.3\ Ro + 1.3\ Feqo$ | 68

L is considered for its full value or its complete absence.

2. Load Combinations for Factored Load Conditions

   a. If elastic working stress design methods are used, the following load combinations are satisfied:

      1) 1.6 S = D + L + To + Ro + Feqs

      2) 1.6 S = D + L + Ta + Ra + Pa

      3) 1.6 S = D + L + Ta + Ra + Pa + 1.0 (Yj + Yr + Ym) + Feqo

Q130.18 |

1 | 4) 1.7 S = D + L + Ta + Ra + Pa + 1.0 (Yj + Yr + Ym) + Feqs

   b. If plastic design methods are used, the following load combinations are satisfied:

      1) .90 Y = D + L + To + Ro + Feqs

      2) .90 Y = D + L + Ta + Ra + 1.5 Pa

      3) .90 Y = D + L + Ta + Ra + 1.25 Pa + 1.0 (Yj + Yr + Ym) + 1.25 Feqo

      4) .90 Y = D + L + Ta + Ra + 1.0 Pa + 1.0 (Yj + Yr + Ym) + 1.0 Feqs

In these combinations, thermal loads are neglected when they are secondary and self-limiting in nature and when the material is ductile.

In combinations shown in Items 2.a.2), 3), and 4), and in Items 2.b.2) 3), and 4), the maximum values of Pa, Ta, Ra, Yj, Yr, and Ym, including an appropriate dynamic factor, are used unless a time history analysis is performed to justify otherwise.

In determining the equivalent static load for the differential pressure Pa, the impulsive nature of the load is taken into account by considering the time history of the applied pressure and the natural frequencies of the structures to which the pressure is applied (including the secondary shield walls and operating and intermediate floors). The steel is designed so that the maximum stress for any load combination, which includes differential pressure, is less than the yield stress, thus assuring elastic behavior.

For combinations shown in Subsection 3.8.3.3.3, Items 2.a.3) and 4), and in Subsection 3.8.3.3.3, Items 2.b.3) and 4), local stresses caused by the concentrated loads Yr, Yj, and Ym exceed the allowables when there is no loss of function of any safety-related system. Furthermore, in computing the required section strength, the plastic section modulus of steel shapes is used.

3.8.3.3.4 Variable Loads

For loads which vary, the values (within the possible range) which produce the most critical combination of loading are used in design.

3.8.3.3.5 Interrelated Loads

For loads which are interrelated as a function of time, such as accident-induced pressure and jet and thermal effects, the maximum values of these effects do not necessarily occur simultaneously. Recognition is given to the time increments associated with these postulated failure conditions.

Table 3.9B-10

(Sheet 11)

ACTIVE VALVES

Valve Identification or Location No	System	Valve Type and Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function	
HV-2459	AF	Globe/Air	3	3	Self-Actuated	Open	AFW Flow Path, Steam Generator Isolation	66
HV-2460	AF	Globe/Air	3	3	Self-Actuated	Open	AFW Flow Path, Steam Generator Isolation	66
HV-2461	AF	Globe/Air	3	3	Self-Actuated	Open	AFW Flow Path, Steam Generator Isolation	66
HV-2462	AF	Globe/Air	3	3	Self-Actuated	Open	AFW Flow Path, Steam Generator Isolation	66
HV-2484	AF	Butterfly Motor	12	3	Auto Trip	Open	Condensate Storage Tank Isolation	66
HV-2485	AF	Butterfly Motor	12	3	Auto Trip	Open	Condensate Storage Tank Isolation	66
AF-045	AF	Check	3	3	Self-Actuated	Closed	Pump Recirculation Flow Path	66
AF-057	AF	Check	3	3	Self-Actuated	Closed	Pump Recirculation Flow Path	66
AF-069	AF	Check	3	3	Self-Actuated	Closed	Pump Recirculation Flow Path	66
AF-167	AF	Check	8	3	Self-Actuated	Closed	Pump Recirculation Flow Path	66
HV-4777	CT	Gate/Motor	16	2	Auto Trip	Closed	CI, Discharge Path Flow	66
CT-145	CT	Check	16	2	Self-Actuated	Closed	CI, Discharge Path Flow	66
HV-4776	CT	Gate/Motor	16	2	Auto Trip	Closed	CI, Discharge Path Flow	66
CT-142	CT	Check	16	2	Self-Actuated	Closed	CI, Discharge Path Flow	66
HV-4782	CT	Gate/Motor	16	2	Auto Trip	Closed	CI, Recirculation Flow Path	55
HV-4783	CT	Gate/Motor	16	2	Auto Trip	Closed	CI, Recirculation Flow Path	55
CT-076 (HV-4758)	CT	Gate/Motor	16	2	Remote Manual	Open	CT Flow Path from RWST, Train A, Recirculation Boundary	66
CT-024 (HV-4759)	CT	Gate/Motor	16	2	Remote Manual	Open	CT Flow Path from RWST, Train B, Recirculation Boundary	66

Table 3.9B-10

(Sheet 12)

ACTIVE VALVES

Valve Identification or Location No.	System	Valve Type and Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function	
CT-135 (LV-4755)	CT	Diaphragm Motor	3	3	Auto Trip	Closed	Chemical Additive Flow to Suction Train B	: 68
CT-136 (LV-4754)	CT	Diaphragm Motor	3	3	Auto Trip	Closed	Chemical Additive Flow to Suction Train A	: 68
CT-072	CT	Check	2	2	Self-Actuated	Closed	Isolation of Chemical Additive Subsystem During Recirculation Train A	: 66
CT-082	CT	Check	2	2	Self-Actuated	Closed	Isolation of Chemical Additive Subsystem During Recirculation Train A	: 66
CT-029	CT	Check	2	2	Self-Actuated	Closed	Isolation of Chemical Additive Subsystem During Recirculation Train B	: 66
CT-031	CT	Check	2	2	Self-Actuated	Closed	Isolation of Chemical Additive Subsystem During Recirculation Train B	: 66
CT-065	CT	Check	10	2	Self-Actuated	Closed	Containment Spray Discharge Flow Isolation, Train A	: 66
CT-094	CT	Check	10	2	Self-Actuated	Closed	Containment Spray Discharge Flow Isolation, Train A	: 66
CT-013	CT	Check	10	2	Self-Actuated	Closed	Containment Spray Pump Discharge Flow Isolation, Train B	: 66
CT-042	CT	Check	10	2	Self-Actuated	Closed	Containment Spray Pump Discharge Flow Isolation, Train B	: 66

Table 3.9B-10

(Sheet 13)

ACTIVE VALVES

Valve Identification or Location No.	System	Valve Type and Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function	
CT-025	CT	Check	16	2	Self-Actuated	Closed	Containment Spray Pump Suction Flow Isolation, Train B	: 66
CT-148	CT	Check	16	2	Self-Actuated	Closed	Containment Spray Pump Suction Flow Isolation, Train B	: 66
CT-077	CT	Check	16	2	Self-Actuated	Closed	Containment Spray Pump Suction Flow Isolation, Train A	: 66
CT-149	CT	Check	16	2	Self-Actuated	Closed	Containment Spray Pump Suction Flow Isolation, Train A	: 66
LV-4752	CT	Globe/Air	3	3	Auto Trip	Open	Chemical Additive Tank Isolation	: 11
LV-4753	CT	Globe/Air	3	3	Auto Trip	Open	Chemical Additive Tank Isolation	: 11
FV-4772-1	CT	Globe Motor	4	2	Auto Trip	Open	Pump Recirculation Flow Path	: 66
FV-4772-2	CT	Globe Motor	4	2	Auto Trip	Open	Pump Recirculation Flow Path	: 66
FV-4773-1	CT	Globe Motor	4	2	Auto Trip	Open	Pump Recirculation Flow Path	: [illegible]
FV-4773-2	CT	Globe Motor	4	2	Auto Trip	Open	Pump Recirculation Flow Path	: 66
HV-4710	CC	Globe/Air	4	2	Auto Trip	Open	Containment Isolation	: 66
HV-4711	CC	Globe/Air	4	2	Auto Trip	Open	Containment Isolation	

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(Sheet 14)

ACTIVE VALVES

Valve Identification or Location No.	System	Valve Type and Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function	
HV-4725	CC	Globe/Air	2	2	Auto Trip	Open	Containment Isolation	
HV-4726	CC	Globe/Air	2	2	Auto Trip	Open	Containment Isolation	
CC-1067	CC	Safety	3/4	2	Self-Actuated	Closed	Pressure Relief During Containment Isolation	; 68
HV-4708	CC	Globe/Air	8	2	Auto Trip	Open	Containment Isolation	
HV-4701	CC	Globe/Air	8	2	Auto Trip	Open	Containment Isolation	
CC-629	CC	Check	2	2	Self-Actuated	Closed	Containment Isolation	
HV-4700	CC	Gate/Motor	8	2	Auto Trip	Open	Containment Isolation	
CC-713	CC	Check	8	2	Self-Actuated	Open	Containment Isolation	
HV-4699	CC	Gate/Motor	8	2	Auto Trip	Open	CCW System Isolation	; 55
HV-4709	CC	Gate/Motor	4	2	Auto Trip	Open	Containment Isolation	
HV-4696	CC	Gate/Motor	4	2	Auto Trip	Open	Containment Isolation	
CC-831	CC	Check	1	2	Self-Actuated	Closed	Containment Isolation	
HV-4514	CC	Butterfly Motor	24	3	Auto Trip	Open	CCW Loop Isolation, Train A	; 55
HV-4515	CC	Butterfly Motor	24	3	Auto Trip	Open	CCW Loop Isolation, Train B	; 55
HV-4572	CC	Butterfly Motor	18	3	Auto Trip	Open	CCW Flow Path RHR Loop, Train A	
HV-4574	CC	Butterfly Motor	18	3	Auto Trip	Open	CCW Flow Path Containment Spray Loop, Train A	
HV-4573	CC	Butterfly Motor	18	3	Auto Trip	Open	CCW Flow Path RHR Loop, Train B	
HV-4575	CC	Butterfly Motor	18	3	Auto Trip	Open	CCW Flow Path Containment Spray Loop, Train B	
HV-4512	CC	Butterfly Motor	24	3	Auto Trip	Open	CCW Loop Isolation, Train A	; 55
HV-4513	CC	Butterfly Motor	24	3	Auto Trip	Open	CCW Loop Isolation, Train B	; 55

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(Sheet 15)

ACTIVE VALVES

Valve Identification or Location No.	System	Valve Type and Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function	
HV-4631A/B	CC	Globe/Air	2	3	Auto Trip	Open	Primary Sampling System Loop Isolation	55
HV-4663A/B	CC	Globe	2	3	Manual	Open	Instrument Air Control Isolation	55
FV-4536	CC	Butterfly/Air	10	3	Auto Trip	Open	Recirculation Loop Isolation	66
FV-4537	CC	Butterfly/Air	10	3	Auto Trip	Open	Recirculation Loop Isolation	66
HV-4524	CC	Butterfly Motor	24	3	Auto Trip	Open	CCW Non-safeguard Loop Isolation	66
HV-4525	CC	Butterfly Motor	24	3	Auto Trip	Open	CCW Non-safeguard Loop Isolation	66
HV-4526	CC	Butterfly Motor	24	3	Auto Trip	Open	CCW Non-safeguard Loop Isolation	66
HV-4527	CC	Butterfly Motor	24	3	Auto Trip	Open	CCW Non-safeguard Loop Isolation	66
LV-4500	CC	Globe/Air	3	3	Remote Manual/ Local Manual	Closed	Emergency Makeup Water Path	66
LV-4500-01	CC	Globe/Air	3	3	Remote Manual	Closed	Emergency Makeup Water Path	66
LV-4501	CC	Globe/Air	3	3	Remote Manual/ Local Manual	Closed	Emergency Makeup Water Path	66
CC-003	CC	Check	3	3	Self-Actuated	Closed	Emergency Makeup Water Path	66
CC-004	CC	Check	3	3	Self-Actuated	Open	Emergency Makeup Water Path Boundary	66
CC-031	CC	Check	24	3	Self-Actuated	Open	Discharge Flow Path	66
CC-061	CC	Check	24	3	Self-Actuated	Open	Discharge Flow Path	66

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(Sheet 16)

ACTIVE VALVES

Valve Identification or Location No.	System	Valve Type and Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function	
FV-4650A/B	CC	Butterfly/Air	10	3	Auto Trip	Open	Ventilation Chiller Isolation	: 66
XPV-3583	CC	Plug/Motor	3	3	Auto Trip	Open	Control Room A/C CCW Flow	: 66
XPV-3584	CC	Plug/Motor	3	3	Auto Trip	Open	Control Room A/C CCW Flow	: 66
XPV-3585	CC	Plug/Motor	3	3	Auto Trip	Open	Control Room A/C CCW Flow	: 66
XPV-3586	CC	Plug/Motor	3	3	Auto Trip	Open	Control Room A/C CCW Flow	: 66
PV-4552	CC	Ball/Air	3	3	Auto/Local Manual	Open	Safety Chiller Condenser CCW regulating valve	: 66
PV-4553	CC	Ball/Air	3	3	Auto/Local Manual	Open	Safety Chiller Condenser CCW regulating valve	: 66
HV-4286	SW	Butterfly Motor	24	3	Auto Trip	Open	SW Flow Path, Train A	
HV-4287	SW	Butterfly Motor	24	3	Auto Trip	Open	SW Flow Path, Train B	
HV-4393	SW	Butterfly Motor	10	3	Auto Trip	Open	SW Flow Path, Train A	: 66
HV-4394	SW	Butterfly Motor	10	3	Auto Trip	Open	SW Flow Path, Train B	: 66
HV-4395	SW	Butterfly Motor	10	3	Remote Manual	Closed	Alternate AFW Flow Path, Train A	: 55
HV-4396	SW	Butterfly Motor	10	3	Remote Manual	Closed	Alternate AFW Flow Path, Train B	: 55
PV-4252	SW	Butterfly/Air	10	3	Auto Trip	Closed	SSW Pump-01 Recirc. Line	: 66
PV-4253	SW	Butterfly/Air	10	3	Auto Trip	Closed	SSW Pump-02 Recirc. Line	: 66
SW-373	SW	Check	24	3	Self-Actuated	Open	Discharge Flow Path	: 66
SW-374	SW	Check	24	3	Self-Actuated	Open	Discharge Flow Path	: 66
SW-016	SW	Check	10	3	Self-Actuated	Open	Discharge Flow Path	: 66
SW-017	SW	Check	10	3	Self-Actuated	Open	Discharge Flow Path	: 66

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(Sheet 17)

ACTIVE VALVES

Valve Identification or Location No.	System	Valve Type and Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function	
SW-388	SW	Check	10	3	Self-Actuated	Closed	Alternate AFW Flow Path	: 66
SW-389	SW	Check	10	3	Self-Actuated	Closed	Alternate AFW Flow Path	: 66
XSF-003	SF	Check	10	3	Self-Actuated	Open	Discharge Flow Path	: 66
XSF-004	SF	Check	10	3	Self-Actuated	Open	Discharge Flow Path	: 66
XSF-160	SF	Check	3	3	Self-Actuated	Closed	Makeup Path	: 66
XSF-180	SF	Check	3	3	Self-Actuated	Closed	Makeup Path	: 66
SI-166	SI	Check	3/4	3	Self-Actuated	Closed/Open	Close on Non-Safety Line Break to maintain Accumulator nitrogen pressure.	: 68
SI-167	SI	Check	3/4	3	Self-Actuated	Closed/Open	Close on Non-Safety Line Break to maintain Accumulator nitrogen pressure.	: 68
SI-168	SI	Check	3/4	3	Self-Actuated	Closed/Open	Close on Non-Safety Line Break to maintain Accumulator nitrogen pressure.	: 68
SI-169	SI	Check	3/4	3	Self-Actuated	Closed/Open	Close on Non-Safety Line Break to maintain Accumulator nitrogen pressure.	: 68
HV-5365	DD	Globe/Air	3	2	Auto Trip	Closed	Containment Isolation	
HV-5366	DD	Globe/Air	3	2	Auto Trip	Closed	Containment Isolation	
DD-430	DD	Safety	3/4	2	Self-Actuated	Closed	Pressure Relief During Containment Isolation	: 68
DD-016	DD	Check	1	3	Self-Actuated	Open	Recirculation Flow Path	: 66
DD-018	DD	Check	3	3	Self-Actuated	Open	Discharge Flow Path	: 66

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ACTIVE VALVES

Valve Identification or Location No.	System	Valve Type and Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function	
XDD-044	DO	Check	1	3	Self-Actuated	Open	Recirculation Flow Path	: 66
XDD-048	DO	Check	3	3	Self-Actuated	Closed	Discharge Flow Path	: 66
DO-005	DO	Check	2	3	Self-Actuated	Closed	Discharge Flow Path	
DO-004	DO	Check	2	3	Self-Actuated	Closed	Discharge Flow Path	
DO-016	DO	Check	2	3	Self-Actuated	Closed	Discharge Flow Path	
DO-017	DO	Check	2	3	Self-Actuated	Closed	Discharge Flow Path	
DD-349	DO	Check	2	3	Self-Actuated	Closed	Fuel Oil Flow Path	: 66
DD-350	DO	Check	2	3	Self-Actuated	Closed	Fuel Oil Flow Path	: 66
HV-3467	CA	Globe/Air	3	2	Auto Trip	Open	Containment Isolation	: 66
CI-030	CA	Check	3	2	Self-Actuated	Open	Containment Isolation	: 66
								: 66
CI-644	CI	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to Control Room Damper After A Non-Safety Air Line Break.	: 68
CI-645	CI	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to Control Room Damper After A Non-Safety Air Line Break.	: 68
CI-646	CI	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to Control Room Damper After A Non-Safety Air Line Break.	: 68

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ACTIVE VALVES

Valve Identification or Location No.	System	Valve Type and Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function	
CI-647	CI	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to Control Room Damper After A Non-Safety Air Line Break.	68
HV-4168	PS	Globe/Air	3/4	2	Auto Trip	Closed	Containment Isolation	
HV-4169	PS	Globe/Air	3/4	2	Auto Trip	Closed	Containment Isolation	
HV-4170	PS	Globe/Air	3/4	2	Auto Trip	Closed	Containment Isolation	
HV-4167	PS	Globe/Air	3/4	2	Auto Trip	Closed	Containment Isolation	
HV-4166	PS	Globe/Air	3/4	2	Auto Trip	Closed	Containment Isolation	
DD-058	DO	Check	1 1/2	3	Self-Actuated	Closed	Starting Air Pressure Boundary	66
DD-059	DO	Check	1 1/2	3	Self-Actuated	Closed	Starting Air Pressure Boundary	66
DD-060	DO	Check	1 1/2	3	Self-Actuated	Closed	Starting Air Pressure Boundary	66
DD-061	DO	Check	1 1/2	3	Self-Actuated	Closed	Starting Air Pressure Boundary	66
DD-062	DO	Check	1 1/2	3	Self-Actuated	Closed	Starting Air Pressure Boundary	66
DD-063	DO	Check	1 1/2	3	Self-Actuated	Closed	Starting Air Pressure Boundary	66
DD-064	DO	Check	1 1/2	3	Self-Actuated	Closed	Starting Air Pressure Boundary	66
DD-065	DO	Check	1 1/2	3	Self-Actuated	Closed	Starting Air Pressure Boundary	56
HV-4176	PS	Globe/Air	3/4	2	Auto Trip	Closed	Containment Isolation	

Table 3.9B-10

(Sheet 20)

ACTIVE VALVES

Valve Identification or Location No.	System	Valve Type and Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function	
HV-4165	PS	Globe/Air	3/4	2	Auto Trip	Closed	Containment Isolation	
HV-4175	PS	Globe/Air	3/4	2	Auto Trip	Closed	Containment Isolation	
HV-4171	PS	Globe/Air	3/4	2	Auto Trip	Closed	Containment Isolation	
PS-193	PS	Safety	3/4	2	Self-Actuated	Closed	Pressure Relief During Containment Isolation	: 66
HV-4172	PS	Globe/Air	3/4	2	Auto Trip	Closed	Containment Isolation	
HV-4173	PS	Globe/Air	3/4	2	Auto Trip	Closed	Containment Isolation	
HV-4174	PS	Globe/Air	3/4	2	Auto Trip	Closed	Containment Isolation	
HV-4178	PS	Angle/Air	3/4	2	Auto Trip	Closed	RHR Loop to Primary Sampling System	: 11
HV-4179	PS	Angle/Air	3/4	2	Auto Trip	Closed	RHR Loop to Primary Sampling System	: 11
HV-4182	PS	Angle/Air	3/4	2	Auto Trip	Closed	ECCS Operation	: 66
CA-016	CA	Check	3	2	Self-Actuated	Closed	Containment Isolation	: 66
HV-3486	CA	Globe/Air	3	2	Auto Trip	Closed	Containment Isolation	
HV-6084	CH	Gate/Motor	6	2	Auto Trip	Open	Containment Isolation	
CH-024	CH	Check	6	2	Self-Actuated	Open	Containment Isolation	: 68
HV-6082	CH	Gate/Motor	6	2	Auto Trip	Open	Containment Isolation	
HV-6083	CH	Gate/Motor	6	2	Auto Trip	Open	Containment Isolation	
XCH-302	CH	Check	1 1/2	3	Self-Actuated	Open	Flow Path	: 66
XCH-305	CH	Check	1 1/2	3	Self-Actuated	Open	Flow Path	: 66
CH-271	CH	Safety	3/4	2	Self-Actuated	Closed	Pressure Relief During Containment Isolation	: 68
CH-272	CH	Safety	3/4	2	Self-Actuated	Closed	Pressure Relief During Containment Isolation	: 68
CH-376	CH	Check	1 1/2	3	Self-Actuated	Open	Flow Path	: 66
CH-436	CH	Check	1 1/2	3	Self-Actuated	Open	Flow Path	: 66
CH-360	CH	Check	2	3	Self-Actuated	Open	Flow Path	: 66

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Table 3.9B-10
(Sheet 21)

ACTIVE VALVES

Valve Identification or Location No.	System	Valve Type and Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function	
CH-383	CH	Check	2	3	Self-Actuated	Open	Flow Path	: 66
CH-420	CH	Check	2	3	Self-Actuated	Open	Flow Path	: 66
CH-444	CH	Check	2	3	Self-Actuated	Open	Flow Path	: 66
CH-456	CH	Check	2	3	Self-Actuated	Open	Flow Path	: 66
CH-466	CH	Check	2	3	Self-Actuated	Open	Flow Path	: 66
CH-343	CH	Check	3	3	Self-Actuated	Open	Flow Path	: 66
Ch-346	CH	Check	3	3	Self-Actuated	Open	Flow Path	: 66
CH-403	CH	Check	3	3	Self-Actuated	Open	Flow Path	: 66
CH-406	CH	Check	3	3	Self-Actuated	Open	Flow Path	: 66
CH-452	CH	Check	3	3	Self-Actuated	Open	Flow Path	: 66
CH-463	CH	Check	3	3	Self-Actuated	Open	Flow Path	: 66
HV-6720	CH	Globe/Air	1	3	Remote Manual	Closed	Emergency Makeup Water Path	: 66
CH-300	CH	Check	1	3	Self-Actuated	Closed	Emergency Makeup Water Path	: 66
CH-301	CH	Check	1	2	Self-Actuated	Open	Emergency Makeup Water Path Boundary	: 66
HV-5542	VA	Butterfly Motor	12	2	Auto Trip	Closed	Containment Isolation	
HV-5543	VA	Butterfly Motor	12	2	Auto Trip	Closed	Containment Isolation	
HV-5563	VA	Butterfly Motor	12	2	Auto Trip	Closed	Containment Isolation	
HV-5540	VA	Butterfly Motor	12	2	Auto Trip	Closed	Containment Isolation	
HV-5541	VA	Butterfly Motor	12	2	Auto Trip	Closed	Containment Isolation	
HV-5562	VA	Butterfly Motor	12	2	Auto Trip	Closed	Containment Isolation	
HV-5556	PS	Globe Solenoid	1	2	Auto Trip	Closed	Containment Isolation	: 66
HV-5557	PS	Globe Solenoid	1	2	Auto Trip	Closed	Containment Isolation	: 66
HV-5544	RM	Globe Solenoid	1	2	Auto Trip	Open	Containment Isolation	: 66
HV-5545	RM	Globe Solenoid	1	2	Auto Trip	Open	Containment Isolation	: 66
HV-5558	PS	Globe Solenoid	1	2	Auto Trip	Closed	Containment Isolation	: 66

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Table 3.9B-10

(Sheet 22)

ACTIVE VALVES

Valve Identification or Location No.	System	Valve Type and Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function	
HV-5559	PS	Globe Solenoid	1	2	Auto Trip	Closed	Containment Isolation	: 66
HV-5560	PS	Globe Solenoid	1	2	Auto Trip	Closed	Containment Isolation	: 66
HV-5561	PS	Globe Solenoid	1	2	Auto Trip	Closed	Containment Isolation	: 66
HV-5546	RM	Globe Solenoid	1	2	Auto Trip	Open	Containment Isolation	: 66
HV-5547	RM	Globe Solenoid	1	2	Auto Trip	Open	Containment Isolation	: 66
HV-5536	VA	Butterfly/Air	48	2	Auto Trip	Closed	Containment Isolation	: 66
HV-5537	VA	Butterfly/Air	48	2	Auto Trip	Closed	Containment Isolation	
HV-5538	[illegible]	Butterfly/Air	48	2	Auto Trip	Closed	Containment Isolation	
HV-5539	VA	Butterfly/Air	48	2	Auto Trip	Closed	Containment Isolation	
HV-5548	VA	Butterfly/Air	18	2	Auto Trip	Closed	Containment Isolation	
HV-5549	VA	Butterfly/Air	18	2	Auto Trip	Closed	Containment Isolation	
XHV-5526	VA	Butterfly Motor	12	3	Remote Manual	Closed	H2 Purge Flow Path	: 66
XHV-5529	VA	Butterfly Motor	12	3	Remote Manual	Closed	H2 Purge Flow Path	: 66
XHV-5579	VA	Gate/Motor	3	3	Remote Manual	Closed	H2 Purge Flow Path	: 66
XHV-5580	VA	Gate/Motor	3	3	Remote Manual	Closed	H2 Purge Flow Path	: 66
HV-5157	VD	Diaphragm Air	4	2	Auto Trip	Open	Containment Isolation	
HV-5158	VD	Diaphragm Air	4	2	Auto Trip	Open	Containment Isolation	
HV-7311	WP	Angle/Air	3/4	2	Auto Trip	Closed	Containment Isolation	: 66
HV-7312	WP	Angle/Air	3/4	2	Auto Trip	Closed	Containment Isolation	: 66

TABLE 3.9B-10
(Sheet 23)

ACTIVE VALVES

SYMBOL	SYSTEM	
AF	AF Auxiliary Feedwater	
CA	Compressed Air-Service Air	
CC	Component Cooling Water	
CH	Chilled Water	
CI	Compressed Air - Instrument Air	
CS	Chemical and Volume Control System	: 55
CT	Containment Spray	
DD	Demineralized and Reactor Makeup	
DO	Diesel Generator Fuel Oil	
FP	Fire Protection	: 55
FW	Steam Generator Feedwater	
MS	Main Steam, Reheat and Dump	
PS	Process Sampling Primary Plant	
RHR	Residual Heat Removal	
SF	Spent Fuel Pool Cooling and Clean-Up	
SW	Service Water	
VA	Heating and Ventilation	
VD	Vents and Drains	
WP	Waste Processing	: 55
RM	Radiation Monitoring	: 66

GENERAL

RWST	Refueling Water Storage Tank	
ATM	Atmospheric	
N/A	Not Applicable	

Note: 1) See Section 3.9N, Table 3.9N-10 for NSSS Active Valve List. : 66

2) Unit 1 and Unit 2 Tag Numbers are generally the same except for the prefix. : 66

Any Unit 2 difference will be identified in a future amendment. : 66

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10. Refueling

At the end of plant cooldown the fluid in the RCS is at 140°F. At this time the vessel head is removed and the refueling canal is filled. This is done by pumping water from the refueling water storage tank, which is outside and conservatively assumed to be at 32°F, into the loops by means of the residual heat removal pumps. It is conservatively assumed that the cold water is replaced with the colder water within 10 minutes.

This operation is assumed to occur twice per year or 80 times over the life of the plant.

Upset Conditions

The following primary system transients are considered upset conditions.

1. Loss of load (without immediate reactor trip).
2. Loss of power.
3. Partial loss of flow.
4. Reactor trip from full power.
5. Inadvertent Reactor Coolant System depressurization.
6. Inadvertent startup of an inactive loop.
7. Control rod drop.
8. Inadvertent Emergency Core Cooling System actuation.
9. Operating Basis Earthquake.

52 | 10. Excessive Feedwater Flow

1. Loss of Load (without immediate reactor trip)

This transient applies to a step decrease in turbine load from full power (turbine trip) without immediately initiating a reactor trip and represents the most severe pressure transient on

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the RCS under upset conditions. The reactor eventually trips as a consequence of a high pressurizer level trip initiated by the Reactor Protection System (RPS). Since redundant means of tripping the reactor are provided as a part of the RPS, transients of this nature are not expected, but are included to ensure a conservative design.

The number of occurrences of this transient is specified at 80 times or 2 times per year for the 40 year plant design life.

2. Loss of power

This transient applies to a blackout situation involving the loss of outside electrical power to the station, assumed to be operating initially at 100 percent power, followed by reactor and turbine trips. Under these circumstances, the reactor coolant pumps are de-energized and, following coastdown of the reactor coolant pumps, natural circulation builds up in the system to some equilibrium value. This condition permits removal of core residual heat through the steam generators which at this time are receiving feedwater, assumed to be at 32°F, from the Auxiliary Feedwater System operating from diesel generator power. Steam is removed for reactor cooldown through atmospheric relief valves provided for this purpose.

The number of occurrences of this transient is specified at 40 times or 1 per year for the 40 year plant design life.

3. Partial loss of flow

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

The number of occurrences of this transient is specified at 80 times or 2 times per year for the 40 year plant design life.

4. Reactor trip from full power

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

Various moderator cooldown transients associated with reactor trips can occur as a result of excessive feed or steam dump after trip or large load increase. For design purposes, reactor trip is assumed to occur a total of 400 times or 10 times per year over the life of the plant. The various types of trips and the number of occurrences for each are as follows:

a. Reactor trip with no inadvertent cooldown - 230 occurrences.

b. Reactor trip with cooldown but no safety injection - 160 occurrences.

c. Reactor trip with cooldown actuating safety injection - 10 occurrences.

5. Inadvertent Reactor Coolant System depressurization

Several events can be postulated as occurring during normal plant operation which will cause rapid depressurization of the RCS. These include:

a. Actuation of a single pressurizer safety valve.

b. Inadvertent opening of one pressurizer power operated relief valve due either to equipment malfunction or operator error.

c. Malfunction of a single pressurizer pressure controller causing one power operated relief valve and two pressurizer spray valves to open.

d. Inadvertent opening of one pressurizer spray valve, due either to equipment malfunction or operator error.

e. Inadvertent auxiliary spray.

Of these events, the pressurizer safety valve actuation causes the most severe transients, and is used as an "umbrella" case to conservatively represent the reactor coolant pressure and temperature variations arising from any of them.

When a pressurizer safety valve opens, and remains open, the system rapidly depressurizes, the reactor trips, and the Emergency Core Cooling System (ECCS) is actuated. Also, the passive accumulators of the ECCS are actuated when pressure decreases by approximately 1600 psi, about 12 minutes after the depressurization begins. The depressurization and cooldown are eventually terminated by operator action. All of these effects are completed within approximately 18 minutes. It is conservatively assumed that none of the pressurizer heaters are energized.

With pressure constant and safety injection in operation, boiloff of hot leg liquid through the pressurizer and open safety valve will continue.

For design purposes this transient is assumed to occur 20 times during the 40 year design life of the plant.

6. Inadvertent startup of an inactive loop

This transient can occur when a loop is out of service. With the plant operating at maximum allowable level the reactor coolant pump in the inactive loop is started as a result of operator error. Reactor trip occurs on high nuclear flux. This transient is assumed to occur 10 times during the life of the plant.

7. Control rod drop

This transient occurs if a bank of control rods drop into the fully inserted position due to a single component failure. The reactor is tripped on either low pressurizer pressure or negative flux rate, depending on time in core life and magnitude of the reactivity insertion. It is assumed that this transient occurs 80 times over the life of the plant.

8. Inadvertent Emergency Core Cooling System Actuation

A spurious safety injection signal results in an immediate reactor trip followed by actuation of the high head centrifugal charging pumps. These pumps deliver cold water from Refueling Water Storage Tank to the RCS cold legs. The initial portion of this transient is similar to the reactor trip from full power with no cooldown. Controlled steam dump and feedwater flow after trip removes core residual heat. Reactor coolant temperature and pressure decreases as the control rods move into the core. | 68

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

For design purposes this transient is assumed to occur 60 times over the life of the plant.

9. Operating Basis Earthquake

The mechanical stresses resulting from the Operating Basis Earthquake are considered on a component basis. Fatigue analysis, where required by the codes, if performed by the supplier as part of the stress anal report. The earthquake loads are a part of the mechanical loading conditions specified in the equipment specifications. The origin of their determination is separate and distinct from those transients resulting from fluid pressure and temperature. They are, however, considered in the design analysis. The number of occurrences for fatigue evaluation is assumed to be 20 earthquakes at 10 cycles each (200 cycles total).

in the load and stress evaluation tables for the primary component | 61
support. Reactor coolant loop normal and upset condition thermal |
expansion loads are treated as primary loadings for the primary | 10
component supports. The adequacy of each member of the steam |
generator supports, reactor coolant pump supports, and piping |
restraints is verified by solving the ASME III Subsection NF stress |
and interaction equations. The adequacy of the RPV support |
structure is verified using the WECAN computer program and comparing |
the resultant stresses to the criteria given in ASME III Subsection |
NF. |

| Q112.25

Tables 3.9N-14 through 3.9N-17 present maximum stresses in each member | 10
of the steam generator, reactor coolant pump, and pressurizer support |
structures expressed as a percentage of maximum permissible values for |
all operating condition loadings. The loads on the reactor vessel | 61
supports and the resulting stresses are shown in Table 3.9N-19. The |
above loads and stresses include the effects of loads resulting from | 10
asymmetric subcompartment pressurization. |

3.9N.1.4.5 Analysis of Primary Components | 13

Equipment which serves as part of the pressure boundary in the reactor coolant loop includes the steam generators, the reactor coolant pumps, the pressurizer, and the reactor vessel. This equipment is evaluated for the loading combinations outlined in Table 3.9N-2. The equipment is analyzed for: 1) the normal loads of deadweight, pressure and thermal, 2) mechanical transients of OBE, SSE, and pipe ruptures, including the effects of asymmetric subcompartment pressurization and 3) pressure and temperature transients outlined in Section 3.9N.1.1.

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

Seismic analyses are performed individually for the reactor coolant pump, the pressurizer, and the steam generator. Detailed and complex dynamic models are used for the dynamic analyses. The response spectra corresponding to the building elevation at the highest component/building attachment elevation is used for the component
68 analysis. Seismic analyses for the steam generator and reactor
coolant pump are performed using 2 percent damping for the OBE and 4
percent damping for the SSE. The reactor pressure vessel is
qualified by static stress analysis based on loads that have been
derived from dynamic analysis.

Reactor coolant pressure boundary components are further qualified to ensure against unstable crack growth under faulted conditions by performing detailed fracture analyses of the critical areas of this boundary. Actuation of the ECCS produces relatively high thermal stresses in the system. Regions of the pressure boundary which come into contact with ECCS water are given primary consideration. These regions include the reactor vessel beltline region, the reactor vessel inlet nozzles, and the safety injection nozzles in the piping system.

The valve will be mounted in a manner which will conservatively represent typical valve installations. The valve will include the operator and all accessories normally attached to the valve in service. The operability of the valve during SSE is demonstrated by satisfying the following criteria:

1. All the active valves are designed to have a first natural frequency which is greater than 33 Hz. This is shown by suitable test or analysis.

2. The extended structure of the valve system will be statically deflected an amount equal to that determined by an analysis as representing SSE accelerations applied at the center of gravity of the operator along the direction of the weakest axis of the yoke. The design pressure of the valve will be simultaneously applied to the valve during the static deflection tests.

3. The valve will then be operated while in the deflected position. The valve must perform its safety-related function within the specified operating time limits.

4. Motor operators, pilot solenoid valves and limit switches necessary for operation will be qualified as operable during SSE by appropriate IEEE Seismic Qualification Standards, prior to their installation on the valve.

68 The accelerations which are used for the static valve qualification are equivalent, as justified by analysis, to the simultaneous application of not less than 2.1g in two orthogonal horizontal directions and not less than 2.1g in the vertical direction. Accelerations imparted to the valve assemblies by the piping are not greater than the accelerations used for the static valve qualification.

If the frequency of the valve is less than 33 Hz, a dynamic analysis of the valve will be performed to determine the equivalent acceleration which will be applied during the static test. The analysis will provide the amplification of the input acceleration considering the natural frequency of the valve and the frequency content of the applicable plant floor response spectra. The adjusted accelerations will be determined using the same conservatisms contained in the 2.1 g horizontal and 2.1 g vertical accelerations used for "rigid" valves. The adjusted accelerations will then be used in the static analysis and the valve operability will be assured by the methods outlined in steps (2) to (4) above using the modified acceleration input. 68

The above testing program applies to valves with extended structures. The testing will be conducted on a representative number of valves. Valves from each of the primary safety-related design types will be tested. Valve sizes which cover the range of sizes in service will be qualified by the tests and the results will be used to qualify all valves within the intermediate range of sizes.

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

The check valves are characteristically simple in design and their operation will not be affected by seismic accelerations or the maximum applied nozzle loads. The check valve design is compact and there are no extended structures or masses whose motion could cause distortions which could restrict operation of the valve. The nozzle loads due to maximum seismic excitation will not affect the functional ability of the valve since the valve designed to be isolated from the casing wall. The clearance supplied by the design around the disc will prevent the disc from becoming bound or restricted due to any casing disortions caused by nozzle loads. Therefore, the design of these valves is such that once the structural integrity of the valve is assured using standard design or analysis methods, the ability of the valve to operate is assured by the design features. In addition to these design considerations, the valve will also undergo 1) in-shop hydrostatic tests, 2) in-shop seat leakage test, and 3) periodic in-situ valve exercising and inspection to assure the functional ability of the valve.

The pressurizer safety valves are qualified by the following procedures. These valves are also subjected to tests and analysis similar to check valves: stress and deformation analyses for SSE loads, in-shop hydrostatic and seat leakage tests, and periodic in-situ valve inspection. In addition to these tests, a static load equivalent to the SSE is applied at the top of the bonnet and the pressure is increased until the valve mechanism actuates. Successful actuation within the design requirements of the valve assures its overpressurization safety capabilities during a seismic event.

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

3.9N.3.3 Design and Installation Details in Mounting of Pressure Relief Devices (Pressurizer Safety and Relief System)

20 The pressurizer safety and relief valve discharge piping systems provide overpressure protection for the RCS. The three spring-loaded safety valves, located on top of the pressurizer, are designed to prevent system pressure from exceeding design pressure by more than 10 percent. The two power-operated relief valves, also located on top of the pressurizer, are designed to prevent system pressure from exceeding

TABLE 3.9N-2

LOADING COMBINATIONS FOR ASME CLASS 1
COMPONENTS AND COMPONENT SUPPORTS | 20
(EXCLUDING PIPE SUPPORTS) | 20

Condition Classification	Loading Combination	
Design	Design Pressure, Design Temperature, Deadweight, Operating Basis Earthquake*	68
Normal	Normal Condition Transient, Deadweight	
Upset	Upset Condition Transients, Deadweight, Operating Basis Earthquake	
Emergency	Emergency Condition Transients, Deadweight	
Faulted	Faulted Condition Transients, Deadweight, Safe Shutdown Earthquake or (Safe Shutdown Earthquake and Pipe Rupture Loads)	

* The Operating Basis Earthquake is not considered a design condition for NSSS Class 1 valves and piping. The primary stresses are calculated for the Operating Basis Earthquake loads and compared with the upset condition allowable stresses. | 68

TABLE 3.9N-5

STRESS CRITERIA FOR SAFETY RELATED ASME CLASS 2 AND CLASS 3 TANKS

Condition	Stress Limits	
Design and Normal	$\sigma_m \leq 1.0\ S$	68
	$(\sigma_m \text{ or } \sigma_L) +$	68
	$\sigma_b \leq 1.5\ S$	68
Upset	$\sigma_m \leq 1.1\ S$	
	$(\sigma_m \text{ or } \sigma_L) +$	
	$\sigma_b \leq 1.65\ S$	
Emergency	$\sigma_m \leq 1.5\ S$	
	$(\sigma_m \text{ or } \sigma_L) +$	
	$\sigma_b \leq 1.80\ S$	
Faulted	$\sigma_m \leq 2.0\ S$	
	$(\sigma_m \text{ or } \sigma_L) +$	
	$\sigma_b \leq 2.4\ S$	

TABLE 3.9N-21

MAXIMUM STRESSES IN THE REACTOR COOLANT LOOP PIPING

	Hot Leg		Crossover Leg		Cold Leg		
Evaluation	Maximum	Allowable	Maximum	Allowable	Maximum	Allowable	68
							Q112.14
Eq 9 design stress (ksi) (DW, P, OBE)	21.8	26.7	23.0	26.7	24.8	26.7	68
Eq 9 faulted stress (ksi) (DW, P, SSE, LOCA) [a]	34.8	53.4	41.37	53.4	39.0	53.4	68
Eq 12 stress (ksi)	26.25	53.4	10.6	53.4	19.15	53.4	68
Eq 13 stress (ksi)	44.3	53.4	42.3	53.4	42.8	53.4	68
Fatigue usage factor	.893	1.0	.131	1.0	.312	1.0	68

NOTE [a]: LOCA piping stresses in this table are based on main loop piping breaks (envelope of breaks 1 to 11, Table 3.6B-2), which are larger than the actual LOCA stresses, due to breaks 9, 10 and 11 only, of Table 3.6B-2. 68

68 The environmental qualification program at CPSES includes all Class 1E equipment including that located in mild and potentially harsh environment areas. For CPSES, mild environment areas are defined as areas, outside the containment, that are not potentially harsh following a design basis accident. A potentially harsh environment is defined as an environment where safety-related equipment would experience due to the direct effects of a design basis accident (Loss of Coolant Accident, Main Steam Line Break, High Energy Line Break) or any of the following parameters:

68 a. An ambient pressure increase greater than two pounds per square inch (2psi) above atmospheric, or

66 b. An ambient temperature increase greater than five degrees centigrade ($5^{o}C$ or $9^{o}F$) above the postulated maximum temperature based on normal and anticipated operational occurrences, or

66 c. A total integrated radiation exposure dose of $1x10^4$ rads Gamma.

66 NOTE: For postulated radiation environments between $1x10^3$ and $1x10^4$ rads Gamma, the design/purchase specifications for electronic equipment located in a mild environment will not allow use of semiconductors susceptible to radiation damage less than $1x10^4$ rads Gamma.

66 If radiation is the only harsh environment criterion exceeded for an area and an evaluation concludes that the component materials for a piece of equipment has a radiation threshold level greater than the postulated radiation environment, then, for purposes of environmental qualification, the equipment will be considered to be located in a mild environment. The radiation threshold

4 a. Mechanical equipment (which contains Class 1E electrical components) in Chapter 6

b. Class 1E equipment in Chapter 8

c. Instrumentation and controls in Chapter 7

4 2. ESF and other Safety-Related Equipment Located Outside Containment

68 a. Mechanical equipment (which contains Class 1E electrical components) is described in Chapters 6, 9 and 10.

b. Instrumentation and controls are described in Chapter 7.

c. Class 1E equipment is described in Chapter 8.

38 The safety-related mechanical equipment at CPSES has been designed to withstand environmental effects as required by GDC-4 and Appendix B of 10 CFR Part 50. These requirements are satisfied through the design, specification, procurement, and quality assurance procedures used at CPSES as supplemented by the pump and valve operability programs and the CPSES maintenance and surveillance programs.

66

3.11B.2 QUALIFICATION TESTS AND ANALYSES

4 Qualification tests and analyses are performed on all ESF equipment and components as necessary to ensure their availability during and after a DBA. These tests consist of simulation of actual physical conditions on the equipment or a prototype on a generic basis, or analysis, or a combination of prototype tests and analysis as applicable. Qualification testing is performed under simulated conditions of temperature, pressure, relative humidity, chemistry, and radiation dose in excess of those expected for post-DBA conditions. The testing period is sufficient to ensure the capability to function during and after a DBA.

In order to provide assurance that all ESF and other Safety Related 4
equipment has the capability to meet environmental conditions as
required the appropriate quality assurance programs are established
and implemented.

Q032.31

Class 1E instrumentation and electrical equipment is capable of 66
operating in the worst expected environmental conditions as specified
for each component and its location in Appendix 3A, Table 4-1. The
Class 1E electrical equipment is specified for manufacture in 33
accordance with the criteria listed in Section 3.3. All Class 1E
equipment will be qualified per IEEE 323 [11] and other applicable 8
standards per Section 8.3 and below.

IEEE-323-1974 requirements include the need to establish the qualified 9
life of Class 1E equipment. This requirement has, in many cases,
represented a state-of-the-art challenge in assessing the longevity of
equipment under normal and accident environments. Many of the 33
equipment qualification test reports have identified relatively short
lives of certain components. In some cases, these estimated,
seemingly short, qualified lives may be the result of testing or 9
accelerated aging limitations, rather than due to intrinsic equipment
limitations. TU Electric, as the first operating license applicant 66
with an IEEE-323-1974 commitment has been confronted with an enormous
developmental program in meeting these new requirements.

Consistent with TU Electric's commitment to meet the requirements of 66
IEEE-323-1974 subject to state-of-the-art limitations, TU Electric
will modify any of the existing qualified lives as additional
information and better testing and analytical techniques are
developed. Therefore, the existing qualified lives currently listed 68
in the Environmental Equipment Qualification Summary Packages should
be regarded as decision points with regard to an ongoing aging
evaluation, rather than a fixed component replacement interval.
Changes from those qualified lives currently indicated will be 9
documented by revisions to the appropriate summary packages.

All Class 1E equipment located in a mild environment will be 66
seismically qualified to IEEE Standard 344-1975 and Regulatory Guide
1.100 [19] using the methods, procedures, and documentation described
in FSAR Sections 3.7 and 3.10B.

How the requirements of GDC 1, 24, 23 and 50 are met is addressed in 6
Section 3.1. Specific information concerning how GDC 1 and 4 are met
is reported in Appendix A of Reference [18]. Specific information
concerning how GDC 23 is met can be found in Sections 7.2 and 7.3.

Q040.32

The quality assurance program to be applied to the design, fabrication 6
and testing of all BOP safety related equipment conform to the
requirements of 10CFR part 50 Appendix B. The QA program is
described in Chapter 17.

Q040.39

Specific information concerning how Appendix B of 10 CFR Part 50 is 4
met is discussed in Chapter 17. Level of compliance with NRC 66
Regulatory Guides 1.30, 1.40, 1.63, 1.73, 1.89, 1.100 and 1.131 is
addressed in Appendix 1A(B).

3.11B.3 QUALIFICATION TEST RESULTS

Q040.36

All safety-related equipment and components in a potentially harsh 66
environment are demonstrated to perform their designed safety function
under all normal, abnormal and accident conditions, by appropriate
testing and analyses. The detailed qualification information and 68
test results are documented in the Environmental Equipment
Qualification Summary Packages and are available for an NRC audit.

3.11B.4 LOSS OF VENTILATION

3.11B.4.1 Environmental Design Basis

The plant design features ensure that room ambient temperatures do not 56
exceed the maximum operational temperature limit for instrumentation
and electrical equipment.

1.0 INTRODUCTION

This Equipment Qualification Report (EQR) supports the operating license application for Comanche Peak Steam Electric Station (CPSES). This report provides design information in sufficient detail to allow a definitive evaluation of the equipment environmental qualification program for CPSES.

The CPSES environmental qualification program is described in Section 2.0 and a detailed comparison to NUREG-0588 is provided in Section
3.0. A summary of Class 1E equipment types located in a potentially 68
harsh environment and their postulated environmental extremes are
provided in Table 4-1, Section 4.0.

The basic scope of the program includes all Class 1E equipment at
CPSES. Detailed system and equipment descriptions are included in the 66
appropriate sections of the FSAR; in particular, those systems
required for Hot Standby and Cold Shutdown are described in FSAR
Section 7.4. Section 7.1 identifies all safety-related systems.

A mild environment is an environment that would at no time be 66
significantly more severe than the environment that would occur during
normal plant operation, including anticipated operational occurrences.
For CPSES, mild environment areas are defined as areas, outside the
containment, that are not potentially harsh following a design basis
accident. A potentially harsh environment is defined as an
environment where safety-related equipment would experience, due to
the direct effects of a design basis accident (Loss of Coolant
Accident, Main Steam Line Break, High Energy Line Break), for any of
the following parameters:

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3.11N ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

The mechanical and electrical portions of the Engineered Safety Features and the Reactor Protection Systems are designed to ensure acceptable performance in all environments anticipated under normal, test, and design basis accident conditions. This section presents information on the design basis and qualification verifications for mechanical and electrical equipment in the Engineered Safety Features and the Reactor Protection System that are within the scope of the Westinghouse Nuclear Steam Supply System (NSSS). Section 3.7N presents the seismic design requirements and Section 3.10N presents the seismic qualification of electrical equipment.

3.11N.1 EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS

A list of Class 1E equipment located in potentially harsh environments and their postulated environmental extremes are provided in Appendix 3A, Table 4-1. The definition of potentially harsh and mild environments at CPSES is provided in Section 3.11B.1. | 68

The safety-related mechanical equipment at CPSES has been designed to withstand environmental effects as required by GDC-4 and Appendix B of 10 CFR Part 50. These requirements are satisfied through the design, specification, procurement, and quality assurance procedures at CPSES as supplemented by the pump and valve operability programs and the CPSES maintenance and surveillance programs. | 38

3.11N.2 QUALIFICATION TESTS AND ANALYSIS

| Q032.2

For Westinghouse NSSS Class 1E equipment located in a potentially harsh environment, Westinghouse meets the Institute of Electrical and Electronics Engineers (IEEE) Standard 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," including IEEE Standard 323-1975, the Nuclear Power Engineering Committee (NPEC) Position Statement of July 24, 1975, by an | 68

68 appropriate combination of any or all of the following: type testing, operating experience, qualification by analysis, and ongoing qualification. The Westinghouse approach to satisfying IEEE-Standard 323-1974 is documented in WCAP-8587 [1] which has been accepted by the NRC Staff. Appendix 3A, Table 4-1 identifies the Westinghouse supplied Class 1E equipment located in a potentially harsh environment and the corresponding Equipment Qualification Data Packages.

68 For NSSS Class 1E equipment located in a mild environment area, CPSES demonstrates qualification through design/purchase specifications, test results, operational experience, and/or design data. This information contains a description of the functional requirements for the equipment's specific mild environmental area. Documentation which meets the requirements for a potentially harsh environment is considered acceptable in lieu of the above requirements. However, the replacement of equipment located in a mild environment area is based on a design life rather than a qualified life. The design life may be based on the manufacturer's rating, vendor's design or application analysis, testing or analysis, or an engineering analysis for the specific environment, as long as said determination is based on conditions which are equivalent to or more conservative than the equipment's specific mild environment conditions.

68 All Class 1E equipment located in a mild environment area is seismically qualified to IEEE Standard 344-1975 and Regulatory Guide 1.100 using the methods, procedures, and documentation described in FSAR Sections 3.7 and 3.10N.

4 The overall Class 1E Westinghouse equipment qualification program includes generic environmental conditions, e.g., temperature, pressure, relative humidity, chemistry, radiation, which are established for the various pieces of Westinghouse supplied Class 1E
68 equipment. The conditions vary according to location of the

equipment. The generic environmental conditions for which the 68
equipment is qualified are reported in the specific Equipment
Qualification Data Package and the Environmental Equipment
Qualification Summary Packages. The postulated environmental
extremes used for equipment qualification are provided in Appendix 3A,
Table 4-1.

How the requirements of the General Design Criteria (GDC) 1, 4, 23,
and 50 are met is addressed in Section 3.1. Specific information 68
concerning how GDC 1 and 4 are met is reported in the applicable
Equipment Qualification Data Packages [1]. Specific information
concerning how GDC 23 is met can be found in Section 7.2.2.2.
Specific information concerning how GDC 50 is met is provided in
Section 6.2. Specific information concerning how Appendix B of 10 CFR 4
Part 50 is met is located in Chapter 17. Regulatory Guides 1.30,
1.40, 1.73 and 1.89 are addressed in Appendix 1A(N).

3.11N.3 QUALIFICATION TEST RESULTS

Qualification test results are provided in the Equipment Qualification 68
Data Packages and in the Environmental Equipment Qualification Summary
Packages. The packages are identified in Appendix 3A, Table 4-1.

3.11N.4 LOSS OF VENTILATION

Refer to Section 3.11B.4

3.11N.5 ESTIMATED CHEMICAL AND RADIATION ENVIRONMENT 36

Chemical environments in the primary containment as result of an 68
accident condition are shown in Appendix 3A, Table 4-1 and discussed
in Section 3.11B.5.1. The postulated integrated radiation
environments for normal and accident conditions are shown in Appendix
3A, Table 4-1 .

TABLE 3.11N-1

DELETED | 36

SEE FSAR TABLE 3.11B-2 | 36
AND | 36
FSAR APPENDIX 3A TABLE 4-1 | 68

TABLE 3.11N-2

DELETED | 36

SEE FSAR APPENDIX 3A TABLE 4-1 | 68

TABLE 3.11N-3

DELETED | 68

INFORMATION INCORPORATED IN APPENDIX 3A, TABIE 4-1 | 68

NOTES TO FIGURE 5.1-2

RCS AT STEADY STATE FULL POWER OPERATION

ITEM	FLUID	PRESSURE (psig)	TEMP (^{o}F)	FLOW[a] (gpm)[b]	FLOW[a] (lb/hr)[c]	VOLUME (cu.ft.)	
1	RC	2235.0	618.8	110,250	36.7125	---	68
2	RC	2233.1	618.8	110,250	36.7125	---	68
3	RC	2195.9	559.3	99,839	36.7125	---	68
4	RC	2192.4	559.3	99,839	36.7125	---	68
5	RC	2285.1	559.6	98,900	36.7125	---	68
6	RC	2283.2	559.6	98,900	36.7125	---	68
7-12	See Loop 1 Specifications (1-6)						68
13-18	See Loop 1 Specifications (1-6)						68
19-24	See Loop 1 Specifications (1-6)						68
25	RC	2285.1	559.6	1.0	0.0004	---	68
26	RC	2285.1	559.6	1.0	0.0004	---	68
27	RC	2235.0	559.6	2.0	0.0008	---	68
28	Steam	2235.0	652.7	---	---	720	68
29	RC	2235.0	652.7	---	---	1080	68
30	RC	2235.0	652.7	2.5	0.0008	---	68
31	RC	2235.0	652.7	0	0	---	68
32	Steam	2235.0	652.7	0	0	---	68
33	RC	2235.0	<652.7	0	0	Minimize	68
34	Nitrogen	3.0	120	0	0	---	68
35	RC	2235.0	<652.7	0	0	Minimize	68
36	Nitrogen	3.0	120	0	0	---	68
37	Nitrogen	3.0	120	0	0	---	68
38	Nitrogen	3.0	120	---	---	456	68
39	PRT Water	3.0	120	---	---	1350	68

a) Flow measured at 130^{o}F and 2300 psig; charging and letdown not included | 68

b) At the conditions specified | 68

c) x 10^{6} | 68

Nuclear plants employing the same RHRS design as the CPSES are given in Section 1.3.

5.4.7.1 Design Bases

RHRS design parameters are listed in Table 5.4-7.

The RHRS is placed in operation approximately 4 hours after reactor shutdown when the temperature and pressure of the RCS are approximately 350°F and 425 psig, respectively. Assuming that two heat exchangers and two pumps are in service and that each heat exchanger is supplied with component cooling water at design flow and maximum temperature, the RHRS is designed to reduce the temperature of the reactor coolant from 350°F to 140°F within 24 hours. The time required, under these conditions, to reduce reactor coolant temperature from 350°F to 212°F is 2.8 hours. The heat load handled by the RHRS during the cooldown transient includes residual and decay heat from the core and reactor coolant pump heat. The design heat load is based on the decay heat fraction that exists at 20 hours following reactor shutdown from an extended run at full power. | 68

Assuming that only one heat exchanger and pump are in service and that the heat exchanger is supplied with component cooling water at design flow and maximum temperature, the RHRS is capable of reducing the temperature of the reactor coolant from 350°F to 200°F within 21 hours. The time required under these conditions, to reduce reactor coolant temperature from 350°F to 212°F is 11.9 hours. | 68

The RHRS is designed to be isolated from the RCS whenever the RCS pressure exceeds the RHRS design pressure. The RHRS is isolated from the RCS on the suction side by two motor operated valves in seies on each suction line. Each motor operated valve is interlocked to prevent its opening if RCS pressure is greater than 425 psig and to automatically close if RCS pressure exceeds 750 psig. The RHRS is

Residual Heat Exchangers

Two residual heat exchangers are installed in the system. The heat 68
exchanger design is based on heat load and temperature differences
between reactor coolant and component cooling water existing 28 hours
after reactor shutdown when the temperature difference between the two
systems is small.

The installation of two heat exchangers in separate and independent residual heat removal trains assures that the heat removal capacity of the system is only partially lost if one train becomes inoperative.

The residual heat exchangers are of the shell and U-tube type. Reactor coolant circulates through the tubes, while component cooling water circulates through the shell. The tubes are welded to the tubesheet to prevent leakage of reactor coolant.

The residual heat exchangers also function as part of the ECCS (see Section 6.3).

Residual Heat Removal System Valves

Valves that perform a modulating function are equipped with two sets of packings and an intermediate leakoff connection that discharges to the drain header.

Manual and motor operated valves have backseats to facilitate repacking and to limit steam leakage when the valves are open. Leakage connections are provided where required by valve size and fluid conditions.

5.4.7.2.3 System Operation

Reactor Startup

Generally, while at cold shutdown condition, decay heat from the

Assumptions utilized in the series of heat balance calculations describing plant residual heat removal cooldown are as follows:

1. Residual heat removal operation is initiated 4 hours after reactor shutdown.

2. Residual heat removal operation begins at a reactor coolant temperature of 350°.

3. Thermal equilibrium is maintained throughout the RCS during the cooldown.

4. Component cooling water temperature during cooldown is limited to a maximum of 122°F. | 68

Cooldown curves calculated using this method are provided for the case of all residual heat removal components operable (Figure 5.4-8) and for the case of a single train residual heat removal cooldown (Figure 5.4-9).

5.4.7.4 Preoperational Testing

Preoperational testing of the RHRS is addressed in Chapter 14.

5.4.8 REACTOR WATER CLEANUP

This section is not applicable to the CPSES.

5.4.9 MAIN STEAM LINE AND FEEDWATER PIPING

The main steam line is described in Section 10.3 and the feedwater piping is described in Section 10.4.7.

TABLE 5.4-1
(Sheet 1 of 2)

REACTOR COOLANT PUMP DESIGN PARAMETERS

Unit design pressure (psig)	2485
Unit design temperature ($^{\circ}$F)	650[a]
Unit overall height (ft)	26.93
Seal water injection (gpm)	8
Seal water return (gpm)	3
Cooling water flow (gpm)	495
Maximum continuous cooling water inlet temperature ($^{\circ}$F)	108 | 68

Pump

Capacity (gpm)	99,000
Developed head (ft)	288
NPSH required (ft)	Figure 5.4-2
Suction temperature ($^{\circ}$F)	557.8
Pump discharge nozzle, inside diameter (in)	27-1/2
Pump suction nozzle, inside diameter (in)	31
Speed (rpm)	1183
Water volume (ft^3)	78.6[b]
Weight, dry (lbs)	205,330

---

a Design temperature of pressure retaining parts of the pump assembly exposed to the reactor coolant and injection water on the high pressure side of the controlled leakage seal shall be that temperature determined for the parts for a primary loop temperature of 650°F.

b Composed of reactor coolant in the casing and of injection and cooling water in the thermal barrier.

TABLE 5.4-7

DESIGN BASES FOR RESIDUAL HEAT REMOVAL SYSTEM OPERATION

Residual Heat Removal System startup	~4 hours after reactor shutdown	
Reactor Coolant System initial pressure (psig)	~425	
Reactor Coolant System initial temperature (^oF)	~350	
Component cooling water maximum temperature (^oF)	122	| 68
Cooldown time (hours after initiation of Residual Heat Removal System operation)	24	| 68
Reactor Coolant System temperature at end of cooldown (^oF)	140	
Decay heat generation at 20 hours after reactor shutdown (Btu/hr)	78.2×10^6	

TABLE 5.4-8

RESIDUAL HEAT REMOVAL SYSTEM COMPONENT DATA

Residual Heat Removal Pump

Number	2
Design pressure (psig)	600
Design temperature (^{o}F)	400
Design flow (gpm)	3800
Design head (ft)	350
NPSH required at 3800 gpm (ft)	18
Power (hp)	400

Residual Heat Exchanger

Number	2
Design heat removal capacity (Btu/hr)	39.1×10^6
Estimated UA (Btu/hr-^{o}F)	2.3×10^6

	Tube side	Shell side	
Design pressure (psig)	600	150	
Design temperature (^{o}F)	400	200	
Design flow (lb/hr)	1.9×10^6	3.956×10^6	68
Inlet temperature (^{o}F)	140	109.6	68
Outlet temperature (^{o}F)	121.9	118.3	68
Material	Austenitic stainless steel	Carbon steel	
Fluid	Reactor coolant	Component cooling water	

NOTES TO FIGURE 5.4-7

(Sheet 1 of 4)

MODES OF OPERATION

Mode A - Initiation of RHR Operation

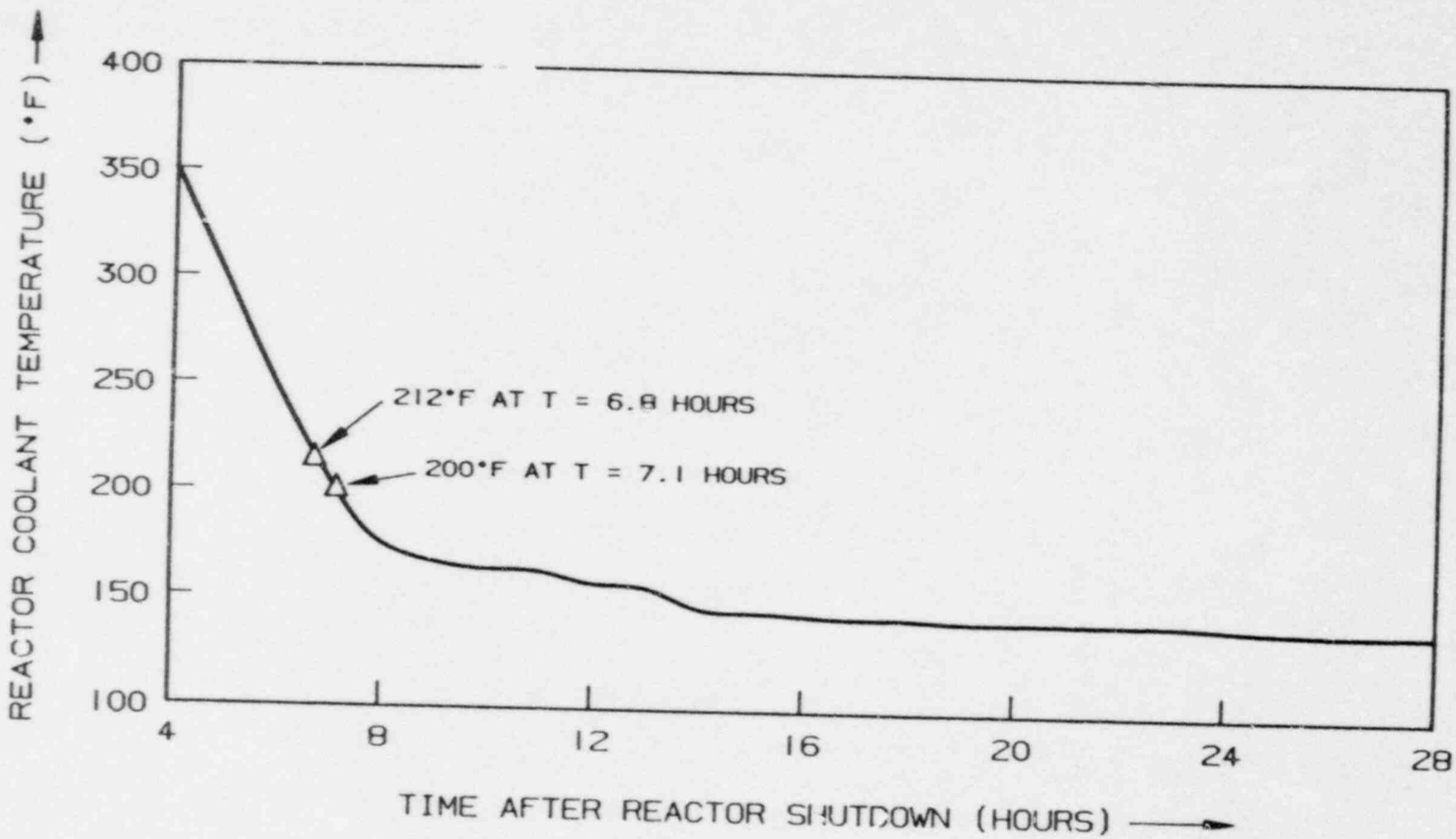
When the reactor coolant temperature and pressure are reduced to 350°F and 400 psig, approximately 4 hours after reactor shutdown, the second phase of plant cooldown starts with the RHRS being placed in operation. Before starting the pumps, the inlet isolation valves are opened, the heat exchanger flow control valves are set a minimum flow, and the outlet valves are verified open. The automatic miniflow valves are open and remain so until the pump flow exceeds 1000 gpm at which time they trip closed. Should the pump flow drop below 500 gpm the miniflow valves open automatically.

Startup of the RHRS includes a warm-up period during which time reactor coolant flow through the heat exchangers is limited to minimize thermal shock on the RCS. The rate of heat removal from the reactor coolant is controlled manually by regulating the reactor coolant flow through the residual heat exchangers. The total flow is regulated automatically by control valves in the heat exchanger bypass line to maintain a constant total flow. The cooldown rate is limited | 68
to 50°F/hr based on equipment stress limits and a 122°F maximum component cooling water temperature.

Mode B - End Conditions of a Normal Cooldown

This situation characterizes most of the RHRS operation. As the reactor coolant temperature decreases, the flow through the residual heat exchanger is increased until all of the flow is directed through the heat exchanger to obtain maximum cooling.

Note: For the safeguards functions performed by the RHRS refer to Section 6.3.

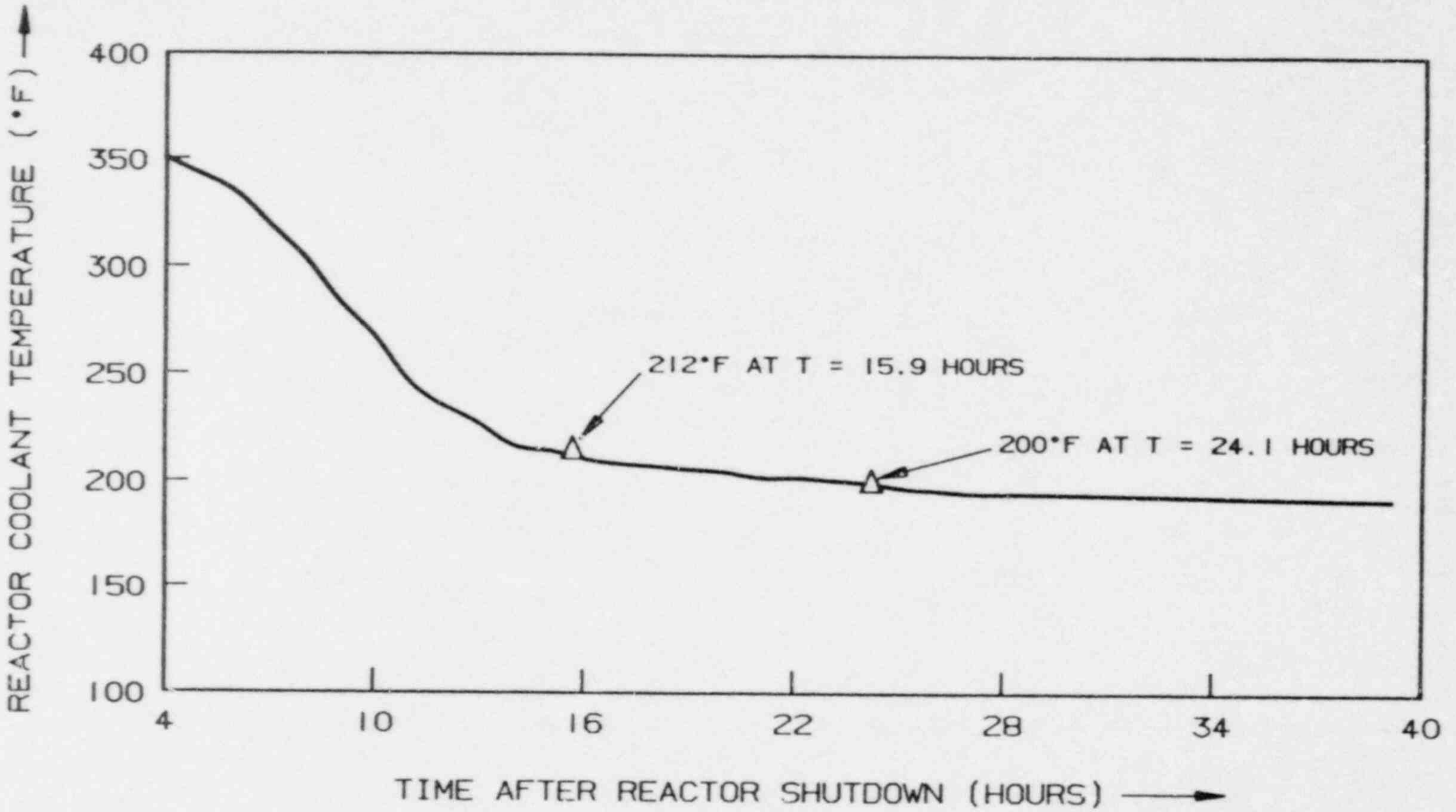


037-A-24384-1

COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

Normal RHR Cooldown

FIGURE 5.4-8



037-A-24384-2

COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

Single Train RHR Cooldown

FIGURE 5.4-9

5. Spray Nozzles and Ring Headers

The nozzles are of one-piece construction, with a 3/8-in. Diameter orifice and produce a hollow cone spray pattern.

The spray nozzles are installed on ring headers in the Containment dome and spray headers at lower elevations in the Containment. The nozzle arrangement is designed to provide maximum coverage inside the Containment. The spray header piping is designed in accordance with the ASME, B&PV Code, Section III, Class 2, and the requirements of seismic Category I. The spray nozzles are designed in accordance with manufacturer's standards and have the following characteristics:

Design flow	15.2 gpm $\pm$5 percent
52 Design operating pressure 52 drop	40 psi

6. Piping and Valves

The piping and valves of the CSS are designed in accordance with the ASME B&PV Code, Section III, Class 2, and seismic Category I requirements. They have the following characteristics:

a. The piping and valves for the CSS are designed to 325 psig
68 and 300°F. The pump suction lines from the recirculation sumps and the RWST are designed to 70 psig and 300°F.

b. The motor-operated isolation valves in the recirculation lines are totally enclosed in valve isolation tanks. The recirculation lines that run from the sumps to the Containment spray pumps are protected by concentric guard
40 piping up to the valve isolation tanks. These tanks are used to control leakage and are not tested at containment design conditions.

Any concrete particles or miscellaneous steel such as ductwork stairs or grating which may be displaced by differential pressure or jet impingement tend to sink in the low-velocity area approaching the screens.

All temporary materials used during refueling or maintenance outages, such as paper, plastic sheeting, or temporary wooden scaffolding, are removed from the Containment prior to operation.

6.2.2.3.4 Net Positive Suction Head

Sufficient net positive suction head (NPSH) is available to the Containment spray pumps for both the injection and recirculation modes of operation.

During the injection phase, the NPSH is calculated using the atmospheric pressure in the RWST, the static head between low-low RWST level and the pumps elevation, the piping losses, and the vapor pressure of water at 120°F. | 52

During the recirculation phase, adequate NPSH for the Containment spray pump is ensured by the design of the CSS in accordance with NRC Regulatory Guide 1.1. It is assumed that the Containment ambient pressure is equal to the vapor pressure of the sump liquid. | 68

Figure 6.2.2-2 shows the relationship between the available NPSH and the pump flow during the injection and recirculation phases and shows the required NPSH. Design parameters for the pumps are shown in Table 6.2.2-1. | 68

6.2.2.4.3 Operational Testing

Routine testing is performed periodically to verify the operability of active CSS components.

1. Sequencing of valves and pumps is tested by shutting the manual valve on the Containment spray line inside the Containment, shutting the manual valve on the chemical additive supply line, and triggering a dummy actuation signal. All automatic valves and the pumps are checked for proper sequencing.

2. Each pump is run at full flow and the flow is directed back to the RWST.

3. The following provisions are made for periodic testing of the spray nozzles:

   a. Compressed air test connections

   b. Special facility permitting access to all ring headers

   Operators will confirm that all nozzles are free and unobstructed.

6.2.2.5 Instrumentation Requirements

CSS operation is automatically initiated by signals generated by the
68 ESF Actuation System as described in Section 7.3. The minimum flow recirculation lines are automatically opened upon start of the Containment spray pumps. The recirculation lines are automatically closed when 25 percent of the spray pump design flow is achieved and the containment spray isolation valve for that pump is open. Flow instruments are provided in each pump discharge line. Four RWST level channels are provided and used for ECCS and CSS operation. (Refer to Section 6.3.2 for use with the ECCS.)

Level instruments provide local and remote indication, and high, low, low-low and empty level alarms.

A low-low [two out four channels] level signal is provided for 68
automatic initiation of ECCS changeover from injection mode to recirculation mode when approximately 270,000 gallons from the total usable volume of approximately 460,000 gallons have been drawn by the ECCS and the CSS.

A empty level alarm is provided to annunciate the necessity to 40
manually switch the CSS to the recirculation phase. This alarm is initiated when approximately 390,000 gallons have been exhausted.

Each Containment spray header riser is provided with a pressure switch to alarm in the Control Room if the level in the riser falls below the minimum level required.

Two water level indicator channels are provided for each Containment sump. Both are indicated in the Control Room.

The Containment spray pump discharge pressures and flows are indicated and recorded in the Control Room.

Safety-related display instrumentation is provided as described in Table 7.5-1 for the following CSS parameters:

RWST levels

Containment spray pump pressure

Containment spray pump flow

Containment sump level

68 Control Room. Lines which fall into this category include low head safety injection lines (Figure 6.2.4-1, Valve Arrangement 8), high head safety injection lines (Figure 6.2.4-1, Valve Arrangement 8, 9 and 33), Containment spray lines (Figure 6.2.4-1, Valve Arrangement 25), hydrogen purge lines (Figure 6.2.4-1, Valve Arrangement 20 and 21), Auxiliary Feedwater Lines (Figure 6.2.4-1, Valve Arrangement 36) and Main Steam Supply to the Auxiliary Feedwater Pump Turbine (Figure 6.2.4-1, Valve Arrangement 17).

2. Containment Sump Recirculation Lines

The containment sump recirculation lines, which supply suction to the low-head safety injection (RHR) pumps, and the containment spray pumps (Figure 6.2.4-1, Valve Arrangement 2) are each provided with a single remote-manual gate valve outside
38 the Containment. This valve is enclosed in a valve isolation tank (Figure 6.2.2-3) and the piping from the sump to the valve is enclosed in a concentric guard pipe.

38 The guard pipe and valve isolation tank are not considered part of the barrier between containment and external environment and are not tested at containment design conditions. The reason for this is that these moderate energy lines are designed to meet the requirements of Branch Technical Position MEB 3-1 (SRP 3.6.2) with stress levels less than 0.4 ($1.2S_h + S_a$). The penetrations are designed and fabricated per ASME Section III CL2 and MC (Article NE1000) Summer '76.

38 The valves and isolation tanks are located in the safeguard building which are served by the ESF emergency filtration system. System arrangement also provides for valve leakage control. Valve stem leakage is piped and routed via the valve stem leakoff connections to the safeguards building sumps and then to the waste holdup tanks. Valve bonnet leakage will be collected at the bottom of the tanks which are equipped with normally closed drain connections. These are also piped and routed via the safeguard building sumps to the waste holdup

seals inside the Containment, or a bolted blind flange both inside and outside the Containment. The arrangement provided satisfies the functional requirement of GDC 56 by providing redundant isolation barriers.

68 7. Turbine Driven Auxiliary Feedwater (TDAFW) Pump Steam Supply and MSIV Bypass Valves (Figure 6.2.4-1, Valve Arrangement 17)

65 The warm-up bypass valves around the steam supply valves in the TDAFW pump steam supply lines (loop 1 & 4 only, see Figure 6.2.4-1 Sheet 5 of 10) are normally locked closed. These valves are opened manually to heat-up the steam supply lines during surveillance testing to minimize degradation of the TDAFW pump due to condensation in the supply lines. The bypass warm-up valves are not required during an emergency cold start of the TDAFW pump.

68 Similarly, the manual bypass valves around the Main Steam Isolation Valves (MSIVs) are normally locked closed, and are opened during plant startup, per Tecnnical Specification requirements, to warm up the system piping downstream of the MSIVs.

66 8. Local Vent, Drain and Test Connection Valves

66 To ensure that containment integrity is maintained, local vent, drain, and test connection valves are locked closed, have capped ends and are under administrative control.

66 9. SIS lest Line Pressure Indicator

66 The SIS test line subsystem is described in Section 6.3.4. The SIS test lines are provided with containment isolation valves 1-8871, 1-8964, and 1-8888. Local pressure indicator 1-PI-929 is provided outside the reactor containment in the piping between

The Containment Isolation System is designed in accordance with 10 CFR Part 50, Appendix A, GDC 54, 55, 56, 57 requirements discussed in Subsection 6.2.4.1.2.

Instrument lines penetrating the Containment are designed in accordance with NRC Regulatory Guide 1.11 as described in Subsection 6.2.4.1.4.

A detailed discussion of missile protection can be found in Section 3.5. The design features and measures against jet forces and pipe whip are described in Section 3.6, and the seismic design bases for the safety-related systems are provided in Section 3.7.

In case of loss of control voltage to the associated solenoid valve, or in case of loss of air, the air-operated containment isolation valves move to the position of greatest safety. Motor-operated containment isolation valves fail in the as is position.

Emergency power from the standby diesel generators is provided to the AC valves to ensure system operation in the event of a loss of offsite power.

15 All containment isolation valves which receive signals to close from containment isolation phase A, containment isolation phase B, steamline isolation or feedwater isolation have valve closure times as fast as practical, consistent with the type of valves and valve
68 operators, with consideration given to water hammer effects. Those lines which provide a direct connection from the Containment atmosphere to the environment are equipped with isolation valves having closing times of five seconds or less or are locked closed.
15 Valve closure time for each valve is shown in Table 6.2.4-3. To ensure operability, the Containment Isolation System is designed to meet the single failure criteria with no loss of function.

The possibility of debris becoming entrained in escaping fluid and preventing tight closure of an isolation valve is of concern only for penetrations which are open to the Containment atmosphere during power operation. The Containment pressure relief line is the only penetration which falls in this category. The following provisions are made to ensure that debris does not become entrained in escaping fluid and prevent tight closing of the isolation valves in this line:

1. Pipe provided for this function is separate from other duct systems inside of the Containment and is seismic Category I; no registers or other potential sources of debris are used in this ductwork. 41

2. The pipe is routed to a clear area inside Containment as close to the penetration as possible; the clear area is free of potential sources of debris such as piping insulation, and so forth. 41

3. The entrance to the pipe is bell mouthed to minimize entrance velocity. A debris screen is provided as described in Section 9.4A. 41

4. The isolation valves are fast closing valves which close in less than five seconds.

The redundancy requirement is satisfied by having two isolation barriers in series, one on each side of Type A and Type B penetrations. Reliability is assured by conducting periodic tests to check the operability of the isolation valves, actuators, and controls. Furthermore, a fail-safe feature is incorporated into air-operated and solenoid-operated isolation valve design, so that in the event of actuating power loss, the valve assumes the position that ensures safety.

The power operated isolation valves may be operated manually from the Control Room to provide a secondary means of actuation. To confirm performance capabilities, i.e., closing time and valve status (open or closed), indicator lights are checked and observed during the periodic testing.

The primary and secondary modes of actuation are shown in Table 6.2.4-2.

68 | As shown in the Figure 6.2.4-1, penetrations which require leakage testing are provided with appropriate test connections.

6.2.4.3 Design Evaluation

The design of the Containment Isolation System meets the requirements for system integrity, response, operation, and reliability. Isolation valve and piping design and their location ensure containment integrity for any postulated single failure.

Regular functional testing of the Containment Isolation System during shutdown periods assures operability of all isolation valves. Leakage rate testing during the same periods ensures that the leakage through isolation valves and piping penetrations does not exceed values commensurate with offsite radiation doses under accident conditions, given in the Technical Specification. The use of double isolation barriers ensures that no single failure of any active or passive component renders the Containment Isolation System either partially or wholly inoperable. Open or closed isolation valve status during normal plant operation is regularly checked and controlled, particularly with regard to manually-operated isolation valves. In addition, automatic isolation valves, whether actuated remote-manually or by isolation signals, are designed to assume a fail safe position. These are tabulated in Tables 6.2.4-1, 2 and 3.

TABLE 6.2.2-1
(SHEET 1 of 3)

CONTAINMENT SPRAY SYSTEM
COMPONENT DESIGN PARAMETERS

1. Containment Spray Pump

Quantity	4	
Type	Horizontal centrifugal	
Design pressure, psig	325	
Design temperature, F	300	
Design flow rate, gpm	3000	
Total design head, ft	585	
NPSH required, ft	12	

2. Containment Spray Nozzle

Quantity	759/764 Unit 1/Unit 2	| 68
Type	Spraco 1713A	
Flow per nozzle at 40 psi, gpm	15.2 ± percent	

3. Refueling Water Storage Tank

Quantity, per unit	1	
Usable volume, gal	450,000	
Total content, gal	503,180	
Boric acid concentration, ppm	2000	
Design pressure, ft	50	
Design temperature, F	120	| 68
Operating pressure, ft	41 (Hydraulic head)	

TABLE 6.2.2-1
(SHEET 2)

CONTAINMENT SPRAY SYSTEM
COMPONENT DESIGN PARAMETERS

4. Containment Spray Heat Exchanger

Quantity	2	
Type	Shell and U tube	
Overall heat transfer (Btu/hr ft^2 F) coefficient	580.0	| 68
Flow, gpm		
Shell side	6080	
Injection	5800	
Recirculation	7200	
Inlet Temperature, F		
Shell side, maximum	135	
Tube side, maximum	243	
Design Pressure, psig		
Shell	150	
Tube	325	
Design Temperature, F		
Shell	200	
Tube	300	

TABLE 6.2.2-1

(SHEET 3)

CONTAINMENT SPRAY SYSTEM

COMPONENT DESIGN PARAMETERS

5. Piping and Valves

Spray Discharge Lines		
Design pressure, psig	325	
Design temperature, F	300	
Spray Suction Lines		
Design pressure, psig	70	| 68
Design temperature, F	300	
Chemical Supply Line		
Design pressure, psig	20	
Design temperature, F	150	
Eductor Piping		
Design pressure, psig	325	
Design temperature, F	300	

6. Valve Isolation Tank

Quantity, per unit	2
Type	Vertical
Design pressure, psig	50
Design temperature, F	280

TABLE 6.2.2-2

CONTAINMENT SUPPLY SYSTEM

DESIGN PARAMETERS

Number of spray trains	2	
Flow rate per spray train, gpm (minimum)		68
Injection	5800	
Recirculation	7200	
Number of headers per spray train	7	
Spray initiation - time after LOCA (sec.)	60	
Spray recirculation initiation - time after LOCA with one 100 percent spray, one ECCS train (min.)	56.2	
Spray recirculation initiation - time after LOCA with two 100 percent sprays, one ECCS train (min.)	30 6	
Spray recirculation initiation - time after LOCA with one 100 percent spray, two ECCS trains (min.)	50.9	
Spray recirculation initiation - time after LOCA with two 100 percent sprays, two ECCS trains (min.)	28.1	

TABLE 6.2.4-1
(Sheet 2 of 10)

CONTAINMENT ISOLATION VALVING APPLICATION

Item	Penetration Number	System	Line or Services	Line Size (Inches)	NRC General Design Criterion or Reg. Guide Met	Isolation Valving Arrangement (Fig. 6.2.4-1)	Fluid Contained	Engineered Safeguard Feature	FSAR Figure Number	
			Generator #4							
16	MI-4	MS	Drain From Main Steam Line #4	2	57	17	Sat. Water	No	10.3-1	
17	MI-4	MS	Main Steam to Aux. F.P.T. From Steam line #4	4	57	17	Sat. Steam	Yes	10.3-1	42
17a	MI-4	MS	TDAFW Pump Bypass Warm-up Valve	1	57	17	Sat. Steam	No	10.3-1	65
18	MI-4	MS	Main Steam From Steam Generator #4	8	57	17	Sat. Steam	No	10.3-1	
18a	MI-4	MS	N_2 Supply to Steam Generator #4	3/4	57	17	Sat. Steam	No	10.3-1	
18b	MI-4	MS	Main Steam Safety Valves From Steam Generator #4	6	57	17	Sat. Steam	No	10.3-1	
19	MI-5	FW	Feedwater to Steam Generator #1	18	57	16	Water	No	10.4-9	
20	MI-5	FW	Feedwater Sample (FW to Steam Generator #1)	3/4	57	16	Water	No	10.4-9	66
20a	MV-18	FW	Auxiliary Feedwater to Steam Generator #1	4	57	36	Water	Yes	10.4-11	68
20b	MI-5	FW	N2 Supply to Steam Generator #1	3/4	57	16	Water	No	10.4-9	
20c	MV-18	FW	Feedwater Preheater Bypass Line	6	57	36	Water	No	10.4-9	
20d	MI-5	FW	Feedwater Bypass Line	3	57	16	Water	No	10.4-9	
21	MI-6	FW	Feedwater to Steam Generator #2	18	57	16	Water	No	10.4-9	
22	MI-6	FW	Feedwater Sample (FW to Steam Generator #2)	3/4	57	16	Water	No	10.4-9	
22a	MV-19	FW	Auxiliary Feedwater to Steam Generator #2	4	57	36	Water	Yes	10.4-11	68

TABLE 6.2.4-1
(Sheet 3 of 10)

CONTAINMENT ISOLATION VALVING APPLICATION

Item	Penetration Number	System	Line or Services	Line Size (Inches)	NRC General Design Criterion or Reg. Guide Met	Isolation Valving Arrangement (Fig. 6.2.4-1)	Fluid Contained	Engineered Safeguard Feature	FSAR Figure Number	
22b	MI-6	FW	N2 Supply to Steam Generator #2	3/4	57	16	Water	No	10.4-9	
22c	MV-19	FW	Feedwater Preheater Bypass Line	6	57	36	Water	No	10.4-9	66
22d	MI-6	FW	Feedwater Bypass Line	3	57	16	water	No	10.4-9	
23	MI-7	FW	Feedwater to Steam Generator #3	18	57	16	Water	No	10.4-9	
										68
24a	MV-20	FW	Auxiliary Feedwater to Steam Generator #3	4	57	36	Water	Yes	10.4-11	
24b	MI-7	FW	N2 Supply to Steam Generator #3	3/4	57	16	Water	No	10.4-9	
24c	MV-20	FW	Feedwater Preheater Bypass Line	6	57	36	Water	No	10.4-9	
24d	MI-7	FW	Feedwater Bypass Line	3	57	16	Water	No	10.4-9	66
25	MI-8	FW	Feedwater to Steam Generator #4	18	57	16	Water	No	10.4-9	
										68
26a	MV-17	FW	Auxiliary Feedwater to Steam Generator #4	4	57	36	Water	Yes	10.4-11	
26b	MI-8	FW	N2 Supply to Steam Generator #4	3/4	57	16	Water	No	10.4-9	66
26c	MV-17	FW	Feedwater Preheater Bypass Line	6	57	36	Water	No	10.4-9	

TABLE 6.2.4-1

(Sheet 4 of 10)

CONTAINMENT ISOLATION VALVING APPLICATION

Item	Penetration Number	System	Line or Services	Line Size (Inches)	NRC General Design Criterion or Reg. Guide Met	Isolation Valving Arrangement (Fig. 6.2.4-1)	Fluid Contained	Engineered Safeguard Feature	FSAR Figure Number	
26d	MI-8	FW	Feedwater Bypass Line	3	57	16	Water	No	10.4-9	
27	MI-9	MS	Blowdown From Steam Generator #3	3	57	35	Sat. Water	No	10.3-1	
28	MI-10	MS	Blowdown From Steam Generator #2	3	57	35	Sat. Water	No	10.3-1	
29	MI-11	MS	Blowdown From Steam Generator #1	3	57	35	Sat. Water	No	10.3-1	2
30	MI-12	MS	Blowdown From Steam Generator #4	3	57	35	Sat. Water	No	10.3-1	
31	MI-13	-	Spare	12	N/A	-	-	-	-	
32	MII-1	CS	Letdown Line to Letdown Heat Exchanger	3	55	37	Water	No	9.3-10	
33	MII-2	RH	R.H.R From Hot Leg Loop #4	12	55	11	Water	No	5.4-6	
34	MII-3	RH	R.H.R From Hot Leg Loop #1	12	55	11	Water	No	5.4-6	
35	MII-4	SI	R.H.R to Cold Leg Loops #1 and #2	10	55	8	Water	Yes	6.3-1	2
36	MII-5	SI	R.H.R To Cold Leg Loops #3 and #4	10	55	8	Water	Yes	6.3-1	
37	MII-6	-	Spare	12	N/A	-	-	-	-	
38	MII-7	-	Spare	24	N/A	-	-	-	-	
39	MII-8	-	Spare	24	N/A	-	-	-	-	
40	MII-9	-	Maintenance Penetration	12	56	31	Air	No	N/A	46
41	MIII-1	RC	Reactor Make Up Water to Pressurizer Relief Tank & R.C. Pump Stand Pipe	3	56	4	Water	No	5.1-1	55
41a	MIII-1	RC	Penetration Thermal Relief	3/4	56	4	Water	No	5.1-1	66
42	MIII-2	SI	S.I. To Cold Leg Loops #1, #2, #3, & #4	3	55	9	Water	Yes	6.3-1 (Sh. 1)	2

TABLE 6.2.4-1
(Sheet 9 of 10)

CONTAINMENT ISOLATION VALVING APPLICATION

Item	Penetration Number	System	Line or Services	Line Size (Inches)	NRC General Design Criterion or Reg. Guide Met	Isolation Valving Arrangement (Fig. 6.2.4-1)	Fluid Contained	Engineered Safeguard Feature	FSAR Figure Number	
108	MIV-12(c)	-	Spare	2	N/A	-	-	-	-	
109	MV-1	VA	Containment Purge Air Supply	48	56	18	Air	No	9.4-6	2
110	MV-2	VA	Containment Purge Air Exhaust	48	56	19	Air	No	9.4-6	
111	MV-3	CC	C.C. Supply To Excess Letdown & R.C. Drain Tank Heat Exchanger	4	57	32	Water	No	9.2-3 (Sh. 3)	66
111a	MV-3	CC	Penetration Thermal Relief	3/4	57	32	Water	No	9.2-3 (Sh. 3)	68
111b	MV-3	CC	Penetration Thermal Relief	3/4	57	32	Water	No	9.2-3	
112	MV-4	CC	C.C. Return From Excess Letdown & R.C. Drain Tank Heat Exchanger	4	57	12	Water	No	9.2-3 (Sh. 3)	
113	MV-5	CA	Service Air to Containment	3	56	7	Air	No	9.3-1	2
114	MV-6	CC	Containment CCW Drain Tank Pumps Discharge	2	56	26	Water	No	9.2-3 (Sh. 3)	
114a	MV-6	CC	Penetration Thermal Relief	3/4	56	26	Water	No	9.2-3 (Sh. 3)	66
115	MV-7	LT	Containment Leak Rate Test Pressure Sensing	8	56	38	Air	No	9.4-6	
116	MV-8	RC	Nitrogen Supply to PRT	1	56	13	Nitrogen	No	5.1-1 (Sh. 2)	
117	MV-9	CC	CC Return From R.C.P's Motors	8	56	24	Water	No	9.2-3 (Sh. 3)	2
118	MV-10	CC	CC Supply to R.C.P's Motors	10	56	25	Water	No	9.2-3 (Sh. 3)	66
119	MV-11	CC	CC Return From R.C.P's Thermal Barrier	4	56	24	Water	No	9.2-3 (Sh. 3)	2
120	MV-12	CH	Chilled Water Supply to Containment Coolers	6	56	25	Water	No	9.4-11	
120a	MV-12	CH	Penetration Thermal Relief	3/4	56	25	Water	No	9.4-11	66

CONTAINMENT ISOLATION VALVING APPLICATION (Note 8) 66

Item	Isolation Valve No.	Location in Relation to Containment	Type of Leakage Rate Test	Length of Pipe to Outermost Isolation Valve (ft)	Valve Type/Operator	Method of Actuation Primary	Method of Actuation Secondary	
19	HV-2134	Outside	Note 1	10	Gate/Hydr. N2 Actuator	Auto close	Remote Manual	42
20	1HV-2154	Outside	Note 1	-	Globe/Air	Auto close	Remote Manual	
20a	1HV-2491A	Outside	Note 1	50	Gate/Motor	Remote Manual	Local Manual	
	1HV-2491B	Outside	Note 1	50	Gate/Motor	Remote Manual	Local Manual	
20b	1FW-106	Outside	Note 1	-	Globe/Manual	Local Manual	N/A	
20c	1FW-2193	Outside	Note 1	29'-6"	Globe/Air	Auto close	Remote Manual	66
20d	1HV-2185	Outside	Note 1	12'-7"	Globe/Air	Auto close	Remote Manual	
21	HV-2135	Outside	Note 1	10	Gate/Hydr. N2 Actuator	Auto close	Remote Manual	
22	1HV-2155	Outside	Note 1	-	Globe/Air	Auto close	Remote Manual	
22a	1HV-2492A	Outside	Note 1	50	Gate/Motor	Remote Manual	Local Manual	42
	1HV-2492B	Outside	Note 1	50	Gate/Motor	Remote Manual	Local Manual	
22b	1FW-104	Outside	Note 1	-	Globe/Manual	Local Manual	N/A	
22c	1FV-2194	Outside	Note 1	29'-6"	Globe/Air	Auto close	Remote Manual	
22d	1HV-2186	Outside	Note 1	12'-8"	Globe/Air	Auto close	Remote Manual	66
23	HV-2136	Outside	Note 1	10	Gate/Hydr. N2 Actuator	Auto close	Remote Manual	
								68
24a	1HV-2493A	Outside	Note 1	50	Gate/Motor	Remote Manual	Local Manual	42
	1HV-2493B	Outside	Note 1	50	Gate/Motor	Remote Manual	Local Manual	

CONTAINMENT ISOLATION VALVING APPLICATION (Note 8) 66

Item	Isolation Valve No.	Location in Relation to Containment	Type of Leakage Rate Test	Length of Pipe to Outermost Isolation Valve (ft)	Valve Type/Operator	Method of Actuation Primary	Method of Actuation Secondary	
24b	1FW-102	Outside	Note 1	-	Globe/Manual	Local Manual	N/A	66
24c	1FV-2195	Outside	Note 1	30'-9"	Globe/Air	Auto close	Remote Manual	
24d	1HV-2187	Outside	Note 1	12'-8"	Globe/Air	Auto close	Remote Manual	
25	HV-2137	Outside	Note 1	10	Gate/Hydr. N2 Actuator	Auto close	Remote Manual	42
								68
26a	1HV-2494A 1HV-2494B	Outside Outside	Note 1 Note 1	50 50	Gate/Motor Gate/Motor	Remote Manual Remote Manual	Local Manual Local Manual	
26b	1FW-108	Outside	Note 1	-	Globe/Manual	Local Manual	N/A	
26c	1FV-2196	Outside	Note 1	31'-10"	Globe/Air	Auto close	Remote Manual	
26d	1HV-2188	Outside	Note 1	12'-8"	Globe/Air	Auto close	Remote Manual	66
27	HV-2399	Outside	Note 1	17	Globe/Air	Auto close	Remote Manual	
28	HV-2398	Outside	Note 1	17	Globe/Air	Auto close	Remote Manual	
29	HV-2397	Outside	Note 1	17	Globe/Air	Auto close	Remote Manual	42
30	HV-2400	Outside	Note 1	17	Globe/Air	Auto close	Remote Manual	
31	-	-	-	-	-	-	-	
32	1-8152 1-8160	Outside Inside	C C	7 -	Globe/Air Globe/Air	Auto close Auto close	Remote Manual Remote Manual	
33	1-8701B, 1-8708B	Inside Inside	Note 5 Note 5	N/A N/A	Gate/Motor Relief	Remote Manual Self Actuated	Remote Manual N/A	
34	1-8701A, 1-8708A	Inside Inside	Note 5 Note 5	N/A N/A	Gate/Motor Relief	Remote Manual Self Actuated	Local Manual N/A	

CONTAINMENT ISOLATION VALVING APPLICATION (Note 8) | 66

Item	Isolation Valve No.	Location in Relation to Containment	Type of Leakage Rate Test	Length of Pipe to Outermost Isolation Valve (ft)	Valve Type/Operator	Method of Actuation Primary	Method of Actuation Secondary	
94	HV-5544	Outside	C	4'-0"	Globe/Solenoid	Auto close	Remote Manual	12
	HV-5545	Inside	C	-	Globe/Solenoid	Auto close	Remote Manual	
95	None	-	A	-	N/A	-	-	
96	-	-	-	-	-	-	-	
97	HV-5558	Outside	C	5'-0"	Globe/Solenoid	Auto close	Remote Manual	12
	HV-5559	Inside	C	-	Globe/Solenoid	Auto close	Remote Manual	
98	None	-	A	-	N/A	-	-	
99	-	-	-	-	-	-	-	
100	HV-5560	Outside	C	4'-0"	Globe/Solenoid	Auto close	Remote Manual	12
	HV-5561	Inside	C	-	Globe/Solenoid	Auto close	Remote Manual	
101	None	-	A	-	N/A	-	-	
102	HV-5546	Outside	C	2'-0"	Globe/Solenoid	Auto close	Remote Manual	12
	HV-5547	Inside	C	-	Globe/Solenoid	Auto close	Remote Manual	
103	-	-	-	-	-	-	-	42
104	1-8880	Outside	C	28'	Globe/Air,	Auto close	Remote Manual	5
	1SI-8968	Inside	C	-	Check	Self Actuated	N/A	
105	1-7126	Inside	C	-	Diaphragm/Air	Auto close	Remote Manual	
	1-7150	Outside	C	19'	Diaphragm/Air	Auto close	Remote Manual	
106	-	-	-	-	-	-	-	
107	-	-	-	-	-	-	-	
108	-	-	-	-	-	-	-	
109	HV-5536	Outside	C	1'-0"	Butterfly/Air	Auto close	Remote Manual	12
	HV-5537	Inside	C	-	Butterfly/Air	Auto close	Remote Manual	
110	HV-5538	Outside	C	2'	Butterfly/Air	Auto close	Remote Manual	
	HV-5539	Inside	C	-	Butterfly/Air	Auto close	Remote Manual	
111	HV-4710	Outside	Note 1	12'	Globe/Air	Auto close	Remote Manual	42
111a	1CC-611	Inside	Note 1	-	Relief	Self-Actuated	N/A	68
111b	1CC-618	Inside	Note 1	-	Refief	Self-Actuated	N/A	

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CONTAINMENT ISOLATION VALVING APPLICATION (Note 1) 42

Item	Containment Isolation Signal	Valve Position Normal	Shutdown	Post-Accident	Valve Power Failure	Valve Closure Time (Sec.)	Power Source	Remarks	
									14
24a	-	Opened	Opened	Opened	FAI	N/A	A		68
	-	Opened	Opened	Opened	FAI	N/A	B		66
24b	-	Closed	Opened	Closed	Closed	N/A	-		
24c	Feedwater Isolation Signal	Closed	Closed	Closed	Closed	5	A/B		
24d	Feedwater Isolation Signal	Closed	Closed	Closed	Closed	5	A/B		
25	Feedwater Signal	Opened	Closed	Closed	FAI	5	A/B		
26a	-	Opened	Opened	Opened	FAI	N/A	A		68
	-	Opened	Opened	Opened	FAI	N/A	B		
26b	-	Closed	Opened	Closed	Closed	N/A	-		66
26c	Feedwater Isolation Signal	Closed	Closed	Closed	Closed	5	A/B		

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TABLE 6.2.4-3
(Sheet 6 of 14)

CONTAINMENT ISOLATION VALVING APPLICATION (Note 1)

Item	Containment Isolation Signal	Valve Position Normal	Valve Position Shutdown	Valve Position Post-Accident	Valve Power Failure	Valve Closure Time (Sec.)	Power Source	Remarks	
37	-	-	-	-	-	-	-		
38	-	-	-	-	-	-	-		
39	-	-	-	-	-	-	-		
40	-	N/A	N/A	N/A	N/A	N/A	N/A		
41	Phase A	Closed	Closed	Closed	Closed	10	B		
41a	-	Closed	Closed	Closed	-	-	-		66
42	-	Closed	Closed	Opened	FAI	N/A	A		
	-	Closed	Closed	Opened	FAI	N/A	B		
	Phase A	Closed	Closed	Closed	Closed	10	B	Air operated valve on test line	68
43	-	Closed	Closed	Opened	FAI	N/A	A		
	Phase A	Closed	Closed	Closed	Closed	10	A	Air operated valve on test line	66
44	-	Closed	Closed	Opened	FAI	N/A	B		
	Phase A	Closed	Closed	Closed	Closed	10	A	Air operated valve on test line	
45	-	Opened	Opened	Opened	FAI	N/A	B		
	Phase A	Closed	Closed	Closed	Closed	10	A	Air operated valve on test line	
46	Safety Injection	Opened	Closed	Closed	FAI	10	B		
47	-	Opened	Closed	Open/ Closed	FAI	N/A	B		

CONTAINMENT ISOLATION VALVING APPLICATION (Note 1)

Item	Containment Isolation Signal	Valve Position Normal	Valve Position Shutdown	Valve Position Post-Accident	Valve Power Failure	Valve Closure Time (Sec.)	Power Source	Remarks	
110	Containment Vent Isolation	Closed	Opened	Closed	Closed	5	B		
	Containment Vent Isolation	Closed	Opened	Closed	Closed	5	A		
111	Phase A	Opened	Closed	Closed	Closed	5	B		
111a	-	Closed	Closed	Closed	-	-	-		
111b	-	Closed	Closed	Closed	-	-	-		68
112	Phase A	Opened	Closed	Closed	Closed	5	B		66
113	Phase A	Closed	Opened	Closed	Closed	5	B		
114	Phase A	Opened	Opened	Closed	Closed	5	A		
	Phase A	Opened	Opened	Closed	Closed	5	B		
114a	-	Closed	Closed	Closed	-	-	-		
115	-	N/A	N/A	N/A	N/A	N/A	N/A		
116	Phase A	Closed	Closed	Closed	Closed	10	B		66
	Phase A	Closed	Closed	Closed	Closed	10	A		
117	Phase B	Opened	Closed	Closed	FAI	10	B(HV-4708)		
	Phase B	Opened	Closed	Closed	FAI	10	A(HV-4701)		
118	Phase B	Opened	Closed	Closed	FAI	10	B		
119	Phase B	Opened	Closed	Closed	FAI	10	B(HV-47[illegible])		
	Phase B	Opened	Closed	Closed	FAI	10	A		66
120	Phase A	Opened	Opened	Closed	FAI	10	B		
120a	-	Closed	Closed	Closed	-	-			66

TABLE 6.2.4-3
(Sheet 14 of 14)

CONTAINMENT ISOLATION VALVING APPLICATION (Note 1)

Item	Containment Isolation Signal	Valve Position			Valve Power Failure	Valve Closure Time (Sec.)	Power Source	Remarks
		Normal	Shutdown	Post-Accident				
121	Phase A Phase A	Opened Opened	Opened Opened	Closed Closed	FAI FAI	10 10	B(HV-6082) A(HV-6083)	
121a		Closed	Closed	Closed	-	-		66
122	Containment Vent Isolation Containment Vent Isolation	Closed Closed	Closed Closed	Closed Closed	Closed Closed	3 3	- -	Intermittently opened in normal operation to function as part of the containment pressure relief system.
123	-	N/A	N/A	N/A	N/A	N/A	N/A	
124	Phase A Phase A	Closed Closed	Closed Closed	Closed Closed	FAI FAI	10 10	B(4075B) A(4075C)	
125	-	Closed	Closed	Opened	FAI	N/A	A	
126	-	Closed	Closed	Opened	FAI	N/A	B	
127	-	Closed	Closed	Opened	FAI	N/A	A	
128	-	Closed	Closed	Opened	FAI	N/A	B	
129	N/A	N/A	N/A	N/A	N/A	N/A	-	

Note 1 - Check valves are not identified; their position is determined by system pressure and flow conditions.

TABLE 6.2.4-5
(Sheet 2)

ISOLATION VALVES (TESTING ARRANGEMENTS)

Item	Penetration No.	Isolation Valve No.	Test Connection	Test Vent	Block Valve or Test Barriers	Direction of Test (Note 11)	
20b	MI-5	1FW-106	N/A	N/A	N/A	N/A	66
20c	MV-18	1FV-2193	N/A	N/A	N/A	N/A	
20d	MI-5	1HV-2185	N/A	N/A			
21	MI-6	HV-2135	N/A	N/A	N/A	N/A	
22	MI-6	HV-2155	N/A	N/A	N/A	N/A	
22a	MV-19	HV-2492A	1AF-164 (Note 13)	N/A	N/A	N/A	
		HV-2492B	1AF-164 (Note 13)	N/A	N/A	N/A	
22b	MI-6	1FW-104	N/A	N/A	N/A	N/A	
22c	MV-19	1FV-2194	N/A	N/A	N/A	N/A	
22d	MI-6	1HV-2186	N/A	N/A	N/A	N/A	66
23	MI-7	HV-2136	N/A	N/A	N/A	N/A	
							68
24a	MV-20	HV-2493A	1AF-163 (Note 13)	N/A	N/A	N/A	
		HV-2493B	1AF-163 (Note 13)	N/A	N/A	N/A	
24b	MI-7	1FW-102	N/A	N/A	N/A	N/A	66
24c	MV-20	1FV-2195	N/A	N/A	N/A		
24d	MI-7	1HV-2187	N/A	N/A			

TABLE 6.2.4-5
(Sheet 3)

ISOLATION VALVES (TESTING ARRANGEMENTS)

Item	Penetration No.	Isolation Valve No.	Test Connection	Test Vent	Block Valve or Test Barriers	Direction of Test (Note 11)	
25	MI-8	HV-2137	N/A	N/A	N/A	N/A	68
26a	MV-17	HV-2494A	1AF-166 (Note 13)	N/A	N/A	N/A	
		HV-2494B	1AF-166 (Note 13)	N/A	N/A	N/A	65
26b	MI-8	1FW-108	N/A	N/A	N/A	N/A	
26c	MV-17	1FV-2196	N/A	N/A	N/A	N/A	
26d	MI-8	1HV-2188	N/A	N/A	N/A	N/A	
27	MI-9	HV-2399	1MS-154 (Note 13)	N/A	N/A	N/A	
28	MI-10	HV-2398	1MS-152 (Note 13)	N/A	N/A	N/A	
29	MI-11	HV-2397	1MS-150 (Note 13)	N/A	N/A	N/A	
30	MI-12	HV-2400	1MS-148 (Note 13)	N/A	N/A	N/A	
31	MI-13	Spare					
32	MII-1	1-8152	1CS-078 (or 1CS-029 (Note 13))	1CS-001	1-8149A 1-8149B 1-8149C 1-8117 1CS-029 (Note 13)	Yes	55
		1-8160	1CS-078	1CS-001 (or 1CS-029 (Note 13))	1-8149A 1-8149B 1-8149C 1-8117	Yes	55

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CPSES/FSAR
TABLE 6.2.4-5
(Sheet 13)

ISOLATION VALVES (TESTING ARRANGEMENTS)

Item	Penetration No.	Isolation Valve No.	Test Connection	Test Vent	Block Valve or Test Barriers	Direction of Test (Note 11)	
110	MV-2	HV-5539 &	1VA-004 (Note 13)	Note 4	N/A	No	55
		HV-5538		Note 3	N/A	Yes	
111	MV-3	HV-4710	N/A	N/A	N/A	N/A	
111a	MV-3	1cc-611	N/A	N/A	N/A	N/A	68
111b	MV-3	1cc-618	N/A	N/A	N/A	N/A	
112	MV-4	HV-4711	N/A	N/A	N/A	N/A	
113	MV-5	1CA-016	1CA-036	1CA-019 (Note 13) (or 1CA-018 (Note 13) or 1CA-032))	1CA-031	Yes	66
		HV-3486	1CA-019 (Note 13) (or 1CA-018 (Note 13))	1CA-032	1CA-031 1CA-036 1CA-018 (Note 13) (or 1CA-019 (Note 13))	Yes	
114	MV-6	HV-4725	1CC-730	Note 8	1CC-743 1CC-741	Yes	55
		HV-4726	1CC-730	Note 8	1CC-743 1CC-741	Yes	
114a	MV-6	1CC-1067	1CC-730	N/A	1CC-741 1CC-743 1HV-4726	Yes	66
115	MV-7	N/A	1LT-004 (Note 13) (or 1LT-005 (Note 13)) (or 1LT-006 (Note 13))	N/A	Note 14	N/A	55
116	MV-8	1-8026	1RC-006	1RC-004 (or 1RC-005 (Note 13))	1RC-8025	Yes	
		1-8027	1RC-006 (or 1RC-005 (Note 13))	1RC-004	1RC-8025 1RC-005 (Note 13) (or 1RC-006)	Yes	66

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TABLE 6.2.4-6
(Sheet 3 of 13)

CLASSIFICATION OF SYSTEMS PATHS PENETRATION CONTAINMENT WALL

Item	Penetration Number	System	Normal Operating Function	Classification	Post-Accident Function	
20b	MI-5	FW	N2 Supply to Steam Generator #1	non-essential	none	66
20c	MV-18	FW	Tempering Line	non-essential	none	
20d	MI-5	FW	Isolation Valve Bypass	non-essential	none	
21	MI-6	FW	Feedwater to Steam Generator #2	non-essential	none	
22	MI-6	FW	Secondary Sample (FW to Steam Generator #2)	non-essential	none	
22a	MV-19	FW	Auxiliary Feedwater to Steam Generator #2	essential	Auxiliary Feedwater to Steam Generator	
22b	MI-6	FW	N2 Supply to Steam Generator #2	non-essential	none	
22c	MV-19	FW	Tempering Line	non-essential	none	66
22d	MI-6	FW	Isolation Valve Bypass	non-essential	none	
23	MI-7	FW	Feedwater to Steam Generator #3	non-essential	none	

68

TABLE 6.2.4-6
(Sheet 4 of 13)

CLASSIFICATION OF SYSTEMS PATHS PENETRATION CONTAINMENT WALL

Item	Penetration Number	System	Normal Operating Function	Classification	Post-Accident Function	
24a	MV-20	FW	Auxiliary Feedwater to Steam Generator #3	essential	Auxiliary Feedwater to	
24b	MI-7	FW	N2 Supply to Steam Generator #3	non-essential	none	
24c	MV-20	FW	Tempering Line	non-essential	none	66
24d	MI-7	FW	Isolation Valve Bypass	non-essential	none	
25	MI-8	FW	Feedwater to Steam Generator #4	non-essential	none	
						68
26a	MV-17	FW	Auxiliary Feedwater to Steam Generator #4	essential	Auxiliary Feedwater to Steam Generator	
26b	MI-8	FW	N2 Supply to Steam Generator #4	non-essential	none	
26	MV-17	FW	Tempering Line	non-essential	none	66
26d	MI-8	FW	Isolation Valve Bypass	non-essential	none	

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CLASSIFICATION OF SYSTEMS PATHS PENETRATION CONTAINMENT WALL

Item	Penetration Number	System	Normal Operating Function	Classification	Post-Accident Function	
101	MIV-10(b)	ESFAS	Containment Pressure Sensing PT-937	essential	Containment Pressure measurement	
102	MIV-10(c)	RM	Radiation Monitoring Sample	non-essential	none	
103	MIV-11(a)	-	Spare	-	-	42
104	MIV-11(b)	SI	N2 Supply To Accumulators	non-essential	none	
105	MIV-11(c)	WP	H2 Supply to RC Drain Tank	non-essential	none	
106	MIV-12(a)	-	Spare	-	-	
107	MIV-12(b)	-	Spare	-	-	
108	MIV-12(c)	-	Spare	-	-	
109	MV-1	VA	Containment Purge Air Supply	non-essential	none	
110	MV-2	VA	Containment Purge Air Exhaust	non-essential	none	
111	MV-3	CC	C.C. Supply to Excess Letdown & RC Drain Tank Heat Exchanger	non-essential	none	
111a	MV-3	CC	None	non-essential	none	68
111b	MV-3	CC	None	non-essential	none	
112	MV-4	CC	C.C. Return from Excess Letdown & RC Drain Tank Heat Exchanger	non-essential	none	
113	MV-5	CA	Service Air to Containment	non-essential	none	

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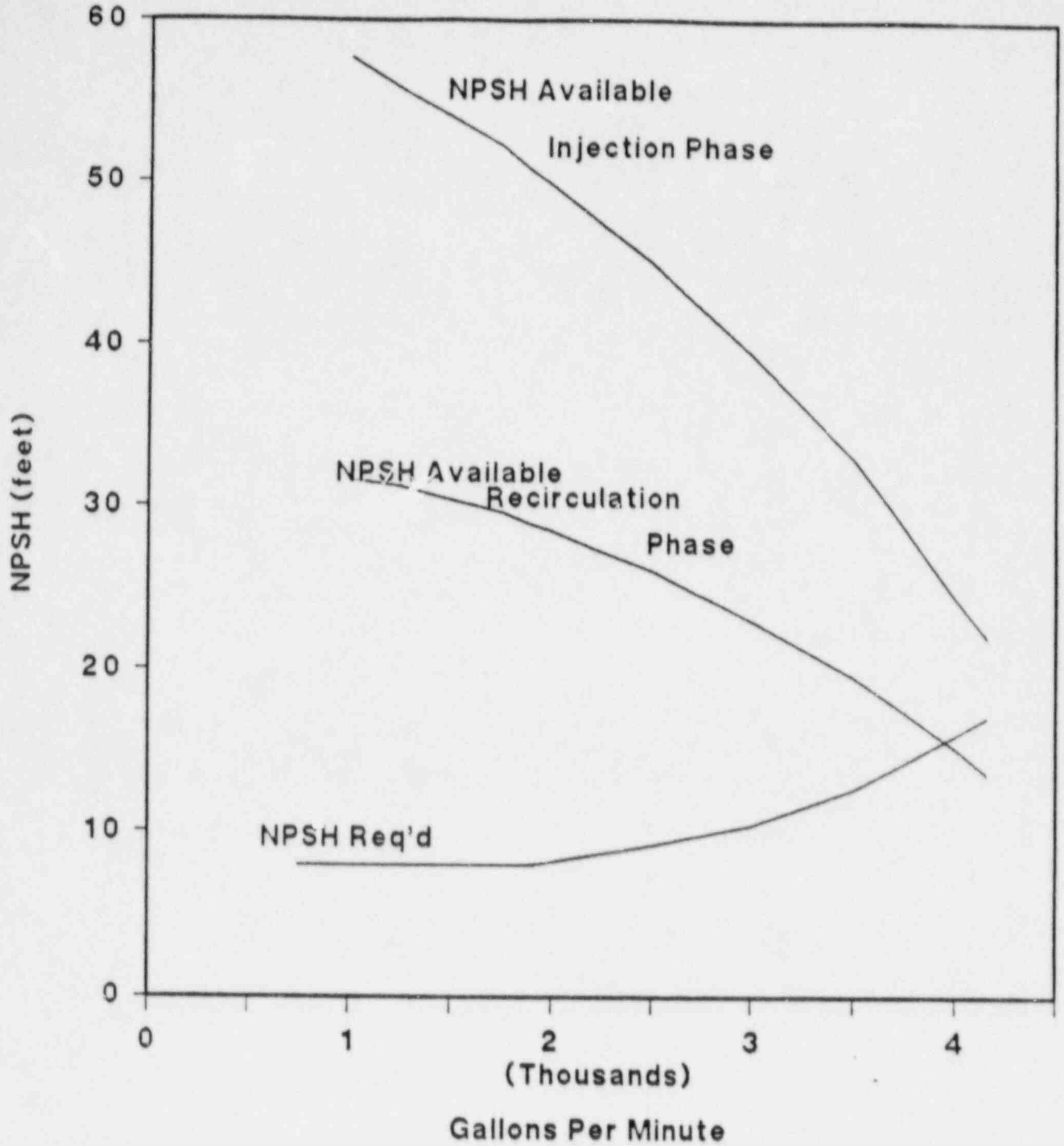
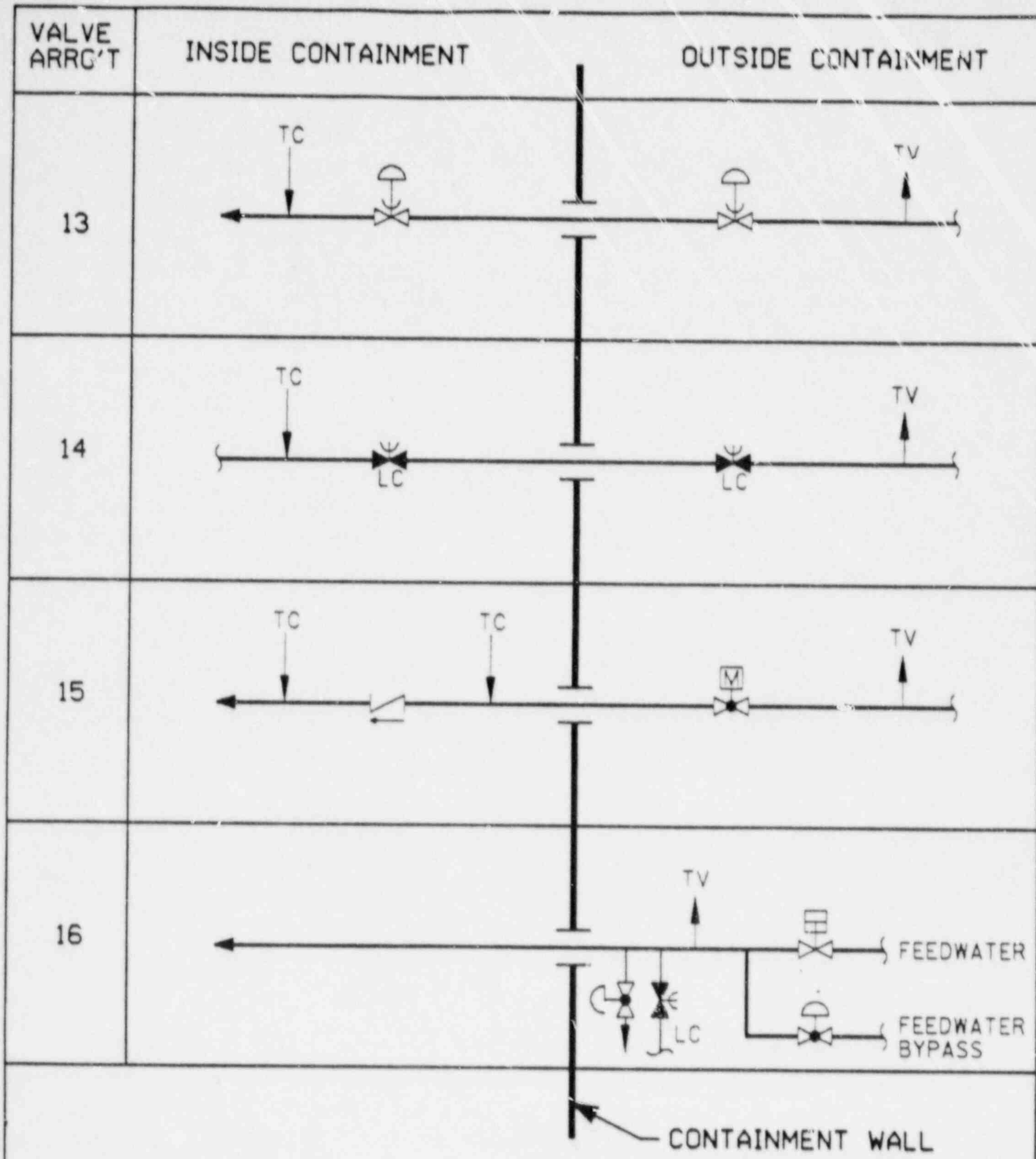


COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

CONTAINMENT SPRAY PUMPS
AVAILABLE NPSH

FIGURE 6.2.2-2



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COMANCHE PEAK S E S
FINAL SAFETY ANALYSIS REPORT
UNITS 1 AND 2

CONTAINMENT ISOLATION VALVING

FIGURE 6.2.4-1 SH. 4 OF 10

FSAR87005

Each residual heat removal heat exchanger bypass line has an air operated butterfly valve which is normally closed and is designed to fail closed. These valves are used during normal cooldown to avoid thermal shock to the residual heat exchanger.

6.3.2.2.10 Net Positive Suction Head

Available and required net positive suction head for ECCS pumps are shown in Table 6.3-1. The safety intent of Regulatory Guide 1.1 is met by the design of the ECCS such that adequate net positive suction head is provided to system pumps. In addition to considering the static head and suction line pressure drop, the calculation of available net positive suction head in the recirculation mode assumes that the vapor pressure of the liquid in the sump is equal to the Containment ambient pressure. This assures that the actual available net positive suction head is always greater than the calculated net positive suction head.

6.3.2.2.11 Accumulator Motor Operated Valve Controls

As part of the plant shutdown administrative procedures, the operator is required to close these valves. This prevents a loss of accumulator water inventory to the RCS and is done shortly after the RCS has been depressurized below the safety injection unblock setpoint. The redundant pressure and level alarms on each accumulator would remind the operator to close these valves, if any were inadvertently left open. Control power is disconnected to these valves after closure by locking open the valve breakers. | 68

During plant startup, the operator is instructed via procedures to energize and open these valves when the RCS pressure reaches the safety injection setpoint. Monitor lights in conjunction with an audible alarm will alert the operator should any of these valves be left inadvertently closed once the RCS pressure increases beyond the safety injection unblock setpoint.

TABLE 6.3-3

(SHEET 1 of 4)

MOTOR OPERATED ISOLATION VALVES IN THE
EMERGENCY CORE COOLING SYSTEM

Location	Valve Identification	Interlocks	Automatic Features	Position Indication	Alarms	
Accumulator isolation valves	8808 A,B,C,D	Cannot be closed with "S" signal or when manually locked out.	Opens on "S" signal, or RCS pressure greater than unblock.	MCB	Yes-out of position	; 31 ; 31 ; 31
Safety injection pump suction from RWST	[illegible]06	Power lockout	None	MCB	Yes-out of position	; 31
	[illegible] A&B	None	None	MCB	Yes-out of position	; 31
RHR suction from RWST	[illegible] A&B	Cannot be opened unless sump valve closed	None	MCB	Yes-out of position	
RHR discharge to safety injection/charging pump suction	[illegible] A&B	Cannot be opened unless safety injection pump miniflow isolated and RHR suction from RCS isolated	None	MCB	Yes-out of position	; 31 ; 31 ; 31
Safety injection hot leg injection	[illegible] A&B	Power lockout	None	MCB	Yes-out of position	; 31 ; 31
RHR hot leg injection	[illegible]	Power lockout	None	MCB	Yes-out of position	; 31
Containment sump isolation valve	[illegible]11 A&B	Cannot be opened in normal operation unless RHR suction valves from RWST and one of two RCS suction series valves are closed.	Opens on RWST lo-lo with "S" signal	MCB	Yes-out of position	; 31 ; 31 ; 31 ; 31 ; 31

MCB - Main Control Board

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TABLE 6.3-3

(SHEET 2)

MOTOR OPERATED ISOLATION VALVES IN THE EMERGENCY CORE COOLING SYSTEM

Location	Valve Identification	Interlocks	Automatic Features	Position Indication	Alarms	
CVCS suction from RWST	LCV-112 D&E	None	Opens on "S" signal, VCT Lo-Lo level or source range flux doubled	MCB	Yes-out of position	; 68 ; 68 ; 68
CVCS normal suction	LCV-112 B&C	None	With CVCS suction from RWST open, closes on "S" signal, VCT Lo-Lo level or source range flux doubled	MCB	Yes-out of position	; 68 ; 68 ; 68 ; 68
Safety injection pump to cold leg	8835	Power lockout	None	MCB	Yes-out of position	; 31 ; 31
CVCS normal discharge	8105 8106	None	Closes on "S" signal	MCB	Yes-out of position	; 68 ; 68
High head safety injection isolation valves	8801 A&B	None	Opens on "S" signal	MCB	Yes-out of position	; 31 ; 31 ; 31
Charging and safety injection pump header from RHR	8807 A&B 8924	None None	None None	MCB MCB	Yes-out of position Yes-out of position	
RHR to RCS cold legs	8809 A&B	Power lockout	None	MCB	Yes-out of position	; 31

MCB - Main Control Board

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TABLE 6.3-3

(SHEET 3)

MOTOR OPERATED ISOLATION VALVES IN THE EMERGENCY CORE COOLING SYSTEM

Location	Valve Identification	Interlocks	Automatic Features	Position Indication	Alarms	
Safety injection pump miniflow	8811	Power lockout. Cannot be opened unless RHR discharge to SI/CHG pump valves closed.	None	MCB	Yes-out of position	: 31 : 31 : 31
	8814 A&B	Cannot be opened unless RHR discharge to SI/CHG pump valves closed.	None	MCB	Yes-out of position	: 31 : 31 : 31
RHR cross connect	8716 A&B	None	None	MCB	Yes-out of position	
Safety injection pump cross connect	8821 A&B	None	None	MCB	Yes-out of position	
Charging pump miniflow isolation valves	8110 8111	None	Closes on "S" signal	MCB	Yes-out of position	: 68 : 68 : 68
RHR hot leg suction isolation valves	8701 A,B 8702 A,B	Cannot be opened unless there is no RCS High-1 pressure, and Containment sump isolation valve, RHR suction from RWST valve and RHR discharge to safety injection/charging pump suction valves are closed.	Closes on RCS High 2 pressure	MCB	Yes-out of position	: 68 : 68 : 68 : 68 : 68 : 68 : 68

MCB - Main Control Board

TABLE 6.3-3

(SHEET 4)

MOTOR OPERATED ISOLATION VALVES IN THE

EMERGENCY CORE COOLING SYSTEM

Location	Valve Identification	Interlocks	Automatic Features	Position Indication	Alarms	
Charging pump miniflow isolation bypass valves	8511 A,B	Cannot be opened manually from MCB unless the CVCS suction from the VCT valve is closed and the RHR discharge to safety injection/charging pump suction valves are closed.	Opens on "S" signal	MCB	Yes-out of position	68 68 68 68 68 68
Charging pump relief isolation valve	8512 A,B	Cannot be opened unless the RHR discharge to safety injection/ charging pump suction valves are closed.	None	MCB	Yes-out of position	68 68 68 68
RHR pump recirculation	FCV 610, 611	None	Opens when flow from the RHR pump is low. Closes when flow from the RHR pump is High 10 sec after the pump has started.	MCB	None	68 68 68 68

Power lockout - Control board control power cut-off switch to prevent spurious movement. Valve remains "as is" when control power is cut-off.

MCB - Main Control Board

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68 Airborne radioiodine is limited to levels compatible with the dose
criteria given in Subsection 6.4.1.1, based on the radioiodine
releases given in Table 6.5-6, and a Containment leak rate of 0.1-
percent for the first 24 hr following an accident and one-half this
value for the balance of the accident. Refer to Subsection 15.6.5.4
for an analysis of the inhalation dose to the Control Room operators.
In the event of a toxic gas release, the Control Room envelope is
completely isolated from the outside environment, and the Control Room
68 HVAC system enters a complete recirculation mode of operation. For
CPSES, the probability of simultaneous occurrence of a toxic gas
release and radiological release caused by a loss of coolant accident
(LOCA) is assumed to be extremely low. Therefore, the event of
concurrent releases is not considered in the design basis.

Airborne radioactive material in the Control Room atmosphere is controlled after an accident by the emergency recirculation filtration units and emergency pressurization filtration units. These atmosphere cleanup units are used in the event of a release of airborne radioactive material. The Control Room Ventilation System is described in detail in Section 9.4. The limitations of the Control Room environment following a LOCA are listed in Table 6.4-3.

6.4.1.3 Respiratory, Eye, and Skin Protection for Emergencies

Portable self-contained breathing apparatus and protective clothing are provided in the Control Room envelope for use by the plant personnel required to leave this controlled zone during the emergency recirculation mode of operation. There will be an adequate supply of air to sustain the five-man emergency team for a six-hr period. At least one portable self-contained breathing apparatus will be provided for each member of the emergency team. Replenishment capability for the breathing apparatus is also located offsite.

Further air supplies will be replenished as needed by the emergency organization.

6.4.1.4 Habitability System Operation During Emergencies

A detailed description of the Control Room Air-Conditioning System emergency modes of operation is presented in Section 9.4.

6.4.1.5 Emergency Monitors and Control Equipment

Radiation monitors used to switch the Control Room Air-Conditioning 68
System into the emergency recirculation mode are located in the
Control Room outside-air intakes. The outside-air intake monitors,
located at opposite sides of the Control Building, are used to sample 46
makeup and pressurization air flows introduced into the Control Room
envelope. Chlorine gas monitors are also located at the outside-air
intakes to switch the air-conditioning system to the isolation mode in
the event of a toxic gas release. See Subsection 2.2.3. In this
mode, the affected outside-air intake is isolated and its associated
Control Room emergency pressurization unit is automatically stopped.
During a postulated chlorine gas release, the Control Room operates at
ambient pressure.

6.4.1.6 Fire Protection Criteria

The Fire Protection System is designed to safeguard equipment and
personnel. Combustible materials are excluded as far as practical
from the Control Room to lessen the possibility of a fire. The fire
stops serve a dual function. Fire stops are incorporated on all
cables entering the Control Room to prevent the entry of a fire
originating outside the Control Room. They also form a leak boundary 46
which limits exfiltration of air from the Control Room envelope.
Because any fire in the control panels would be very limited, due to
the amount of combustible materials present, Control Room evacuation
is not considered a necessity; however, remote shutdown capability is
available as described in Section 7.4. Codes and guides used in the
design of the Fire Protection System are given in Subsection 9.5.1.

6.4.2 SYSTEM DESIGN

6.4.2.1 Definition of Control Room Envelope

68 The Control Room pressurized envelope consists of the following areas where continuous or frequent operator occupancy may be required during emergency operation:

	Space	Elevation
68	East Control Room	830'-0"
68	West Control Room	830'-0"
68	Console and Control Room Unit 1	830'-0"
68	Console and Control Room Unit 2	830'-0"
68	Instrument Room Unit 1	830'-0"
68	Instrument Room Unit 2	830'-0"
68	Computer Room Unit 1	830'-0"
68	Computer Room Unit 2	830'-0"
68	File Room	830'-0"
68	Production Supervisor's Office	830'-0"
68	Corridor	830'-0"
68	Toilet	830'-0"
68	Locker Room	830'-0"
68	Kitchen and Janitor Closet	830'-0"
68	Charts and Supplies Storage Room	830'-0"
68	TSC* Observation Area, Office and Corridor	840'-6"
68	Offices (2)	840'-6"
68	Electrical Equipment Corridors (2)	840'-6"

68 * Technical Support Center

68 The Control Room Air Conditioning System (CRACS) mechanical equipment rooms, Trains A and B, located in the Control Building above the Control Room complex at elevation 854 ft 4 in., are pressurized and may require infrequent access by a Control Room operator during an emergency condition. The components located in the CRACS mechanical equipment rooms are described in detail in Section 9.4.

6.4.2.2 Ventilation System Design

The Control Room Air-Conditioning and Ventilation System is a recirculation system during post-LOCA operation. The system is designed to control the level of airborne contamination in the Control Room atmosphere and to control the temperature and humidity for personnel safety and comfort. The flow diagrams of the system are 68
shown on Figure 9.4-1. These diagrams include equipment, dampers, instrumentation, and flow paths for normal and emergency operation. Redundant atmosphere cleanup units (emergency filtration units) are 68
used to remove particulate matter and other contaminants from the Control Room air. The design of emergency filtration units is in accordance with NRC Regulatory Guide 1.52 [3]. Each filtering unit consists of a particulate, HEPA, iodine adsorber, and HEPA filters, and a booster fan to draw the air through the unit. See Section 9.4 for design parameters and capacities of the filters and related equipment. See Figure 6.4-1 for filtration unit drawing.

The system operation is discussed in Section 9.4. The performance objectives of the Control Room Air Conditioning and Ventilation System and the associated design basis necessary to ensure habitability during and after a LOCA are given in Table 6.4-3.

Provisions for monitoring intake air and filtering Control Room recirculated air are provided by the air-conditioning system. Upon actuation of the system to isolation mode of operation as outlined in Section 9.4, the exhaust dampers and the contaminated outside-air intake close; the air-conditioning system switches to complete recirculation. Closure time is in accordance with NRC Reg. Guide 1.95. [5] 46

Q312.8

Redundant emergency pressurization units are used to pressurize the Control Room envelope during emergency recirculation except in the event of a high chlorine release signal or smoke detection signal. 29

6.4.4.2 Toxic Gas Protection

A hazards analysis for each toxic material was performed as
recommended in NRC Regulatory Guide 1.78 [4] and is presented in
Section 2.2. The habitability of the Control Room envelope was
evaluated to determine if a site-related accident involving a release
of hazardous chemicals exceeds the toxicity limits as specified in NRC
Regulatory Guide 1.78. Based on the analysis, monitors are provided in 46
the Control Room outside air intakes (a total of two per intake) of
the Control Room envelope to automatically switch to the isolation
mode of operation. Chlorine sensors are placed at the outside-air
intakes which are approximately 600 feet from the nearest chlorination
storage facility. These automatically switch the Air-Conditioning 68
System to the isolation mode of operation when chlorine levels unsafe
for Control Room personnel (as suggested in NRC Regulatory Guide 1.95
are detected. Chlorine is used as a biocide in the plant Circulating
and Service Water systems. The Chlorination System is described in
Subsection 10.4.5.

A plant specific analysis based on Reference 9 has been performed to 64
demonstrate that the chlorine concentration in the control room would
be well within the protective action limit of 15 ppm based on a
maximum isolation air exchange rate of 800 cfm. The acceptance test
to verify the above isolation air exchange rate and the pressurization
flow rate is that the control room can be maintained at greater than
or equal to 0.125 in. wg with the pressurization flow rate less than
or equal to 800 cfm.

The Computer Rooms for Unit 1 and Unit 2 and the Technical Support 68
Center, which are located inside the Control Room pressure boundary,
employ ten non-seismic non-safety related supplementary cooling
units. These areas do not contain safety related equipment and are
not needed for continuous occupancy. An analysis based on Reference
10 has been performed to demonstrate that refrigerant concentrations
in these areas due to the release of the total refrigerant inventory
associated with these units after a seismic event (DBE) will be within
the limits specified in ANSI/ASHRAE 15-78 [10].

64 9. Jacobus, M. J. NUREG/CR-3786, A Review of Regulatory Requirements Governing Control Room Habitability Systems, August 1984.

68 10. ANSI/ASHRAE 15-78, Safety Code for Mechanical Refrigeration.

TABLE 6.4-1

NOBLE GAS AND HALOGEN INVENTORIES | 68

RELEASED AS THE RESULT OF A MAXIMUM CREDIBLE ACCIDENT | 68

Noble Gases		Halogens		
	Curies Released From Core		Curies Released From Core	68
Nuclide	(100% Core Inventory)	Nuclide	(50% Core Inventory)[a]	68
Kr-83m	1.2×10^7	Br-82	1.5×10^5	68
Kr-85m	2.7×10^7	Br-83	6.0×10^6	68
Kr-85	6.6×10^5	Br-84	1.0×10^7	68
Kr-87	4.9×10^7			68
Kr-88	7.0×10^7	I-130	8.8×10^5	68
Kr-89	8.7×10^7	I-131	4.9×10^7	68
		I-132	7.2×10^7	68
Xe-131m	7.0×10^5	I-133	1.0×10^8	68
Xe-133m	2.9×10^7	I-134	1.1×10^8	68
Xe-133	1.9×10^8	I-135	9.4×10^7	68
Xe-135m	4.0×10^7			68
Xe-135	4.2×10^7		Kilograms Released	68
Xe-138	1.6×10^8	Nuclide	From Core	68
		I-127	1.4	68
		I-129	5.7[b]	68

Note: | 68

a. Half of the released halogens are assumed plated out with the remainder airborne and available for release from containment immediately. | 68

b. Equivalent to 1.007 Ci. | 68

TABLE 6.4-2

This table has been deleted. | 68

TABLE 6.4-3

LIMITATIONS OF CONTROL ROOM ENVIRONMENT

Ambient pressure, in. wg	+0.125*	| 68
Ambient temperature, FDB	75 ±5	
Ambient relative humidity, percent	35 to 50	| 54
Noxious substances	None	
Maximum Radiation doses to operators:		| 68
Whole Body Gamma Dose	5 rem	| 68
Beta Skin Dose	75 rem**	| 68
Thyroid Inhalation Dose	30 rem	| 68

* Except during detection of smoke or chlorine at which time the control room is maintained at atmospheric pressure. | 68

** The 75 rem limit for beta skin dose is allowed due to the use of protective clothing and eye protection. | 68

7. The iodine removal coefficient is calculated in Subsection 6.5.2.3.

6.5.2.2 System Design (for Fission Product Removal)

The CSS has the dual function of removing heat as well as fission-product iodine from the containment atmosphere. Equipment descriptions and principal design parameters for those system components required for the heat removal function are delineated in Section 6.2.2.

The Containment spray chemical additive subsystem is composed of one chemical additive tank, four chemical eductors, connecting piping, and valves. (The Containment spray pumps supply kinetic energy to the motive fluid in the eductors.) The flow diagram of the CSS, including the chemical additive subsystem, is presented on Figure 6.2.2-1.

6.5.2.2.1 Major Components

The mechanical components described in this section are either additions to the CSS or part of the system which has the dual function of heat removal and fission product removal.

1. Chemical Additive Subsystem

   The chemical additive subsystem is comprised of the following components and methods which are used to ensure adequate delivery and mixing of the spray additive:

   a. Chemical Additive Tank

52 The chemical additive tank is sized to hold a sufficient
quantity of 30 weight percent sodium hydroxide solution to
sustain sump water alkalinity within a design pH range of
8.6 to 10.5. The equilibrium sump water is above 8.6.
52 (See Table 6.5-3.) An inert blanket of nitrogen gas is
used to cover the sodium hydroxide solution at a pressure
of 1 to 2 psig.

b. Containment Spray Chemical Eductors

The 30 weight percent sodium hydroxide solution is added to the spray water mainstream by means of liquid jet eductors. A portion of the Containment spray pump discharge is recirculated through the eductor, where the sodium hydroxide is drawn from the chemical additive tank and discharged into the pump suction. The eductors are 52
designed to limit the sodium hydroxide addition to the extent that the spray solution during the injection phase has a pH of approximately 9.5 and the final sump solution has a pH between 8.6 and 10.5. Furthermore, the eductors and piping are designed to have a pressure drop compatible with the Containment spray pump suction line in order not to affect the pump NPSH.

2. Source of Water Supply

The sources of water supply for the Containment spray are as follows:

a. RWST during the injection phase

b. The Containment recirculation sumps during the recirculation phase

Additional information on the modes of operation is given in Subsection 6.5.2.2.2.

3. Containment Spray Headers

The CSS, including the spray headers, is divided into two redundant trains. Each train has headers and nozzles in four regions (A, B, C, and D) of the Containment. For information on the volumes covered by the Containment spray, see Subsection 6.5.2.2.3.

Region A consists of four ring headers per train located within
2 ft of the Containment dome and a minimum of 115 ft 9 in. above
46 the operating floor. Each train in region A contains 274
nozzles for Unit 2 and a minimum 272 nozzles for Unit 1.

Ring 4 is located at a radius of 61 ft 8 in. from the centerline
of the Containment with the upper ring (train A) elevation of
1022 ft 10 in. and a lower ring (train B) elevation of 1021 ft 6
46 in. This ring has 120 (119 for Unit 1) equally spaced nozzles
per header, (59 for Unit 1) of which face vertically downward.
Of the remaining 60 nozzles, 20 point upward and toward the
center of the Containment at a 45 degree angle, with the
remaining 40 pointing horizontally toward the center of the
Containment.

Ring 3 is located at a radius of 51 ft 4 in. from the
Containment centerline with a upper ring elevation at 1041 ft 3
in. and a lower ring elevation of 1039 ft 11 in. This ring
consists of 60 equally spaced nozzles per header, 40 of which
are pointed horizontally toward the center of the Containment.
The remaining 20 nozzles are pointed upward and toward the
center of the Containment at a 45 degree angle. Ring 2 is
located at a radius of 30 ft 3 in. from the Containment
centerline with an upper ring elevation of 1058 ft 8 3/8 in. and
68 a lower ring elevation of 1057 ft 4 3/8 in. This ring contains
48 (47 for Train A of Unit 1) equally spaced nozzles per header,
32 (31 for Train A of Unit 1) of which are pointed downward.
The remaining 16 point upward and toward the center of the
Containment at a 45 degree angle.

66 Ring 1 consists of 46 nozzles and is located at a radius of 13
ft 3 in. from the Containment centerline with an upper ring
elevation of 1065 ft 2 in. and a lower ring elevation of 1063
ft 10 in. This ring contains 46 nozzles per header with 24
pointing downward and away from the center of the containment at
a 45 degree angle, 18 pointing downward, and 4 pointing upward
and toward the center of the Containment at a 45 degree angle.

Regions B, C, and D are large open areas below the operating floor of the Containment where if not for Containment spray, airborne fission products could collect during an accident. For additional information on spray volumes, see Subsection 6.5.2.2.3.

The two spray headers in region B contain 66 spray nozzles per header. The header on Train B has 31 pointing downward, 16 pointing downward 45 degrees from the horizontal, and 19 pointing downward 60 degrees from vertical. The header on Train A has 66 spray nozzles with 28 pointing downward, 15 pointing downward at 45° below the horizontal, 18 pointing downward at 30° from vertical, 4 pointing downward at 20° from vertical and 1 pointing downward at 40° from vertical. These headers are located at an elevation of 900 ft 0 in. and spray an open volume between the secondary shield walls and the Containment liner between elevation 905 ft 9 in. and 860 ft 0 in. 68

The two headers in region C contain 14 nozzles per header, all pointing downward and away from the center of the Containment at a 45 degree angle; they are located at an elevation of 853 ft 6 in. These headers spray an open volume between the secondary shield walls and the Containment liner between elevations 860 ft 0 in. and 832 ft 6 in.

The chemical additive system is isolated automatically when chemical additive tank isolation valves close on chemical additive tank low level or manually upon changeover to recirculation.

In case of LOCA and loss of offsite power, the worst case, the maximum time to close Containment spray pump breakers, including 10 sec for a diesel generator to come up to speed and voltage, is 25 sec after the safety injection signal ("S" signal). The first delivery through the spray nozzles occurs in less than 60
12 sec after the initiation of the safeguard sequence. At the end of the injection phase, the recirculation phase is manually initiated. Duration of the Containment spray phase is determined by the operators and depends on Containment environmental conditions.

The pressure set point at which the CSS is actuated is 20 psig (40 percent of the Containment design pressure). However, the Containment analysis used the value of 60 sec mentioned previously. This delay time includes the following sequence, assuming a double-ended pump suction guillotine break, with maximum safeguards and loss of offsite power:

14 a. The "Hi 3" signal is actuated at 20 psig Containment pressure approximately 6.5 seconds after the break, initiating the opening of the spray pump isolation valves upon receipt of power to the valve motors.

14 b. The Containment spray pumps are started 25 seconds after an "S" signal as mentioned previously. A "P" signal is also used to confirm the pump startup.

68 c. The spray pump isolation valves have a 112 second opening time.

b. Region B

66 This volume is 0.168×10^6 ft^3, of which 0.041×10^6 ft^3 (24.4 percent) are covered by spray. The volume is located between 905 ft 9 in. and 860 ft 0 in. (between the secondary shield wall and the Containment wall).

c. Region C

66 This volume is 0.073×10^6 ft^3, of which 0.007×10^6 ft^3 (9.8 percent) are covered by spray. The volume is located between 860 ft 0 in. and 832 ft 6 in. (between the secondary shield wall and the Containment wall).

d. Region D

66 This volume is 0.125×10^6 ft^3, of which 0.005×10^6 ft^3 (4.3 percent) are covered by spray. The volume is located between 832 ft 6 in. and 808 ft 0 in. (between the secondary shield wall and the Containment wall).

5 e. Region E

66 All sub-volumes included in the Containment total free volume and not calculated to be part of Regions A-D are combined together in Region E as "unsprayed volume". These sub-volumes are separate from each other, but linked to Regions A-D by flow paths which permit varying degrees of convective mixing. The sub-volumes which are less open to convective mixing, such as the reactor cavity, are separated from the Containment liner (and potential leakage paths) by the six inch radial gap which is open to convective mixing. Region E has a total volume of 0.356×10^6 ft^3 and includes the following compartments:

1) Cavity beneath reactor vessel

2) Elevator

3) Rod position indication room (860 ft 0 in.)

4) Pressurizer relief tank compartment

5) Pressurizer compartment

6) In-core instrumentation room (849 ft 0 in.)

7) Seal table room (832 ft 6 in.)

8) Neutron flux detector operating room (808 ft 0 in.)

9) Heat exchanger compartments

10) Miscellaneous passageways | 66

66 The effective spray volume consists of the sprayed volumes in regions A, B, C, and D.

68 The total sprayed volume from all sprayed regions is 1.717×10^6 ft^3. This volume represents 56.7 percent of the Containment free volume.

66 Regions A, B, C and D are linked to each other and to the various sub-regions which comprise region E via numerous air flowpaths (e.g. six inch radial gap between the concrete floors and the inner wall of the containment building , gated doors, floor grating, etc.). These flowpaths are sufficient to allow an assumption that convective mixing occurs between the sprayed and unsprayed volumes at a rate of two turnovers per hour.

TABLE 6.5-6

RADIOACTIVE IODINE ISOTOPES AVAILABLE | 68
FOR RELEASE FOLLOWING A LOCA | 68

Isotope	Activity* (Ci) ($x10^7$)	
I-131	2.45	68
I-132	3.6	68
I-133	5.0	68
I-134	5.5	68
I-135	4.7	68

* Based on NRC Regulatory Guide 1.4 assumptions, this is the activity available for release from containment atmosphere.

components and subsystems comprising the Engineered Safety Features System.

2. Manual actuation requirements

   The ESFAS must have provisions in the Control Room for manually initiating the functions of the Engineered Safety Features System.

7.1.2.1.3 Class 1E 118V AC Uninterruptable Power System | 66

The UPS provides continuous, reliable, regulated single phase alternating current (AC) power to all Reactor Protection System instrumentation and control equipment required for plant safety. Details of this system are provided in Section 7.6. The design bases are given below: | 66

1. The inverter shall have the capacity and regulation required for the AC output for proper operation of the equipment supplied.

2. Redundant loads shall be assigned to different distribution panels which are supplied from different inverters.

3. Auxiliary devices that are required to operate dependent equipment shall be supplied from the same distribution panel to prevent the loss of electric power in one protection set from causing the loss of equipment in another protection set. No single failure shall cause a loss of power supply to more than one distribution panel.

4. Each of the distribution panels shall have access only to its respective inverter supply and a standby power supply.

5. The system shall comply with IEEE Standard 308-1974 Section 5.4. | 68

preserve the redundancy and to ensure that no single credible event will prevent operation of the associated function due to electrical conductor damage. Critical circuits and functions include power, control and analog instrumentation associated with the operation of the RTS or ESFAS. Credible events shall include, but not be limited to, the effects of short circuits, pipe rupture, missiles, fire, etc. and are considered in the basic plant design. Control board details are given in Section 7.7.1.12. In the control board, separation of redundant circuits is maintained as described in Section 7.1.2.2.2.

11 7.1.2.2.1 General

The criteria for the installation and routing of cables is discussed in Section 8.3.1.4.

The physical separation criteria for redundant safety-related system sensors, sensing lines, wireways, cables, and components on racks within Westinghouse NSSS scope meet recommendations contained in Regulatory Guide 1.75 with the following comments:

Q032.55

Q032.12

6 1. The Westinghouse design of the protection system does not rely on over-current devices to prevent malfunctions in one circuit from causing unacceptable influences on the functioning of the protection system. The protection system uses redundant instrumentation channels and actuation trains and incorporates physical and electrical separation to prevent faults in one
27 channel from degrading any other protection channel. For discussion on control grade instrumentation power circuits powered from protection grade power supplies, see paragraph 8.3.1.2.1, item 7 "Compliance with NRC Regulatory Guide 1.75[15] and IEEE 384 [31]".

2. Separation recommendations for redundant instrumentation racks are not the same as those given in Regulatory Position C.16 of Regulatory Guide 1.75, Revision 1, for the control boards because of different functional requirements. Main control boards contain redundant circuits which are required to be physically separated from each other. However, since there are no redundant circuits which share a single compartment of an NSSS protection instrumentation rack, and since these redundant protection instrumentation racks are physically separated from each other, the physical separation requirements specified for the main control board do not apply.

However, redundant, isolated control signal cables leaving the protection racks are brought into close proximity elsewhere in the plant, such as the control board. It could be postulated that electrical faults, or interference, at these locations might be propagated into all redundant racks and degrade protection circuits because of the close proximity of protection and control wiring within each rack. Regulatory Guide 1.75 (Regulatory Position C.4) and IEEE Standard 384-1974 (Section 4.5(3)) provide the option to demonstrate that the absence of physical separation could not significantly reduce the availability of Class 1E circuits.

The Nuclear Instrumentation System and Solid State Protection System were included in the "Westinghouse Protection System Noise Tests" report submitted and accepted by the Nuclear Regulatory Commission (NRC) in support of the Diablo Canyon application (Docket No. 50-275 and 50-323). The tests on the Process Control System - 7300 Series are reported in Reference [2], the conclusions having been accepted by the NRC.

5 The protection and surveillance upgrade package cabinets that house the protection equipment and components including isolators are 7300 series process control equipment similar to that reported in Reference [2].

Provisions will be made to provide assurance that maximum credible fault voltages and conditions which could be postulated in the Comanche Peak Steam Electric Station (CPSES), as a result of BOP cable routing design, will not exceed those used in the tests.

These Westinghouse tests demonstrated that protection systems performance would not be degraded even if subjected to abnormal electrical conditions which far exceed those which can be reasonably postulated.

3. The physical separation criteria for instrument cabinets (including the protection and surveillance upgrade package cabinets) within Westinghouse NSSS scope meet the recommendations contained in Section 5.7 of IEEE Standard 384-1974.

4. The BOP Analog Instrumentation System utilizes redundant safety trains and incorporates physical and electrical separation to prevent faults in one safety train from degrading any other safety train. Each BOP Analog Rack cabinet contains only single safety related train wiring and devices and therefore the separation requirement (between trains) is not applicable in this case. 66

66 The train orientation and wiring techniques used in the 7300
series process control system has been demonstrated by tests to
meet regulatory requirements per Regulatory Guide 1.75 (Ref.
66 WCAP-8892-A, "Westinghouse 7300 Series Process System Noise
Tests," June 1977). In the BOP Analog Instrumentation System,
68 Westinghouse utilized similar wiring techniques. Westinghouse
certificate of qualification, CQ-W9525, establishes an auditable
link between hardware furnished under the BOP Analog
Instrumentation System and the 7300 Series Process Control
Equipment. Additionally it has been demonstrated that close
66 proximity of safety train wiring to non-safety train wiring
within individual BOP Analog Instrumentation cabinets will not
degrade the safety functions of Class 1E circuits should faults
occur in the non-safety train circuits.

66 Cables external to each BOP Analog Instrumentation cabinet are
routed in separate raceways appropriate to their train
orientation.

Q032.17

11 m. Monitoring of system operation

Q032.17

66 The Containment Spray System is provided with Post Accident
68 Monitors as described in Section 7.5. In addition, local indication is provided for spray-additive tank level and pressure, spray-additive flow and Containment Spray pump discharge pressure. The spray additive tank is provided with level indication, low-low level alarm, and a high/low pressure alarm in the Control Room. Additional Control
11 Room alarms are provided for valve isolation tank high level, refueling water storage tank low temperature and spray additive low flow.

Q032.17

11 Each power operated valve is provided with position indicating lights in the Control Room. ESF monitor lights are provided for pumps and valves.

Q032.17

11 n. Sequencing

Q032.17

11 The sequencing of containment spray pumps upon loss of offsite power concurrent with a LOCA is shown in Tables 8.3-1A and 8.3-1B.

Q032.17

11 2. Containment Isolation System

Q032.17

11 Containment isolation is carried out in two phases: Phase A isolation closes all nonessential process lines penetrating the Containment, and Phase B isolation closes remaining process
29 lines, except for those lines required for ESFAS. Containment Ventilation Isolation is initiated by Phase A isolation or a high containment radioactivity signal. The Containment Isolation System is discussed in further detail in Section 6.2.4 and 7.3.1.1.5.

injection initiation functions associated with one actuation train (e.g., train A) shares portions of the automatic initiation circuitry logic of the same logic train; however, a single failure in shared functions does not defeat the protective action of the redundant actuation train (e.g., train B). A single failure in shared functions does not defeat the protective action of the safety function. It is further noted that the sharing of the logic by manual and automatic initiation is consistent with the system level action requirements of the IEEE Standard 279-1971, Section 4.17, and the minimization of complexity.

Manual actuation of main steam line isolation (all valves), containment isolation (Phase A), and containment spray actuation conforms to the same criteria herein described for the manual safety injection manual actuation functions.

56 7.3.2.2.8 Component Control Switches

56 The control switches for ESF final actuators, e.g., pumps, valves and dampers, are the spring-return-to-automatic type. This design feature assures the completion of the protective function once it has been initiated, regardless of the operational status of the final actuator prior to the initiation of the protective function.

56 When operating conditions necessitate, the operator can manually
68 override the automatic operation of individual components. The
manual overrides are accomplished in the following manner:

68 A. Valves and Dampers - The control switch must be held by the operator in the alternate position for the duration of the period that the alternate action is required. The automatic action is restored as soon as the operator releases the control switch.

B. Pumps - For those pumps that receive an auto start signal from the ESF sequencer, the operator can stop the pump by holding the control switch in the stop position. After the sequencer has cycled, the pump will not restart when the operator releases the control switch. This override will be continuously indicated in the control room and both the override and the indication will be automatically removed whenever permissive conditions for the bypass are not met. 68

The alternate operating mode is provided with permissives or operator lockouts that prevents the operator from intervening in the automatic operation of the final actuator when the ESF sequencer is in operation. 56

This manual override meets the intent of IEEE Std 279 requirements for operating bypasses. The channel is not removed from service during the period that the operator is manually operating the component and the operator has status indicating lamps to inform him of the status of the final actuator. 56

suitable procedures and secondary controls as described in Section 7.4.1.3.3. | 32

2. Except for a fire in the alternate shutdown areas (Control Room, Cable Spreading Room, or Control Room HVAC Mechanical Equipment Room), Control Room evacuation is initiated for an undefined cause (for example, control room environment not habitable) and the event which causes Control Room evacuation does not degrade Control Room equipment. | 68

3. Design basis accidents are not assumed to occur simultaneously with control room evacuation. However, loss of offsite power has been considered.

4. For five specific plant fires, shutdown capability is provided as follows: | 57

Fire Location	Loss	Shutdown From	32
Cable Spreading Room (CSR)	Both Trains	Hot Shutdown Panel using Train A, Shutdown Transfer Panel transfer and local controls.	66
Control Room (CR)	Both Trains	Hot Shutdown Panel using Train A, Shutdown Transfer Panel transfer and local controls.	66
Control Room HVAC Mechanical Equipment Room	Both Trains	Hot Shutdown Panel using Train A, Shutdown Transfer Panel transfer and local controls.	66

	Fire Location	Loss	Shutdown From
32			
32	Hot Shutdown Panel (HSP)	Train B	Control Room using Train A
32	Shutdown Transfer Panel (STP)	Train A	Control Room using Train B

57 For any other postulated plant fire, shutdown is accomplished from the Control Room utilizing surviving equipment.

5. Except for a fire in the alternate shutdown areas, loss of safety 66
system redundancy does not occur as a result of the event
requiring control room evacuation and all equipment in the
control room and all automatic controls continue to function.

6. The Hot Shutdown Panel (HSP) and the Shutdown Transfer Panel 52
(STP) including Class 1E equipment mounted on them, are designed
to withstand an SSE with no loss of Class 1E function. The
essential local control stations are also designed to withstand 32
an SSE with no loss of essential functions.

7. The Hot Shutdown Panel is normally unattended and is surrounded 32
by a locked enclosure. Opening the enclosure doors will
initiate an alarm in the Control Room. The Shutdown Transfer 66
Panel is also normally unattended and access to it is restricted
via normally locked doors. Opening the enclosure doors will
initiate an alarm in the Control Room. 32

8. The Hot Shutdown Panel, located in the switchgear area of the 66
Safeguards Building, at elevation 831'-6", the Shutdown Transfer
Panel, located one floor below at elevation 810'-6", and other
local controls are easily accessible to Control Room operators
through controlled access areas.

Q032.36
Q040.23
Q040.64

9. Electrical separation for the Hot Shutdown Panel and the Shutdown 32
Transfer Panel follows the same criteria as corresponding Control
Room equipment. Loss of control or indication for one train for
any reason will not affect its redundant counterpart.

Consistent with the definition of Type A variables in Regulatory Guide 68
1.97 Revision 2, the verification of the actuation of safety systems
has been excluded from the definition of Type A. The variables which
provide this verification are included in the definition of Type D. 56

Variables in Type A are restricted to pre-planned actions for Design Basis Accident Events. Contingency actions and additional variables which might be utilized will be in Types B, C, D, and E.

7.5.1.2.2 Type B

Those variables that provide to the Control Room operating staff 66
information to assess the process of accomplishing or maintaining
critical safety functions, i.e., reactivity control, reactor coolant
system integrity, RCS inventory, reactor core cooling, heat sink, and
containment integrity.

7.5.1.2.3 Type C

Those variables that provide to the Control Room Operating Staff information to monitor (1) the extent to which variables, which indicate the potential for causing a gross breach of a fission product barrier, have exceeded the design basis values and (2) that the in-core fuel cladding, the reactor coolant system pressure boundary or

7.5.2 DESCRIPTION OF VARIABLES

7.5.2.1 Type A Variables

Type A variables are defined in Section 7.5.1.2.1. They are the variables which provide primary information required to permit the Control Room operating staff to:

1. Perform the diagnosis specified in the applicable CPSES ERG's; | 42

2. Take specified pre-planned manually controlled actions for which no automatic control is provided, that are required for safety systems to accomplish their safety function to recover from the Design Basis Accident Event (verification of actuation of safety systems is excluded from Type A variables and is included as Type D); | 68

3. Reach and maintain a safe shutdown condition.

Key Type A variables have been designated Category 1. These are the variables which provide the most direct measure of the information required. The KEY Type A variables are:

1. RCS Wide Range Pressure

2. Wide Range Hot Leg Reactor Coolant Temperature (T(Hot))

3. Wide Range Cold Leg Reactor Coolant Temperature (T(Cold))

4. RCS Subcooling (Saturation Margin) | 66

5. Narrow Range Steam Generator Water Level (NR)

6. Pressurizer Water Level | 66

7.5.2.2 Type B Variables

Type B variables are defined in Section 7.5.1.2.2. They are the variables that provide to the Control Room Operating Staff information to assess the process of accomplishing or maintaining critical safety functions, i.e.:

1. Subcriticality (Reactivity Control) | 66

2. Reactor Coolant System integrity | 66

3. Reactor Coolant System Inventory | 68

4. Reactor Core Cooling

5. Heat Sink | 66

6. Containment integrity | 66

Variables which provide the most direct indication (i.e., Key variable) to assess each of the 6 critical safety functions have been designated Category 1. Preferred backup variables have been designated Category 2. All other backup variables are Category 3. These are listed in Table 7.5-3.

7.5.2.3 Type C Variables

Type C variables are defined in Section 7.5.1.2.3. Basically, they are the variables that provide to the Control Room Operating Staff information to monitor the potential for breach or actual gross breach of:

Table 7.5-5

(Sheet 3)

System/Component	Variable	Variable Function	Type/ Category	
Emergency Core	RWST Level	Key	D2	
Cooling System	Safety Injection Pump	Key	D2	66
(ECCS)	flow	Key	D2	66
	Safety Injection Pump	Key	D2	66
	Status			66
	RHR flow	Key	D2	41
	RHR Pump Status	Key	D2	66
	Centrifugal Charging Pump	Key	D2	66
	injection flow			41
	Centrifugal Charging Pump	Key	D2	68
	Status			
	Containment Water	Key	D2	66
	Level			66
	ECCS valve status	Key	D2	
	SI accumulator	Key	D2	41
	isolation valve			41
	status			41
	SI accumulator tank	Key	D2	64
	pressure			
	SI accumulator tank level	Backup	D3	
	Control Rod Position	Backup	D3	66
Auxiliary	Flow to each S/G	Key	D2	66
Feedwater	Pump Status	Key	D2	66
	Valve status	Key	D2	
	CST level	Key	D2	
	AFW Pump Turbine Main	Key	D2	66
	Steam Header Isolation			66
	Valve Status			66

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Table 7.5-5
(Sheet 5)

System/Component	Variable	Variable Function	Type/ Category	
ESF	CR A/C Units	Key	D2	66
Ventilation	CR Vent Damper Position	Key	D2	66
	CR Emergency Filtration	Key	D2	66
	Fans			66
	CR Emergency	Key	D2	66
	Pressurization Fans			66
	Primary Plant Exhaust	Key	D2	68
	Fan Cooler			68
	Electrical Area Fan Cooler	Key	D2	66
	Diesel Generator Fuel Oil	Key	D2	66
	Day Tank Area Vent Fan			66
	Diesel Generator Area	Key	D2	66
	Vent Fan			66
	SSW Intake Structure	Key	D2	66
	Exhaust Fan			66
	Battery Room Exhaust Fan	Key	D2	66
	RHR Room Fan Cooler	Key	D2	66
	Motor Driven AFW	Key	D2	66
	Pump Room Fan Cooler			66
	CCW Pump Room Fan Cooler	Key	D2	66
	SI Pump Room Fan Cooler	Key	D2	66
	Containment Spray Pump	Key	D2	66
	Room Fan Cooler			66
	Centrifugal Charging Pump	Key	D2	66
	Room Fan Cooler			66
	UPS Ventilation	Key	D2	66
Safety Chilled	Flow	Key	D2	66
Water System				66

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pressure above the safety injection unblock pressure (P-11). The manual block or "maintain closed" position is required when performing periodic check valve leakage test when reactor is at pressure. The maximum permissible time that an accumulator valve can be closed when the reactor is at pressure is specified in the Technical Specifications.

Administrative control is required to ensure that any accumulator valve, which has been closed at pressures above the safety injection unblock pressure, is returned to the "AUTO" position. Verification that the valve automatically returns to its normal full open position would also be required.

During plant shutdown, the accumulator valves are in a closed position. To prevent an inadvertent opening of these valves during that period the accumulator valve breakers should be opened or removed. Administrative control is again required to ensure that these valve breakers are closed during the pre-startup procedures.

These normally open motor operated valves have alarms, indicating a malpositioning (with regard to their Emergency Core Cooling System function during the injection phase). The alarms sound in the control room.

An alarm will sound for either accumulator isolation valve under the following conditions when the RCS pressure is above the "SI unblocking pressure.":

68 | 1. Valve motor operator limit switch indicates valve not open.

2. Valve stem limit switch indicates valve not open. The alarms on this switch will report itself at given intervals.

10. NEMA FU 1-1972, Low-Voltage Cartridge Fuses

11. NEMA PB 1-1971, Panelboards With Revision 1

12. NEMA PB 2-1972, Dead-Front Distribution Switchboards With Revision 1

8.1.4.7 National Fire Protection Association (NFPA)

No. 70-1971, National Electrical Code

8.1.4.8 Underwriters' Laboratories, Inc. (UL)

1. UL-50, Electrical Cabinets and Boxes (1975)

2. UL-67, Electric Panelboards (Revision, 10/75)

3. UL-891, Dead-Front Electrical Switchboards (1975)

8.1.4.9 Illuminating Engineering Society (IES)

IES Lighting Handbook, Application Volume, 1981. | 68

8.1.5 COMPLIANCE WITH NRC REGULATORY GUIDES AND IEEE STANDARDS

The extent to which the recommendations of the NRC regulatory guides and IEEE standards are complied with is described in the following sections:

8.1.5.1 NRC Regulatory Guides

For description of compliance to the Regulatory Guides, see Appendix 1A(B) and 1A(N).

8.1.5.2 IEEE Standards

1. IEEE 338-1971, Trial-Use Criteria for Periodic Testing of Nuclear Power Generating Station Protection Systems

   Periodic testing of protection systems conforms to the requirements of this standard.

2. IEEE 344-1975, Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations

   All Class 1E equipment is seismically qualified in accordance with this standard.

   For details, see Sections 3.10B and 3.10N

3. IEEE 387-1977, Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Generating Stations

   The design criteria and the qualification and testing requirements of the diesel generator units conform to the requirements of this standard.

60 4. IEEE 420-1973, Trial-Use Guide for Class 1E Control Switchboards for Nuclear Power Generating Station.

3 All Class 1E control boards are designed in accordance with IEEE 420-1973 requirements with following clarification:

Lens and Buttons for approximately forty (40) safety system inoperable 3
indication lights and for sixteen (16) non-safety valve control
devices are constructed of "LEXAN" (poly carbonate) and "CELON"
respectively. These indicating lights/buttons are grouped and
enclosed in a metal housing to provide seperation between them and
other equipment on the main control board. The housings are flush
mounted on the front of the main control board. Cable connectors are
provided on the back plate of the housing for external wiring.
Internal fire (unlikely event) in the housing will not significatly
degrade the integrity of the main control board.

Wire splices are used in limited applications on field cables that 44
terminate in certain Class 1E panels, cabinets or racks. The normal
design is to terminate field cables without the use of wire splices.
The wire splices are only used where additional length is required for
the field wire and it was not judged reasonable to pull a new field
cable. The use of such wire splices has been minimized.

The wire splices are butt splices. The crimping technique, device 44
and materials used for the splices are identical to those used for the
terminal lugs in that panel. The wire splices are only allowed on
low power applications such as control cables. The inspection 60
procedure verifies the operability of the spliced circuit by
completion of a continuity check. The wire splices used are
qualified for anticipated service conditions, and the splices are
staggered or separated within the panel so that they are not adjacent
to each other or pressing against one another in the same wire bundle.
Since previously accepted crimping methods and materials are used,
the splices are limited to low power circuits and to field cables that
already terminate in the panel, and the required wire separation and
wire bundle support is maintained, the wire splices are not expected
to significantly alter the heat load in the panel, the probability of
a fire or the operability of any equipment or cables in that panel.

68 5. IEEE 494-1974, Standard Method for Identification of Documents Related to Class 1E Equipment for Nuclear Power Generating Stations.

68 All station design documents and drawings containing Class 1E equipment and systems, in whole or in part, are identified with the term "Nuclear Safety Related" in accordance with this standard or, as an alternative, with the equivalent term "Class 1". Vendor supplied documents and drawings are used in conjunction with station design documents and/or drawings which are properly identified as to their safety classification; therefore, vendor supplied documents and drawings are not necessarily identified as "nuclear safety related" in accordance with IEEE 494-1974.

8.3.1.1.5 Automatic Transfers, Loading and Load Shedding

8.3.1.1.5.1 Non-Class 1E Bus - Automatic Transfer Q040.44 30

When Unit 1 trips, the power source for Bus 1A1 is automatically transferred from main generator 1G to transformer 1ST. Q040.44 66

The logic, and typical electric schematics for Bus 1A1 are shown in Figure 10.2-1 and G&H drawing 2323-E1-0032 (sheets 5 and 7) respectively. Depending on the system conditions, this automatic transfer will be either "fast" or "slow" as described below: Q040.44 12

Energizing of relays 86-1/1G, 86-2/1G, 94-1B/1G and 94-2B/1G results in tripping of the main generator. Upon operation of any of these relays, the normally closed incoming breaker 1A1-1 receives a trip signal. Q040.44 12

An "early 52-b" contact of this breaker will send a signal to automatically close the offsite source incoming breaker 1A1-2 by: Q040.44 12

a. Fast transfer, provided the Q040.44 12

   1) Offsite source voltage is available (this voltage is monitored by undervoltage relays 27/1ST set to drop out at approximately 85% of nominal system voltage of 6900 volts) and, Q040.44 66

   2) Offsite source and the bus 1A1 voltages are not out of phase by more than 40 degrees (as seen by the synchronism check relay 25/1A1). Q040.44 68

   3) This transfer is completed in the minimum time possible (maximum of 10 cycles), however, it is disabled after 15 cycles from initiation to eliminate the possibility of accidental closure in case of any contact malfunction in the close circuit. Q040.44 68

30 A successful fast transfer assures maintenance of voltage on bus 1A1 without interruption.

Q040.44

12 b. Slow transfer, if the requirements of a. are not satisfied within approximately 15 cycles after the closure of the "early 52-b" contact of the normally closed inccming breaker, provided:

Q040.44

12 1) Voltage on the bus (to be transferred) has decreased to approximately 35% of the nominal voltage of 6900 volts,

Q040.44

12 2) Offsite source voltage is available as described in a. 1), and

Q040.44

30 3) All motor feeder breakers on bus 1A1 have tripped as evidenced by closure of their "52-b" contacts. Tripping
12 of these motor feeder breakers is accomplished by bus undervoltage relays set at approximately 35% of the nominal voltage of 6900 volts via timers set at approximately 0.5 seconds.

Q040.44

68 The main generator 1G tripping is delayed by 30 seconds subsequent to a reactor trip (see FSAR Figure 7.2-1 Sheet 16 and Figure 10.2.1). This ensures full coolant flow for 30 seconds after a reactor trip before any bus transfer is made.

Q040.44

66 The automatic transfers of the other non-Class 1E buses (1A2, 1A3, 1A4, 2A1, 2A2, 2A3, and 2A4) are the same as discussed for 1A1 above.

8.3.1.1.5.2 Class 1E Buses Automatic Transfer 30

1. Undervoltage Sensors 30

Q040.5

The sensors that detect loss of preferred and alternate offsite sources are located in 6.9 kV safety related switchgear buses 1EA1 and 1EA2 for Unit 1 and 2EA1 and 2EA2 for Unit 2. The scheme consists of instantaneous undervoltage relays which initiate tripping of the source breakers via time delay pick up relays. 1

Q040.5

The approximate setting for the loss of voltage detection sensors are shown on Figure 8.3-6 and are as follows: 30

Q040.5

Undervoltage relay dropout - 70% of 6900V 30

Q040.5

Time delay pick up relay 0.5 seconds 30

Q040.5

Loss of voltage detector relay setting is selected to preclude operation during momentary low voltage conditions that prevail during large motor starting. The selected setting is lower than the lowest momentary voltage fluctuation and plant load conditions. Settings for the time delay pickup relay is set long enough to preclude their operation during a three phase fault when system voltage could drop below the undervoltage relay setting, for a few cycles, until the fault is removed by opening the appropriate circuit breakers. 30

2. Automatic Transfer 30

When the source breakers for the preferred offsite power source are tripped, the Class 1E buses are transferred to the alternate power source by a slow transfer in a manner similar to that described for the non-Class 1E buses. 33

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66 The alternate power source of Unit 1 safety-related buses 1EA1 and 1EA2 is startup transformer XST1 and the alternate power source of Unit 2 safety-related buses 2EA1 and 2EA2 is startup transformer XST2.

30 In the event that the preferred offsite source is lost, the Class 1E buses would automatically transfer to the alternate source.
66 Operation of the unit could continue for a limited period of time based upon the constraints defined in the plant Technical Specifications.

66

30 Low voltage (after an appropriate time delay) on either the preferred power source or the Class 1E bus shall start the respective emergency diesel generators for that bus. If both the preferred and alternate power sources are not available, the emergency diesel generator will power the bus after it has
65 reached rated voltage and frequency. The bus will be powered by the emergency diesel generator within 10 seconds after the diesel receives a starting signal.

30 8.3.1.1.5.3 Sequencer Loading

30 1. Sequencer

Q040.59

66 For each unit, two independent and redundant safeguards sequencer cabinets are provided for sequential loading of the safeguard buses, one for Train A and one for Train B. Each cabinet houses two sequencers, one for the safety injection mode sequencing and one for Loss of Offsite Power (blackout) only mode sequencing.

Each sequencer is basically a group of output electromechanical relays operated by solid state logic circuits and timers. The separate and independent timer circuits are initiated for each loading step by solid state logic circuits. Reliability of each sequencer is assured by continuous testing from the input diode matrix through the logics and timers to the coils of the output relays. If a fault is detected it is alarmed in the control room. A review of the sequencer circuitry reveals that the sequencer design is relatively direct and simple, not subject to sneak circuits and is highly reliable. 20

2. Loading 30

In the unlikely event of a DBA the following sequence of operation is initiated.

a. The diesel generator sets receive starting signals.

Q040.44

b. All 480V non-Class 1E loads connected to Class 1E buses are tripped or isolated in accordance with Section 8.3.1.2.1.7.a. 68

Q040.44

c. Power source for the Class 1E buses is established as discussed above. The diesel generator breaker will close, if required, after the diesel generator rated voltage and frequency have been established and there is no bus fault. Large loads required during a DBA are started in sequence by the safety injection sequencer. Small loads (less than 20 hp) are generally started in accordance with their respective circuit logic. See Tables 8.3-1 A & B for details. 12

Q040.44

12 If both offsite sources are lost, but no "S" signal is present, the following sequence of operation is initiated:

Q040.44

12 a. The diesel generator sets are started.

Q040.44

12 b. Undervoltage relays shed the required loads from the Class 1E buses.

Q040.44

12 c. Power source for the Class 1E buses is established as discussed above. Diesel generator breaker will close, if required, after the diesel generator rated voltage and frequency have been established and there is no bus fault.
66 Large loads required during a Loss of Offsite Power (blackout) mode are started in sequence by the blackout sequencer. Small loads (less than 20 hp) are generally
12 started in accordance with their respective circuit logic. See Table 8.3-2 for details.

66

41 8.3.1.1.6 Safety-Related Power System Equipment Identification

41 Safety-related power system electrical equipment is uniquely numbered so that identification as safety equipment is evident. The numbering sequence used for Class 1E equipment is different from the one used for non-Class 1E equipment. The equipment numbering system is discussed in Subsection 8.3.1.3. In addition, color-coded nameplates conspicuously identify the major equipment as Class 1E. Plant personnel can determine the train or channel that the equipment is associated with by this color.

Color delineation of the trains and reactor protection instrumentation channels is as follows:

1. Engineered Safety Features (ESF) Systems

Train A	orange
Train B	green

2. Reactor Protection System and ESF Systems at Channel Level

Channel I	red
Channel II	white
Channel III	blue
Channel IV	yellow

These color designations are consistent throughout the plant to allow for immediate identification. Cable and cable tray identifications are discussed in Subsections 8.3.1.3 and 8.3.1.4. These standards do not apply to the identification of control panels. | 41

8.3.1.1.7 System Instrumentation and Control

Remote instrumentation for the 6900-V Class 1E switchgear consists of ammeters in the Control Room and on the hot shutdown panel for the preferred power source and the alternate power source. The remote instrumentation for the standby power source (diesel generator unit) consists of ammeter, voltmeter, frequency meter, wattmeter, and varmeter in the Control Room and an ammeter on the hot shutdown panel. | 66
Various annunciators and bus voltmeters and frequency meters are located in the Control Room. Instrumentation for each 480-V Class 1E switchgear consists of ammeters on the 6900-V switchgear breaker for that switchgear and a bus voltmeter located on the 480-V switchgear. This is shown on Figures 8.3-6 and 8.3-8.

The diesel generator breaker control and synchronizing switches are
located in the Control Room for corrective action by the operator.
66 Selector switches located on the shutdown transfer panel and on the
hot shutdown panel, and diesel generator breaker control switches
located on the hot shutdown panel, are provided for corrective action
by the operator in the unlikely event the Control Room becomes
inaccessible. Control switches for the 6900-V Class 1E incoming
supply breakers, bus tie breakers, and feeder breakers for Class 1E
loads, are located on the main control board and the 6900-V
68 switchgear. Control switches for the Class 1E 480-V switchgear
incoming supply and bus tie circuit breakers are located on the main
control board and the 480-V switchgear. All feeder breakers have
66 control switches at their respective switchgear. For remotely
controlled circuit breakers the local control switches (at the
respective switchgears) are operable only when the circuit breaker is
in the test position. Control required for safe shutdown in the
event of Control Room uninhabitability is discussed in Section
7.4.1.3.

The described control and instrumentation is used in testing the diesel generator and in monitoring 6900-V and 480-V Class 1E switchgear during normal and Loss of Offsite Power conditions.

Control power for Class 1E equipment is provided as follows:

Class 1E Equipment	Class 1E Control Power Source
6900-V breaker and associated protective relaying	125-VDC system
480-V load center breakers and relaying	125-VDC system
480-V motor control center starters	120-VAC derived through control transformers

No single failure can prevent operation of the minimum number of required safety loads and loss of any one group will not prevent
68 the minimum safety functions from being performed. Each Class 1E AC bus has access to two offsite power sources and an onsite standby power source. There are no automatic or manual ties between redundant buses.

Two diesel generators are provided for each unit. Each diesel generator is connected exclusively to its associated 6.9-kV Class 1E bus, which ensures independence in the onsite standby power sources.

Each Class 1E DC bus can be energized either by a battery or by one of two battery chargers (one spare) or combination of battery and battery charger. There are no automatic or manual ties between Class 1E redundant DC load groups. Arrangement of the AC and DC systems is described in Subsections 8.3.1 and 8.3.2, respectively.

Because there are no bus ties between redundant load groups, interlocks are not required.

4. Compliance With NRC Regulatory Guide 1.9 [3]

The rating of the diesel generators is based on the maximum continuous load demand. This rating exceeds the sum of the conservatively rated loads. Motor loads are based on nameplate rating, pump runout conditions, or flow pressure conditions. 6600-V motor efficiency is based on design data. Low-voltage motor efficiency is assumed to be 80 percent.

During preoperational testing, the maximum continuous load demand is verified by tests.

Each diesel generator set is capable of starting and accelerating to rated speed all Class 1E loads in the required sequence.

| Q040.60

Sequencing of large loads at 5-sec intervals ensures that large motors have reached rated speed and that voltage and frequency have stabilized before the succeeding loads are applied. The voltage may dip below 75 percent of nominal voltage when the diesel generator breaker closes and energizes the two 2000/2666 kVA, 6.9 kV/480-V unit substation transformers supplied from each diesel generator. This dip is due to magnetizing inrush current which exists for two to three cycles. The diesel generators are designed to recover to 80 percent of nominal voltage within 10 cycles for this transient. The effect on the first load group would, therefore, be a maximum possible delay of 12 to 13 cycle after closure of the diesel generator breaker. However, the objective of first load group and subsequent load groups is not
affected. During recovery from transients caused by step load | 36
increases or resulting from the disconnection of the largest |
single load, the speed of the diesel generator set should not |
exceed the nominal speed plus 75 percent of the difference |
between nominal speed and the overspeed trip setpoint or 115 |
percent of nominal, whichever is lower. The voltage is restored | 66
to within 10 percent of nominal; and the frequency is restored to |
within two percent of nominal in less than 40 percent of each |
load sequence time interval. The diesel generator supplier has |
successfully performed these tests in his facility on one CPSES | 7
diesel generator set. |

The prototype qualification test program of | 7

| Q040.60

a) Start and load capability at full load, and | 8

| Q040.60

b) 300 valid start and load tests | 7

on the diesel generator are discussed in Section 8.3.1.1.11. | 7

5. Compliance With Regulatory Guide 1.32 [7]

66 The offsite power system includes the preferred design stated in NRC Regulatory Guide 1.32: namely, two immediate access circuits from the transmission network are available to the emergency (Class 1E) bus systems.

Each battery charger is sized to handle the combined steady-state loads while recharging the battery from the design minimum charge state to the fully charged state under all modes of plant operation.

6. Compliance With NRC Regulatory Guide 1.63 [12]

The electric penetration assembly design complies with the intent of NRC Regulatory Guide 1.63.

Q040.69

In reference to Regulatory Position C.1 of NRC Regulatory Guide 1.63, the electric penetration assembly design is capable of withstanding, without loss of mechanical integrity, the maximum current versus time conditions permitted by backup protective
68 devices. The adequacy of penetration protective devices to protect the penetrations is established by detailed calculations which demonstrate that the fault current-versus time conditions for which the penetrations are designed and qualified will not be exceeded.

Q040.69

8 The electrical distribution system design incorporates backup
68 protective devices for all power circuits. Control circuits have also been provided with backup protective devices.

Fuses or fusible links within the penetration assembly are not incorporated in the design because of the physical limitations of the standard penetration designs available.

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y. Rod Position Indication Cabinets - | 68

Power circuits from 120V distribution panels. Breakers | 68
are used as primary and backup protection devices. |

| 66

| 68

7. Compliance With NRC Regulatory Guide 1.75 [15] and IEEE 384 [31]

The CPSES design complies with the intent of NRC Regulatory Guide | 60
1.75 and IEEE 384 (Refer to Appendix 1A(B)). Physical |
separation of redundant safety-related equipment and wiring is
achieved by location in separate rooms or by providing barriers.
Isolation devices are provided to preclude interaction between
Class 1E and associated circuits and non-Class 1E circuits, as
described in the following paragraphs.

Electrical isolation methods are used as required in power, | 68
control and instrumentation circuits to maintain the independence |
of redundant circuits and equipment such that protective |
functions required during and following any design basis event is |
accomplished. Different types of isolation devices are used for |
power, control and instrumentation circuits. |

Isolation devices meet the criteria and performance requirements | 68
specified in IEEE 279-1971, Revision 1, and are qualified in |
accordance with IEEE 323-1974 and 344-1975. |

Associated circuits shall comply with one of the following: | 68

(1) They shall be uniquely identified as such and shall | 68
remain with, or be separated the same as, those Class |
1E circuits with which they are associated. |

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68 (2) They shall be in accordance with (1) above from the Class 1E equipment to and including an isolation device. Beyond the isolation device a circuit is not subject to the requirements of IEEE Std. 384-1974, provided it does not again become associated with a Class 1E system.

68 (3) They shall be analyzed or tested to demonstrate that Class 1E circuits are not degraded below an acceptable level.

68 The following paragraphs describe the various conditions:

68 a. Power Circuits

68 The following types of devices are used in the CPSES design for isolation of power circuits:

68 1) Circuit breaker tripped by a safety injection signal.

68 2) Starter contactor opened by a safety injection signal.

68 3) Two circuit breakers, two fuses or a breaker and a fuse in series both coordinated with an upstream circuit breaker and periodically tested.

Q040.62
68 All non-Class 1E loads connected to Class 1E buses are tripped on safety injection signal or justification is provided. These loads are identified in Table 8.3-11 and in the figures as follows:

Q040.62
6

Figure No.	Figure Note No.
8.3-6	8
8.3-8 thru 12	2 and 4

Q040.62
6
Q040.62
6

b. Control Circuits

The isolation devices mentioned below are classified as 68
Class 1E for control circuits. Separation between the Class 1E and non-Class 1E wiring is maintained within the cabinets in which these devices are mounted. Maximum wiring separation at the isolation device is limited by the physical design of the device.

1) Auxiliary Relays provide isolation of control circuits 68
between Class 1E and non-Class 1E circuits for both coil to contact and contact to coil isolation.

2) Phototransistor coupled pairs (light emitting diode 68
and transistor) provide isolation of Class 1E contacts to non-Class 1E monitoring devices.

3) A SIAS signal provides isolation between Class 1E and 68
non-Class 1E control circuits by relay actuation, which opens the circuit.

Safety-related equipment, exposed raceways, and cables are identified by distinct color markers so that the plant personnel can distinguish, without resorting to any reference material, between the various redundant Class 1E systems and between redundant Class 1E systems and non-Class 1E circuits.

The physical separation and identification of circuits are described in detail in Subsections 8.3.1.4 and 8.3.1.3, respectively.

c. Instrumentation Circuits

Instrumentation power for Unit 1 NSSS is provided from 46
distribution panels 1PC1, 1PC2, 1PC3 and 1PC4. See Figure 8.3-15. These panels power safety loads as well as non-safety 27
loads.

27 Each non-safety circuit powered from these panels will have a non-safety circuit breaker or fuse connected in series with the panel circuit breaker.

27 Protection channel wiring, safety-train wiring and non-safety train wiring within panels 1PC1, 1PC2, 1PC3 and 1PC4 will be in different wire bundles. These bundles will be separated to the maximum extent practicable.

46 The same criteria described above is also applicable to Unit 2 instrumentation circuits and their respective distribution panels.

66 d. Lighting System

68 The non-Class 1E security lighting circuits are isolated from their Class 1E power source with two separate Class 1E breakers connected in series. These breakers are coordinated with their supply breakers and will be tested periodically to ensure that coordination is maintained.

66 The non-Class 1E AC essential lighting circuits are isolated from Class 1E power sources with two separate Class 1E breakers (i.e., main breaker and feeder breaker within the Class 1E lighting distribution panel) connected in series. These breakers are coordinated with their supply breaker and will be tested periodically to ensure that coordination is maintained.

66 The non-Class 1E AC essential lighting circuits use interconnecting cable (i.e., from the lighting distribution panel feeder breaker to the lighting load) routed in conduit. The routing of the circuits in conduit ensures the physical and electrical independence from Class 1E circuits beyond the second isolation breaker.

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The basis, criteria, and analysis of potential effects of radiation on all Class 1E equipment from accident conditions, plus normal condition for long-term operation, are described in Section 3.11B.

Class 1E equipment is designed, fabricated, and qualified in 66
accordance with the requirements of IEEE 323 [22] and applicable IEEE standards for particular equipment (e.g., IEEE 382 [29] for valve motor operators, IEEE 383 [30] for cables, and IEEE 317 [21] and ASME Boiler and Pressure Vessel Code [45] for electric penetrations).

8.3.1.3 Physical Identification of Class 1E Power Systems Equipment 41

The identification method by which onsite power system equipment can be distinguished as redundant Class 1E systems, associated Class 1E circuits, and non-Class 1E systems is described below:

1. Equipment Tag No.

   Electrical equipment has its own tagging scheme developed by 68
   equipment type. Many equipment types follow the tagging scheme for mechanical equipment with a modification of the eighth and ninth character. For equipment using a modified mechanical equipment tagging scheme, the eighth character is generally either "E" or "N". "E" designates equipment which is Class 1E and "N" designates equipment which is non safety-related. The ninth character typically indicates bus voltage.

   Motors supplied with pumps are identified by the same tag numbers 68
   as the pumps, usually with the suffix "M" appended.

2. Cables and Raceway Tag Nos.

   All Class 1E system cables and the seismic Category I raceway system are identified by nine alphanumeric character tag numbers as follows:

a. Raceway System

The fourth character of a cable tray identification number identifies whether or not the given raceway contains safety-related cables. The fourth character is obtained from the first letter of the applicable train or reactor protection channel color code as described below. For non-safety-related raceways, the fourth character is K.

27 | b. Cables

27 | The second character in the cable number identifies the cable color code. In addition, the first character of each cable number indicates whether the cable is safety train or channel oriented (E), associated train (A), or non-safety related (N).

41 | 3. Color Coding (Identification) System for Equipment, Raceways, Conduits, and Cables (excluding control panels)

In addition to the tag numbers, Class 1E equipment, raceways, and cables in raceways are identified by the following color coding system:

a. ESF System

68 |

Train A	Orange	(O with a slash)
Train B	Green	(G)

b. Reactor Protection System and ESF Systems at Channel Level

Channel I	Red	(R)
Channel II	White	(W)
Channel III	Blue	(B)
Channel IV	Yellow	(Y)

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c. Associated Circuits

Associated Train A (Train AA)	Orange with white stripes (O with a slash)	68
Associated Train B (Train BB)	Green with white stripes (G)	68

d. Non-Class 1E Circuits

Train C	Black (K)

e. Non-Class 1E Cable Raceways and Equipment	Natural raceway color or natural equipment color

4. Method of Equipment, Raceway System, and Cable Color Coding (Excluding Control Panels) | 41

a. Equipment

The preceding color coding is applied to the Class 1E equipment on an area which is readily visible (e.g., close to the nameplate or identification tag number).

b. Raceways

Exposed raceways containing Class 1E cables are marked by the color codes described previously in a distinct permanent manner at intervals not to exceed 15 ft and at points of entry to and exit from enclosed areas. These raceways are marked prior to the installation of their cables.

c. Cables

In general, all Class 1E cables and associated cables are jacket color-coded throughout their entire length. Cable | 45
jackets that require field color coding prior to

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45 installation, will be so worked at intervals not to exceed five feet. Cable jackets that require field color coding after installation (due to reclassification of cables from associated Class 1E to Class 1E, or vice versa, after their initial pull) will be field color coded as follows:

45 i) Where entering and exiting equipment, raceway and inside junction/pull boxes.

45 ii) All exposed portions of the cables will be worked at intervals not to exceed five feet.

45 iii) Portions of installed cables in conduit or trays will not be field color coded.

d. Non-Class 1E Equipment, Raceways, and Cables

Non-Class 1E equipment, raceways, and cables in raceways are not marked by color code, but are left in their natural
17 color. In general all non-Class 1E cables have a black outer jacket. However, non-Class 1E cables whose natural color is not black will be field color coded black at intervals not to exceed five feet.

8.3.1.4 Independence of Redundant Systems

The criteria which have been used to establish the minimum requirements for preserving the independence of redundant Class 1E systems are stated in IEEE 308 [20] and 384 [31] and NRC Regulatory
68 Guides 1.6 [2] and 1.75 [15]. Class 1E equipment and circuits are clearly identified on documents and drawings in accordance with IEEE 494-1974 [36] as discussed in Section 8.1.5.2 (Item 5). The electrical cable system for Class 1E systems is described in the following subsections.

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64 a. Area Radiation Monitoring detectors utilize Geiger-Muller Tubes and Ionization Chambers. These devices require very low currents but at high voltage levels. Since the power supplies need to supply 1 milliamp or less, they are designed to provide 2 watts under normal operation conditions and have a design limit of 5 watts. The power supply is not capable of igniting a shorted detector cable because of its "fold over regulation" characteristic which turns off even this low current flow. Therefore any damage will be limited to internal damage and will not be propagated to nearby Class 1E circuits.

64 b. The Public Address System speaker wire is a similar case with both low current and low voltage requirements. (The National Electric Code recognizes such low power circuits under article 725's Class II wiring). The speaker amplifier is designed to provide a maximum of 12 watts at 14 VDC, which is not capable of igniting the circuit under a fault condition. Therefore any damage in speaker wire shall be limited to internal damage and will not be propagated to nearby Class 1E circuits.

66 Fiber optic cables used in non-Class 1E monitoring circuits carry no electrical energy by themselves and therefore are not required to maintain physical separation from Class 1E circuits.

The raceways of one train are separated from those of the other train by locating them in separate structures or on opposite sides of large rooms or spaces. Where this is not possible, separation is maintained as described below or by providing barriers. The Class 1E cables are routed such that any single failure in one train system does not cause a failure in another train system. The separation of associated circuit cables is maintained on a train basis in the same manner and degree as Class 1E circuit cables, with which they are associated.

68

Where cables are exposed to such potential hazards as pipe whip, flammable material, and missiles, separation requirements are evaluated on a case-by-case basis to ensure an acceptable level of redundant circuit independence.

5. Minimum Separation Requirement

In plant areas which are free from potential hazards such as missiles, external fires, and pipe whip, the minimum separation between redundant cable trays is three ft between trays separated
horizontally and five ft between trays separated vertically. The 60
minimum separation between safety-related conduit and redundant cable tray in these areas is three ft in both horizontal and vertical directions whenever the conduit elevation is above that of the tray side rails.

Q040.12

In the cable spreading area and the Control Room, the minimum 1
separation between redundant cable trays is one ft between trays separated horizontally and three ft between trays separated
vertically. The minimum separation between safety-related 60
conduit and redundant cable tray in this area is one ft between these raceways separated horizontally and two ft between these raceways separated vertically whenever the conduit elevation is above that of the tray side rails.

In all plant areas free of potential hazards as described above, 60
the minimum separation required in any direction between redundant tray and conduit is one inch whenever the conduit is non safety-related or its elevation is not above the tray side rails.

64 All of the above conduit to cable tray separation distances are based on testing and analysis.

41 All Nuclear Instrumentation System (NIS) cables are routed in conduit according to their channel assignment. A minimum separation of 6 feet is maintained between NIS conduits and
42 raceway containing 6.9 kV circuits. Also, a minimum separation of 2 feet is maintained from NIS conduits running parallel to raceways containing electrical noise sources such as low voltage power and rod control cables.

The minimum separation distance between redundant Class 1E equipment and circuits internal to the main control boards is six inches. In this case, the wire and cables are flame-retardant
66 with self-extinguishing and nonpropagating characteristics. Other components such as terminal blocks, wire troughs, wire cleats, raceways, cable ties, glastic barriers, and so forth are manufactured from self-extinguishing material.

66 Separation within the NSSS Inverters listed in Table 8.3-10 between Class 1E train related input cables and the Class 1E channel related output cables is not required since these cables are integrally associated with each other (Refer to Figure 8.3-15).

66 Separation within the BOP inverters listed in Table 8.3-10 between associated circuits and non-divisional circuits is not required since the two circuits are isolated by a Class 1E breaker tripped by an "S" signal.

65 Where plant arrangements preclude maintaining the minimum separation distance as stated above, tray covers, bottoms, or other barriers are provided between the redundant circuits.
60 The minimum distances between redundant enclosed raceways and between barriers and raceways are in accordance with NRC Regulatory Guide 1.75 [15] and IEEE-384 [31].

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Inside panels, for control and instrumentation cables or raceways, minimum separation is 1". Conduit to conduit minimum separation is 0". Control cables #10 AWG and larger maintain 6" separation or are enclosed in a conduit, wrapped with woven silicon dioxide, separated by a glastic barrier or routed in enclosed wire ways. | 68

Outside panels, for control and instrumentation cables or raceways where plant arrangements preclude maintaining minimum separation, a single barrier is provided as required through use of a single conduit, woven silicon dioxide, or tray cover on top or bottom. Conduit to conduit minimum separation is 1/8". | 68

Control and instrumentation cables size #12 AWG or smaller in BISCO fire sealant maintain a minimum separation of 1". | 68

Approved conduits are rigid, EMT, Servicair Flexible, American Boa Flexible, or Anaconda Sealtite Flexible. | 68

The above separation criteria has been demonstrated by testing and analysis (refer to References 41 and 42) to meet or exceed Regulatory Guide 1.75 [15] and IEEE-384 [31]. | 62

For the purpose of electrical cable separation, acceptable enclosed raceway includes rigid metal conduit, electrical metallic tubing (EMT) and flexible metallic conduit. Ventilated tray covers are considered equivalent to solid non-ventilated tray covers. Cable bus enclosures are considered the same as enclosed raceway for separation purposes. | 65

A wrap of woven silicon dioxide is equivalent to a metal enclosed raceway with respect to protection from electrical failures. | 65

Testing performed by other utilities has demonstrated the adequacy of the above materials to be used as enclosed raceway and barriers for Regulatory Guide 1.75 [15] separation purposes. | 65 between redundant NIS penetrations is six feet. The minimum center line separation between penetrations of redundant channels of RPS, between RPS and NIS, and between these channels and any other electrical penetrations, is 5 feet. The minimum centerline separation between any two (2) non-Class 1E penetrations or any two (2) same train penetrations is approximately 2-1/2 feet. The minimum centerline separation between any Class 1E penetration and non-Class 1E penetration is 3 feet. | 41

8. Hostile Environments

Routing of cables for Class 1E systems through an area where there is potential for accumulation of meaningful quantities of oil or other combustible material is avoided. Where such routing is unavoidable, only one system of redundant cables is allowed in any such area, and the cables are protected by installing them in conduits or solid bottom trays with solid covers. In areas containing potential missiles, physical arrangement, protective barriers, or pipe restraints preclude loss of redundant systems.

9. Sharing of Cable Trays

Non-Class 1E cables are separated from Class 1E cables and from associated cables. Non-Class 1E cables, when they share the same raceway with Class 1E cables or by virtue of power supply connection to Class 1E buses, are associated cables and are so designated. As described in preceding sections, these associated cables are separated from non-Class 1E and from Class 1E channels and redundant train cables with which they are not associated. | 68

This separation is maintained throughout the length of the cable (or circuit) until it passes through an isolation device.

accordance with IEEE 450-1980 [35], and Regulatory Guide 1.129
49 [18D]. Visual checks and performance tests are also scheduled for the battery chargers.

41 The periodic tests of all Class 1E DC equipment are performed to satisfy the requirements of GDC 18 and 21 [1].

7. Identification of Safety Loads

68 The safety loads connected to the Class 1E 125-VDC systems are identified in Tables 8.3-4, 8.3-4A, 8.3-4B and 8.3-4C. In
33 addition, the tables indicate the maximum length of operating time required for each load upon loss of AC power. The method of distinguishing between Class 1E and non-Class 1E loads is defined in Subsection 8.3.1.3.

8.3.2.2 Analysis

27 The design of the Class 1E 125-VDC systems is in accordance with the requirements of GDC-17 [1], GDC-18 [1], NRC Regulatory Guides 1.6 [2], 1.32 [7], 1.75 [15] and IEEE 308 [20]. Compliance with these criteria is described in Subsection 8.3.1.2.1. The seismic requirements are specified in Section 3.10B.

Redundant power supplies and equipment satisfy GDC 17 for a single failure. A failure mode analysis is presented in Table 8.3-7. Quality assurance, cable routing, separation, and equipment identification are covered in Subsections 8.3.1.2 and 8.3.1.4.

1. Surveillance and Monitoring

Each battery charger is equipped with a DC voltmeter, a DC ammeter, AC failure relay, and low battery voltage relays. Malfunction of the chargers annunciates in the Control Room. A

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system level inoperable status indication is provided for the Class 1E 125-VDC system in accordance with NRC Regulatory Guide 1.47 [9].

Battery float and discharge current and distribution bus voltage are monitored at the switchboard and in the Control Room. Ground detection, undervoltage relays, and overvoltage relays are provided on the main distribution buses; off-normal conditions are annunciated in the Control Room.

The overall system design including function requirements, redundancy, capacity, and availability is in conformance with IEEE 308 [20] criteria for Class 1E systems. | 27

2. Physical Identification and Separation

The physical identification of the Class 1E DC power systems is combined with the identification of Class 1E AC Power Systems and is described in Subsection 8.3.1.3. The physical separation of the Class 1E DC systems is maintained in accordance with the intent of NRC Regulatory Guide 1.75 [15].

3. Independence of Redundant Class 1E 125-VDC Systems

Redundancy of power sources and distribution equipment is provided in the DC system. This redundancy extends from the station batteries and battery chargers through distribution panels, cabling, and switchgear. Each redundant DC system and its associated distribution equipment can independently provide the required DC power for safe shutdown of the plant. | 33

Each redundant Class 1E 125-V battery system is located in a separate seismic Category I battery room. The Class 1E battery chargers and main distribution buses associated with each Class 1E 125-V battery system are located in a separate seismic | 33

33 Category I room. The local distribution panels, feeders, and control and instrumentation cables raceway system associated with each Class 1E battery system are separated as described in the previous section. Separation of train A and train B battery systems is maintained up through the Class 1E 6900-VAC buses and is consistent with the train designations of such equipment.

The quality assurance program discussed in Section 17 ensures compliance with established criteria in IEEE 336-1971 [24] and
68 NRC Regulatory Guide 1.30 [6]. The Class 1E 125-VDC equipment and circuits are identified on documents and drawing in accordance with the requirements of IEEE 494-1974 [36] as discussed in Section 8.1.5.2 (Item 5).

8.3.3 FIRE PROTECTION FOR CABLE SYSTEMS

20 The Fire Protection System for cables is a part of the integrated system of the fire detection and protection for the entire plant. Cable fire prevention measures such as power cable derating, restricted percentage of cable tray fill, and the installation of nonpropagating and self-extinguishing type cable, insulation, and jacket material are used. These measures are described in detail in the following sections. In general, ionization and thermal type fire detectors are located in areas of heavy cable concentration to detect a cable fire. These fire detectors form a part of the Fire Detection System installed throughout the plant. The system provides early warning at local panels and in the Control Room to alert the operator and subsequently the plant fire brigade to take immediate action to extinguish a fire.

52 Portable fire extinguishers and hose stations located throughout the plant are available to extinguish a cable fire.

22 The fire suppression systems and specific equipment used to extinguish a cable fire are described in detail in Section 9.5.1 for each area containing cables.

Conduit fill for power, control and instrumentation cables is limited to the requirements delineated below: 66

66

Number of contained cables	1	2	3 or more	
Percent Fill Limit	53	31	40	66

However, these percentages may be exceeded provided the following conditions are satisfied: 66

1. Cable pulling tension is not exceeded. 66

2. For power cables - the thermal rating of the cable is not exceeded. 66

3. For fire seal application the limits of conduit fill for the qualified seal are not exceeded. 66

4. Conduit support design is adequate. 68

8.3.3.2 Fire Detection and Protection Devices

As stated previously, fire detection equipment and protection equipment in the area where cables are installed is described in detail in Section 9.5.1.

8.3.3.3 Fire Barriers and Tray Supports

The fire barriers, where required between redundant trays, are described in detail in Section 9.5.1. The separation between redundant trays is described in Subsection 8.3.1.4.

TABLE 8.3-1A

(Sheet 3)

LOADING REQUIREMENTS FOR LOSS-OF-COOLANT ACCIDENT

COINCIDENT WITH LOSS OF OFFSITE POWER

INJECTION PHASE(7)

Component**	Nameplate hp (kW if Indicated)	Auto-Sequence Start	Start Time After SIAS (sec) (1)	Number Installed Per Unit	Number Required	hp Req.	Volts	(13) FLA	(13) P.F.	(12) Eff.	kW	Basis for hp Required*	Time To Stop	
Technical support center isolation transformer	30 (3)	yes	10	2	1	-	480	-	0.95	-	43	(d)	(4)	66
Safety injection pump	450	yes	15	2	1	425 (8)	6600	38.5	.914	-	345.4	(b)	(4)	66
Chilled water recirculation pump NSR Class 3	25	yes	15	2	1	25	460	-	-	.8	23.3	(a)	(4)	
SIS pump area fan coil unit	3	yes (10)	15	2	1	3	460	-	-	.8	2.8	(a)	(4)	
RHR pump	450	yes	20	2	1	435	6600	31.7	.93	-	349.5	(c)	(4)	66
RHR pump area fan coil unit	[illegible]	yes (10)	20	2	1	3	460	-	-	.8	2.8	(a)	(4)	
Containment spray pumps	1500	yes	25	4	2	1500 (8)	6600	116.6	.945	-	1180.6	(b)	(4)	66
Containment spray pump area fan coil units	[illegible]	yes (10)	25	4	2	6	460	-	-	.8	5.6	(a)	(4)	
CCWP	1000	yes	30	2	1	1000	6600	74.6	.918	-	782.9	(a)	(4)	36
CCWP area fan coil unit	3	yes (10)	30	2	1	3	460	-	-	.8	2.8	(a)	(4)	36

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TABLE 8.3-1B

(Sheet 7)

LOADING REQUIREMENTS FOR LOSS-OF-COOLANT ACCIDENT

COINCIDENT WITH LOSS OF OFFSITE POWER

RECIRCULATION PHASE[illegible]

Component**	Nameplate hp (W if Indicated)	Auto-Sequence Start	Start Time After SIAS (sec) (1)	Number Installed Per Unit	Number Required	hp Req.	Volts	(13) FLA	(13) P.F.	(12) Eff.	kW	Basis for hp Required*	Time To Stop	
														: 66
Hydrogen purge electric heater	20 (3)	no	as required after 5 days	2 (2)	1	-	480	-	-	-	(16)	(a)	(5)	
Hydrogen purge supply fan	4	no	as required after 5 days	2 (2)	1	40	460	-	-	-	(16)	(a)	(4)	
Hydrogen purge exhaust fan	10	no	as required after 5 days	2 (2)	1	-	460	-	-	-	(16)	(a)	(4)	
Total kW											6158			: 68

TABLE 8.3-2

(Sheet 3)

EMERGENCY ELECTRICAL LOADING REQUIREMENTS FOR LOSS OF OFFSITE POWER (BLACKOUT) CONDITIONS

Component***	Nameplate hp (kW if Indicated)	Auto-Sequence Start	Start Time After Blackout (sec) (1)	Number Installed Per Unit	Number Required	hp Req.	Volts	(11) FLA	(11) P.F.	(10) Eff.	kW	Basis for hp Required*	Time To Stop	
BOP inverter (Non-Class 1E)	[illegible]	yes	10	2	1	-	460	-	0.9	-	21	(a)	(4)	66
NSSS Computer inverter	[illegible]	yes	10	1	-	-	460	-	0.9	-	18.7	(a)	(4)	66
Control room rad. monitoring system sample pump	[illegible]	yes	10	2 (2)	1	1.5	460	-	-	.8	1.4	(a)	(4)	
Vent. stack rad. monitoring system sample pump	1.5	yes	10	2 (2)	1	1.5	460	-	-	.8	1.4	(a)	(4)	
Containment air rad. monitoring system sample pump	1.5	yes	10	1 (15)	1	1.5	460	-	-	.8	1.4	(a)	(4)	68
SUPS bypass transformer	[illegible]	yes	10	2	1	-	480	-	-	-	7.5	estimated	(4)	66
Auxiliary building isolation transformer	[illegible]	yes	10	2 (2)	1	-	480	-	-	-	64.0	(d)	(4)	66
Control room A/C unit transformer	[illegible]	yes (13)	10	4 (2)	2	-	480	-	0.95	-	14	estimated	(4)	66

TABLE 8.3-2

(Sheet 10)

EMERGENCY ELECTRICAL LOADING REQUIREMENTS FOR LOSS OF OFFSITE POWER (BLACKOUT) CONDITIONS

Component***	Nameplate hp (kW if Indicated)	Auto-Sequence Start	Start Time After Blackout (sec) (1)	Number Installed Per Unit	Number Required	hp Req.	Volts	(11) FLA	(11) P.F.	(10) Eff.	kW	Basis for hp Required*	Time To Stop	
RHR pump	[illegible]	no (3)	as required after 4 hrs.	2	1	435	6600	31.7	.93	-	349.5	(c)	(4)	: 66
RHR pump area fan coil unit	[illegible]	no (3)	as required after 4 hrs.	2	1	3	460	-	-	.8	2.9	(a)	(4)	
Total kW											6934			: 68

EMERGENCY ELECTRICAL LOADING REQUIREMENTS FOR L SS OF OFFSITE POWER (BLACKOUT) CONDITIONS | 66

Total Load on Diesel Generator (kW) Train A 6889 | 66

Total Load on Diesel Generator (kW) Train B 6705 | 68

*Basis for hp Required

***Where "Pump (Fan)" components are listed, it is understood that the pump (fan) motor is the actual electrical load component.

(a) Nameplate Rating

(b) Pump Runout Condition

(c) Estimated Flow-Pressure Condition or Bhp

(d) Maximum (current limiting) output

NOTES:

(1) Maximum time to breaker including 10 sec for diesel to come up to speed and voltage; the delay times for automatic start are 10 sec. less when offsite power is available.

(2) Equipment is shared between two units and number shown is for two units.

(3) Manual start when required.

(4) Manually stopped.

(5) Stops automatically with assigned diesel or pump, or temperature, or pressure, and so forth.

(6) Motor stops automatically when valve action is completed, or receives signal to stop (e.g., sump pump stops on low water level).

(7) Motor Service Factor of 1.15 meets this HP requirement.

(8) Starts automatically with assigned load or upon failure of engine-driven pump.

(9) Starts automatically upon temperature, pressure, level switch signal, etc.

(10) The following assumptions were made for 460V motors in calculating total load on diesel generator:

a) Less than

Power factor) efficiency 80%

Table 8.3-4

125-Vdc CLASS 1E BATTERY LOAD REQUIREMENTS* | 66

Load Description	Amperes Required Per Time Interval After Loss of AC Power			
	0 to 1 Min.	1 to 239 Min.	239 to 240 Min.	66
1. BOP Inverter	82.18	82.18	82.18	68
2. NSSS Inverters (two)	134.78	134.78	134.78	68
3. DELETED	0	0	0	66
4. 6.9 kV and 480 V Switchgear	200.9#	6.9	96.9##	66
5. Diesel Generator Control and Engineer Panels	7.4	7.4	7.4	66
6. Diesel Generator Field Flash	14.0	-	-	66
7. Reactor Trip Switchgear	12.41	.41	.41	66
8. Turbines Driven Auxiliary Feedwater Pump Control Panel	11.79	.88	.88	66
9. Hot Shutdown Panel	.160	.160	.160	66
10. Termination racks and H&V panels	51.60	32.70	32.70	66
11. Solid State Protection Cabinet	3.0	2.0	2.0	66
12. DELETED	0	0	0	66
13. Sample Valve Control Panel	3.3	3.3	3.3	66
14. DELETED	0	0	0	66
15. Shutdown Transfer Panel	.450	.450	.450	66
TOTALS	521.97	271.16	361.16	68

*NOTE: THE ABOVE LOADING APPLIES TO BT1ED1 ONLY. | 66

\# Includes: 50A UV Trip for 6.9kV Bkrs. | 66

120A UV Trip for 480V Bkrs. | 66

\## Includes: 90A for 6.9kV SWGR BKR Closure | 66

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Table 8.3-4A

125-Vdc CLASS 1E BATTERY LOAD REQUIREMENTS* | 66

Load Description	Amperes Required Per Time Interval After Loss of AC Power	
	0 To 240 Minutes	33
1. BOP inverter	76.15	68
2. Deleted	-	66
3. Deleted	-	66
4. Deleted	-	66
5. PASS Cntmt Isol Vlv Cnt Pnl	.35	66
Total	76.50	68

(Header lines marked 33 | 33)

* Note: The above loading applies to BT1ED3 only. | 66

TABLE 8.3-4B

125-Vdc CLASS 1E BATTERY LOAD REQUIREMENTS* | 66

Load Description	Amperes Required for Time Interval After Loss of AC Power			
	0 to 1 Min.	1 to 239 Min.	239 to 240 Min.	
1. BOP Inverter	75.95	75.95	75.95	| 68
2. NSSS Inverters (two)	154.44	154.44	154.44	| 68
3. DELETED	0	0	0	| 66
4. 6.9 kV AND 480 V	176.628#	6.628	96.628##	| 66
Switchgear				| 66
5. Diesel Generator				| 66
Control and				| 66
Engineer Panels	7.4	7.4	7.4	| 66
6. Diesel Generator				| 66
Field Flash	14.0	-	-	| 66
7. Reactor Trip				| 66
Switchgear	12.41	.41	.41	| 66
8. DELETED	0	0	0	| 66
9. Hot Shutdown Panel	.19	.19	.19	| 68
10. Termination Racks				| 66
and H&V Panels	48.44	32.24	32.24	| 66
11. Solid State Protection				| 66
tion Cabinet	3.0	2.0	2.0	| 66
12. DELETED	0	0	0	| 66
13. Sample Valve Control	1.78	1.78	1.78	| 66
Panel				| 66
14. Deleted	0	0	0	| 66
TOTALS	494.24	281.04	371.04	| 68

*NOTE: THE ABOVE LOADING APPLIES TO BT1ED2 ONLY. | 66

Includes: 50A UV Trip for 6.9kV Bkrs. | 66

120A UV Trip for 480V Bkrs. | 66

##Includes: 90A for 6.9kV SWGR BKR Closure | 66

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TABLE 8.3-4C

125-Vdc CLASS 1E BATTERY LOAD REQUIREMENTS* | 66

Load Description	Amperes Required for Time Interval After Loss of AC Power 0 to 240 Minutes	
1. BOP Inverter	80.87	68
2. DELETED	0	66
3. DELETED	0	66
4. DELETED	0	66
5. PASS Cntmt Isol Vlv Cnt Pnl	.35	66
TOTAL	81.22	68

*NOTE: THE ABOVE LOADING APPLIES TO BT1ED4 ONLY. | 66

TABLE 8.3-7

(Sheet 2)

FAILURE MODE AND EFFECT ANALYSIS FOR

AUXILIARY DC POWER SYSTEM

Item	Description	Function	Failure Mode	Causes of Failure	Effects of Failure	How Failure Is Detected	Effects on System	
4	Fusible Switch	protects Train A buses and batteries	Blows at less than rated current	material defect	battery cannot supply power to bus	annunciation and safety system inoperable indication (SSII) in control room	None: redundant safety-related equipment is is connected to Train B buses	33
4A	Fusible Switch	protects Train B buses and batteries	Blows at less than rated current	material defect	battery cannot supply power to bus	annunciation and safety system inoperable indication (SSII) in control room	None: redundant safety-related equipment is connected to Train A buses	33
5	Circuit Breaker	protects Train A bus and charger	fails to open; fails to close	mechanical failure; stuck contacts	fails to open: could damage charger; fails to close: power not available from battery charger	annunciation and safety system inoperable indication (SSII) in control room	None: Redundant safety-related equipment is connected to Train B bus	68
5A	Circuit Breaker	protects Train B bus and charger	fails to open; fails to close	mechanical failure; stuck contacts	fails to open: could damage charger; fails to close: power not available from battery charger	annunciation and safety system inoperable indication (SSII) in control room	None: Redundant safety-related equipment is connected to Train A bus	68
6	Bus 1ED1 and 1ED3	distributes Train A dc power	fails to deliver power	short circuit or overload; fire	loss of Train A load group	annunciation in control room	None: redundant loads are supplied by Train B buses	33

TABLE 8.3-11 | 68

(Sheet 1 of 12) | 68

NONSAFETY RELATED EQUIPMENT CONNECTED | 68

TO SAFETY RELATED POWER CIRCUITS | 68

EQUIPMENT ID NO.	DESCRIPTION	POWER SOURCE ID NO.		NOTE	
					68
CPX-CHCICE-03	HVAC CENTRIFUGAL WATER CHILLER #13	CP1-EPSWEA-01	2	(1)	68
CPX-CHCICE-01	HVAC CENTRIFUGAL WATER CHILLER #11	CP1-EPSWEA-01	4	(1)	68
CPX-CHCICE-04	HVAC CENTRIFUGAL WATER CHILLER #14	CP1-EPSWEA-02	15	(1)	68
CPX-CHCICE-02	HVAC CENTRIFUGAL WATER CHILLER #12	CP1-EPSWEA-02	7	(1)	68
CPX-EPMCNB01-06	AUX BLDG NON SAFEGUARD MCC XEB1-3	CP1-EPSWEB-01	5B	(2)	68
CP1-EPMCNB03-01	AUX, SG & FUEL BLDGS NON-SAFEGUARDS MCC 1EB1-3	CP1-EPSWEB-01	5D	(2)	68
CP1-EPMCNB01-06	CONTAINMENT NON-SAFEGUARDS MCC 1EB1-2	CP1-EPSWEB-01	5C	(2)	68
CP1-VAFNAV-01	CONTAINMENT RECIRC FAN	CP1-EPSWEB-01	3B	(1)	68
TBX-CSAAPPD-01	POSITIVE DISPLACEMENT CHARGING PUMP	CP1-EPSWEB-01	2B	(1)	68
TBX-RCPCPR-03	ISOL. TRANSFORMER FOR PRESSURIZER HTR CTRL GROUP-C	CP1-EPSWEB-01	6D	(2)	68
TBX-RCPCPR-01	ISOL. TRANSFORMER FOR PRESSURIZER HTR BK-UP GRP-A	CP1-EPSWEB-03	11B	(2)	68
CP1-VAFNAV-03	CONTAINMENT RECIRC FAN	CP1-EPSWEB-03	9B	(1)	68
CP1-VAFNCB-01	CRDM VENT FAN	CP1-EPSWEB-03	8B	(1)	68
CPX-EPMCNB03-04	FUEL BLDG NON-SAFEGUARD MCC XEB3-1	CP1-EPSWEB-03	11C	(2)	68
CPX-FPAPFP-01	FIRE PUMP (FIRE BRIGADE TRAINING/EMERG FILL PUMP)	CP1-EPSWEB-03	9C	(1)	68
CP1-EPMCNB04-01	AUX, SG & FUEL BLDGS NON-SAFEGUARD MCC 1EB2-3	CP1-EPSWEB-02	5D	(2)	68
CP1-EPMCNB02-06	CONTAINMENT NON-SAFEGUARD MCC IEB2-2	CP1-EPSWEB-02	5C	(2)	68
CP1-VAFNAV-02	CONTAINMENT RECIRC FAN	CP1-EPSWEB-02	3B	(1)	68

TABLE 8.3-11 | 68

(Sheet 2) | 68

NONSAFETY RELATED EQUIPMENT CONNECTED | 68

TO SAFETY RELATED POWER CIRCUITS | 68

EQUIPMENT ID NO.	DESCRIPTION	POWER SOURCE ID NO.		NOTE	
TBX-RCPCPR-02	ISOL. TRANSFORMER FOR PRESSURIZER HTR BK-UP GRP-B	CP1-EPSWEB-02	5B	(2)	68
TBX-RCPCPR-04	ISOL. TRANSFORMER FOR PRESSURIZER HTR BK-UP GRP-D	CP1-EPSWEB-04	11B	(2)	68
CP1-VAFNAV-04	CONTAINMENT RECIRC FAN	CP1-EPSWEB-04	9B	(1)	68
CP1-VAFNCB-02	CRDM VENT FAN	CP1-EPSWEB-04	8B	(1)	68
CPX-EMPCNB04-05	FUEL BLDG NON-SAFEGUARD MCC XEB4-1	CP1-EPSWEB-04	11C	(2)	68
CP1-MESCCP-01	CONTAINMENT POLAR CRANE	CP1-EPSWEB-04	10B	(2)	68
CP1-ELTRNT-26	LTG TRANSFORMER XFS1 POWER SUPPLY	CP1-EPMCEB-01	9E	(5)	68
TBX-ASELIV-01	COMPUTER POWER SUPPLY INVERTER 1V1C1	CP1-EPMCEB-01	1BR	(1)	68
MOV 1-HV-2452	AUXILIARY FEED PUMP TURBINE TRIP	CP1-EPMCEB-01	2KL	(1)	68
CP1-VAFNCB-03	POSITIVE DISPLACEMENT CHARGING PUMP ROOM FAN	CP1-EPMCEB-03	4F	(1)	68
CP1-CICACO-01	INSTRUMENT AIR COMPRESSOR 1-01	CP1-EPMCEB-03	1M	(1)	68
CP1-VAFNID-11	BATTERY ROOM 1-3 (C) EXHAUST FAN	CP1-EPMCEB-03	3C	(1)	68
MOV 1-8109	POSITIVE DISPLACEMENT PUMP BYPASS VALVE	CP1-EPMCEB-03	10J	(1)	68
CP1-CIDYIA-01	AIR DRYER CONTROL PANEL FDR	CP1-EPMCEB-03	2BR	(1)	68
CP1-ECIVNC-02	INSTR INVERTER 1V1C3/INSTR DIST PNL BD 1C3	CP1-EPMCEB-03	1CR	(1)	68

TABLE 8.3-11 | 68

(Sheet 3) | 68

NONSAFETY RELATED EQUIPMENT CONNECTED | 68

TO SAFETY RELATED POWER CIRCUITS | 68

EQUIPMENT ID NO.	DESCRIPTION	POWER SOURCE ID NO.		NOTE	
1RE-5502/03/66	RADIATION MONITOR	CP1-EPMCEB-02	11D	(1)	68
CP1-ELTRNT-27	LTG TRANSFORMER XFS2 POWER SUPPLY	CP1-EPMCEB-02	10D	(5)	68
CP1-EPBCND-02	BATTERY CHARGER BC1D2 FOR BATTERY BT1D2	CP1-EPMCEB-02	1FR	(1)	68
CP1-VAFNID-12	BATTERY ROOM 1-3(C) EXHAUST FAN 12	CP1-EPMCEB-04	3C	(1)	68
CP1-ECIVNC-01	INSTR INVERTER 1V1C2/INSTR DIST PNL BD 1C2	CP1-EPMCEB-04	10DL	(1)	68
CP1-EPBCND-04	BATTERY CHARGER BC1D4 FOR BATTERY BT1D4	CP1-EPMCEB-04	10DR	(1)	68
CP1-CICACO-02	INSTRUMENT AIR COMPRESSOR CONTROL PANEL	CP1-EPMCEB-04	12B	(1)	68
CP1-CIDYIA-02	INSTRUMENT AIR DRYER CONTROL PANEL	CP1-EPMCEB-04	12E	(1)	68
CP1-VAFNAV-09	NEUTROL DETECTOR WELL FAN 09	CP1-EPMCEB-05	8M	(1)	68
CP1-ELTRNT-18	CONTAINMENT LTG TRANSFORMER/LTG PNLS SC1 & SC3	CP1-EPMCEB-05	7M	(1)	68
MOV 1-HV-6074	VENT CONTROL COOL UNIT COOLER/CHILLED WATER	CP1-EPMCEB-05	5G	(1)	68
MOV 1-HV-6076	VENT CONTROL COOL UNIT COOLER/CHILLED WATER	CP1-EPMCEB-05	4G	(1)	68
MOV 1-HV-6078	VENT CONTROL COOL UNIT COOLER/CHILLED WATER	CP1-EPMCEB-05	4M	(1)	68
EP1-EPTRET-07	ISOL. TRANFORMER TIEC3-3	CP1-EPMCEB-05	2BR	(5)	68

TABLE 8.3-11 | 68

(Sheet 4) | 68

NONSAFETY RELATED EQUIPMENT CONNECTED | 68

TO SAFETY RELATED POWER CIRCUITS | 68

EQUIPMENT ID NO.	DESCRIPTION	POWER SOURCE ID NO.		NOTE	
CP1-VAFNAV-10	NEUTRON DETECTOR WELL FAN 10	CP1-EPMCEB-06	7M	(1)	| 68
CP1-ELTRNT-19	CONTAINMENT LTG TRANSFORMER/LTG PNLS SC2 & SC4	CP1-EPMCEB-06	6M	(1)	| 68
MOV 1-HV-6075	VENT CONTROL COOL UNIT COOLER/CHILLED WATER	CP1-EPMCEB-06	4G	(1)	| 68
MOV 1-HV-6077	VENT CONTROL COOL UNIT COOLER/CHILLED WATER	CP1-EPMCEB-06	3G	(1)	| 68
MOV 1-HV-6079	VENT CONTROL COOL UNIT COOLER/CHILLED WATER	CP1-EPMCEB-06	3M	(1)	| 68
CP1-MEDGEE-02K	AUXILIARY JACKET WATER PUMP	CP1-EPMCEB-10	1M	(1)	| 68
CP1-MEDGEE-02J	AUXILIARY LUBE OIL PUMP	CP1-EPMCEB-10	3M	(1)	| 68
CP1-MEDGEE-02P	JACKET WATER HEATER	CP1-EPMCEB-10	4M	(1)	| 68
CP1-MEDGEE-02V	LUBE OIL HEATER	CP1-EPMCEB-10	2M	(1)	| 68
CP1-MECAED-03	DG AIR COMPRESSOR #1	CP1-EPMCEB-10	1G	(1)	| 68
CP1-MECAED-04	DG AIR COMPRESSOR #2	CP1-EPMCEB-10	2G	(1)	| 68
CP1-MEDGEE-02M	PRELUBE PUMP	CP1-EPMCEB-10	6M	(1)	| 68
CP1-MEDGEE-02L	JACKET WATER KEEP WARM PUMP	CP1-EPMCEB-10	7J	(1)	| 68
CP1-MEDGEE-02N	FUEL OIL BOOSTER PUMP	CP1-EPMCEB-10	7M	(1)	| 68

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NONSAFETY RELATED EQUIPMENT CONNECTED | 68

TO SAFETY RELATED POWER CIRCUITS | 68

EQUIPMENT ID NO.	DESCRIPTION	POWER SOURCE ID NO.		NOTE	
CP1-MEDGEE-02G	FUEL OIL DRIP RETURN PUMP	CP1-EPMCEB-10	8M	(1)	68
CP1-MECAED-03	COMPRESSOR AFTERCOOLER 1	CP1-EPMCEB-10	7C	(1)	68
CP1-MECAED-04	COMPRESSOR AFTERCOOLER 2	CP1-EPMCEB-10	8C	(1)	68
CP1-VAEHOH-30	ROOM ADJACENT TO REFUEL WATER STORAGE TANK HTR	CP1-EPMCEB-10	2BR	(1)	68
CP1-VAEHOH-28	ROOM ADJACENT TO REACTOR MK-UP STORAGE TANK HTR	CP1-EPMCEB-10	4BL	(1)	68
CP1-VAEHOH-29	ROOM ADJACENT TO COND. WATER STORAGE TANK HTR	CP1-EPMCEB-10	4BR	(1)	68
CPX-CHAPCP-03	VENT CHILLED WATER PUMP 03	CPX-EPMCEB-01	3M	(1)	68
CPX-VAFNCB-09	PRIMARY PLANT VENT EXHAUST FAN 09	CPX-EPMCEB-01	4M	(1)	68
CPX-VAFNCB-11	PRIMARY PLANT VENT EXHAUST FAN 11	CPX-EPMCEB-01	5G	(1)	68
CPX-VAFNCB-13	PRIMARY PLANT VENT EXHAUST FAN 13	CPX-EPMCEB-01	5M	(1)	68
CPX-VAFNCB-03	CONTAINMENT HYDROGEN PURGE SUPPLY FAN 03	CPX-EPMCEB-01	3E	(1)	68
CPX-CHAPCP-04	VENT CHILLED WATER PUMP 1	CPX-EPMCEB-02	3M	(1)	68
CPX-VAFNCB-10	PRIMARY PLANT VENT EXHAUST FAN 10	CPX-EPMCEB-02	4M	(1)	68
CPX-VAFNCB-12	PRIMARY PLANT VENT EXHAUST FAN 12	CPX-EPMCEB-02	5G	(1)	68
CPX-VAFNCB-14	PRIMARY PLANT VENT EXHAUST FAN 14	CPX-EPMCEB-02	5M	(1)	68

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NONSAFETY RELATED EQUIPMENT CONNECTED | 68

TO SAFETY RELATED POWER CIRCUITS | 68

EQUIPMENT ID NO.	DESCRIPTION	POWER SOURCE ID NO.		NOTE	
CPX-VAFNCB-04	CONTAINMENT HYDROGEN PURGE SUPPLY FAN 04	CPX-EPMCEB-02	3E	(1)	| 68
CPX-EPPRET-02	AUX BLDG ISOLATION TRANSFORMER TXEC1	CPX-EPMCEB-02	5BL	(5)	| 68
CPX-CHAPCP-01	VENT CHILLED WATER PUMP 1	CPX-EPMCEB-07	4M	(1)	| 68
CPX-VAFNAV-27	AUXILIARY BLDG VENT EQUIP ROOM EXHAUST FAN 27	CPX-EPMCEB-07	3M	(1)	| 68
XRE-5895	CONTROL ROOM RADIATION MONITOR SYS SAMPLE PUMP	CPX-EPMCEB-07	1F	(1)	| 68
CPX-EPTRET-03	ISOLATION TRANSFORMER TXEC3	CPX-EPMCEB-07	1C	(5)	| 68
CPX-EPTRET-01	ISOLATION TRANSFORMER TXEC1	CPX-EPMCEB-07	1BR	(5)	| 68
CPX-CHAPCP-02	VENT CHILLED WATER PUMP 2	CPX-EPMCEB-08	3M	(1)	| 68
CPX-VAFNAV-28	AUXILIARY BLDG VENT EQUIP ROOM EXHAUST FAN 28	CPX-EPMCEB-08	4M	(1)	| 68
X-RE-5896	CONTROL ROOM RADIATION MONITOR SYS SAMPLE PUMP	CPX-EPMCEB-08	6F	(1)	| 68
X-RE-5568/75/67A	VENT STACK "1" RADIATION MONITOR SYS SAMPLE PUMP	CPX-EPMCEB-03	5C	(1)	| 68
CPX-VAFNCB-15	PRIMARY PLANT VENT EXHAUST FAN 15	CPX-EPMCEB-03	4G	(1)	| 68
CPX-FAFNCB-17	PRIMARY PLANT VENT EXHAUST FAN 17	CPX-EPMCEB-03	4M	(1)	| 68
CPX-VAFNCB-19	PRIMARY PLANT VENT EXHAUST FAN 19	CPX-EPMCEB-03	5G	(1)	| 68
CPX-VAFNCB-21	PRIMARY PLANT VENT EXHAUST FAN 21	CPX-EPMCEB-03	5M	(1)	| 68

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NONSAFETY RELATED EQUIPMENT CONNECTED | 68

TO SAFETY RELATED POWER CIRCUITS | 68

EQUIPMENT ID NO.	DESCRIPTION	POWER SOURCE ID NO.		NOTE	
CPX-CHCICE-01	HVAC CENTRIFUGAL WATER CHILLER OIL PUMP	CPX-EPMCEB-03	5BL	(1)	68
CPX-CHCICE-03	HVAC CENTRIFUGAL WATER CHILLER OIL PUMP	CPX-EPMCEB-03	5BR	(1)	68
X-RE-5568/75/67B	VENT STACK "2" RADIATION MONITOR SYS SAMPLE PUMP	CPX-EPMCEB-04	5G	(1)	68
CPX-VAFNCB-16	PRIMARY PLANT VENT EXHAUST FAN 16	CPX-EPMCEB-04	4G	(1)	68
CPX-VAFNCB-18	PRIMARY PLANT VENT EXHAUST FAN 18	CPX-EPMCEB-04	4M	(1)	68
CPX-VAFNCB-20	PRIMARY PLANT VENT EXHAUST FAN 20	CPX-EPMCEB-04	5G	(1)	68
CPX-VAFNCB-22	PRIMARY PLANT VENT EXHAUST FAN 22	CPX-EPMCEB-04	5M	(1)	
CPX-EPTRET-04	ISOLATION TRANSFORMER TXEC4	CPX-EPMCEB-04	7C	(5)	68
CPX-CHCICE-02	HVAC CENTRIFUGAL WATER CHILLER OIL PUMP	CPX-EPMCEB-04	5BL	(1)	68
CPX-CHCICE-04	HVAC CENTRIFUGAL WATER CHILLER OIL PUMP	EXP-EPMCEB-04	5BR	(1)	68
CP1-MEDGEE-01M	PRELUBE PUMP	CP1-EPMCEB-09	6MV	(1)	68
CP1-MEDGEE-01L	JACKET WATER KEEP WARM PUMP	CP1-EPMCEB-09	7J	(1)	68
CP1-MEDGEE-01N	FUEL OIL BOOSTER PUMP	CP1-EPMCEB-09	7M	(1)	68
CP1-MEDGEE-01G	FUEL OIL DRIP RETURN PUMP	CP1-EPMCEB-09	9M	(1)	68
CP1-MECAED-01	COMPRESSOR AFTERCOOLER 1	CP1-EPMCEB-09	7C	(1)	68

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NONSAFETY RELATED EQUIPMENT CONNECTED | 68

TO SAFETY RELATED POWER CIRCUITS | 68

EQUIPMENT ID NO.	DESCRIPTION	POWER SOURCE ID NO.		NOTE	
CP1-MECAED-02	COMPRESSOR AFTERCOOLER 2	CP1-EPMCEB-09	5C	(1)	68
CP1-VAEHUH-27	RM ADJ TO REACTOR MK-UP WATER STORAGE TANK HTR	CP1-EPMCEB-09	2BR	(1)	68
CP1-VAEHUH-25	RM ADJ TO REACTOR MK-UP WATER STORAGE TANK HTR	CP1-EPMCEB-09	4BL	(1)	68
CP1-VAEHUH-26	RM ADJ TO REACTOR MK-UP WATER STORAGE TANK HTR	CP1-EPMCEB-09	4BR	(1)	68
CP1-MEDGEE-01K	AUXILIARY JACKET WATER PUMP	CP1-EPMCEB-09	1M	(1)	68
CP1-MEDGEE-01J	AUXILIARY LUBE OIL PUMP	CP1-EPMCEB-09	3M	(1)	68
CP1-MEDGEE-01P	JACKET WATER HEATER	CP1-EPMCEB-09	2M	(1)	68
CP1-MEDGEE-01V	LUBE OIL HEATER	CP1-EPMCEB-09	4M	(1)	68
CP1-MECAED-01	DG AIR COMPRESSOR #1	CP1-EPMCEB-09	1G	(1)	68
CP1-MECAED-02	DG AIR COMPRESSOR #2	CP1-EPMCEB-09	2G	(1)	68
CPX-EPMCEB-05	MOTOR CONTROL CENTER XEB3-3	CP2-EPMCEB-07	2M	(4)	68
CPX-EPMCEB-05	MOTOR CONTROL CENTER XEB3-3	CP1-EPMCEB-07	2E	(4)	68
CPX-SWTSTS-01	SERVICE WATER TRAVELING SCREEN	CPX-EPMCEB-05	3C	(4)	68
CP1-SWAPTS-01	SERVICE WATER SCREEN WASH PUMP	CPX-EPMCEB-05	3F	(4)	68

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NONSAFETY RELATED EQUIPMENT CONNECTED | 68

TO SAFETY RELATED POWER CIRCUITS | 68

EQUIPMENT ID NO.	DESCRIPTION	POWER SOURCE ID NO.		NOTE	68
CPX-ELTRET-01	LIGHTING TRANSFORMER/LTG PNL SWP1	CPX-EPMCEB-05	3ML	(4)	68
WRCPT2	WELDING RECEPTACLE	CPX-EPMEB-05	1L	(3)	68
CPX-MESCSW-01	SERVICE WATER INTAKE STRUCTURE CRANE	CPX-EPMEB-05	1FL	(3)	68
CPX-VAEHUH-08	SERVICE WATER PUMPHOUSE UNIT HEATER 08	CPX-EPMEB-05	1FR	(3)	68
CPX-MEMBCH-12	SERVICE WATER TRAVELING SCREEN HOIST	CPX-EPMEB-05	1BL	(3)	68
CPX-VAEHUH-06	SERVICE WATER PUMPHOUSE UNIT HEATER 06	CPX-EPMEB-05	1BR	(3)	68
CPX-VAEHUH-09	SERVICE WATER PUMPHOUSE UNIT HEATER 09	CPX-EPMEB-05	1DL	(3)	68
CPX-RUDMSW-01	ROLL UP DOOR	CPX-EPMEB-05	1DR	(3)	68
CPX-ELTRET-02	LIGHTING TRANSFORMER/LTG PNL SWP2	CPX-EPMEB-06	2HL	(4)	68
CPX-VAFNAV-41	DIESEL FIRE PUMP ROOM VENTILATION EXH FAN 41	CPX-EPMEB-06	5F	(3)	68
CPX-SWAPCA-01	SERVICE WATER RESIDUAL SAMPLE PUMP	CPC-EPMRN-06	5ML	(3)	68
CPX-SWEHSG-01	SCREEN STOP GATE HOIST (2HP)	CPX-EPMEB-06	5MR	(3)	68
WRCPT1	WELDING RECEPTACLE (60 AMPS)	CPX-EPMEB-06	4L	(3)	68
CPX-ELTRNT-02	SWIS CHLOR. BLDG DIST TRANSFORMER/LTG PNL SWN1	CPX-EPMEB-06	4BL	(3)	68
CPX-VAEHUH-04	SERVICE WATER PUMPHOUSE UNIT HEATER 04	CPX-EPMEB-06	4BR	(3)	68

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NONSAFETY RELATED EQUIPMENT CONNECTED | 68

TO SAFETY RELATED POWER CIRCUITS | 68

EQUIPMENT ID NO.	DESCRIPTION	POWER SOURCE ID NO.		NOTE	
CPX-VAEHUH-05	SERVICE WATER PUMPHOUSE UNIT HEATER 05	CPX-EPMEB-06	4DL	(3)	68
CPX-ECCPRT-05	SWIS CATHODIC PROTECTION RECTIFIER #5	CPX-EPMEB-06	4DR	(3)	68
CPX-EPTRNT-21	HEAT TRACING TRANSFORMER TXHT-2/HT DIST PNL XHT-2	CPX-EPMEB-06	4FL	(3)	68
CPX-VAEHUH-07	SERVICE WATER PUMPHOUSE UNIT HEATER 07	CPX-EPMEB-06	4FR	(3)	68
CPX-EPMCEB-06	MOTOR CONTROL CENTER XEB4-3	CP2-EPMCEB-08	2M	(4)	68
CPX-EPMCEB-06	MOTOR CONTROL CENTER XEB4-3	CP1-EPMCEB-8	2E	(4)	68
CPX-SWTSTS-02	SERVICE WATER TRAVELING SCREEN 02	CPX-EPMCEB-06	2C	(4)	68
CPX-SWAPTS-02	SERVICE WATER SCREEN WASH PUMP 02	CPX-EPMCEM-06	2F	(4)	68
CPX-FPAPFP-03	JOCKEY FIRE PUMP	CPX-EPMCEB-06	3C	(4)	68
CP1-ELTRET-01	XFMR FOR LTG DIST PNL ESB1 & ESB3	CP1-EPMCEB-01	1DL	(5)	68
CP1-ELTRET-02	XFMR FOR LTG DIST PNL ESB5, ESB7 & ESB9	CP1-EPMCEB-01	1DR	(5)	68
CP1-ELTRET-03	XFMR FOR LTG DIST PNL ESB2 & ESB4	CP1-EPMCEB-02	1BL	(5)	68
CP1-ELTRET-04	XFMR FOR LTG DIST PNL ESB6, ESB8 & ESB10	CP1-EPMCEB-02	1BR	(5)	68
CPX-ELTRET-09	XFMR FOR LTG DIST PNEL EAB5 & EAB7	CPX-EPMCEB-01	4BL	(5)	68

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NONSAFETY RELATED EQUIPMENT CONNECTED | 68

TO SAFETY RELATED POWER CIRCUITS | 68

EQUIPMENT ID NO.	DESCRIPTION	POWER SOURCE ID NO.		NOTE	
CPX-ELTRET-06	XFMR FOR LTG DIST PNL ECB2 & EAB10	CPX-EPMCEB-02	6C	(5)	68
CPX-ELTRET-10	XFMR FOR LTG DIST PNL EAB6 & EAB8	CPX-EPMCEB-02	4BL	(5)	68
CP1-ELTRET-05	XFMR FOR LTG DIST PNL ECB5 & ECB3	CP1-EPMCEB-07	5BL	(5)	68
CP2-ELTRET-05	XFMR FOR LTG DIST PNL 2ECB3	CPX-EPMCEB-07	5BR	(5)	68
CPX-ELTRET-05	XFMR FOR LTG DIST PNL EAB9 & ECB1	CPX-EPMCEB-07	1BL	(5)	68
CP2-ELTRET-06	LTG DIST PNL 2ECB4	CPX-EPMCEB-08	2BL	(5)	68
CPX-ELTRET-06	XFMR FOR LTG DIST PNL ECB4 & ECB6	CPX-EPMCEB-08	2BR	(5)	68
CPX-ELTRET-07	XFMR FOR LTG DIST PNL EAB1, EAB3, & EAB11	CPX-EPMCEB-03	4BL	(5)	68
CPX-ELTRET-08	XFMR FOR LTG DIST PNL EAB2, EAB4 & EAB12	CPX-EPMCEB-04	4BL	(5)	68

| 68

NOTES: (1) In accordance with Regulatory Guide 1.75, January 1975, Position C.1, Automatically Tripped on SIAS (accident signal). Reconnection requires operator action to reset after SIAS signal. | 68

(2) Breaker trips on SIAS, requires operator action to reset, and connecting cable is black and routed separately. In addition, these loads are also tripped on LOOP. | 68

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NONSAFETY RELATED EQUIPMENT CONNECTED | 68

TO SAFETY RELATED POWER CIRCUITS | 68

(3) This portion of the non-Class 1E MCC is tripped on SIAS or Blackout (Loss of Offsite Power) | 68
signal, and cable is in dedicated raceway. | 68

(4) Same as No. 1, except MCC's are tripped from either Unit 1 or 2 independent power supply, thus | 68
MCC's, associated and non-Class 1E loads are isolated by SIAS signal. During normal operation | 68
MCC's are powered by Unit 2, and Unit 1 power is locked out. | 68

(5) Non-Class 1E loads fed from Class 1E supplies are protected by two separate Class 1E breakers | 68
connected in series. These breakers are coordinated with their supply breakers, and will be | 68
tested and calibrated periodically to ensure coordination. | 68

APPENDIX 8A

ANALYSIS TO JUSTIFY CABLE SPLICES IN RACEWAYS

1.0 Purpose

The purpose of this analysis is to show that the limited use of splices in raceways, incorporated in the CPSES design, does not degrade the Class 1E circuits and does not pose any undue hazard of a fire. This analysis is developed to satisfy the requirements of NRC Regulatory Guide 1.75, Rev. 1, Regulatory Position C.9. 68

2.0 Scope

This analysis covers all cable splices in raceways utilized in the CPSES plant design. The term "raceways" shall mean to include open or enclosed trays, rigid or flexible steel conduits and condulets and site installed junction boxes in raceway runs. This analysis does not include splices inside an equipment enclosure, splices in junction/terminal boxes furnished as an integral part of an equipment and vendor furnished splices not in raceways. Where an enclosure has been provided and space exists, the splices are located within the equipment enclosure, e.g., field cables for motor leads. Where this is not the case, the splices are located in raceways nearby.

In CPSES cable splices which are made in raceways can be categorized in the following five groups: 68

a. Group 1 - Field routed power, control and instrumentation cables which are connected to Electric Penetration Assembly (EPA) and Thermocouple Reference Junction Boxes pigtail cables by means of in-line splices located in trays. 68

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b. Group 2 - Pigtail cables from local mounted devices (LMDs - solenoid valves, limit switches, level switches, etc.) which are connected to field routed control and instrument grade cables by means of parallel or in-line butt splices located in flexible conduit or condulets.

c. Group 3 - Pigtail cables from local mounted devices which are connected to the pigtail conductors of the Electric Conductor Seal Assembly (ECSA) by means of in-line butt splices located in the flexible conduit of the ESCA.

d. Group 4 - Field routed power cables which are spliced inside junction boxes (splices made within an equipment enclosure are outside the scope of this analysis) to a smaller size cable in order to accomodate equipment connection provisions (i.e., the equipment can only accept a cable size smaller than the field cable).

68 | e. Group 5 - Field routed power cables which are spliced in manholes by means of an in-line splice located in cable trays.

3.0 Regulatory Position Requirement

The CPSES FSAR commits to IEEE Std. 384-1974 and NRC Regulatory Guide 1.75, Revision 1 dated January 1975. Section 5.1.1.3 of IEEE Standard 384, with the Regulatory Guide 1.75, Regulatory Position C.9 supplement, reads as follows:

"5.1.1.3 The minimum separation distances specified in Section 5.1.3 and 5.1.4 are based on open ventilated cable trays of either the ladder or trough type as defined in NEMA VE1-1971, Cable Tray Systems. Where these distances are used to provide adequate physical separation:

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originating at a cable splice and (2) cable splices in raceways do not pose any undue hazard of initiating a fire.

4.0 Determination of Acceptability of Cable Splices In Raceways

4.1 Independence of Redundant Trains

Minimum separation distances utilized in CPSES design to achieve independence between redundant trains meet the requirements of Sections 5.1.3 and 5.1.4 of IEEE Standard 384. Per the requirements of IEEE Standard 384, as modified by Regulatory Guide 1.75, these minimum separation distances are adequate provided (1) cable and raceway materials involved are flame retardant, (2) cable trays are not filled above their side rails, (3) hazards are limited to fire originating in the raceway and (4) there should be no cable splices in raceways.

Since the design may not comply with the fourth requirement, the question would be whether the specified minimum separation distances are adequate for providing independence between redundant trains in case of a fire electrically generated at a cable splice.

Where there are no splices, the specified separation distances per the Standard are adequate to satisfy this objective for the case of an electrically generated cable fire. If the fire generated at a cable splice is no worse than a cable fire, the specified minimum separation distances would be adequate. We will show that the degree of potential damage from a cable splice fire is no more than the degree of potential damage from a fire in a cable without splices.

All cable splices within the scope of this analysis are made of essentially two types of material: one is the metallic connector part for conducting current and the second is the non-metallic insulation part. The metallic part is non-combustible like the copper conductor in a cable. The insulation materials applied at the splice are of a non-fire propagating type similar to the insulation and jacket material of a cable.

The insulation and jacket materials of CPSES cables in raceways consist of EPR, XLPE, CPE, ETFE and CSPE. The materials, in their constructed configurations, do not propagate fire and are self extinguishing. This has been shown by documented tests performed by the cable vendors in order to qualify their cables per Section 2.5 test requirements of IEEE Standard 383-1974. The oxygen index of these cable insulating materials ranges from 22 to 34. The higher the oxygen index, the quicker the fire extinguishes. For cable splices, the insulation materials involved are furnished by two vendors, namely AMP Special Industries and Raychem Corporation. Groups 1 through 5 | 68
uninsulated splices (see scope) employ Raychem heat-shrink type insulating material. For some group 2 splices, AMP has furnished pre-insulated in-line butt type connectors, with PVF2 (Kynar) insulation material. The materials involved in the Raychem splices consist of WCSF for sleeves, "-52" for molded parts and S119 used as a sealant and adhesive. Except for the S119, the other three materials are flame retardant and meet the flame test requirements of IEEE Standard 383. The S119 adhesive/sealant, although not flame retardant as a material, has passed the flame test requirement of the IEEE 383 Standard in the installed configuration. In the installed configuration, the major portion of the S119 material does not have access to air. Access to air is esential for a burning process to continue. The oxygen index of the remaining three products ranges from 28-44, which is higher than the minimum oxygen index of the cable insulation and jacket material. Therefore, it can be seen that all cable splices, as installed, are flame retardant, non-propagating and self-extinguishing to a degree equal to or greater than the cable insulating material itself.

Based on the above, it is concluded that the degree of potential damage due to a fire at a cable splice is no more than that from a fire in a cable. Therefore, use of the minimum required separation distances specified in Sections 5.1.3 and 5.1.4 of IEEE Standard 384-1974 is justified for the splices utilized in the CPSES design.

4.2 Assessment of Fire Hazard Due to Cable Splices in Raceways

4.2.1 Approach

In Section 4.1 of this anlaysis we have shown that the consequences of a postulated fire originating at a cable splice are acceptable. In this Section we will show that the likelihood of such a postulated fire in a splice is no greater than that in a cable itself.

A fire can originate in the insulating materials of a cable splice due to (1) excessive heating of the internal current-carrying metallic parts and (2) breakdown of the dielectric property of the insulating material. In order to assess the probability of such a fire, we need to evaluate the control of splices which have been used in raceways and those attributes which can cause such heating and dielectric breakdown.

The purpose of this analysis is not to prove that a cable splice would not generate a fire under any circumstances. The attributes which have the potential of starting a fire in a cable (e.g., failure of a circuit protective device to interrupt a fault current) may also start a fire at a cable splice. What will be shown is that introduction of a good quality cable splice in a continuous cable run does not create a weak link; that is, it is at least as reliable as the continuous cable run. We will therefore analyze all the attributes of a splice which are essential to ensure that consistently good quality splices are provided in the CPSES design and construction.

4.2.2 Review of Attributes and Their Effects

a. Attribute: Limited and controlled use of splices.

Discussion: The electrical erection specification, used for construction of CPSES, states that "Cables shall be installed without splices unless splices are specifically called for on the drawings, cable and raceway schedule, or when approved by the Engineer." All splices in CPSES are made per the electrical physical construction drawings. These drawings show typical details for each group of splices. The details identify the materials to be used to make field splices. Per procedures, the Engineer's approval of design changes is recorded via Design Change Authorizations. These procedures ensure application of quality control for site performed splices.

b. Attribute: Quality assurance and quality control.

Discussion: All materials involved in the splice connections are procured under the quality assurance program in compliance with the applicable ANSI N45.2 series of standards. Splices are made only by trained personnel as per the installation procedures. This procedure ensures adherence to splice manufacturer's recommended methods for installation and all construction drawings and specifications.

Use of the proper compression tool is assured since all AMP tools are matched by design to specific connectors. The tools are serialized and periodically checked for calibration by the on-site calibration laboratory. The tools are of the ratchet type and each crimp is brought to a full compression before the tool can be released. This ensures compliance with the manufacturer's crimping requirement and excludes any possibility of an under or over crimp condition. The tools are logged in and out of a tool room each day and are checked with a "go/no go" gauge at the end of each day. The craftsman records on the tool card the use made of the tool on each occasion. Tools that do not pass the "go/no go" test are taken out of service and the connectors installed with them that day are removed.

All Class 1E splices are inspected in accordance with quality control procedures and each splice inspection is documented by an inspection report. QC inspects all physical attributes that make up the splice, e.g., selection of proper connector and crimping tool, proper stripping of cable insulation and jacket, proper insertion of conductor into the connector barrel, adequacy of crimp, etc. If a bolted connection is being made, QC verifies correct bolting materials and witnesses torquing. For splices which are insulated at site by installation of heat shrink tubing, QC verifies proper cleaning of cable, proper selection of heat shrink material and proper installation. All these steps ensure that the splices are made per manufacturers' installation procedures and thereby produce qualified high quality splices. The QC inspection reports and the cable connection sign-off cards are maintained as a permanent QA record.

The above shows that adequate quality assurance and control measures are implemented to ensure a good quality splice.

c. Attribute: Effect of temperature on splice materials.

68 Discussion: All CPSES splices utilize uninsulated type connectors except for some splices which are covered by Group 2 (see scope). On smaller size wires, up to #12AWG, Group 2 permits use of Nuclear Preinsulated (Kynar insulated) Environmental Sealed splices. All uninsulated connectors are of "solistrand" or "ampower" type and furnished by AMP. All connectors of uninsulated or

9.1.3 SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

9.1.3.1 Design Bases

The Spent Fuel Pool Cooling and Cleanup System, a common system for | 68
both units, is designed in compliance with Title 10, Code of Federal |
Regulations, Part 50 Appendix A, General Design Criteria (GDC) 1, 2, |
4, 5, 44, 45, 46, 56, 61 and 63 [1], [2], [3], [4], [5], [6], [7] to |
perform the following principal functions: |

1. To remove heat generated by stored spent fuel elements from the station's spent fuel pools

2. To maintain the clarity and purity of water in the spent fuel | 68
pools, the transfer canal, the wet cask pit, the RWST, and the |
refueling cavities |

Two cooling loops are provided, each capable of simultaneously servicing both of the station spent fuel pools. Two cleanup loops are also provided [14]. System design parameters are presented in Table 9.1-1.

The water depth above the top of the fuel assemblies as well as the removal of fission products and other contaminants by the system's purification loop limits the dose rate at the surface of the pools to 2.5 mr/hr.

Two damaged fuel containers are provided to limit the fission product | 4
release from gross failed fuel assemblies. |

9.1.3.1.1 Spent Fuel Pool Cooling

The Spent Fuel Pool Cooling and Cleanup System is designed to limit the temperature of the spent fuel pools in the following cases:

1. The spent fuel pool water temperatures are maintained at less than 135°F and 120°F for one- and two-loop operation, respectively, based on decay heat generation from one-third of a core at 150 hr after shutdown, plus decay heat from one-third of a core 480 hr after shutdown, plus decay heat from fuel assemblies from a maximum number of previous refuelings in one spent fuel pool. At least 193 spaces in the spent fuel pools remain available to accept one full core in accordance with ANSI N18.2 [15].

68 2. Spent fuel pool water temperatures are maintained at less than 137°F for two-loop operation for Case No. 1 plus the decay heat load imposed by a full core at 150 hr after shutdown placed
68 in the remaining 193 spaces. For fuel assembly loading in the spent fuel pools versus time, see Table 9.1-4. One loop operation is not postulated for this case because a single active failure need not be considered for a full core unload.

3. The spent fuel pool water temperatures are maintained at less than 129°F and 117°F for one- and two-loop operation, respectively, with a heat load based on the decay heat generation from two thirds of a core (one third of a core at 150 hr after shutdown, and one third of a core at 480 hr after shutdown).

68 4. The spent fuel pool water temperatures are maintained at less than 135°F for two-loop operation with a heat load based on the decay heat generation from the Case No. 3 and an additional complete core.

The spent fuel pool water temperature in the above four cases is based on the component cooling water temperature of 105°F at the inlet to the spent fuel pool heat exchanger. In the unlikely event both units are in cooldown or one is in LOCA and the other in cooldown, the component cooling water temperature may be as high as 120°F. The effect would be 15°F high spent fuel pool temperatures for case Nos. 1 and 3, which raises the maximum spent fuel pool temperature from 135°F to 150°F for one-loop operation (Case No. 1) after normal refuelings of the two units. (Note that after full core unloading, the component cooling water temperature will be 105°F because of the reduced heatload caused by core removal).

Cases No. 3 and No. 4 assume no fuel is accumulating in the spent fuel | 68
pools. See Table 9.1-1 for water temperature in various cases (one |
loop out of service, two loops out of service).

9.1.3.1.2 Water Purification

Should a leaking fuel assembly have to be transferred from the fuel transfer canal to a spent fuel pool, a small quantity of fission products may enter the pool water. Two purification loops are provided for removal of such fission products and other contaminates by means of filtration and ion exchange. Each purification loop is capable of purifying flow from either the spent fuel pool cooling pumps or the refueling water purification pumps. The use of two loops ensures maintenance of acceptable activity and purity levels in the
spent fuel pools in the event of failure of one loop. Each | 68
purification loop limits the activity of fission and corrosion |
products in the spent fuel water to a maximum of 5×10^{-9} Ci/cm^3, |
exclusive of tritium, as stated in Table 9.1-1. Purification is |
sufficient to permit unrestricted access to the spent fuel storage area.

9.1.3.3 Safety Evaluation

Spent fuel pool water is cooled by two redundant cooling loops, each of which contains a pump, heat exchanger, piping, valves, and instrumentation. Normally only one pool is loaded and one-half of the installed cooling equipment is in service. In the event of a failure of spent fuel pool cooling pump or heat exchanger, the standby loop ensures the continuity of effective cooling.

In case of a closely spaced refueling of both reactors, the two cooling loops are used. In the event of a failure of one loop, the second loop ensures a minimum cooling and limits the water temperature below 135°

In the case of loading one and two thirds cores, the two cooling loops are required to operate and limit the water temperature below 137°F. | 68

To detect leakage through the spent fuel pool liner welds, a channel is provided in back of the welds to form a leak chase. Concrete troughs are formed under the welds in the floor plate. Sections of welds which are leaking can be determined by observing which leak chase the water is coming from before the leak chases merge into a common drain header. Once a section of weld has been determined to be leaking, the exact location can be determined by draining the pool and purging the leak chase with a gas other than air. A gas detection device can be used to pinpoint the exact location in the weld from which the gas is leaking.

Furthermore, as indicated in Table 9.1-1, the pool capabilities are sufficiently large so that an extended cooling outage is required before pool temperatures reach 180°F. Thus the system can be shut down safely for reasonable time periods for maintenance or replacement of malfunctioning components. The effect of the evaporation rate from the pools due to a loss of cooling are described in Section 9.4.

TABLE 9.1-1

SPENT FUEL POOL COOLING AND CLEANUP SYSTEM DECAY HEAT PARAMETER

	After Year 5 Refueling		After Year 8 Full Core Unloading		Spent Fuel Not Accumulating in Pools		
	Pool No. 1	Pool No. 2	Pool No. 1	Pool No. 2	2/3 Core in One Pool	1-2/3 Core in One Pool	: 68
Number of fuel assemblies stored	514	0	533*	560	130	323	
Decay heat produced	26.8	0	3.2	58.5	21.7	54.4	: 68
Number of cooling loops	2 1 0	2 1 0	2 1 0	2 1 0	2 1 0	2 1 0	
Maximum SFP temperature (F)	120 135 -	- - -	- 109 -	137 - -	117 129 -	135 - -	: 68
**Temperature rise time (hr)	- - 4.8	- - -	- - 55.8	- - -	- - 6.5	- - -	: 68

NOTES:

1. Storage capacity of each pool is 560 fuel assemblies.
2. The decay heat produced includes +10 percent uncertainty margin.
3. Number of fuel assemblies added to the pools by year is from Table 9.1-4.
4. Spent fuel pool water volume is 350,000 gal per pool.
5. Spent fuel activity level (exclusive of Tritium) is $\leq 5 \times 10^9$ Ci/cm^3

* Assumes that 19 fuel assemblies are stored in Year 6 refueling: all other fuel assemblies in SFP #1 are from prior to Year 6 refuelings. : 68

** Assuming both SFP cooling loops are inoperable, the temperature rise time is the time for the SFP water temperature to rise to 180 F from the SFP water temperature of one SFP cooling operation.

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TABLE 9.1-3

(Sheet 1 of 4)

SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

MAJOR COMPONENT PARAMETERS

Spent Fuel Pool Cooling Water Pump

Quantity (shared)	2	
Design pressure, psig	150	
Design temperature, F	200	
Design flow, gpm	3600	
Total dynamic head, ft water	209	68
Material	SS	

Refueling Water Purification Pumps

Quantity (per unit)	2
Design pressure, psig	150
Design temperature, F	200
Design flow, gpm	250
Minimum developed head, ft water	165
Material	SS

Spent Fuel Pool Skimmer Pump

Quantity (shared)	1
Design pressure, psig	150
Design temperature, F	200
Design flow, gpm	200
Fluid	Spent fuel pool water
Material	SS

Spent Fuel Pool Heat Exchanger

Quantity (shared)	2
Design heat transfer, btu/hr	13.6×10^0

TABLE 9.1-3

(Sheet 2 of 4)

SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

MAJOR COMPONENT PARAMETERS

	Shell	Tube	
Design pressure, psig	150	150	
Design temperature, F	200	200	
Design flow, lb/hr	2×10^{0}	1.80×10^{0}	
Inlet temperature, F	105	120	
Outlet temperature, F	111.8	112.5	
Fluid circulated	Component cooling water	Spent fuel pool water	
Material	CS	SS	

Spent Fuel Pool Demineralizer

Quantity (shared)	2	
Design pressure, psig	200	| 68
Design temperature, ^{o}F	200	
Design flow, gpm	150 (maximum = 273)	| 66
Resin volume, ft^3	50	
Material	SS	
Resin type	Rohm and Hass Amberlite IRN-150 or equivalent	

Spent Fuel Pool Filter

Quantity (shared)	2	
Design pressure, psig	150	
Design temperature, ^{o}F	200	
Design flow, gpm	150 (maximum = 273)	| 66
Filtration requirement	90% retention of particles above 5 microns	
Material, vessel	SS	

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TABLE 9.1-3

(Sheet 3 of 4)

SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

MAJOR COMPONENT PARAMETERS

Spent Fuel Pool Skimmer Filter

Quantity (shared)	1	
Design pressure, psig	150	
Design temperature, °F	200	
Design flow, gpm	200	
Filtration requirement	98% retention of particles above 5 microns	

Spent Fuel Pool Suction Screens

Quantity (shared)	4 (2 per pool)	
Design flow, gpm	3600	
Perforation, in.	0.08, slotted	| 66
Material	SS	

Spent Fuel Pool Skimmer

Quantity (shared)	4 (2 per pool)
Design flow, gpm	50

Spent Fuel Pool Skimmer Strainer

Quantity	1
Design flow, gpm	200
Maximum particle size strainer will pass	150 microns
Material	SS

Purification Loop Resin Trap

Quantity	2	
Design flow, gpm	250	
Perforation, mm	0.15	| 68
Material	SS	

TABLE 9.1-3

(Sheet 4 of 4)

SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

MAJOR COMPONENT PARAMETERS

Cask Pit and Transfer Canal Drain Pump

Quantity (shared)	1
Design pressure, psig	150
Design temperature, F	200
Design flow, gpm	100
Fluid	spent fuel pool water
Material	SS

Piping and Valves (Nuclear and Non-Nuclear)

Design pressure, psig	150
Design temperature, F	200
Material	SS

Refueling Cavity Skimmer Pump

Quantity (per unit)	1
Design pressure, psig	150
Design temperature, F	200
Design flow, gpm	100
Fluid	refueling water
Material	SS

TABLE 9.1-4

(Sheet 1 of 2)

NUMBER OF FUEL ASSEMBLIES IN SPENT FUEL POOLS BY YEAR

Number of Assemblies Added Per Year After Plant is Operational 68

	0	1	2	3	4	5	6	7	8	Total Normal	+ 1 Core**	Total Abnormal	68
Unit 1													
Unit 2													
Total	0									0	193	193	
Unit 1		64											
Unit 2		0											
Total	0	64								64	193	257	
Unit 1			64										
Unit 2			0										
Total	0	64	64							128	193	321	
Unit 1				65									
Unit 2				64									
Total	0	64	64	129						257	193	450	
Unit 1					64								
Unit 2					64								
Total	0	64	64	129	128					385	193	578	
Unit 1						64							
Unit 2						65							
Total	0	64	64	129	128	129				514	193	707	
Unit 1							65						
Unit 2							64						
Total	0	64	64	129	128	129	129			643	193	836	

TABLE 9.1-4

(Sheet 2 of 2)

NUMBER OF FUEL ASSEMBLIES IN SPENT FUEL POOLS BY YEAR

Number of Assemblies Added Per Year After Plant is Operational | 68

Unit 1	0	1	2	3	4	5	6	647	8	Total Normal	+ 1 Core**	Total Abnormal	
Unit 2								64					
Total	0	64	64	129	128	129	129	128		771	193	964	
Unit 1									64				
Unit 2									65				
Total	0	64	64	129	128	129	129	128	129	900	193	1093*	

| 68 (header row)

* Space in lieu of 27 storage spaces remains unfilled and can be used for storage of failed fuel, irradiated components, and so on.

** Number of fuel storage cells reserved for emergency removal of entire core from one unit only

Assumptions:

1. Unit 1 starts up in Year 0. The first refueling takes place after one year of plant operation . | 68
2. Unit 2 starts up two years after Unit 1. The first refueling takes place after one year of plant operation. | 68
3. Approximately one-third of a core is unloaded every year. One full core per unit is unloaded every 3 years for inspection of reactor vessel and internals.
4. No fuel is shipped offsite during this time.

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68 The maximum allowable SSWS supply temperature is 115°F. The maximum calculated SSWS supply temperature is 115°F during LOCA conditions. One system for each unit is provided as described in
10 Section 9.2.1.2.

The SSWS is designed on the basis of the following:

1. Flow from one of the two redundant trains is continuously delivered to one of the two emergency diesel generators of each unit.

68 2. Flow is continuously delivered to one of two lube oil coolers per unit for the safety injection and centrifugal charging pump and two of four containment spray pump bearing oil coolers as shown on Figure 9.2-1 (Sheet 2 of 2).

68 3. Normal cooldown of one unit is accomplished with two SSWS pumps delivering water to two CCWS heat exchangers to cool the reactor coolant system (RCS) from 350°F to 140°F in 24 hours. CCWS outlet temperature from the heat exchangers is maintained below 122°F and it will gradually resume the 108°F design value towards the end of the cooldown time. The SSWS outlet temperature is maintained below 122°F.

4. An orderly shutdown can be achieved with one SSWS pump and one CCWS heat exchanger, although at a slower cooldown rate than outlined in Item 3 above.

5. One SSWS pump in conjunction with one CCWS heat exchanger provides one unit with enough cooling capacity for safe recovery after a LOCA (see Section 9.2.2.1).

underground with an amount of earth cover sufficient to protect the piping from tornado missiles. Other components are located inside the Auxiliary Building, which is a Category I structure and is designed against tornadoes. Pump motors, valve operators, and controls in the pumphouse are located above the highest postulated water level in the SSI. Valve operators and controls inside the Auxiliary Building also are located above the highest water level that might occur due to equipment failures within the building.

All of the previously mentioned conditions make the system capable of withstanding adverse environmental conditions, such as postulated earthquakes, tornadoes, and tornado missiles [1],[2]. (For details on seismic Category I structures, see Sections 3.2.1, 3.3, and 3.5.)

The SSWS is a moderate energy piping system. For a discussion of postulated pipe ruptures in this system, refer to Section 3.6 [14].

Pumps and pump motors inside the pumphouse are physically separated | 68
from each other by walls, as shown on Figures 1.2-45 and 1.2-46 . |
These walls are designed to preclude coincident damage to redundant equipment in the event of a postulated pipe rupture, equipment failure, or missile generation.

Flooding is not considered possible because the floor of the pump | 41
compartments is at el. 796 ft, which is above the probable maximum |
water level of 791.3 ft. In the event of a pipe rupture within the |
pump compartment, the water drains back to the SSI through large holes |
in the compartment floor.

The single-intake bay for all pumps, downstream of the intake channels and traveling screens, ensures flow to all pumps in the service water pumphouse.

The CCWS is required to be operating during all phases of plant operation including startup, power operation, shutdown, refueling, and the injection and recirculation phases following a loss-of-coolant accident (LOCA).

The design cooldown rate, based on reducing the temperature of the 68
reactor coolant from 350°F to 140°F in 24 hours will be achieved
using two CCWS pumps and two CCWS heat exchangers. A slower but
acceptable cooldown rate may be accomplished using only one CCWS pump
and one CCWS heat exchanger. Under both conditions the CCWS supplies 68
cooling water to its RHR heat exchanger at a maximum temperature of
122°F during its early part of cooldown. The supply water
temperature will gradually return to 108°F towards the end of
cooldown.

Because the system is required to perform its safety function during the short-term and long-term plant accident conditions, the safety related passive components as well as the active components are designed to meet the single failure criteria [12]. An analysis of postulated cracks in moderate energy systems is in Section 3.6. All ANSI Class 2 and 3 components of the CCWS are designed to meet seismic Category I requirements. Equipment that is necessary for shutdown and equipment that is required to mitigate the effects of an accident is supplied with emergency diesel power, should normal and offsite power sources fail.

The CCWS is designed in accordance with the following criteria, regulatory guides, and codes:

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criteria for Nuclear Power Plants GDC 2 [1], 4 [2], 5 [3], 44 [5], 45 [6], 46 [7], 56 [8], and 57 [9]

2. NRC Regulatory Guides 1.26 [10], 1.29 [12], and 1.48 [13]

52 f. Two spent fuel pool heat exchangers (common)

52 g. One boron recycle evaporator package consisting of one distillate cooler, one evaporator condenser, and one vent condenser (common)

52 h. One waste evaporator package consisting of one distillate cooler, one evaporator condenser, and one vent condenser (common)

52 i. One floor drain evaporator package consisting of one distillate cooler, one evaporator condenser, and one vent condenser (common)

j. Four reactor coolant pump packages, each consisting of one upper bearing lube oil cooler, two motor air coolers, one lower bearing lube oil cooler, and one thermal barrier cooler (located inside Containment)

k. One excess letdown heat exchanger and one reactor coolant drain tank heat exchanger (CVCS equipment located inside Containment)

52 l. Four ventilation chillers (common)

m. One letdown chiller package condenser

n. One rotary instrument air compressor package consisting of one intercooler, one aftercooler and one oil cooler. | 68

o. One reciprocating instrument air compressor and one aftercooler | 68

p. Process sample coolers

q. Reactor coolant post accident sampling system sample cooler | 66

Items a to k inclusive above are ANSI Safety Class 3 components; items 1 to q inclusive above are non-safety-class-components. Common | 52

a. All pumps and remote-operated valves are operated from the Control Room.

b. All remote-operated valves are supplied with open-close position indicating lights in the Control Room. Both lights are on when the valve is in the intermediate position.

68 c. CCWS pumps for both trains A & B have control switches located on the hot shutdown panel (HSP). In addition, CCWS Pump Train A transfer switch (CR-HSP) is located on the shutdown transfer panel, while CCWS pump train B transfer switch is located on the hot shutdown panel. Local control position alarms are provided in the Control Room.

d. If a low CCWS pressure signal is received, the redundant CCWS pump and its associated SSWS pump automatically start. The CCWS pump and the SSWS pumps are train associated pumps.

e. If a SSWS pump automatic start on low SSWS pressure occurs, the corresponding CCWS pump is automatically started to ensure correct alignment of the safeguard train.

2. Makeup

Makeup to the CCWS Surge tank is normally obtained from the Demineralized Water Storage Tank (DWST) by remote manual operation of valves for either surge tank compartment. The operator is informed of the need for makeup by a low-level alarm for each compartment. The demineralized water pumps are used to transfer makeup water. If the alarm signal does not reach the operator because of equipment failure or operator inattention, makeup is automatically supplied by the emergency makeup system from the Reactor Makeup Water Storage Tank (RMWST). A low-low-level signal opens valves for either surge tank compartment and actuates a low-low-level alarm in the Control

60 6. Ventilation Chillers and Letdown Chillers

60 Flow to the ventilation chillers is stopped automatically when it reaches an abnormally high value indicative of leak or system malfunction. In addition, the CCWS inlet and outlet valves to the ventilation chiller and letdown chiller condensers are closed automatically on an "S" signal.

66 The cooling of the ventilation chiller is based on the condenser refrigerant pressure. A pressure indicating controller (PIC) on the refrigerant side of the condenser adjusts the CCW regulating valve (200-2000 gpm) to the condenser and inversely the CCW bypass valve (1800 - 0 gpm) for a total flow around/across the condenser of 2000 gpm.

7. Thermal Barrier Cooler

66 Flow to the thermal barrier of any reactor coolant pump is stopped automatically with control grade equipment whenever the outlet temperature reaches an abnormally high value indicative of a thermal barrier break. Output flow and temperature indication is provided by Class 1E control board equipment to permit operator closure of the containment isolation valves in response
68 to diverse high flow or temperature indication. In addition, control grade alarms are provided for output high flow and temperature.

8. Drain Tank Pumps

CCWS drain tank pumps are interlocked to stop on low drain tank level.

9. S Signal

When an S signal occurs, the following operations take place:

a. Both CCWS pumps are started.

b. The Containment CCWS drain tank pumps stop.

c. The pump recirculation valves close.

d. The RHR heat exchanger CCWS isolation valves open partially.

e. CCWS Containment phase "A" isolation valves close as described in Section 6.2.4. | 52

f. The ventilation chiller condenser's and letdown chiller condenser's isolation valves close. | 60

10. P Signal

When a P signal occurs the following operations take place.

a. The RHR heat exchanger CCWS isolation valves open fully.

b. The Containment spray heat exchanger CCWS isolation valves open.

c. CCWS containment Phase "B" isolation valves close as described in Section 6.2.4. | 52

d. The safeguards loop isolation valves close. | 10

e. The non-safeguards loop isolation valves close. | 52

9.2.2.5.4 Radiation Monitoring

52 Radiation monitors are provided on the return headers in each loop of
66 the CCWS. Any of these monitors can trigger a high-activity-level alarm signal in the Control Room. A temperature switch is provided
46 to trip the monitor sample pump if the temperature exceeds 120°F to protect the detector.

9.2.3 DEMINERALIZED AND REACTOR MAKEUP WATER SYSTEM

9.2.3.1 Design Bases

68 The design of the Demineralized and Reactor Makeup Water System is in accordance with the following criteria: 10CFR50, Appendix-A, General Design Criteria (GDC) 1, 2, 3, 4, 5, 52, 54, and 56; NRC Regulatory Guide 1.26 and 1.29; and Branch Technical Position APCSB 3-1. The related electrical systems are designed to comply with the requirements of regulatory guides, standards and other documents as
68 described in Section 8.1.4. The related Containment Penetrations are designed to comply with the requirements described in Section 6.2.4. The Demineralized and Reactor Makeup Water System is a single system designed to provide an adequate supply of deaerated demineralized water of reactor coolant purity to other systems as makeup and to provide other plant demineralized water requirements for both units of the CPSES.

52 The Demineralized Water Storage Tank (DWST) receives water from the Water Treatment System. When treated water enters the DWST, the quality of water in the tank is consistent with the water chemistry listed in Table 9.2-9 for demineralized water saturated with oxygen. This water normally serves as a source of demineralized water for initial filling of the Reactor Coolant Systems (RCSs) including the

pressurizer relief tanks, the boric acid tanks, the Refueling Water Storage Tanks, the CCWSs, the Turbine Plant Cooling Water (TPCW) Systems, and the spent fuel pools. Demineralized water is also transferred to condenser hot wells, Condensate Storage Tanks, and Reactor Makeup Water Storage Tanks (RMWSTs) as required by their respective levels. The demineralized water is deaerated in a vacuum deaerator to remove dissolved oxygen to 0.1 mg/l before it is transferred to the Condensate Storage Tanks and the RMWSTs.

Demineralized water is also provided via a transfer pump to the 56
containment pre-access filtration units charcoal deluge system, and to
the fire protection standpipe for the containment hose stations. For
a detailed description of the Fire Protection System refer to Section
9.5.1.

Demineralized water is also required to perform turbine generator 66
primary flow rate testing during refueling outages. The source of
the demineralized water is provided at the inlet line to the turbine
plant cooling water head tank.

The RMWST serves as a source of reactor makeup water, containing trace amounts of tritium, which has been recycled for the RCS. Demineralized deaerated water is provided to the RCS and the auxiliary equipment, where the presence of tritium is not objectionable. These include supply to evaporators, gas strippers, pumps, demineralizer tanks, and pipelines for cleaning and flushing operations, and to the RCS as a diluent. The water chemistry for the reactor makeup water is listed in Table 9.2-10.

9.2.3.2 System Description

The Water Treatment System (which includes surface water pretreatment) 68
is used to produce demineralized water, of quality per Table 9.2-9,
through two trains of 150 gpm capacity each, which is fed directly to
the DWST. Figure 9.2-4 shows the paths used for treating lake water
or well water in the Water Treatment System.

The mixed-bed effluent flows directly to the DWST, the amount of which is controlled by the level in the tank.

68 Sulfuric acid and sodium hydroxide are used to regenerate the Water
Treatment System demineralizers. The spent regenerants are pumped to
52 the Wastewater Management System for treatment prior to discharge, as
discussed in Section 9.2.8.2.

Water from the DWST is used as a source for initial filling of the RMWST and Reactor Makeup Water System.

The RMWST accommodates the makeup requirements resulting from a cold
68 shutdown, followed by startup from cold conditions. A minimum
quantity of water must be available during normal plant operation to
achieve and maintain a safe, cold shutdown.

68 The RMWST is also required to provide a seismic category I supply of
makeup water for the Chemical and Volume Control System, the Safety
Chilled Water System, the Component Cooling Water System, and the
Spent Fuel Pool Cooling System.

As a minimum, one reactor makeup water pump per unit operates constantly.

If the system load demand is low, the pump recirculation insures that sufficient flow of tank water passes through the outdoor piping to prevent icing during severe outdoor temperatures.

The Demineralized and Reactor Makeup Water System is shown schematically in Figure 9.2-5.

9.2.3.3 Safety Evaluation

68 The Demineralized and Reactor Makeup Water System has sufficient
storage capacity (400,000 gal in the DWST) to supply makeup water to
the system should the Water Treatment System be out of service for
short periods of time. Spare pumps and two trains of ion exchangers
are provided to ensure the reliability of the Water Treatment System.

The Water Treatment System does not contain, treat, or produce any radioactive material in its operation. Therefore, any waste produced 60
by this system can be deposited directly into the Low Volume Waste Treatment Facility.

The RMWST, reactor makeup water pumps, associated valves and piping, 52
and all connections to the boric acid blenders, the safety chilled water surge tanks, the component cooling water surge tanks, and the spent fuel pools are designated seismic Category I and Safety Class 3. Containment penetrations are described in Section 6.2.4. Components of the Demineralized and Reactor Makeup Water System are compatible with all chemicals used throughout the system. Stainless steel piping is used throughout the system as a deterrent against corrosion in the transfer of corrosive fluids.

The RMWST is designed to withstand the effects of natural phenomena such as safe shutdown earthquake (SSE), probable maximum flood (PMF), and tornado missile. All components located in the Containment, Safeguards, Auxiliary, Fuel and Electrical, and Control buildings are located inside seismic Category I structures, which are also designed to withstand tornado effects. These seismic Category I structures are capable of withstanding adverse environmental conditions such as postulated earthquakes, tornadoes, and tornado missiles. (For details on seismic Category I structures, see Section 3.2.1, 3.3, and 3.5.)

Any failure of non-safety-class-design equipment associated with the 68
Demineralized and Reactor Makeup Water System will not cause any failure of safety-related systems or components.

Flow restriction devices are provided in the discharge of each reactor 52
makeup water pump. These orifices are designed so that with the worst case alignment of the Reactor Makeup Water System to Reactor Coolant System (RCS), via the volume control tank, the maximum rate at which unborated water can be added to the RCS satisfies the design criteria of the Boron Dilution Mitigation System. For further information see section 15.4.6.

9.2.3.4 Testing and Inspection Requirements

68 The equipment in the Demineralized and Reactor Makeup Water System is initially inspected and tested to insure system integrity and completeness. The inspection includes the following:

1. Pumps and Motor Drives

   Each pump is started and runs for sufficient time to insure its proper operation. Maintenance is provided as required.

2. Valves

   Each valve is operated through its complete range to ensure that it is in normal operating condition.

3. Safety Class Isolation Valves

   An electrical signal is transmitted to ensure that isolation valving systems (safety-related) are operating properly.

4. Instruments and Annunciators

   Operation is checked periodically for proper operation and accuracy.

9.2.3.5 Instrumentation Requirements

68 Control for the Demineralized and Reactor Makeup Water System is monitored from the main control panel or from local panels: the potable and demineralized water panel (PDP) or the water treatment panel (WTP), which are located adjacent to each other.

68 The Demineralized and Reactor Makeup Water System is controlled from a local PDP. System operation is designed to be fully automatic, but with provision

for manual control. The final water quality of the Water Treatment System is continuously monitored for conductivity and silica. High conductivity and high silica concentration are alarmed locally and in the Control Room. Deviations from specified water quality trip the demineralizers to prevent poor | 68

9.2.5.3 Safety Evaluation

The heat rejection capabilities of the SSI are a function of ambient conditions and the volume/surface area relationship of this body of water. A thermal analysis was performed, using a longitudinal-vertical, time-varying hydrodynamic transport mode. 10

An analysis of a 30 year record of offsite and onsite meteorological data revealed July and August 1974 as the most severe period for ultimate heat sink performance. The simulated DBA was begun on July 15, 1974, eight days prior to the maximum SSI temperature due to meteorological conditions. The pond takes eight days to reach its maximum temperature due to the DBA heat loads. Following this day of maximum temperature, the simulation was continued for 30 days, for a total 39 days. Onsite meteorological data for this entire period, including a 43 day period preceding the DBA for model initialization, were used. 10

The meteorological data for this period are presented in Table 2.3-7.

Heat rates of the cooling water discharged onto the SSI are based on two safety trains in operation per unit, and are developed from the nuclear steam supply system (NSSS) manufacturer's functional requirements, design criteria for residual decay heat removal, and from balance-of-plant (BOP) heat load requirements. In generating equations for the spent fuel pool heat load, it was assumed that the DBA occurs in one unit during normal two unit operation 8 years after Unit 1 startup and after 1/3 core refuel of both units; specifically 150 hours after refueling of the second unit and 480 hours after refueling of the first unit. 68

The decay heat rate, however, is based on a reactor operating time of 68 16,000 hours for all regions. This results in a high heat input which reflects a conservative spent fuel pool heat loading in conjunction with a conservative containment spray and RHR heat loading.

1. Compliance with NRC Regulatory Guide 1.27

The intent of NRC Regulatory Guide 1.27 is met by SSI. The thermal ability of the SSI to act as an ultimate heat sink for 30 days is covered in Item 3, Thermal Performance Evaluation. As a single source ultimate heat sink serving two units, the SSI is able to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods, taken individually, without loss of capability to perform its safety functions. The natural phenomena and their magnitude are selected in accordance with their probability of occurrence, and designs are based upon the most severe of the natural phenomena recorded for the site, with an appropriate margin to account for uncertainties in historical data. Such phenomena and design criteria are discussed in Sections 2.4, 3.3, and 3.4.

10 Exceptions to Regulatory Guide 1.27, Rev. 2, are discussed in Appendix 1A(B).

2. Hydraulic Performance

The SSI is supplied with a bleed flow from the Circulating Water System during all normal operating conditions. This flow is directed into the service water pump intake structure and creates a blowdown flow from the SSI to the SCR through the equalization channel. If, as a result of an earthquake, the SCR dam fails, the equalization channel invert maintains the water level in the SSI at elevation 769 ft 6 in. Surface area and volume in the SSI as a function of elevation is discussed in Section 2.4.

Hydraulic short circuiting is prevented by the physical separation of the intake structure and discharge piping outfall and the orientation of the discharge. The intake and discharge points are over 1800 ft apart, and the exit velocity of the discharge water carries it upstream initially, away from the area of the intake structure, allowing extra time for the transfer of heat to the atmosphere.

3. Thermal Performance Evaluation

The evaluation of the SSI was performed using a computer code based on a longitudinal-vertical, time-varying hydrodynamic transport model to predict the temperature of a thermally loaded pond. The result predicts maximum SSI water temperature based on conservatively calculated heat loads to the SSI following a LOCA. 68

68

The intake temperature reaches a maximum on the evening of the eighth day at $115^{\circ}F$ (two train). 68

Component cooling water temperature is a function of heat load on the system and the service water temperature. Component cooling water temperature in the DBA unit peaks at $131^{\circ}F$, approximately one half hour after the DBA, with service water temperature at $102^{\circ}F$. The maximum component cooling water temperature of the shutdown unit, with 68 10

68 only one heat exchanger operable, is $122^{\circ}F$ and occurs upon initiation of RHR, four hours after the DBA occurred in the other unit. The preceding maxima of $131^{\circ}F$, and $122^{\circ}F$ are below the design basis temperatures of $135.0^{\circ}F$ and $122^{\circ}F$ respectively, indicating that sufficient cooling is present.

10 During the postulated 100-year drought conditions and after 40 years of sedimentation, the SSI is determined to have 284-acre feet of water. The consumption of SSI water during the 39-day postaccident shutdown cooldown period amounts to 71 acre-feet (one train) and 80 acre-feet (two train), resulting in a decrease in surface elevation of 2.4 ft and 2.7 ft, respectively, allowing adequate margin for post-30-day operation without exceeding the service water pump submergence requirements. Refer to Section 2.4.11.6 for further discussion of the heat sink dependability requirements.

Consumption of SSI water by the Auxiliary Feedwater System for supply to the steam generators, if a failure of the Condensate Storage Tank is assumed, amounts to only 0.63 acre-feet, based on a 60 gal of water per thermal MW rating of the steam supply system.

68

10 The time variation of decay power, based on the ANS 5.1 fission product curve, is corrected for a finite operating time and
68 includes the ANS uncertainty factors. The related curve has been used to develop the values of decay heat rates shown in Table 9.2-13.

TABLE 9.2-9

DEMINERALIZED WATER ANALYSIS

Constituent	Concentration	
Specific conductivity, micromhos/cm	<1.0	57
Soluble silica as SiO2, ppm	<0.1	68
Sodium, as Na, ppm	<0.01	55
Suspended solids, ppm	<0.1	55
pH at 25°C	6.0 to 8.0	55
Chloride, ppm	Negligible	57
Fluoride, ppm	Negligible	57

TABLE 9.2-11

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TABLE 9.2-14

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FIGURE 9.2-2

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FIGURE 9.2-8

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FIGURE 9.2-10

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9.4 AIR CONDITIONING, HEATING, COOLING, AND VENTILATION SYSTEMS

9.4.1 CONTROL ROOM AREA VENTILATION SYSTEM

9.4.1.1 Design Bases

The Control Room HVAC and filtration systems are designed to maintain suitable and safe ambient conditions for operating personnel and equipment during all modes of operation including post-DBA conditions, in the following areas of the Control Building.

Areas on floor elevation 830 ft 0 in.:

East Control Room | 68
West Control Room | 68
Console and Control Room Unit 1 | 68
Console and Control Room Unit 2 | 68
Instrument Room Unit 1 | 68
Instrument Room Unit 2 | 68
Computer Room Unit 1 | 68
Computer Room Unit 2 | 68
File Room | 68
Production Supervisor's Office | 68
Corridor | 68
Toilet | 68
Locker Room | 68
Kitchen and Janitor Closet | 68
Charts and Supplies Storage Room | 68

Areas on floor elevation 840 ft 6 in.:

68 | Technical Support Center
68 | Observation Area
68 | Offices
68 | Corridor
68 | Electrical Equipment Corridor

Areas on floor elevation 854 ft 4 in.:

68 | Control Room Air Conditioning System mechanical equipment rooms, trains A and B.

66 | The Control Room, located on elevation 830 ft 0 in, is maintained at $75^{\circ}F$ ($\pm 5^{\circ}F$) and 35-50 percent relative humidity. The Control Room HVAC and filtration equipment rooms are maintained between $40^{\circ}F$ and $104^{\circ}F$. Miscellaneous areas on elevations 830 ft and 840 ft are also maintained between $40^{\circ}F$ and $104^{\circ}F$. Other system design parameters are presented in Tables 9.4-1 and 9.4-2.

As described in the following paragraphs, the system is provided with sufficient redundancy in equipment and power supplies to enable the system to sustain a single failure of an active component without loss of function.

1. The system is equipped with four modular air-conditioning units. Each unit is rated at 50 percent of the Control Room HVAC and filtration systems capacity. Each pair of air-conditioning units is powered from an independent Class 1E bus and is physically separated by a dividing fire wall in the Control Room HVAC and filtration mechanical equipment room.

Any contaminants that have entered the Control Room prior to full closure of the dampers are removed by the emergency filtration unit. Removal of heavy concentrations of contaminants caused by a fire in the Control Room is accomplished by portable smoke ejectors.

Audible and visual alarms are provided in the Control Room to alert the operator in the event of system malfunction or unsafe conditions.

53 The Control Room air-conditioning system is designed to automatically
switch to the emergency recirculation mode of operation described in
Subsection 9.4.1.2 should the offsite power fail (for operator
68 convenience only). The system also automatically switches to this
mode of operation upon receiving a Control Room ventilation high-
radiation signal or a safety injection signal. The radiation signal
originates from radiation monitors which sample the Control Room
intake air vents. Radiation detectors of this type are discussed in
31 Section 11.5.

68 Two redundant chlorine gas detectors are located at each of the fresh
air intakes which are located in the north and south walls of the HVAC
mechanical equipment room. The sensors are located upstream of their
respective intake damper arrangements and upstream from the branch
takeoffs leading to the makeup air and pressurization isolation
68 dampers. Because of their functional requirements, the sensors
associated with the chlorine detectors are located outside the missile
protected area. No credit is taken for the chlorine detectors during
or after a tornado accompanied by a chlorine release. Two additional
chlorine detectors are provided in the circulating water chlorine
storage area.

The iodine absorbers are manufactured from impregnated activated carbon and are used for radioiodine compound removal.

59 The iodine adsorbers construction and efficiencies comply with NRC
66 Regulatory Guide 1.52 (See Appendix 1A(B)). Additional data
concerning iodine adsorbers is presented in Table 9.4-6.

Anticipated efficiencies of filters are in accordance with Table 9.4-4. | 66

An analysis of postaccident dose levels in the Control Room is presented in Section 15.6.5.4. | 68

Sufficient redundancy in equipment and power supplies enables the system to sustain a single failure without total loss of function. The emergency pressurization and recirculation filter trains, plus supply and exhaust fans and the associated dampers, are completely redundant. Four parallel air-conditioning units are used to provide a 100 percent (two air conditioners) standby feature.

The probability of an electrical fire in the Control Room is low because of the low-voltage and flame-retardant cables used. Fire dampers automatically isolate the affected areas to prevent spreading of the fire and shut off the oxygen supply. The charcoal adsorbers are also provided with a fire protection water deluge system which operates automatically upon receipt of a high- temperature signal from an adsorber bed. Actuation of the system also activates an alarm in the Control Room. A failure mode and effects analysis is presented in Table 9.4-8.

Alarms in the Control Room alert the operator to any system malfunction so that he can manually actuate the necessary standby units. The maximum operational temperature limit for Control Room instrumentation is 120°F. This temperature is not reached during the short periods of system malfunction prior to the actuation of the standby units.

During fuel handling operations, the system is sized to maintain the concentrations of airborne radioisotopes below the MPC levels specified in Appendix B to 10 CFR Part 20. (See Section 12.2.6.) For a discussion of a fuel handling accident, see Section 15.7.4.3.

56 The primary plant ventilation system is described in Section 9.4.3.

68 The Fuel Building air exhaust ductwork is ANS Safety Class 3, seismic
Category I, and NNS, seismic Category II. The emergency fan coil
56 units, which are located in the safety related pump rooms, are seismic
Category I and ANS Safety Class 3. The ventilation failure modes are presented in Table 9.4-9. The spent fuel pool exhaust fans are non-nuclear safety related and non-seismic category I. Operation of these fans is not required to limit Fuel Building exfiltration during normal operation.

9.4.2.4 Inspection and Testing Requirements

See Subsection 9.4.1.4.

Chilled water for the cooling coils is provided by the plant ventilation chilled water system. (See Appendix 9.4E.) Heating coils are electric. The exhaust system is seismic Category I up to the fan 56
discharge. The air supply system is seismic Category I except for the fans and the dampers which are seismic Category II.

9.4.3.3 Safety Evaluation 41

The reliability and safety of the Auxiliary Building ventilation 41
system is ensured by the following features:

1. Instrumentation and controls which incorporate audible and visual 41
alarms in the Control Room facilitate continuous monitoring of performance and alert the operator in the event of system malfunction.

2. Standby supply units can be remotely actuated from the Control 41
Room.

3. Deleted 53

4. The Auxiliary Building exhaust system is of ANS Safety Class 3 68
and seismic Category I, and NNS, seismic Category II design.

5. Failure modes for dampers are set so that they do not render the 56
system inoperable.

6. Adequate cooling of safeguard equipment is ensured by the 56
operation of the emergency fan coil units.

7. Exhaust air is passed through iodine adsorber filter beds prior 56
to its discharge. It is also monitored by an in-line radiation monitor at the ventilation duct outlet.

2. Emergency fan coil units are supplied with chilled water from the 56
safety chilled water system. Each chilled water system train
and the auxiliary cooling units it serves are powered from the
same train Class 1E bus.

3. Instrumentation and controls which incorporate audible and visual
alarms in the Control Room facilitate continuous monitoring or
system performance and alert the operator if the system
malfunctions.

4. Failure modes for isolation valves and dampers are set so that 56
their failure does not render the system inoperable. The
appropriate safeguards building exhaust dampers (Figure 9.4-2 Sh.
1) are locked open.

5. The cooling units, and safety features chilled water system are 68
of seismic Category I and ANS Safety Class 3 design. The
exhaust system is of ANS Safety Class 3, seismic Category I, and
NNS, seismic Category II, design. The supply system components
are designed to appropriate seismic criteria, where necessary, to 56
negate the possibility of these components interfering with the
operation of safety-related components.

6. Exhaust air is passed through iodine adsorber beds prior to its
discharge. Data concerning the ESF and non-ESF iodine adsorbers 4
are presented in Table 9.4-6.

7. In case of a DBA and subsequent operation of the ECCS, the 56
emergency fan coil units operate on a recirculation basis and
provide cooling of the ECCS pumps.

9.4.5.4 <u>Inspection and Testing Requirements</u>

See Subsection 9.4.1.4.

TABLE 9.4-2

(Sheet 1 of 3)

DESIGN CONDITIONS - INDOOR

Building or Area	Normal Plant Conditions: Maximum DB(°F)	Minimum DB(°F)	Relative Humidity (%)	
				68
Auxiliary Bldg. (AB)				66
1. Elevator Machine Room	122	40	*	66
2. Heat Exchanger Area(el. 790'-6")	122	40	*	66
3. Valve & Piping Area (el. 790'-6")	122	40	*	66
4. Operating Valve Rooms (el. 822')	122	40	*	66
5. Auxiliary Steam Drain Tank				66
Equipment Room (el. 790'-6")	122	40	*	66
6. Valve Rooms (el. 810'-6" and				68
831'-6")	122	40	*	68
7. Operating Valve Room (el. 862'-6")	122	40	*	68
8. ESF Pump Rooms	122	40	*	66
9. All other areas	104	40	*	66
Electrical and Control Bldg. (ECB)				66
1. Control Room	80	70	35-50	66
2. Mechanical Equipment Rooms	104	400	*	66
3. UPS & Distribution Rooms	104	40	*	66
4. Uncontrolled Access Area	104	40	*	66
5. Battery Rooms	104	70	*	66
6. Chiller Equipment Areas (el. 778')	122	40	*	66
7. All other Areas	104	40	*	66

*Uncontrolled

TABLE 9.4-2
(Sheet 2)

DESIGN CONDITIONS - INDOOR

Building or Area	Normal Plant Conditions			
	Maximum DB(^{O}F)	Minimum DB(^{O}F)	Relative Humidity (%)	68
Fuel Handling Building				66
1. Stairs (el. 810'-6" & 841')	122	40	*	66
2. Spent Fuel Pool Cooling Heat				66
Exchanger & Pump Rooms	122	40	*	66
3. All other Areas	104	40	*	
Reactor Containment Bldg. (RCB)				66
1. Outside Missile Barrier	120	50	*	66
2. Inside Missile Barrier	120	50	*	66
3. CRDM Shroud (Air Temperature)	163 (outlet)	50	*	66
	120 (inlet)	50	*	66
4. CRDM Platform	140	50	*	66
5. Detector Well and Reactor Cavity	135**	50	*	66
6. Reactor Coolant Pipe Penetrations	200	50	*	66
Safeguards Bldg. (SGB)				66
1. MS & FW Piping Area	104	40	*	66
2. Valve Isolation Tank Rooms				66
(el.790'-6")	122	40	*	66
3. Valve Rooms (el. 790'-6")	122	40	*	66
4. RHR/Cont. Heat Exchanger				66
Rooms (el. 790'-6")	122	40	*	66
5. ESF Pump Rooms	122	40	*	66
6. All Other Areas	104	40	*	66

*Uncontrolled | 66

**Occasional temperature spikes of 175^{O}FDB may occur. | 66

TABLE 9.4-2
(Sheet 3)

DESIGN CONDITIONS - INDOOR

Building or Area	Normal Plant Conditions Maximum DB(°F)	Minimum DB(°F)	Relative Humidity (%)	
				68
Diesel Generator Bldg. (DGB)				66
1. Equipment Room	130	40	*	66
2. All Other Areas	122	40	*	66
Service Water Intake Structure (SWIS)				66
1. SWIS-All Areas	127	40	*	66
Turbine Bldg. (TB)				66
1. Switchgear Area	104	40	*	66
2. ERF-Computer Battery Room	104	40	*	66
3. CAS Room, CPU & UPS Rooms	75	68	35-50	66
4. Battery Rooms & HVAC Equipment Rooms	104	40	*	66
5. Office and Service Area A/C	80	70	35-50	66
6. Laboratories	75	68	35-50	66
7. Hot Shop	85	68	*	66
8. All Other Areas	122	40	*	66
Miscellaneous Building (MB)				66
1. Alternate Access Point (AAP) Bldg.	80	68	*	66
2. Circulating Water Intake Structure Chlorination Bldg.	115	40	*	66
3. SWIS - Chlorination Building	115	40	*	66
4. Switchyard Relay House	80	68	*	66
5. Security Office	80	68	*	66
6. Maintenance and Administration	80	68	*	66

*Uncontrolled | 66

9.4B SERVICE WATER INTAKE STRUCTURE VENTILATION SYSTEM

9.4B.1 DESIGN BASES

The heating and ventilation system is designed to maintain the ambient temperatures within the Service Water Intake Structure between 40°F and 132°F during all normal and emergency modes of operation, including plant shutdown and refueling conditions. | 68

9.4B.2 SYSTEM DESCRIPTION

The Service Water Intake Structure heating and ventilating system consists of the following:

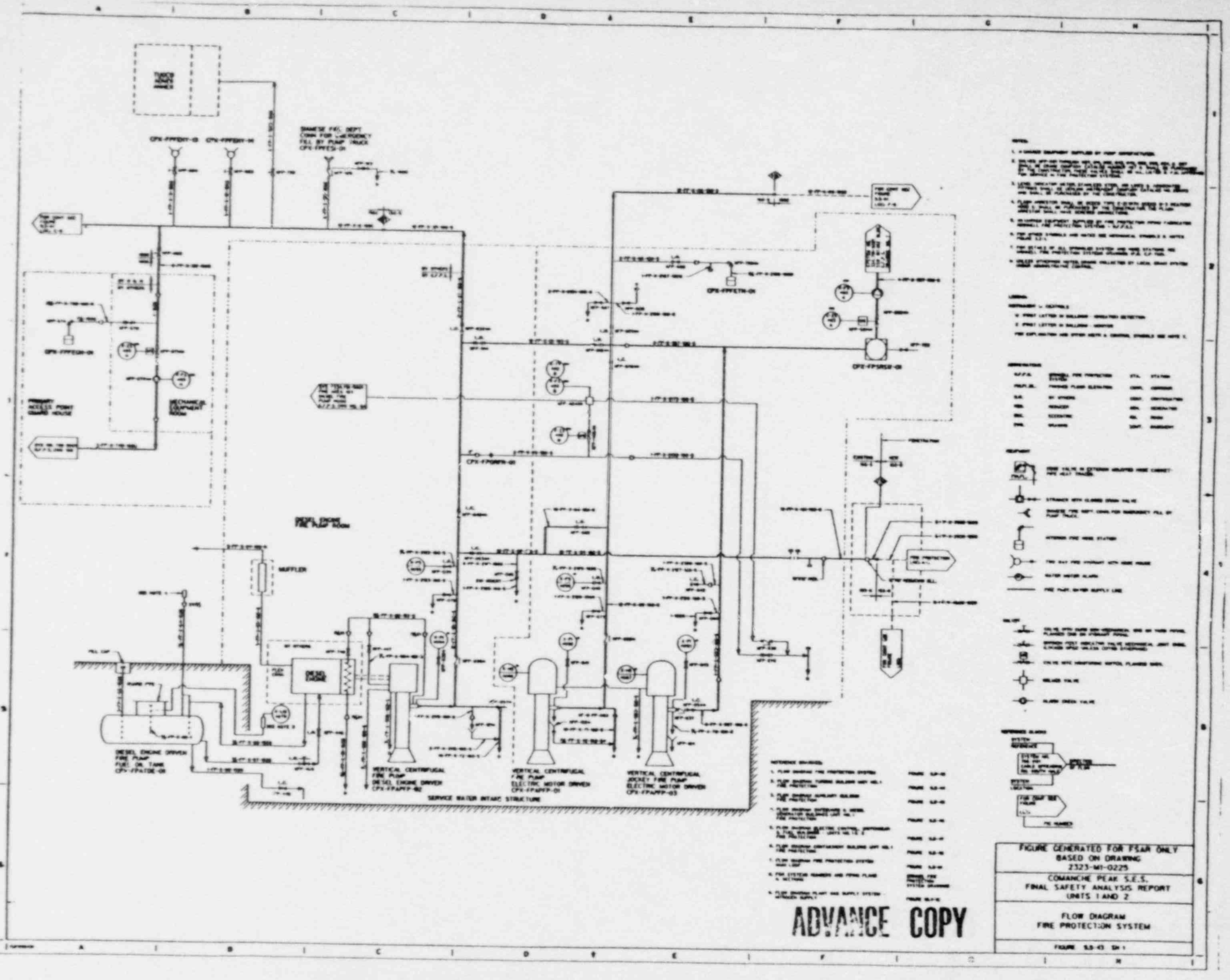
1. Service water pump area exhaust system

2. Intake structure heating system

3. Diesel-driven fire pump room exhaust system

The Service Water Intake Structure ventilation system is shown schematically on Figure 9.4-7.

The service water pump area exhaust system consists of eight manually operated, 50-percent-capacity, propeller-type fans, which are mounted on a missile-protected exhaust plenum located in the overhead of the draft damper. During summer operation, two exhaust fans, each rated at 5000 scfm, are required to dissipate the heat generated by a service water pump motor. The number of fans in operation depends on the number of service water pumps functioning during a given plant operating mode. The number of fans manually placed into operation at any given time is left upon to the discretion of the intake structure


SIAMESE FIRE DEPT.
CONN FOR EMERGENCY
FILL BY PUMP TRUCK
CPX-FPFESI-01
PRIMARY ACCESS POINT GUARD HOUSE
MECHANICAL EQUIPMENT ROOM
DIESEL ENGINE FIRE PUMP ROOM
MUFFLER
DIESEL ENGINE
DIESEL ENGINE DRIVEN FIRE PUMP FUEL OIL TANK CPX-FPATDE-01
VERTICAL CENTRIFUGAL FIRE PUMP DIESEL ENGINE DRIVEN CPX-FPAPFP-02
VERTICAL CENTRIFUGAL FIRE PUMP ELECTRIC MOTOR DRIVEN CPX-FPAPFP-01
VERTICAL CENTRIFUGAL JOCKEY FIRE PUMP ELECTRIC MOTOR DRIVEN CPX-FPAPFP-03
SERVICE WATER INTAKE STRUCTURE
CPX-FPFETN-01
CPX-FPSRSV-01
CPX-FPFESH-01
FIGURE GENERATED FOR FSAR ONLY
BASED ON DRAWING
2323-M1-0225
COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNITS 1 AND 2
FLOW DIAGRAM
FIRE PROTECTION SYSTEM
ADVANCE COPY

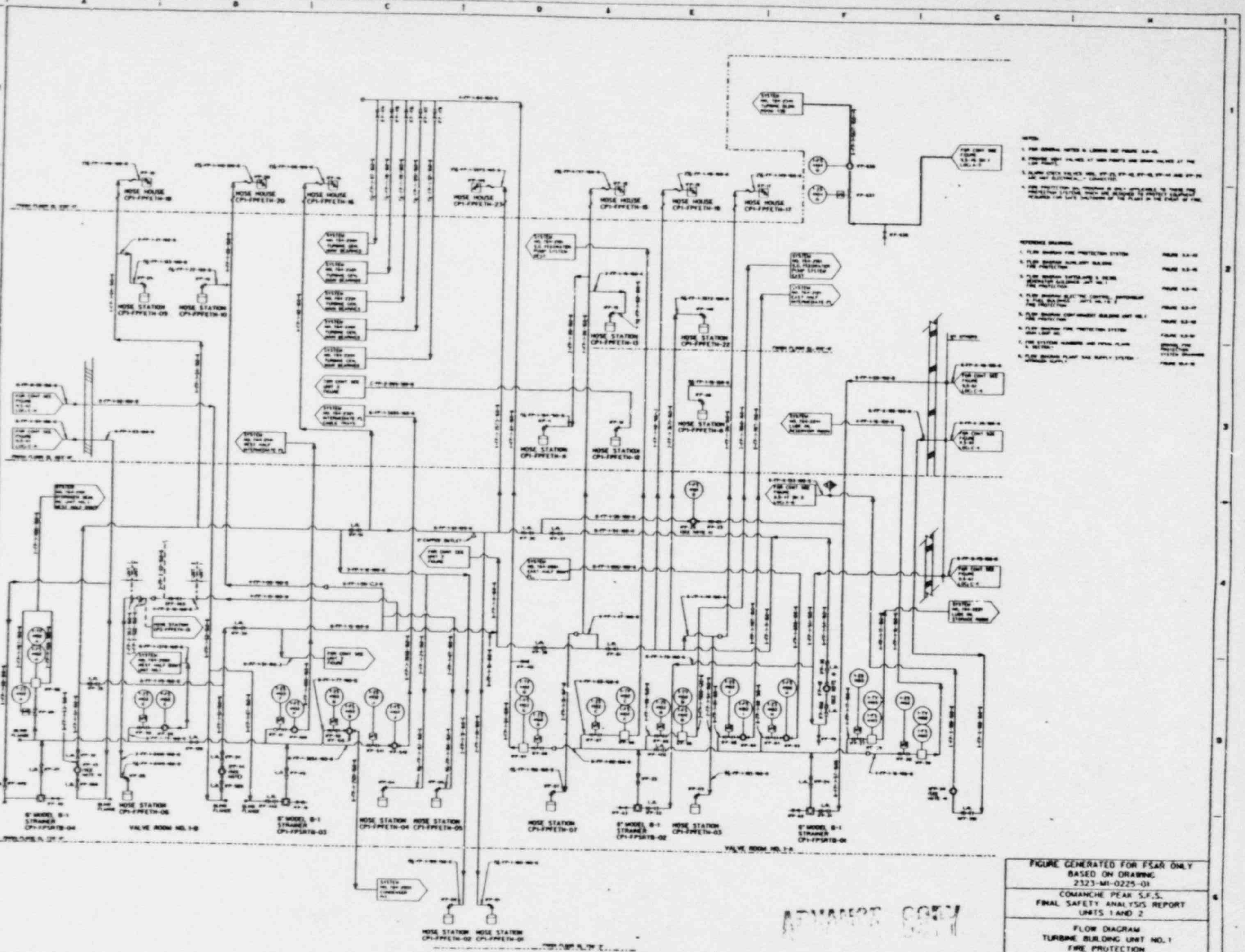


FIGURE GENERATED FOR FSAR ONLY
BASED ON DRAWING
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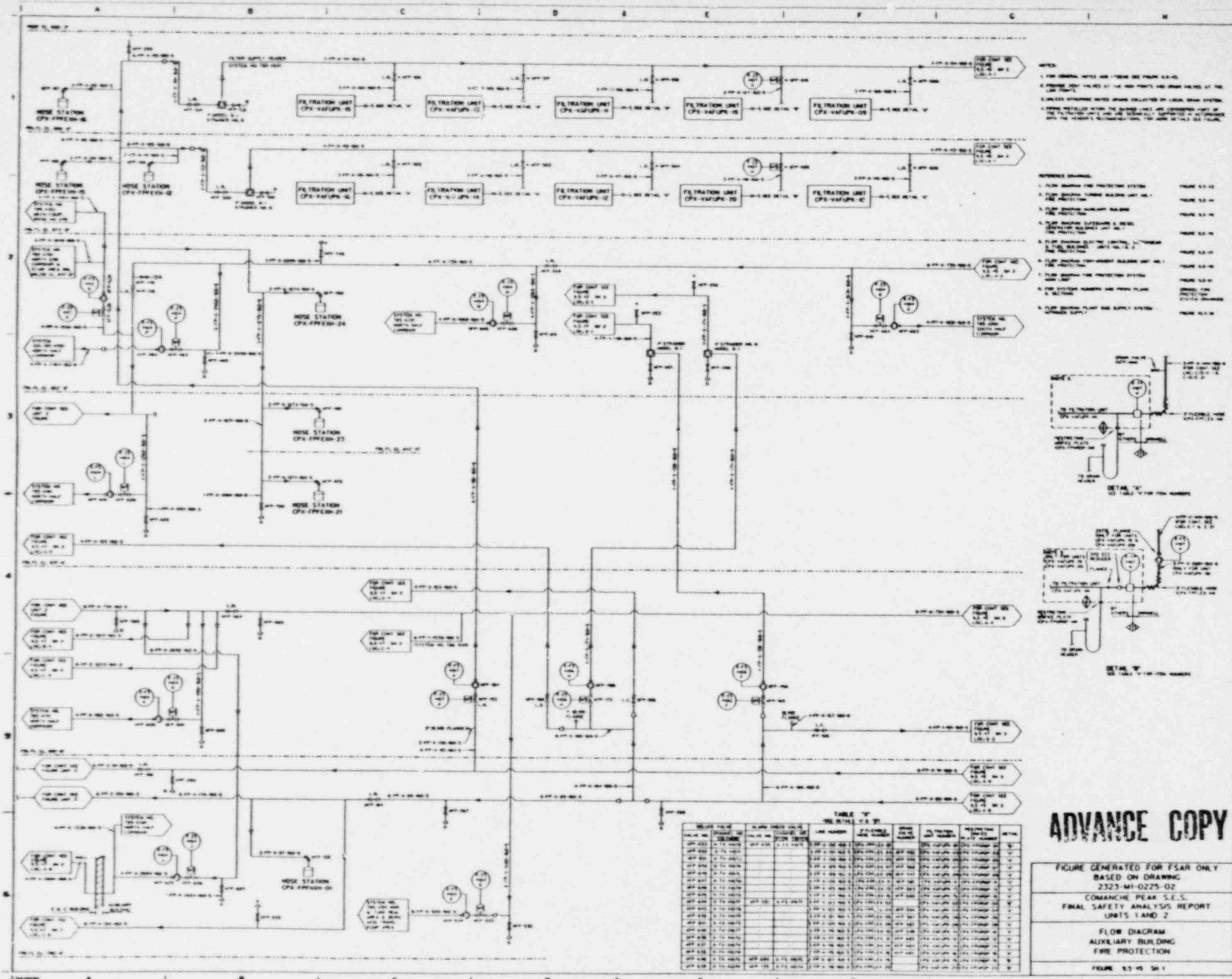
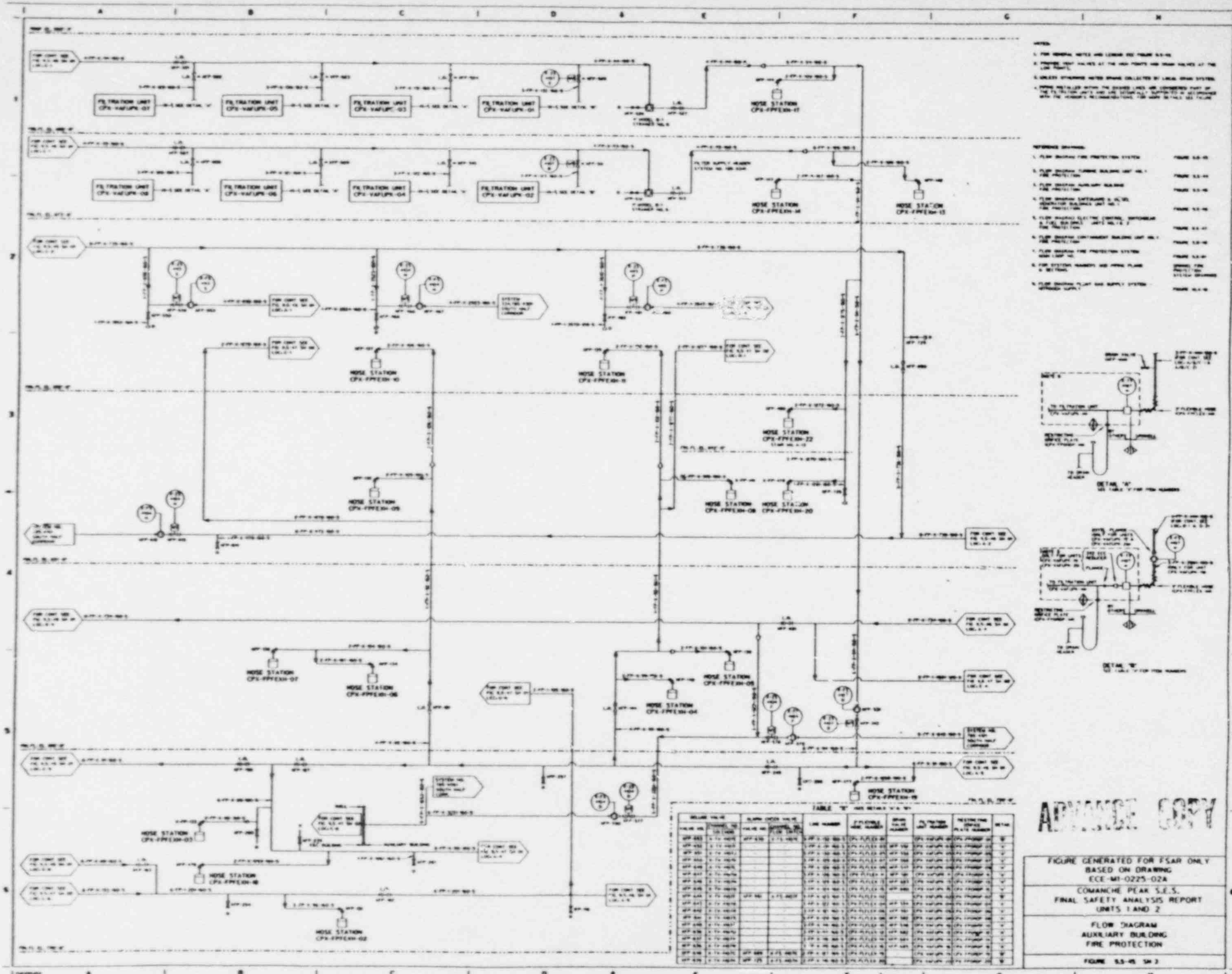
COMANCHE PEAK S.F.S.
FINAL SAFETY ANALYSIS REPORT
UNITS 1 AND 2

FLOW DIAGRAM
TURBINE BUILDING UNIT NO. 1
FIRE PROTECTION

FIGURE 9.5-44 SH 1



HOSE STATION
CPX-FPFEXH-16
HOSE STATION
CPX-FPFEXH-15
HOSE STATION
CPX-FPFEXH-12
FILTRATION UNIT
CPX-VAFUPK-15
FILTRATION UNIT
CPX-VAFUPK-13
FILTRATION UNIT
CPX-VAFUPK-11
FILTRATION UNIT
CPX-VAFUPK-19
FILTRATION UNIT
CPX-VAFUPK-09
FILTRATION UNIT
CPX-VAFUPK-16
FILTRATION UNIT
CPX-VAFUPK-14
FILTRATION UNIT
CPX-VAFUPK-12
FILTRATION UNIT
CPX-VAFUPK-20
FILTRATION UNIT
CPX-VAFUPK-10
HOSE STATION
CPX-FPFEXH-24
HOSE STATION
CPX-FPFEXH-23
HOSE STATION
CPX-FPFEXH-21
HOSE STATION
CPX-FPFEXH-01
DETAIL "X"
DETAIL "Y"
TABLE "Y"
ADVANCE COPY
FIGURE GENERATED FOR FSAR ONLY
BASED ON DRAWING
2323-M1-0225-02
COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNITS 1 AND 2
FLOW DIAGRAM
AUXILIARY BUILDING
FIRE PROTECTION


FILTRATION UNIT
HOSE STATION
DETAIL "A"
DETAIL "B"
ADVANCE COPY
FIGURE GENERATED FOR FSAR ONLY
BASED ON DRAWING
ECE-M1-0225-02A
COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNITS 1 AND 2
FLOW DIAGRAM
AUXILIARY BUILDING
FIRE PROTECTION
FIGURE 9.5-45 SH 2

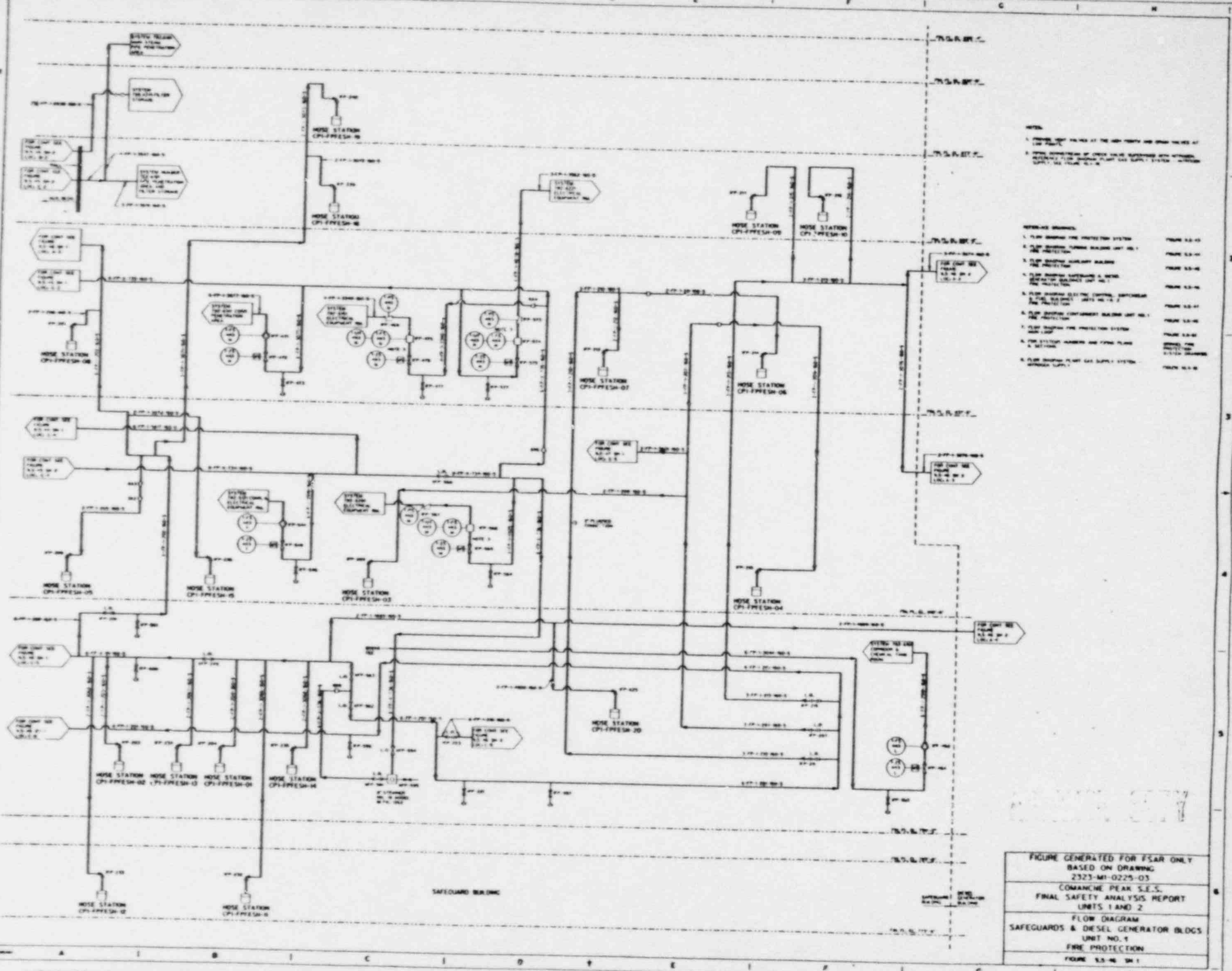


FIGURE GENERATED FOR FSAR ONLY
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COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNITS 1 AND 2

FLOW DIAGRAM
SAFEGUARDS & DIESEL GENERATOR BLDGS
UNIT NO. 1
FIRE PROTECTION

FIGURE 9.5-46 SH 1

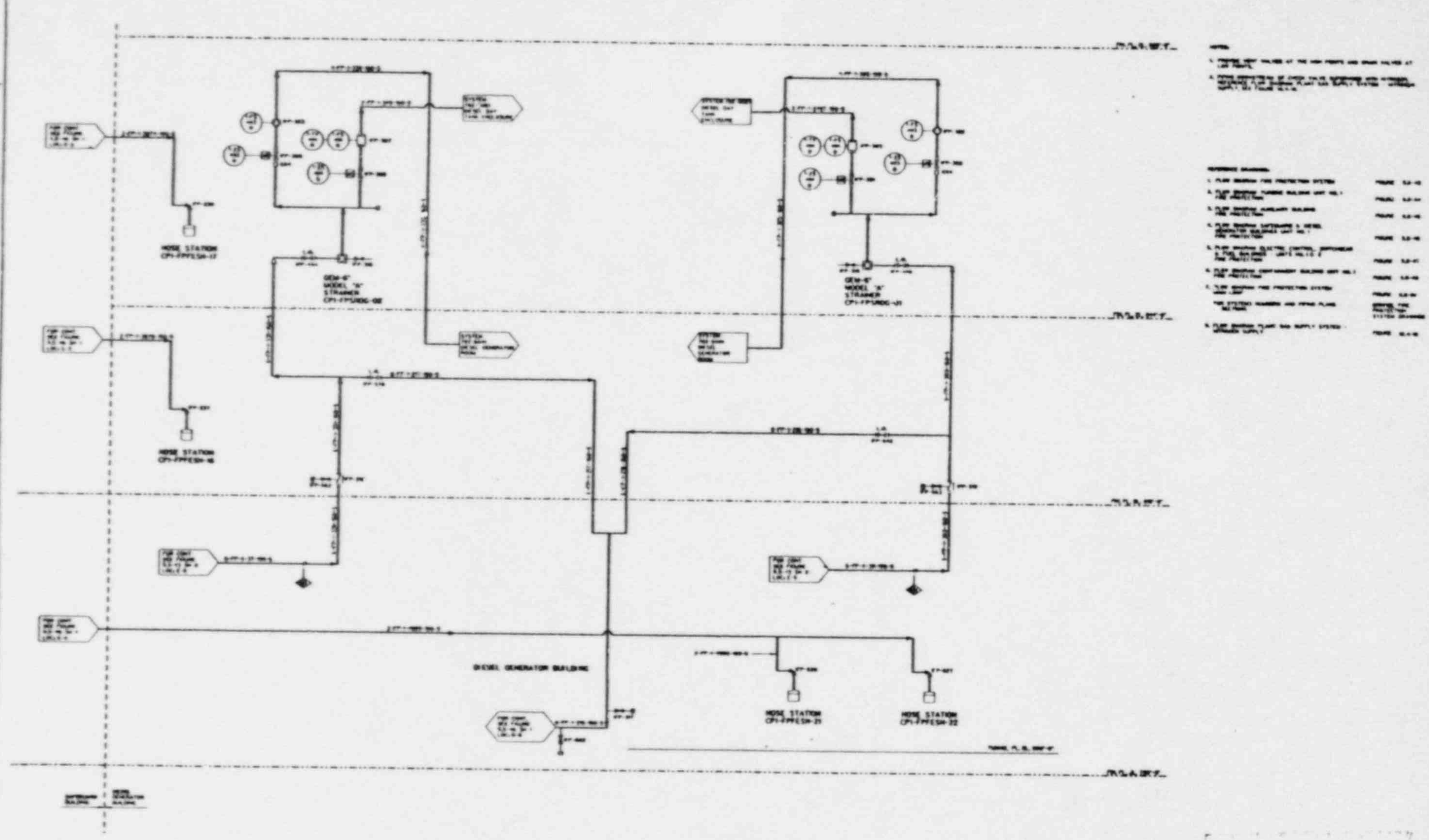

FIGURE GENERATED FOR FSAR ONLY
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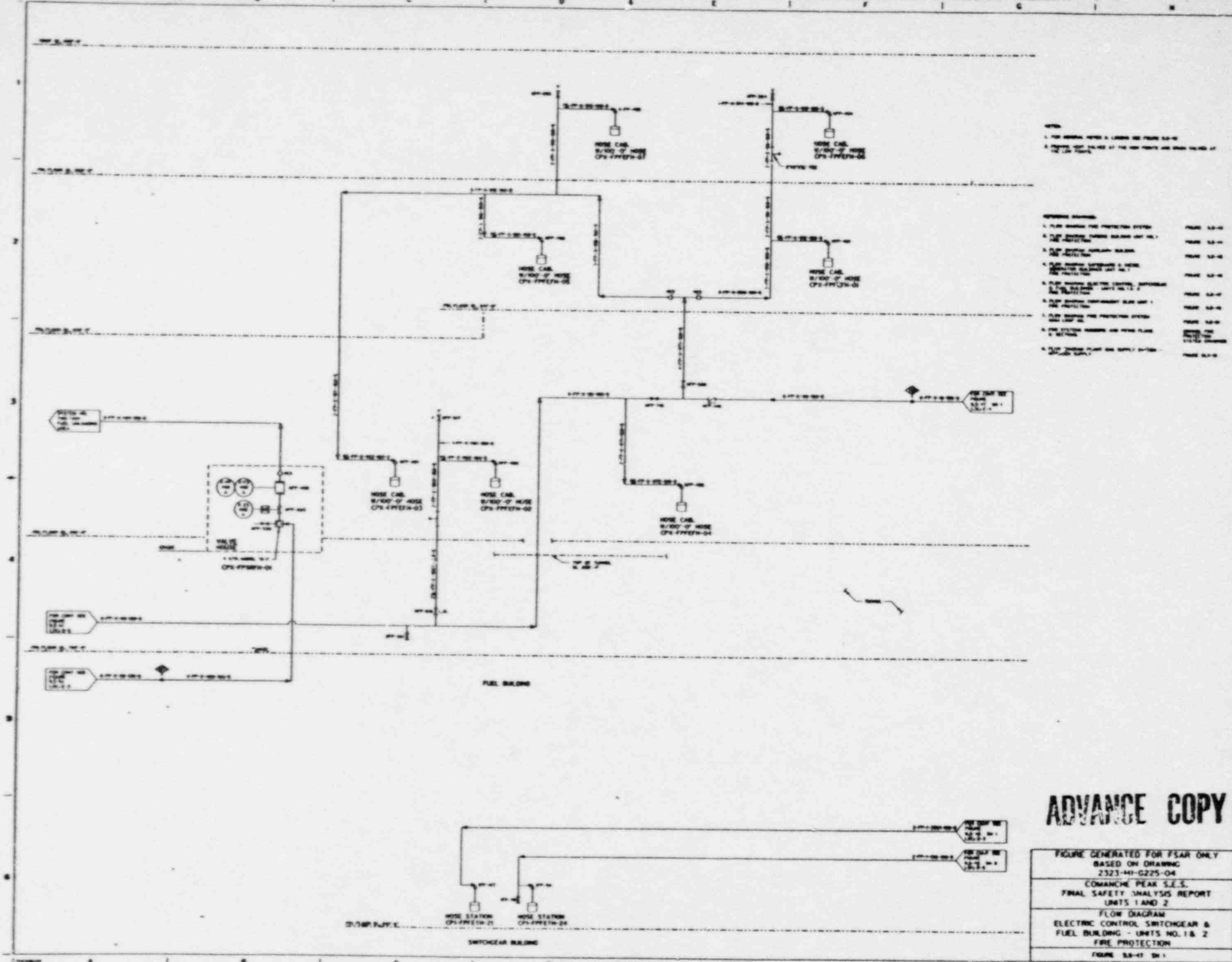
COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNITS 1 AND 2

FLOW DIAGRAM
SAFEGUARD & DIESEL GENERATOR BUILDINGS
UNIT NO. 1
FIRE PROTECTION

FIGURE 9.5-46 SH 2



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COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNITS 1 AND 2

FLOW DIAGRAM
ELECTRIC CONTROL SWITCHGEAR &
FUEL BUILDING - UNITS NO. 1 & 2
FIRE PROTECTION

FIGURE 9.5-41 SH 1

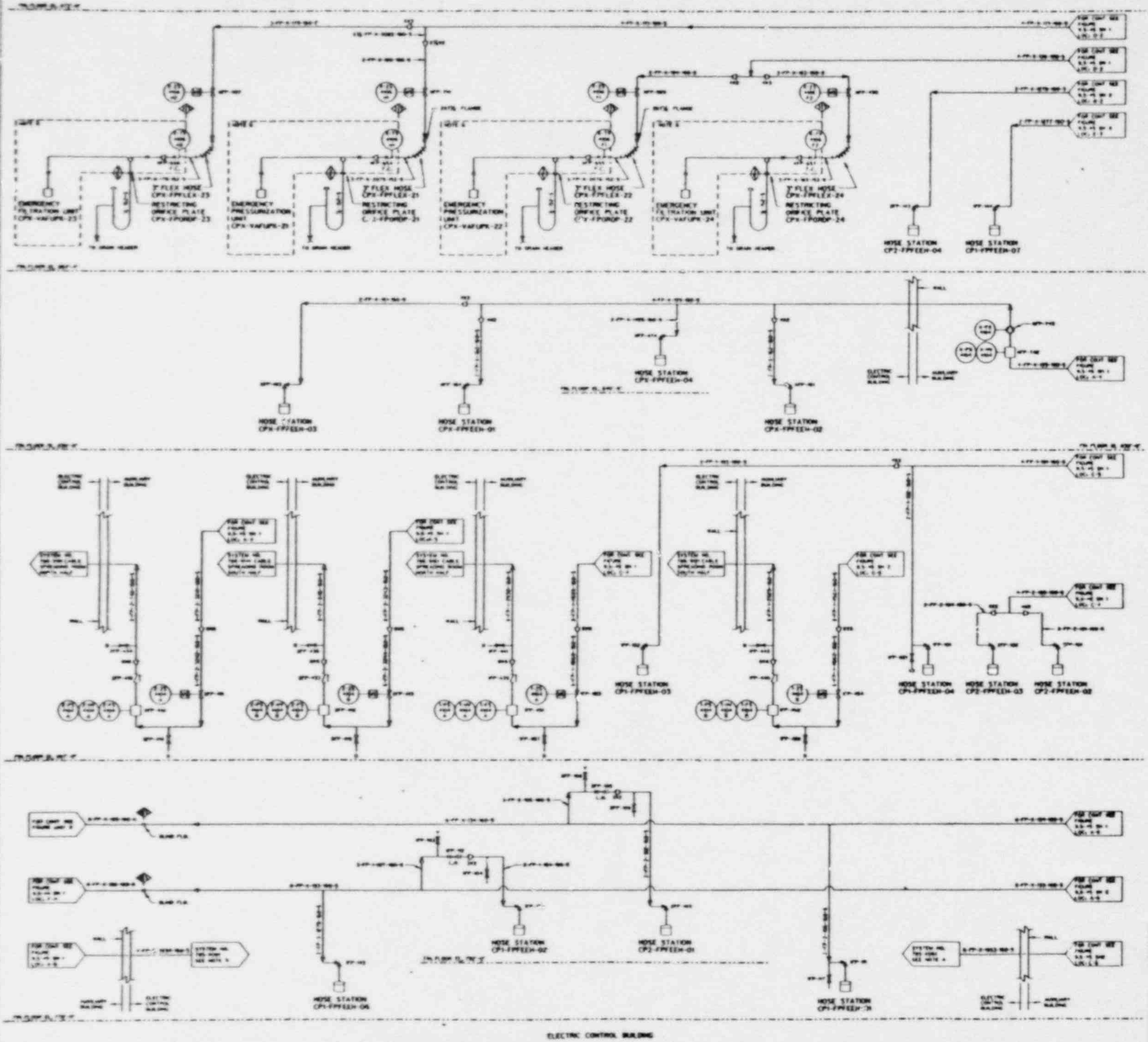

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FIGURE GENERATED FOR FSAR ONLY
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COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNITS 1 AND 2

FLOW DIAGRAM
ELECTRIC CONTROL SWITCHGEAR &
FUEL BUILDINGS - UNITS 1 & 2
FIRE PROTECTION

FIGURE 9.5-47 SH 2

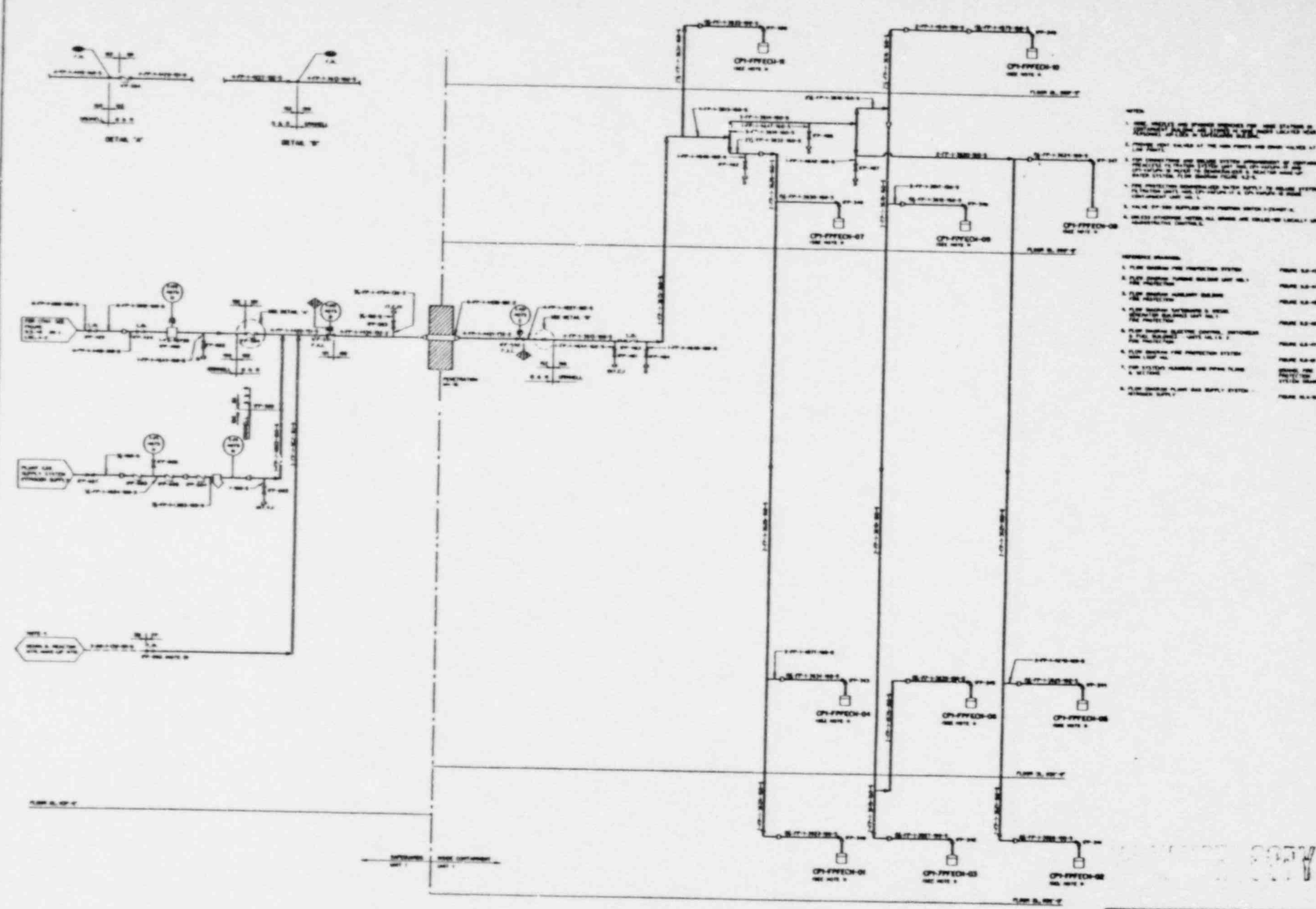

FIGURE GENERATED FOR FSAR ONLY
BASED ON DRAWING
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COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNITS 1 AND 2

FLOW DIAGRAM
CONTAINMENT BUILDING UNIT 1
FIRE PROTECTION

FIGURE 9.5-48 SH 1

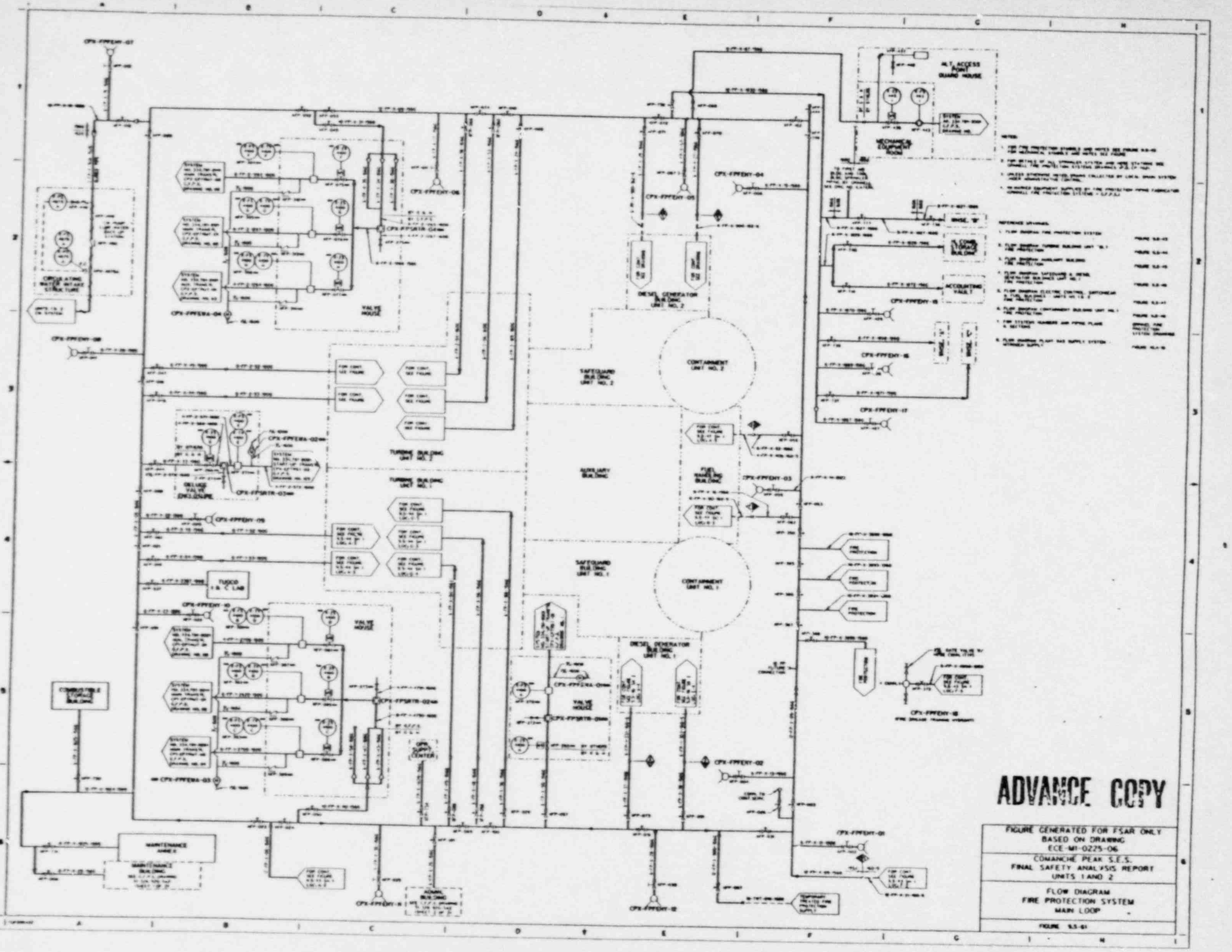

FIGURE GENERATED FOR FSAR ONLY
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ECE-M1-0225-06

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FLOW DIAGRAM
FIRE PROTECTION SYSTEM
MAIN LOOP

FIGURE 9.5-61

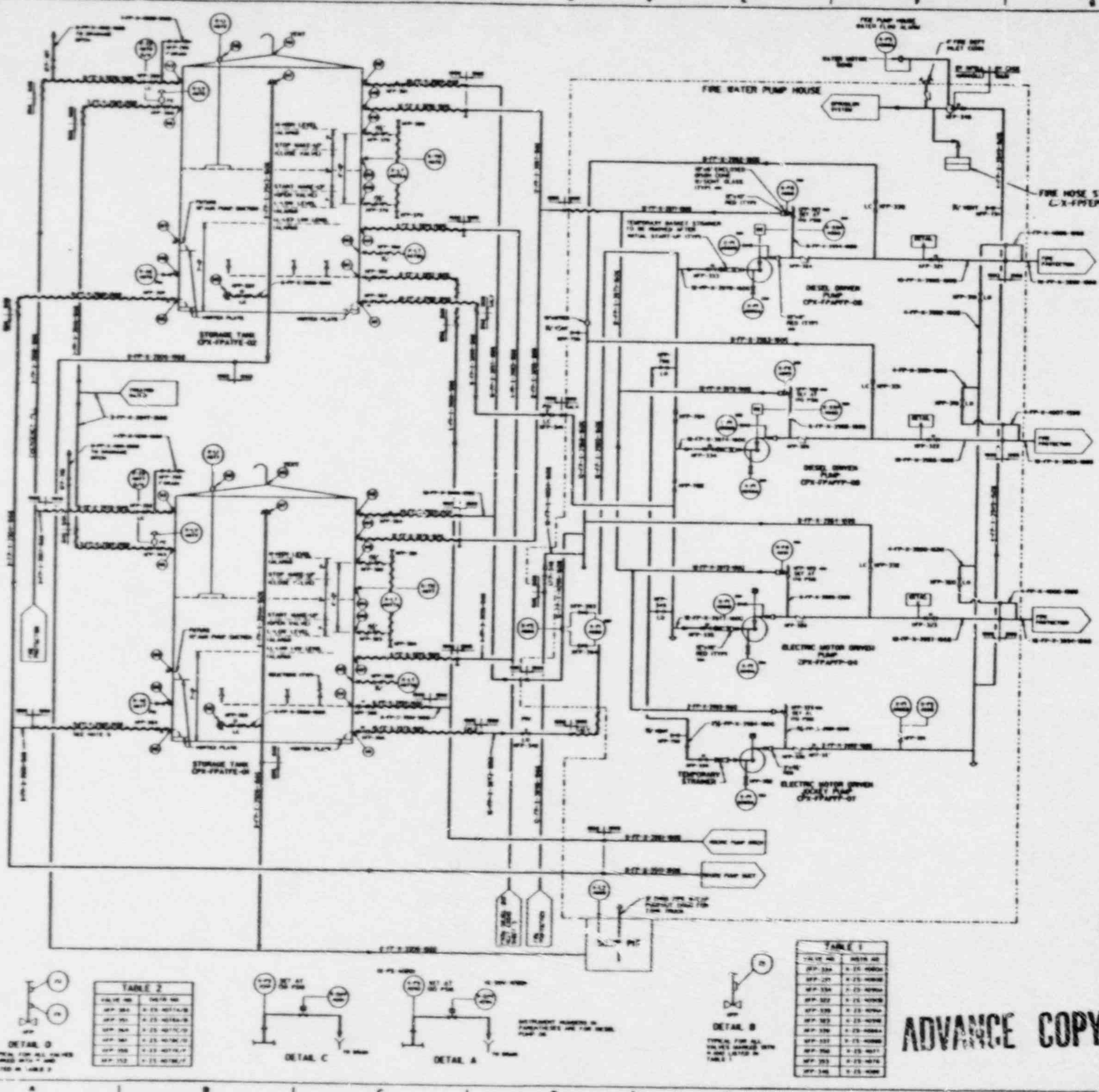

FIGURE GENERATED FOR FSAR ONLY
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15454.15-EB-1 SH 1

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FIRE PROTECTION TREATED WATER
SUPPLY SYSTEM

FIGURE 9.5-62 SH 1

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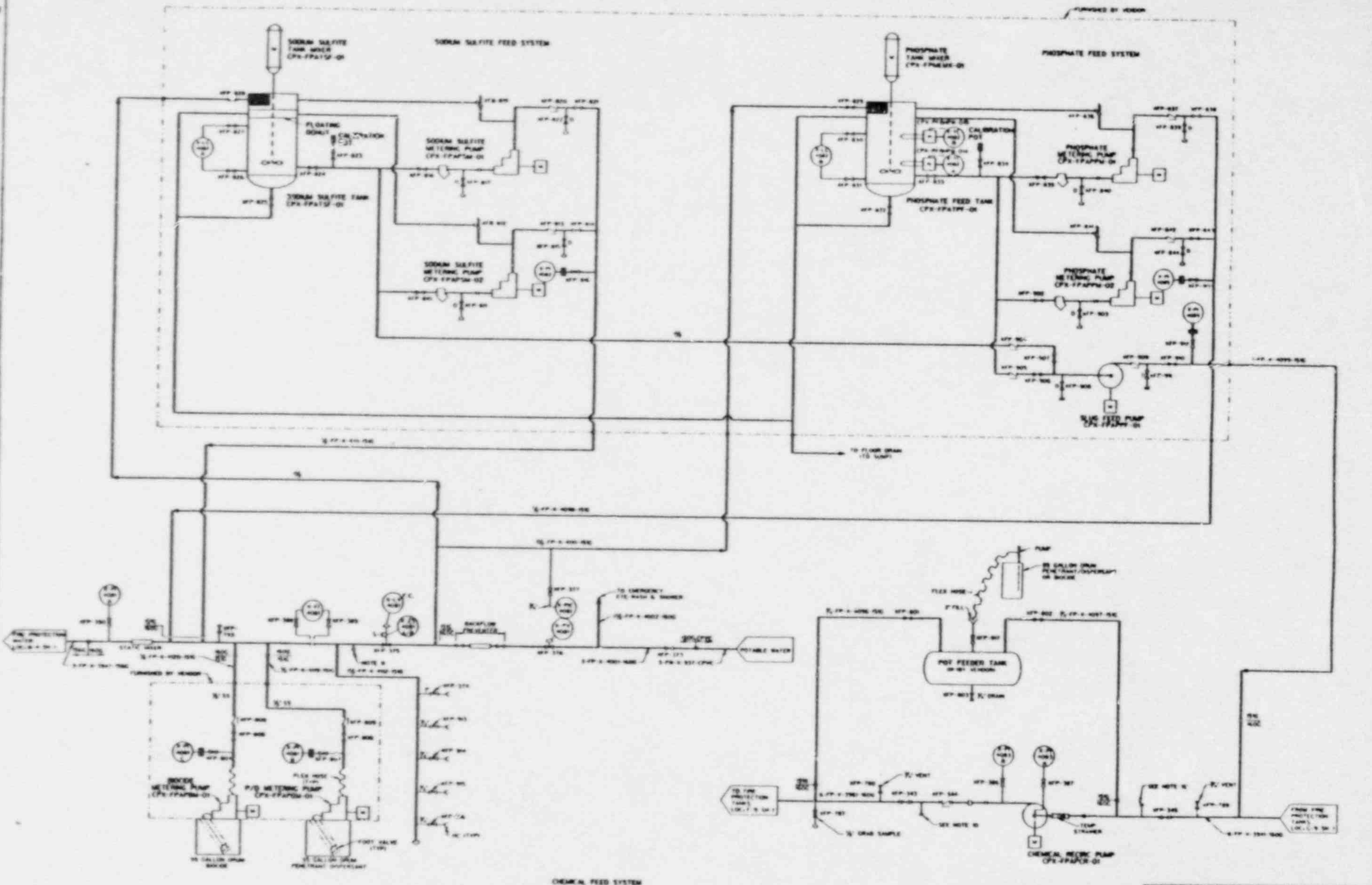

FIGURE GENERATED FOR FSAR ONLY
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15454.15-EB-1SH 2

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UNITS 1 AND 2

FIRE PROTECTION
TREATED WATER SUPPLY SYSTEM

FIGURE 9.5-42 SH 2

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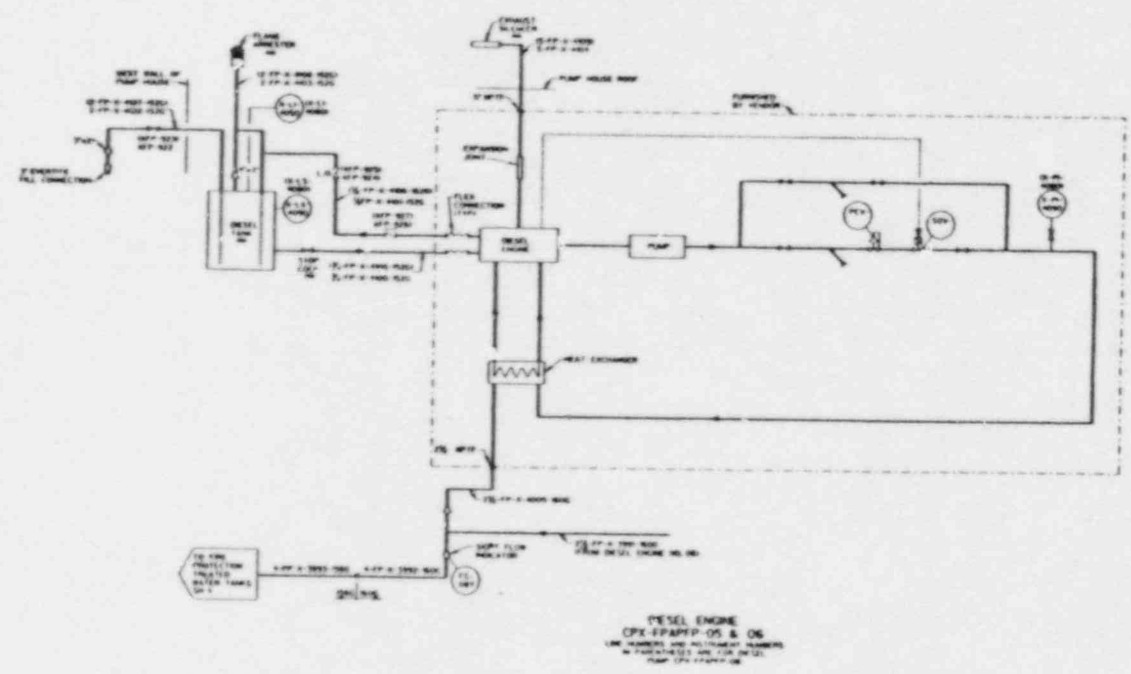

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UNITS 1 AND 2

FIRE PROTECTION TREATED WATER SUPPLY SYSTEM

FIGURE 9.5-62 SH 3

10. Condenser vacuum

11. Stator winding temperature

12. Turbine stress evaluator indicator

13. Temperature load allowance turbine stress evaluator

Q040.129

10.2.2.12 SAFETY CONSIDERATIONS 11

As described in Section 7.2.1.1.2 the turbine generator hydraulic pressure switches and the stop valve limit switches are part of the anticipatory reactor-trip-on-turbine-trip channels which provide turbine tripped signals to the Reactor Protections System. 11

Failure in a high or moderate energy piping or failure of the connection from the low pressure turbine to condenser could damage components of the anticipatory reactor trip channels because no protection is provided for the above mentioned switches. The safety analysis does not take any credit for the anticipatory reactor trip as described in Section 15.2. Consequently, there are no safety implications if the switches are damaged as a result of a pipe break. 11

10.2.3 TURBINE DISK INTEGRITY

This section provides information demonstrating the integrity of the turbine disks and rotor, as failure of these components can result in turbine high-energy missiles as discussed in Section 3.5.1.3. As 68
described in Reference [1], for the purposes of missile formation the most hazardous missiles are generated from the low pressure turbine. The potential missile energy from the high pressure turbine is less than that from the low pressure turbine because of its much smaller potential missile mass and thicker turbine casing. Therefore, the high pressure turbine potential missiles are bounded by the low pressure turbine potential missiles .

corrosion control. Chlorine is added to the water once or twice daily | 68
to control organic and biological growth in the cooling system. A |
single system is shared by both units. | 68

Figure 10.4-6 shows the major components and flow path for the circulating water chlorination subsystem. Chlorine is drawn from one ton containers through the liquid chlorine manifold to the evaporators, which are essentially vaporizing chambers surrounded by a water jacket. The water is heated by an electric immersion heater, which is thermostatically controlled to maintain constant temperature. Liquid chlorine enters the vaporizing chamber through the inlet tube and is piped to the bottom of the chamber. After it emerges from the pipe, the liquid absorbs heat from the hot water and vaporizes. The vapor rises to pass out of the evaporator through the chlorine gas outlet. The demands of the using system for chlorine vapor automatically regulate the level of chlorine liquid inside the vaporizing chamber. As the vapor pressure inside the chamber increases, the rate at which the liquid enters the vaporizing chamber decreases.

If the demand for chlorine vapor increases, the pressure inside the vaporizing chamber decreases, thus permitting chlorine to enter the chamber at a higher rate. An equilibrium condition is soon achieved where the rate at which the liquid is being converted to gas exactly equals the rate at which liquid enters the vaporizing chamber. An automatic pressure-reducing and shutoff valve (controlled by low water temperature in the gas line to the dispensing system) automatically shuts off when the water chamber temperature falls below a preset limit, preventing liquid from entering and flooding the gas dispensing system.

4. NRC Regulatory Guides

66 | 1.26 Quality Group Classification and Standards for Water, Steam & Radioactive-Waste containing components of Nuclear Power Plants. [3]

1.29 Seismic Design Classification [5]

68 | 1.32 "Criteria for Safety Related Electric Power Systems for Nuclear Power Plants" [22]

66 | 1.47 "Bypassed & Inoperable Status Indication for Nuclear Power Plant Safety Systems" [23]

66 | 1.53 "Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems" [24]

66 | 1.75 "Physical Independence of Electric Systems" [25]

5. American National Standards Institute (ANSI)

B31.1 Power Piping

N18.2 Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants [4]

66 | N271 Containment Isolation Provisions for Fluid Systems

Additional requirements for the portion of the Feedwater System which is within the Containment boundary are given in Section 10.3.6.

10.4.7.2 Systems Description

The Condensate and Feedwater System flow diagrams are shown on Figure 10.4-8, and 10.4-9, respectively.

1. Condensate System

All Condensate System Condensate Storage Tank connections are set
68 23 ft 9 in. above the bottom of the tank to prevent drawdown below this level, thus ensuring that a minimum storage of 270,000 gal of water remains in the tank for possible emergency plant shutdown.

68 The design basis for the minimum storage capacity of 270,000 gallons is as follows:

a. Water required to cool down the reactor for two hours at hot shutdown and five hours of cool down

b. Allowance for feedwater line break spillage

68 The minimum capacity required for the AFW system to perform item a and b above is 247,044 gallons. The actual contained volume of water below these condensate connections is 282,554 gallons which includes 270,000 gallons useable capacity and that volume which is not useable.

All heaters are provided with tube side-safety valves to provide thermal relief by guarding against possible overpressurization caused by heating of water trapped between closed isolation valves.

Provisions for the removal of individual feedwater heaters from service have been incorporated in the plant design.

2. Feedwater System

66 The Feedwater System is of the closed-cycle type and receives water from the Condensate System and the Heater Drains Systems (specifically, drains from heaters 1, 2, and 3, and MSR Separator drain tanks). The feedwater is transported through the final two stages of feedwater heating to the steam generators.

41 During power operation each steam generator feedwater pump takes suction from the Condensate System and discharges through a

feedwater nozzle. This line incorporates a restricting flow orifice and an air operated globe type shutoff valve which also serves as a containment isolation valve. 10

Connections from the Condensate System are provided both upstream and downstream of the heaters to permit flushing of the heaters. The upstream connection can also be used to fill the system and the steam generators and to provide feedwater directly from the condensate pumps during the early stages of startup.

Full flow flushing of the Condensate and Feedwater Systems is provided to permit piping cleanup before plant start-up. Full flow flushing is accomplished through a 24" diameter pipe routed from the outlet of the HP feedwater heaters to the condenser. Bypass lines with isolation valves are provided around each steam generator feedwater pump for use during flushing operations. 66

Sampling system connections are provided for Steam Generator #1 & #2 feedwater lines to the main FW nozzles for steam generators 1 & 2 for monitoring feedwater chemistry. The sampling system is also connected to various points in the Condensate, Main Steam, and Heater Drain Systems as shown on Figure 10.4-20. 68

Condensate and feedwater chemistry are controlled as described in Section 10.3.5. This system uses all volatile chemical treatment in conjunction with a Condensate Polishing System, which is described in Section 10.4.6. 66

Chemical feed to each steam generator is introduced at a point downstream of the auxiliary feedwater supply connection to the Feedwater System for adjustments to the water chemistry during steam generator layup periods only.

3. Electrical Systems

Safe shutdown of the plant does not rely upon the availability of either the Condensate System or the Feedwater System. However, the portion of the Feedwater System from the moment restraint upstream from the feedwater isolation valve to the steam

66 The system functions over the full operating pressure range of the steam generators, 100 psia to 1107 psia (maximum), and is capable of supplying the minimum required flow to at least two of the effective steam generators against a back pressure equivalent to the accumulation pressure of the lowest set safety valve (1236 psia) plus the system frictional and static losses.

4

5 The Auxiliary Feedwater System is designed to preclude the effects of hydraulic instability due to water hammer by supplying water to the secondary side of the steam generator through a separate upper auxiliary feedwater nozzle. This permits the cold auxiliary feedwater to be heated as it comes down the side of the steam generator prior to reaching the feedwater preheater. For further discussion of the feedwater system upper nozzle arrangement and considerations see Subsection 10.4.7.

5 The water level in steam generators is maintained at the proper level to prevent a temperature rise in the RCS, which could result in the release of primary coolant through the pressurizer relief valves.

68 With either only onsite or only offsite power available with an assumed single failure, sufficient auxiliary feedwater flow of 430 gpm to two steam generators is provided to permit operation at hot standby for four hours, followed by a cooldown period, at a cooldown rate of $50^{o}F$/hr, to reduce the T_{avg} to $350^{o}F$, at which time the RHR System can be operated. In the event of a main steam or feedwater line break, sufficient auxiliary feedwater is provided to permit operation at hot standby for two hours followed by the above mentioned cooldown period.

57 Two motor-driven pumps and one turbine-driven-pump are provided to ensure an adequate supply of auxiliary feedwater of 430 gpm to two steam generators following a feedwater line break accident coincident with a single active failure.

All redundant components are physically separated from each other by an arrangement of concrete barriers designed to preclude coincident damage to equipment in the event of a postulated pipe rupture, equipment failure, or missile generation [6] [7].

The system has been designed to withstand the adverse environmental conditions delineated in Section 3.11, including the effects of flooding which are further discussed in Subsection 10.4.9.2, System Description.

The system is classified as nuclear-safety-related and consists of ANS Safety Class 2 and 3 piping and equipment, except for the non-nuclear-safety condensate transfer pump and associated piping and valves used to provide makeup and drainage for the Condensate Storage Tank [3] [4]. Seismic Category I design criteria are considered for all ANS Safety Class 2 or 3 components. Seismic requirements are given in Section 3.2 [5]. The piping is designed to meet the requirements of Branch Technical Positions APCSB 3-1 and MEB 3-1. The system is designed in accordance with 10 CFR Part 50, GDC 2, 4, 5, 19, 44, 45, 46, and 57 [8], [9], [10], [11], [12], [13], [14], [17].

10.4.9.2 System Description

The Auxiliary Feedwater System is comprised of two electric motor-driven auxiliary feedwater pumps and associated valves, piping, and controls and a third turbine-driven auxiliary feedwater pump with associated valves, piping, and controls, which is independent of the electrical power supply to the motor-driven pumps. Three pumps are necessary to ensure an adequate supply of auxiliary feedwater following an accident, coincident with the single failure of a pump. The design parameters of the auxiliary feedwater pumps are given in Table 10.4-8.

All three pumps normally draw suction from the Nuclear Safety Class 3 Condensate Storage Tank. A single line supplies water through locked-open valves to the suction of the motor-driven auxiliary feedwater pumps, and a second line supplies water through locked-open valves to the suction for the turbine driven auxiliary feedwater pump. Of the | 68
approximately 500,000-gal capacity, 270,000 gal are available for use |
only as auxiliary |

feedwater. The rest of the tank is used as condensate storage for the Demineralized and Reactor Makeup Water System and Condensate System. The reserved auxiliary feedwater cannot be drained by the non-nuclear-safety systems because of the elevation of the outlet nozzles.

While the Condensate Storage Tank is the preferred water supply, another ANS Safety Class 3 alternate supply is provided. The Auxiliary Feedwater System has the capability to draw suction from the service water system (SWS) in the event of loss of the Condensate Storage Tank. Two normally closed, key-switch activated, motor-operated butterfly valves in the SWS prevent contamination of the auxiliary feedwater by station service water. In addition, a high-point leakoff is provided between the SWS isolation valves and the motor-operated gate valves to allow detection of any station service water inleakage.

68 Each motor-driven auxiliary feedwater pump is capable of delivering 470 gpm to the two steam generators, and the turbine-driven auxiliary feedwater pump is capable of delivering 860 gpm to the four steam generators. All three pumps automatically deliver their flow within one minute following an auxiliary feedwater actuation signal.

Each motor-driven pump normally feeds two steam generators. A
normally closed interconnection between the motor-driven pump
discharge lines permits either pump to feed to all four steam
51 generators. This interconnection provides the operator with the means
to maintain the water level in all steam generators on a long-term
basis following a LOCA by operating either motor driven pump. The
motor driven pumps can be manually started or stopped from the Control
Room or the hot shutdown panel.

4 The turbine-driven pump discharge line branches into four separate
68 lines each feeding one steam generator. The turbine-driven pump can
be manually started from the Control Room or the hot shutdown panel.

Each of the lines that connects the three auxiliary feedwater pumps to 68
the steam generators is provided with: a normally open, pneumatically
operated feed regulator control valve; a flow-limiting orifice; a
check valve; and three isolation valves. Remote manual control of
the feed regulator control valve is provided from the Control Room 4
with provision for local manual operation on the hot shutdown panel.
Air accumulators are provided for the pneumatically operated valves
with sufficient capacity to permit remote valve closure in the event
of a secondary system break where local valve operation cannot be
accomplished within the required time period following the incident.
The valves are located near the auxiliary feedwater pumps to allow 51
local manual operation.

The flow limiting orifices are provided to limit flow to any one steam 66
generator to a maximum of 1380 gpm, in the event of either a main feed
line break or a main steam line break inside containment.

Downstream of the last isolation valve, each line from the motor-driven pumps joins with a corresponding line from the turbine-driven pump to form a common line that connects with an auxiliary feedwater nozzle on the steam generator.

An orifice-type flow measuring device is located in each of the 66
auxiliary feedwater lines to indicate flow to each steam generator and
to provide a means of detecting grossly uneven flow to the steam
generators. Readout for these flow measuring devices is located in
the Control Room and on the hot shutdown panel. To avoid the
possibility of a single active failure stopping all auxiliary
feedwater flow to a steam generator, there are no valves located in
the common main feedwater lines.

The Auxiliary Feedwater System operates over an extended period of time following a LOCA. The two motor-driven pumps start automatically and they provide an additional means for removing core residual heat
68 in the event of a LOCA for small breaks. During large break LOCA conditions, the system is used to maintain an adequate water level above the tubes in the steam generators to prevent primary to secondary leakage. The operator shuts down the pumps at his discretion and manually adjusts feed flow to individual steam generators.

All three auxiliary feedwater pumps start automatically after either a main steam line break or a feedwater line break (see Subsection 10.4.9.5). At an early stage in the accident, the operator isolates the feedwater to the affected steam generator. The system provides for the cooldown of the unaffected steam generators to prevent the RCS from being repressurized. The operator shuts down the pumps at his discretion.

57 After a loss of the main feedwater system , either the two motor-driven auxiliary feedwater pumps together or the turbine-driven pump alone is capable of providing sufficient flow to the steam generators to allow the plant to be taken to a safe shutdown condition. The operator shuts down the pumps at his discretion.

The operation of the Auxiliary Feedwater System following a steam generator tube rupture is manually initiated. The two motor-driven pumps are started manually and are used to maintain the required water level in the steam generators as the plant is shut down. The operator identifies the affected steam generator and isolates it and the operator shuts down the pumps at his discretion.

The operation of the Auxiliary Feedwater System following a Control Room evacuation is manually initiated and is controlled from the hot shutdown panel. The operator maintains water level in the steam generators with either the two motor-driven pumps or the turbine-driven pump. The pumps are used to maintain the required water level 51 in the steam generators as the plant is shut down. Again the operator shuts down the pumps at his discretion.

Each power supply train for the motor-driven pumps, control valves, and instrumentation is supplied from a separate and independent Class 1E bus that is capable of supplying the minimum required power for the safety-related loads required following a LOCA or loss of offsite power (blackout), or both. Each bus can be powered from two independent offsite power sources or by the diesel generator assigned to the bus [15].

The Control panel power for the turbine driven pump is from the Train 51 A Class 1E buses; however, the electrical features of the pump and control panel are not required for any safety function and are classified Associated Class 1E.

10.4.9.3 Safety Evaluation

The Auxiliary Feedwater System is designed to ANS Safety Class 2 and 3 68 requirements, and in the event of loss of offsite power, the backup turbine driven auxiliary feedwater pump operates. The turbine drive does not have any auxiliaries requiring electrical power. For redundancy, steam for the turbine driver is supplied from two steam generators. Either supply can meet the turbine driver requirements. The turbine steam supply valves are fail-open air-operated types each 10 with a pilot solenoid valve supplied from a redundant Class 1E power supply.

The turbine speed control governor is of the mechanical/hydraulic 4 type, which is capable of maintaining the turbine at the high speed setting without any outside sources of power.

66 A low pump discharge pressure along with high flow automatically
trips the control from manual flow control to automatic pressure
control to provide protection against pump cavitation and
excessive load on pump motor or diesel generator. Each
4 auxiliary feedwater regulator control valve is air-operated and
is provided with a nuclear safety-related air accumulator to
permit valves to close in the event of a secondary system break
and an instrument air system failure. The valves fail open on
loss of air or electric failure.

Q032.73

11 All controls for motor-driven pump A are electrical Train A
oriented; all controls for motor-driven pump B are electrical
Train B oriented; all controls for the turbine-driven pump are
fed from the 125VDC System.

Q032.73

68 The turbine-driven pump starts and accelerates to design
conditions within 60 seconds of an automatic actuation signal.
On loss of electrical power or air supply, the pump accelerates
to maximum speed demand. Since the turbine-driven pump is
supplied with a fail-closed trip and throttle valve, this valve
is latched in the open position. Two redundant steam supply
lines, each with an air operated supply valve, provide steam to
start and accelerate the turbine-driven pump. These air-operated
valves fail open, ensuring that the pump accelerates to design
66 speed on loss of air supply or electrical power. Speed control
is accomplished with a Woodward mechanical/hydraulic type
governor. A mechanical overspeed trip device is provided to
11 trip the turbine at 125-percent rated speed. Manual speed
control is from the Control Room, the local control station, or
the Hot Shutdown Panel The manual control from the Hot
Shutdown Panel overrides all other signals. There is speed
indication on the Control Room Panel and Hot Shutdown Panel and
68 at the local panel. Flow from the turbine-driven pump to each
steam generator is regulated by control

| Q032.73

valves under manual control from the Control Room, the Hot | 51
Shutdown Panel, or locally. Each valve has an air accumulator | 68
to permit the valves to close in the event of a secondary system
break and an air system failure.

3. Feedwater Supply Control

Condensate Storage Tank makeup is automatically supplied whenever the tank level is below set point level. Makeup water can be supplied manually from a main control board switch. Tank level | 65
is indicated locally and remotely and HI-HI, LO, and LO-LO tank level alarms are provided. Redundant level transmitters are used.

The Condensate Storage Tank supplies water to the auxiliary | 5
feedwater pumps. The automatic starting of any auxiliary | 11
feedwater pump initiates the automatic isolation of the Condensate Storage Tank from all its other users. This ensures an adequate water supply to the auxiliary feedwater pumps whenever they are started.

The condensate transfer pump is manually started and stopped from a main control board switch. The pump is automatically stopped | 5
in the event of an "S" signal or on low pump suction pressure.

4. Emergency Feedwater Supply Control | 11

Inlet motorized control valves are manually controlled by a key lock switch to admit service water to the suction of the auxiliary feedwater pumps.

5. Display Information, Alarms, and Controls | 11

Control switches and position indication lights are provided for all remotely operated valves.

11. 10 CFR Part 50, Appendix A, GDC 19, Control Room.

12. 10 CFR Part 50, Appendix A, GDC 44, Cooling Water.

13. 10 CFR Part 50, Appendix A, GDC 45, Inspection of Cooling Water System.

14. 10 CFR Part 50, Appendix A, GDC 46, Testing of Cooling Water System.

15. Branch Technical Position APCSB 10-1, Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants.

16. NRC Regulatory Guide 1.70, Standard Format & Content of Safety Analysis Report for Nuclear Power Plants, LWR Edition, Revision 2, September 1975.

17. 10 CFR Part 50, Appendix A, GDC 57, Closed System Isolation Valves.

10 | 18. NUREG-0291, An Evaluation of PWR Steam Generator Water Hammer by Creare, Inc, Dec. 31, 1976.

60 | 19. WCAP-09364, Vol. 1, High Pressure Water Hammer Test Program for the Counterflow Preheat Steam Generator.

41 | 20. Westinghouse Report "Counterflow Preheat Steam Generator Tube Expansion Report," June 1983.

41 | 21. Westinghouse Report "Counterflow Preheat Steam Generator Vibration Summary," June 1983.

68 | 22. NRC Regulatory Guide 1.32, "Criteria for Safety Related Electric Power Systems for Nuclear Power Plants", Revision 2, February 1977, U. S. Nuclear Regulatory Commission.

66 | 23. NRC Regulatory Guide 1.47, Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems, May 1973, U.

TABLE 10.4-9

(Sheet 6 of 6)

FAILURE MODE ANALYSIS

BREAK ON FEEDWATER PIPE TO STEAM GENERATOR NO. 4[1], INSIDE CONTAINMENT AND SINGLE FAILURE

Failure	Equipment/System	Function	Failure Mode	Effect of Failure	Analysis	
15	AO Feed Regulator Valves (7, 9, 11, 13) (Turbine-driven pump)		a) Loss of control power b) Loss of air	a) Remote manual capability lost b) Accumulated supplies 30 min. of air to retain and allow remote control. Then valve fails open.	a) AF supplied by motor-driven pumps or valves manipulated locally. Shut valves 16 & 17 remotely. b) Remote manual control available to regulate AF flow for 30 minutes. Then resort to local manual manipulation of valves. AF flow assured as all valves fail open. Shut valves 16 & 17 remotely.	51 51 51 66

1 Pipe break assumed on line to steam generator 4 for failure mode analysis. Analysis and results for breaks on lines to generators 1, 2 or 3 are similar.

2 The equipment numbers are also shown on Figure 10.4-12.

TABLE 10.4-10

(Sheet 2 of 2)

MAXIMUM SYSTEM HEAT SINK REQUIREMENTS

Component To Be Cooled	Number Provided	Number in Operation	Total Cooling Water Flow Rate (gpm)	Total Operating Heat Load (Btu/hr)	
Service air compressor	1	1	70	1.16×10^6	10
Reciprocating-Instrument air compressor	1	1 (common)	10	0.07×10^6	68
Reciprocating-Instrument air aftercooler	1	1 (common)	15	0.15×10^6	68
Rotary Instrument air compressor package	1	1 (common)	27	0.35×10^6	68 68
Auxiliary steam condensate cooler	1	1 (common)	150	4.65×10^6	10 10
Auxiliary steam condensate sample cooler	1	1 (common)	5	0.025×10^6	10
Secondary sampling system	1	1	50	1.25×10^6	10
Auxiliary boiler sample cooler and conductivity cell	1	1 (common)	3	$.0225 \times 10^6$	10 10
Reverse osmosis chiller	1	1 (common)	20	$.3 \times 10^6$	10
Condensate polishing system	1	1	21	0.158×10^6	10
TOTAL	43	36	16,186	77.6085×10^6	10

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TABLE 11.5-1
(SHEET 1 OF 5)

PROCESS RADIATION MONITORING SYSTEM PARAMETERS

Detector Nos.** Unit 1	Detector Nos.** Unit 2	Detector Type	Monitor Service	Monitor Locations (El., Column Coord., Figure Numbers)	Principal Isotopes Monitored	Monitored Medium	Measurement Made	Specified Instrument Range (uCi/Cm³)	MDC (uCi/cm³)	Bases for Alarm Set Points	
Auxiliary Building											: 66 : 66 : 66
XRE 5568A XRE 5568B		Beta scintillator	Plant vent effluent - air particulate (off-line)	El.873 feet 6 in. (Fig. 1.2-35, 1.5A-JA and 8.5A-JA	I-131, I-133, Cs-134, Cs-137*, Co-58, Co-60	Air	Gross beta	5E-11 to 5E-07	5E-11	Note 2	: 46 : 46 : 46 : 46
XRE 5575A XRE-5575B		Gamma scintillator	Plant vent effluent -- Iodine (off line)	El.873 feet 6 in. (Fig. 1.2-35, 1.5A-JA	I-131* I-133	Air	Isotopic -- I-131	Note 6	Note 6	Note 2	: 52 : 52 : 52
XRE-5567A XRE-5567B		Beta scintillator	Plant vent effluent -- noble gases (off-line)	El.873 feet 6 in. (Fig 1.2-35, 1.5A-JA and 8.5A-JA)	Kr-85, Xe-135 Xe-133*	Air	Gross beta	1E-06 to 1E-02	1E-06	Note 2	: 46 : 68 : 68
XRE 5570A XRE 5570B		Beta scintillator CdTe(Cl)	Plant vent effluent -- noble gases (off-line)	El.873 feet 6 inches (Fig. 1.2-14: 8S-ES and Fig. 1.2-20, 9S-ES) Flow diagram 9.4-9	Kr-85, Xe-135 Xe-133*, Kr-88, Xe-138	Air	Gross beta	1E-06 to 1E+05	1E-06	Note 2	: 66 : 68 : 68 : 68 : 66
XRE 5701	-	Beta scintillator	Auxiliary Building ventilation air-noble gases (in-line)	Vent duct, El.873 feet 6 in. (Fig. 1.2-35, 2.5A-FA) Flow diagram Fig. 9.4-2	Kr-85, Xe-135 Xe-133*	Air	Gross beta	1E-04 to 1E 00	1E-04	Note 2, 4	: 68 : 46 : 46 : 46 : 46
1RE 5637	2RE 5637	Beta scintillator	Main steam and feedwater area ventilation air -- noble gases (in-line)	Vent duct, El.852 ft. 6 in. (Fig. 1.2-13, 4S-CS) and Fig. 1.2-19, 13S-CS) Flow diagram Fig. 9.4-4	Kr-85, Xe-135 Xe-133I	Air	Gross beta	1E-04 to 1E 00	1E-04	Note 5	: 68 : 66 : 66 : 66 : 66
XRE 5250	-	Gamma scintillator	Waste gas (on-line)	GWPS, El.862 feet 6 inches (Fig. 1.2-34, 3A-GA) Flow diagram Fig. 11.3-1	Kr-85, Xe-135 Xe-133*	Gas	Gross gamma	1E-01 to 1E-04	1E-01	Note 1	: 46 : 46 : 46 : 46
1RE 4269 1RE 4270	2RE 4269 2Re 4270	Gamma scintillator	Service water (off-line)	El.790 feet 6 in. (Fig. 1.2-31, 4A-GA) Flow diagram Figure 9.2-1	I-131, I-133 Cs-134, Cs-137, Co-58, Co-60*	Water	Gross gamma	1E-05 to 1E-04	1E-05	Note 7	: 68 : 46 : 46 : 46

* Reference nuclide

** Detector numbers preceded by an "X" (i.e., XRE 5701) are common to both units.

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TABLE 11.5-1
(SHEET 2)

PROCESS RADIATION MONITORING SYSTEM PARAMETERS

Detector Nos.** Unit 1	Unit 2	Detector Type	Monitor Service	Monitor Locations (El., Column Coord., Figure Numbers)	Principal Isotopes Monitored	Monitored Medium	Measurement Made	Specified Instrument Range (uCi/Cm^3)	MDC (uCi/cm^3)	Bases for Alarm Set Points	
											; 66 ; 66 ; 66
1RE 4509 1RE 4510 1RE 4511	2RE 4509 2RE 4510 2RE 4511	Gamma scintillator	Component cooling water (off-line)	El.810 feet 6 inches (Fig. 1.2-32, 4A-FA to JA and 6A-FA to JA) Flow diagram Fig. 9.2-3	I-131, I-133 Cs-134, Cs-137, Co-58, Co-60*	Water	Gross gamma	1E-05 to 1E-01	1E-05	Note 7	; 68
XRE 5380		Gamma scintillator	Boron recycle fluid (off-line)	El. 852 feet 6 inches (Figure 1.2-34, 4A-KA) Flow diagram 9.3-11	I-131, I-133, Cs-134, Cs-137 Co-60*	Water	Gross gamma	1E-05 to 1E-01	1E-05	Note 1	; 46 ; 46 ; 68 ; 46
XRE 5251 XRE 5252		Gamma scintillator	LWPS fluids (in-line)	El. 790 feet 6 inches (Fig. 1.2-31, 6A-HA, 2.5A-HA, Flow diagrams Figs. 11-2.4 and 11.2-5	I-131, I-133, Cs-134, Cs-137, Co-58, Co-60*	Water	Gross gamma	1E-05 to 1E-01	1E-05	Note 1	; 68 ; 46 ; 68 ; 46 ; 46
XRE 5253		Gamma scintillator	Effluent Waste Liquid (in-line)	El. 790 feet 6 in. (Figure 1.2-31, 3A-FA Flow Diagram Fig. 11.2-5	I-131, I-133, Cs-134, Cs-137, Co-58, Co-60*	Water	Gross gamma	1E-05 to 1E-01	1E-05	Note 2	; 68 ; 68 ; 68 ; 68
XRE 4180 XRE 4181		Gamma scintillator	Spent fuel pool demineralizer (off-line)	El.852 feet 6 inches (Fig. 1.2-34 5A-KA and 6A-KA) Flow diagram Fig. 9.1-13	I-131, I-133, Cs-134, Cs-137, Co-58, Co-60*	Water	Gross gamma	1E-05 to 1E-01	1E-05	Note 1	; 46 ; 46 ; 68 ; 46 ; 46
XRE 3230	-	Gamma scintillator	Auxiliary steam condensate (off-line)	El.790 feet 6 inches (Fig. 1.2-31 3A-LA) Flow diagram Fig. 10.4-16	Co-60*, Co-58, Cs-134, Cs-137	Water	Gross gamma	1E-05 to 1E-01	1E-05	Note 1	; 46 ; 46 ; 46 ; 46
1RE 5698	2RE 5698	Beta scintillator	Safeguards building ventilation air (in-line)	Vent duct, El.873 feet 6 inches, (Fig. 1.2-35, 2A-HA and 8A-HA) Flow diagram Fig. 9.4-2	Xe-133*, Xe-135 Kr-85	Air	Gross beta	1E-04 to 1E-00	1E-04	Note 4	; 68 ; 56 ; 56 ; 56 ; 56
XRE 5700	-	Beta scintillator	Fuel building ventilation air (in-line)	Vent duct, El. 886 ft. (Fig. 1.2-35 4A-KA) Flow diagram Fig. 9.4-2	Xe-133*, Xe-135 Kr-85	Air	Gross beta	1E-04 to 1E-00	1E-04	Note 4	; 68 ; 56 ; 56 ; 56
XRE 5702	-	Beta scintillator	HVAC room ventilation	Vent duct, El.873 ft. 6 inches, (Fig. 1.2-35, 4A-KA) Flow diagram Fig. 9.4-2	Xe-133*, Xe-135 Kr-85	Air	Gross beta	1E-04 to 1E-00	1E-04	Note 4	; 68 ; 56 ; 56 ; 56

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PROCESS RADIATION MONITORING SYSTEM PARAMETERS

Detector Nos.** Unit 1	Unit 2	Detector Type	Monitor Service	Monitor Locations (El., Column Coord., Figure Numbers)	Principal Isotopes Monitored	Monitored Medium	Measurement Made	Specified Instrument Range (uCi/Cm^3)	MDC (uCi/cm^3)	Bases for Alarm Set Points	
Safeguards Building											: 66 : 66 : 66
1RE 4200	2RE 4200	Gamma scintillator	Steam generator blowdown sample (off-line)	El. 810 feet 6 inches (Fig. 1.2-11, 65-CS to Fig. 1.2-17, 11S-CS) Flow diagram Fig. 9.3-4	I-131, I-133, Cs-134, Cs-137 Co-58, Co-60*	Water	Gross gamma	1E-05 to 1E-01	1E-05	Note 3	: 46 : 46 : 46 : 46
1RE 5179	2RE 5179	Gamma scintillator	Steam generator Blowdown Processing System Fluid (off-line)	El. 810 feet 6 inches (Fig. 1.2-11, 6S-CS and Fig. 1.2-17, 11S-CS) Flow diagram Fig. 10.4-10	I-131, I-133, Cs-134, CS-137, Co-58, Co-60*	Water	Gross gamma	1E-05 to 1E-01	1E-05	Note 1	: 46 : 46 : 46 : 46 : 46
1RE 5502	2RE 5502	Beta scintillator	Containment air-particulate (off-line)	El. 831 feet 6 inches (Fig. 1.2-12, 6S-DS and Fig. 1.2-18, 11S-DS) Flow diagram Fig. 9.4-6	Cs-137*, Rb-88, I-133	Air	Gross beta	5E-11 to 5E-07	5E-11	Note 7	: 68 : 59 : 59 : 59 : 59
1RE 5566	2RE 5566	Gamma scintillator	Containment air iodine	El. 831 feet 6 inches (Fig. 1.2-12, 6S-DS and Fig. 1.2-18, 11S-DS) Flow diagram Fig. 9.4-6	I-131*, I-133	Air	Isotopic - I-131	Note 6	Note 6	Note 4	: 68 : 59 : 59 : 59 : 59
1RE 5503	2RE 5503	Beta scintillator	Containment air noble gas (off-line)	El. 831 feet 6 inches (Fig. 1.2-12, 6S-DS and Fig. 1.2-18, 11S-DS) Flow diagram Fig. 9.4-6	Xe-133*, Kr-85 Xe-135	Air	Gross beta	1E-06 to 1E-02	1E-06	Note 7	: 68 : 46 : 46 : 46 : 46
1RE 406	2RE 406	Geiger-Mueller tube	Reactor coolant letdown line liquid (off-line)	El.831 feet 6 inches (Fig. 1.2-12, 4.5S-ES and 1.2-18, 12.5S-ES) Flow diagram Fig. 9.3-10	Co-60*, Co-58 Cs-134, Cs-137	Water	Gross gamma	1E-00 to 1E+05	1E-00	Note 1	: 46 : 46 : 46 : 46 : 46
1RE 2325	2RE 2325	Geiger-Mueller tube	Main steam noble gas (on-line)	El. 873 feet 6 inches (Fig. 1.2-14, 6S-ES and Fig. 1.2-20, 11S-ES) Flow diagram Fig. 10.3-1	Xe-133*, Kr-85 Xe-135	Steam	Gross gamma	1E-01 to 1E+03	1E-01	Note 3	: 46 : 46 : 46 : 46 : 46

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TABLE 11.5-1
(SHEET 4)

PROCESS RADIATION MONITORING SYSTEM PARAMETERS

Detector Nos.** Unit 1	Unit 2	Detector Type	Monitor Service	Monitor Locations (El., Column Coord., Figure Numbers)	Principal Isotopes Monitored	Monitored Medium	Measurement Made	Specified Instrument Range (uCi/Cm^3)	MDC (uCi/cm^3)	Bases for Alarm Set Points	; 66 ; 66 ; 66
1RE 2326	2RE 2326	Geiger-Mueller tube	Main steam noble gas (on-line)	El.873 feet 6 inches (Fig. 1.2-14, 6S-ES and Fig. 1.2-20, 11S-ES) Flow diagram Fig. 10.3-1	Xe-133*, Kr-85 Xe-135	Steam	Gross gamma	1E-01 to 1E+03	1E-01	Note 3	; 68 ; 46 ; 46 ; 46 ; 46
1RE 2327	2RE 2327	Geiger-Mueller tube	Main steam noble gas (on-line)	El. 873 feet 6 inches (Fig. 1.2-14 6S-ES) and Fig. 1.2-20, 11S-ES) Flow diagram Fig. 10.3-1	Xe-133*, Kr-85 Xe-135	Steam	Gross gamma	1E-01 to 1E+03	1E-01	Note 3	; 46 ; 46 ; 46 ; 46 ; 46
1RE 2328	2RE 2328	Geiger-Mueller tube	Main steam noble gas (on-line)	El. 873 feet 5 inches (Fig. 1.2-14, 6S-ES and Fig. 1.2-20, 11S-ES) Flow diagram Fig. 10.3-1	Xe-133*, Kr-85 Xe-135	Steam	Gross gamma	1E-01 to 1E+03	1E-01	Note 3	; 46 ; 46 ; 46 ; 46 ; 46
Fuel Building											
XRE 4863 XRE 4864		Gamma scintillator	Spent fuel pool water (off-line)	E. 810 feet 6 inches (Fig. 1.2-38, 5F-CF) Flow diagram Fig. 9.1-13	I-131, Cs-137, Co-60*, Co-58	Water	Gross gamma	1E-05 to 1E-01	1E-05	Note 1	; 46 ; 68 ; 46 ; 46
Electrical and Control Building											
XRE 5895A XRE 5896A XRE 5895B XRE 5896B		Beta scintillator	Control Room ventilation intake	El. 854 feet 4 inches (Fig. 1.2-34, 2.9A-DA and 7.1A-DA) Flow diagram 9 4-1	Xe-133*, Xe-135 Kr-85, I-131	Air	Gross beta	1E-06 to 1E-02	1E-06	Note 4	; 68 ; 68 ; 66 ; 66
Turbine Building											
1RE 2959	2RE 2959	Beta scintillator	Condenser off-gas (off-line)	El. 778 feet (Fig. 1.2-22, 7T-FT and Fig. 1.2-27, 8T-FT) Flow diagram Fig. 10.4-3	Kr-85, Xe-133* Xe-135	Gas	Gross beta	1E-05 to 1E-01	1E-05	Note 3	; 68 ; 56 ; 56 ; 56 ; 56

TABLE 11.5-1
(SHEET 5)

PROCESS RADIATION MONITORING SYSTEM PARAMETERS

Detector Nos.** Unit 1	Detector Nos.** Unit 2	Detector Type	Monitor Service	Monitor Locations (El., Column Coord., Figure Numbers)	Principal Isotopes Monitored	Monitored Medium	Measurement Made	Specified Instrument Range ($\mu Ci/Cm^3$)	MDC ($\mu Ci/cm^3$)	Bases for Alarm Set Points	
											: 66
1RE 5100	2RE 5100	Gamma scintillator	Turbine Building drains liquid (off-line)	El. 775 feet 3 inches (Fig. 1.2-22, 4T-FT and 1.2-27, 11T-FT) Flow diagram Fig. 9.3-8	Co-60*, Co-58, Cs-134, Cs-137	Water	Gross gamma	1E-05 to 1E-01	1E-05	Note 2	: 68

NOTES:

1. Alarm set points are based on process system requirements for normal operation.
2. Alarm set points are based on effluent requirements for normal operation.
3. Alarm set points are based on Steam Generator Tube Rupture (SGTR) detection.
4. Alarm set points are based on airborne radioactivity considerations for personnel protection.
5. Alarm set points are based on verification of containment integrity for accident conditions.
6. Since these iodine detectors monitor the accumulated activity of iodine in the iodine adsorber cartridge, the specified instrument range ($\mu Ci/Cm^3$) and MDC ($\mu Ci/Cm^3$) are dependent on the frequency of the cartridge replacement, flow rate, sensitivity, and detector count rate. : 68
   The activity concentration may be calculated as follows:

   $$\text{Activity } (\mu Ci/cm^3) = \frac{\text{Count Rate (counts/min)}}{\text{Time since cartridge replacement (min)} \times \text{Sample flow rate } (cm^3/min) \times \text{Fractional cartridge efficiency} \times \text{Sensitivity (counts per min}/\mu Ci)}$$

   The sensitivity of the detector is > 4E+4 cpm/μCi.
7. Alarm setpoints are based upon leakage detection. : 68

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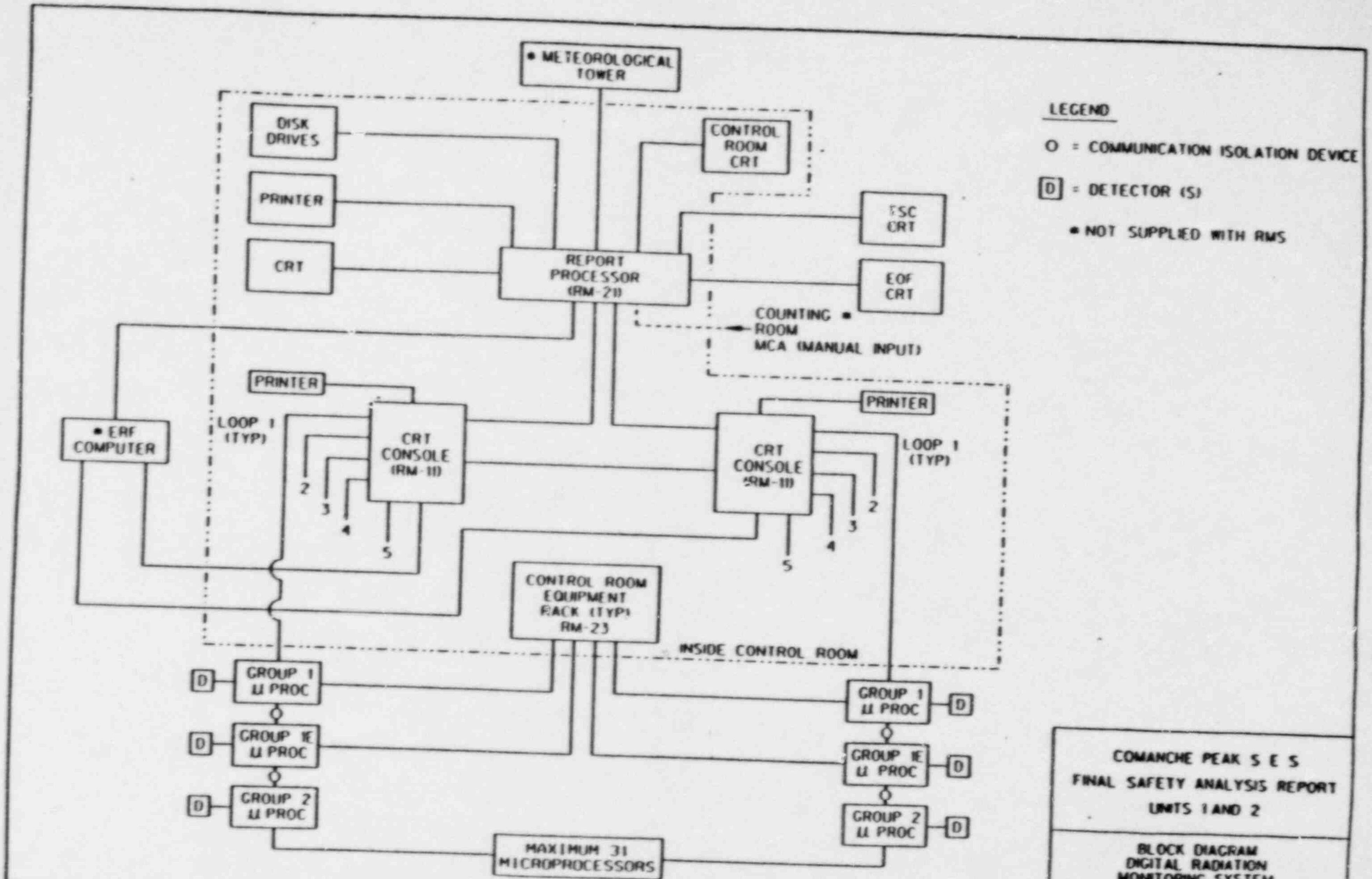

COMANCHE PEAK S E S
FINAL SAFETY ANALYSIS REPORT
UNITS 1 AND 2

BLOCK DIAGRAM
DIGITAL RADIATION
MONITORING SYSTEM

FIGURE 11.5-1

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13.1.1.1.2 Preoperational Activities

Nuclear Operations is responsible for the start-up testing activities | 60
which are those testing activities up to fuel load. These | 68
activities are performed by the CPSES Start-up Staff under the |
direction of the Manager, Start-up and Test. A description of the |
CPSES start-up and test program is presented in Section 14.2. | 60

13.1.1.1.3 Technical Support for Nuclear Operations | 60

Technical services and backup support for Nuclear Operations are | 60
furnished by Nuclear Engineering, Administration, Support and |
Engineering and Construction. Personnel are available who are |
competent in technical matters related to plant safety and other
engineering and scientific support aspects. In the event Nuclear | 60
Operations needs assistance with specific problems, the services of |
qualified individuals will be engaged as appropriate. |

The special capabilities that are available are: | 6

- Nuclear, mechanical, structural, electrical, thermal-hydraulic, | 49
  metallurgical and materials, chemistry, and instruments and |
  controls engineering |

- Nuclear Safety | 6

- Plant Chemistry

- Health Physics

- Fueling and refueling operations support

- Maintenance support

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Manager, Safeteam - The responsibilities of the Manager, Safeteam are: 60

- Manage the Safeteam Program for the review and investigation of employee safety concerns. 60

- Ensure both departing employees and employees with concerns are interviewed. 60

- Maintain the independence and credibility of the Program. 60

- Ensure adequate responses are provided to employees with concerns. 60

13.1.1.2.1 Organization - Nuclear Operations 62

Vice President, Nuclear Operations - The responsibilities of the Vice President, Nuclear Operations are: 60

- Ensure CPSES operation and maintenance activities are conducted in compliance with federal, state, and local laws, regulations, licenses, codes, and within established corporate and NEO policies, plans, and procedures. 62

- Provide direction and guidance to the Manager, Plant Operations; the Director, Nuclear Training; the Manager, Technical Support; the Manager, Plant Support; the Manager, Administrative Support; the Plant Evaluation Manager; and the Manager, Start-up and Test. 68

The CPSES operating organization is discussed in Section 13.1.2 and illustrated in Figure 13.1-3. Organizations which support plant operations are discussed below. 62

- Provide for the review and assessment of reports of nuclear industry operating experiences. | 62

- Develop CPSES plant specific Emergency Preparedness exercise scenarios and conduct the critique of each exercise. | 62

Manager, Technical Support - The responsibilities of the Manager, Technical Support are discussed in Section 13.1.2.2. | 62

Manager, Plant Support - The responsibilities of the Manager, Plant Support are discussed in Section 13.1.2.2. | 62

Manager, Startup and Test - The responsibilities of the Manager, Startup and Test are discussed in Section 13.1.2.2. | 68

13.1.1.2.2 Organization - Administration | 60

Vice President, Administration - The responsibilities of the Vice President, Administration are: | 60

- Provide assistance, as required, in various areas of CPSES construction and operation. | 62

- Provide for engineering and administrative services to support construction and operation activities at CPSES (and for the Operations Review Committee). | 62

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60 - Provide project and site construction management for backfit and betterment projects.

60 - Provide engineering design and drafting services for specifications and design documents.

60 - Provide for external construction personnel and services, as needed.

60 - Provide site facilities and support services for the engineering and construction forces at CPSES.

60 - Provide direction and guidance to the Director of Engineering; the Director of Construction; and the Director of Projects.

60 - Direct management of CPSES engineering and construction contractors.

60 Director of Engineering - The responsibilities of the Director of Engineering are:

64 - Support CPSES managers and the NEO group in the specification, review, evaluation, design, and procurement of plant systems, apparatus, and materials.

60 - Perform site selection studies.

60 Director of Construction - The responsibilities of the Director of Construction are:

60 - Support Engineering and Construction Project Managers by providing field construction management, supervision and labor, and construction-engineering interface for CPSES.

68

Roger D. Walker - Manager, Nuclear Licensing 68

Education: 68

B.S. Operations Management - California State University of Hayward - 1973 68

U.S. Navy Nuclear Power School - 1960 68

Experience: 68

1959 - U.S. Machinist Mate First Class. 68

1967 - Bruning Company, Laboratory Technician. 68

1967 - General Electric Company, Vallecitoes Nuclear Center, Reactor Operator. Licensed Reactor Operator. 68

1973 - General Electric Company, BWR Training Center, Instructor. Licensed Senior Reactor Operator 68

1974 - U.S. Nuclear Regulatory Commission, Office of Inspection and ENforcement, Nuclear Engineer. 68

1978 - U.S. Nuclear Regulatory Commission, Reactor Inspector. 68

1979 - U.S. Nuclear Regulatory Commission, Senior Resident Inspector. 68

1981 - U.S. Nuclear Regulatory Commission , Chief, Projects Section 2C, Division of Reactor Projects. 68

1984 - U.S. Nuclear Regulatory Commission, Chief, Operations Branch, Division of Reactor Safety. 68

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68 1984 - U.S. Nuclear Regulatory Commission, Deputy Director, Division of Reactor Projects.

68 1984 - U.S. Nuclear Regulatory Commission, Director, Division of Reactor Projects

68 1987 - TU Electric, Assigned as Executive Assistant to Executive Vice President, Nuclear Engineering & Operations.

68 1987 - TU Electric, Assigned as Manager, Nuclear Licensing.

68 Activities:

68 Member - American Nuclear Society

68 Owen W. Lowe - Director of Engineering

59 Education:

68 BSCE - Tufts University - 1964

68 MSCE - Tufts University - 1967

59 Experience:

68 1966 - AVCO Corporation, Systems Division, As a Structural Engineer responsible for design of support structures.

68 1971 - Richard J. Donovan, Inc., Consulting, Winchester, Ma, as a Structural Engineer.

68 1972 - Raytheon Company, Mechanical Systems Laboratory, as a Structural Engineer.

68 1973 - San-Vel Concrete Corporation, Littleton, Ma, as a Project Engineer.

68 1974 - Stone and Webster Enineering Corporation (SWEC), Structural Division, as a support engineer (Structural Mechanics).

68 1975 - SWEC, as the Engineering Mechanics Division Lead Engineer for Duquesne Light Company (Beaver Valley 2 Nuclear Power Station).

68 1985 - SWEC, as an Assistant Project Engineer for Northeast Utilities Service Company (Millstone 3 Nuclear Power Station).

68 1986 - SWEC, as an Evaluation Team Member for engineering mechanics activities at Detroit Edison (Enrico Fermi Atomic Power Plant).

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1987 - TU Electric Company as the Manager of Mechanical Engineering. 68

1987 - Assigned to position of Director of Engineering, Comanche Peak Steam Electric Station. 68

Affiliations:

American Society of Civil Engineers - Member 68

Professional Engineer Registrations: Massachusetts, Pennsylvania, Connecticut. 68

62 13.1.2 OPERATING ORGANIZATION

62 The CPSES Operating Organization is divided into two factions:

68 1) Operating Organization - Plant Operations, the Technical Support Department, the Test Department, and the Plant Support Department organizations are described below.

58 2) Management and additional support organizations - The Vice President, Nuclear Operations; Director, Nuclear Training; Manager, Administrative Support; and the Plant Evaluation Manager; in Section 13.1.1.2.1.

62 13.1.2.1 <u>Plant Organization</u>

62 The Manager, Plant Operations reports to the Vice President, Nuclear Operations and is responsible for the operation and maintenance of the Comanche Peak Steam Electric Station (CPSES). Reporting to the Manager Plant Operations are the Operations Manager, the Maintenance
68 Manager and the Instrumentation and Controls Manager. Operating support is provided by the Technical Support, the Test Department and Plant Support Departments as discussed in Sections 13.1.2.2.4 and 13.1.2.2.5. Reporting interfaces and relationships are shown on Figures 13.1-2 and 13.1-3. Quality Assurance is illustrated on
62 Figure 13.1-2 and discussed in Section 13.1.1.2.4 (organization) and Chapter 17 (program).

13.1.2.2 <u>Personnel Functions, Responsibilities, and Authorities</u>

60 In addition to the responsibilities for plant operation and maintenance, the Manager, Plant Operations (see Section 13.1.1.2.1) is chairman of the Station Operations Review Committee (SORC). The SORC serves in an advisory capacity to the Manager, Plant Operations
15 on safety-related matters. The Manager, Plant Operations is a member of the Operations Review Committee (ORC) of CPSES operations. The SORC and ORC are discussed in Section 13.4.

13.1.2.2.1 Operations Department 49

The Operations Manager is responsible for the operation of CPSES and 62
the management and training of Operations Department personnel. He 68
coordinates the generation of power and changes in plant operating
modes and participates in the start-up test program and the refueling
efforts. The Operations Manager is a member of the SORC.

Reporting to the Operations Manager are the Operations Engineering 62
Manager and the Shift Operations Manager. The Operations Engineering
Manager provides technical assistance for the development and
maintenance of Operations Department procedures to ensure CPSES is
operated as prescribed in Section 13.5. The Shift Operations Manager
directs the Shift Supervisors and is responsible for ensuring that
shift operations personnel are trained and qualified (see Section
13.2). He is also responsible for the operation of the radioactive
waste handling systems and for the processing and packaging of
radioactive waste.

Shift Operations

The Shift Supervisors are members of management responsible to the 62
Shift Operations Manager for the operation of the CPSES units. Each
Shift Supervisor is responsible for supervising the evolutions
conducted during his shift and ensuring that they are conducted in
accordance with the operating license, station procedures, and
applicable directives and policies. The Shift Supervisors are 62
responsible for supervising shift operations personnel and for
conducting on-shift training. During periods when senior management
personnel are not on site, the Shift Supervisor assumes responsibility 25
for all station activities. The Shift Supervisors are required to 62
maintain an USNRC Senior Reactor Operator License.

15 The Assistant Shift Supervisors are members of management and assist
the Shift Supervisors in discharging their responsibilities for
25 supervision of the CPSES units. The Assistant Shift Supervisor may
62 assume the duties of the Shift Supervisor in his absence. The
Assistant Shift Supervisors are required to maintain an USNRC Senior
Reactor Operator License.

62 The Reactor Operators are responsible for routine evolutions on their
assigned unit and for monitoring the status of that unit. The
Reactor Operators are supervised by the Shift Supervisor or an
Assistant Shift Supervisor and are required to maintain an USNRC
Reactor Operator License.

15 Auxiliary Operators work under the direction of the Shift Supervisor
62 or an Assistant Shift Supervisor. The Auxiliary Operators'
responsibilities include operating equipment from the Control Room and
operating and servicing equipment remote from the Control Room at the
direction of Control Room operations personnel.

62 Shift Technical Advisors will be on each shift until such time that
shift operations personnel are qualified to meet the applicable
education, experience and training requirements of NUREG-0737, October
31, 1980. They work with the Shift Supervisors, but report to the
Operations Engineering Manager.

62 An Operations Aide functions as the Radwaste Coordinator and is
responsible to the Shift Operations Manager for the operation of the
liquid and gaseous waste systems, and for the operation of the makeup
water systems. He coordinates the various waste processing and
62 packaging activities, as well as makeup water requirements. The
Radwaste Coordinator serves as an interface between the Operations
Department and other CPSES personnel involved in waste or water
handling evolutions.

13.1.2.2.2 Maintenance Department 49

The Maintenance Manager is responsible for all maintenance activities 62
associated with mechanical and electrical equipment including
preventive maintenance programs. The Maintenance Manager ensures
that maintenance personnel are trained and qualified. He ensures 68
that maintenance activities during routine operation and refueling
outages and activities associated with the start-up test program are
conducted in accordance with approved procedures and instructions,
regulatory requirements, and applicable policies and directives. The
Maintenance Manager is responsible for developing and maintaining 62
procedures and instructions as described in Section 13.5. The
Maintenance Manager is a member of the SORC.

Electrical and Mechanical Maintenance 49

The Electrical Maintenance and Mechanical Maintenance Managers are 62
responsible to the Maintenance Manager for the maintenance of
electrical and mechanical plant equipment. They are responsible for
managing their respective areas through the Electrical and Mechanical 49
Maintenance Supervisors who direct the work of the electricians and
mechanics.

Their duties include preparation of maintenance procedures and 49
instructions, preventive and corrective maintenance planning and
scheduling, and ensuring that the maintenance personnel are trained
and supervised.

Maintenance Engineering 49

The Maintenance Engineering Manager is responsible to the Maintenance 62
Manager for providing engineering, technical and administrative
support to the Maintenance Department.

62 His duties include developing the Preventive Maintenance Program and coordinating the implementation of station modifications in maintenance. The Preventative Maintenance Program includes the Managed Maintenance Program, the Station Planning Program, and the Nondestructive Examination Program.

Managed Maintenance Program

25 In 1978, a systematic effort was initiated to find a method for increasing the reliability and availability of the CPSES units. That study resulted in the implementation of a Managed Maintenance Program which is designed to provide the plant staff with the maintenance data and information systems necessary to support proper planning and management of the maintenance activities. This is accomplished by a systematic evaluation of each plant component in which all maintenance activities are identified, and the resources for performing these activities are assessed. Examples of these resources are: manpower, radiation exposure, special tools, spare tools, spare parts, procedure number, and plant condition required for performing the activity.

25 Once all maintenance activities have been identified, two sets of maintenance plans are generated. The first set is the on-line preventive maintenance plan which includes all of those maintenance
64 activities which can be performed with the plant at power. These activities are scheduled on a computer with various printouts and worksheets for the craft and supervisory personnel.

25 The second set of plans includes the outage-related work which will normally be performed concurrent with refueling. These activities, along with the refueling sequence are plotted on a computer which is used for managing the outage.

A significant point to note is that, because of inservice inspection 25
requirements, the outage plan is repetitive with a ten-year cycle.
The plant staff has completed the outage plans for the first ten-year
cycle and, because of the repetitive nature of the work, has a plan
for each year of commercial operation throughout the life of the
plant. The Managed Maintenance Program is designed to be an active
program which will be updated as plant conditions and requirements
change.

13.1.2.2.3 Instrumentation and Controls (I&C) Section 62

The Instrumentation and Controls Manager is responsible for the 62
supervision of the Instrumentation and Controls personnel. He is
responsible for ensuring the proper installation, calibrating,
testing, and maintenance of station instrumentation and control
systems. In discharging these responsibilities, he ensures that I&C
technicians are trained and that safety-related activities are
conducted in accordance with applicable procedures, instructions,
policies, and regulations. The I&C Supervisors are responsible to
the I&C Manager for directing the day-to-day activities of the I&C
Technicians.

13.1.2.2.4 Technical Support 62

The Manager, Technical Support reports to the Vice President, Nuclear 62
Operations and is responsible for managing the Technical Support
Department and for providing technical support and services for plant
operations. He is charged with the responsibility for station 68
chemistry, radiation protection, results engineering, environmental
monitoring, and regulatory compliance. The Manager, Technical
Support is a member of the SORC. 62

25 Radiation Protection

62 The Radiation Protection Manager is responsible for the supervision of Radiation Protection Supervisors, for the transportation of radioactive material, for the CPSES Radiation Protection program (see Section 12.5) and for implementation of the station policy of maintaining operational radiation exposures "as low as reasonably achievable." In discharging these responsibilities, he ensures that his personnel are trained and that safety-related activities are
62 conducted in accordance with applicable procedures, instructions, policies and regulations.

62 The Radwaste Transportation Coordinator is responsible to the Radiation Protection Manager for the shipping and receiving of all radioactive material and the handling, processing and storage of solid radioactive waste. The Radwaste Transportation Coordinator is
46 responsible for implementing the procedures which direct the conduct of radiation protection personnel during these operations. He also insures that radiation protection personnel are trained in applicable DOT and NRC regulations.

25 Chemistry & Environmental

62 The Chemistry & Environmental Manager is responsible for the supervision of chemistry personnel, for monitoring and maintaining the station's fluid systems chemistry and for implementing the station environmental monitoring program. In discharging these responsibilities, he ensures that his personnel are trained and that safety-related activities are conducted in accordance with applicable procedures, instructions, policies and regulations.

Results Engineering

The Results Engineering Manager is responsible for performing duties 68
and providing support in the areas of technical support, testing, and design modification implementation. The Results Engineering Manager supervises a staff of engineers who are responsible for performing the 62
various duties mentioned above. These engineers primarily have experience and training in the areas of electrical, nuclear, or mechanical engineering.

13.1.2.2.5 Plant Support 62

The Manager, Plant Support reports to the Vice President, Nuclear 62
Operations and is responsible for Station Security, Warehouse activities, safety services, and fire protection.

Station Security 62

The Security Manager is responsible for the overall development and 62
implementation of the plant and site security programs. The Plant Security Supervisor is responsible for the activities of the Plant Security Force and for implementing the Station Security Plan as described in Section 13.6. The Industrial Security Supervisor is responsible for the activities of the Industrial Security Force and for implementing site security outside of the protected area.

Safety Services 62

The Safety Services Manager is responsible for industrial safety and 62
fire protection. The Fire Protection Supervisor reports to the Safety Services Manager and is responsible for the fire protection program and for the activities of the CPSES Fire Brigade.

68 13.1.2.2.6 Test Department

68 The Manager, Startup and Test reports to the Vice President, Nuclear Operations and is responsible for managing the Test Department personnel. He is responsible for the Startup program for both units, including the initial startup. The startup program is described in section 14.2. The Manager, Startup and Test is also responsible for design modification acceptance tests, station performance tests, and those assigned station surveillance tests (typically the 18 month interval tests or integrated tests such as ILRT). The Manager, Startup and Test is a member of the SORC for test related activities.

68 13.1.2.2.7 Supervisory Succession

15 The Manager, Plant Operations is responsible for the operation of
62 CPSES. If the Manager, Plant Operations is absent or becomes incapacitated, then, unless otherwise designated, the following members of his staff assume his responsibilities in the order listed:

62 1. Operations Manager
62 2. Maintenance Manager

During back shift and weekend periods when the station staff is not on site, the Shift Supervisor is responsible for all activities at CPSES.

13.1.2.3 Shift Crew Composition

The minimum on-duty shift complement for various modes of single and dual unit operation is shown in Table 13.1-2 and is as follows:

A Shift Supervisor shall be onsite at all times when at least one unit is loaded with fuel.

68 Charles E. Scott - Manager, Start-up and Test

60 Education:

60 BSEE - New Mexico State University - 1972

60 Experience:

60 1970 - Houston Lighting and Power. Employed as a Co-op Engineer in Distribution/Substation Engineering.

60 1971 - El Paso Electric. Employed as a night dispatcher at Mesilla Valley Service Center.

60 1973 - Dallas Power and Light. Operating Division and Construction Coordination. Associate Engineer responsible for coordinating various phases of distribution construction.

60 1975 - Texas Utilities Generating Company, Big Brown SES. Employed as a power engineer providing technical and administrative assistance in all areas of operation for 2-575 MW Lignite-fired generating units.

60 1977 - Monticello SES - Electrical Start-up Lead. Responsible for testing, troubleshooting and initial operation of all plant electrical systems, precipitator and scrubber for 750 MW lignite-fired generating unit.

60 1978 - Comanche Peak SES - Electrical Start-up. Initial testing and operation of electrical systems, plant equipment and final walkdown of all plant electrical turnovers.

1980 - Assigned position of Maintenance Engineer at CPSES Plant
60 Operations. Responsible for technical direction for the maintenance organization, direction of contractors and developing and coordinating plant outages.

1982 - Assigned position of Electrical Maintenance Engineer at CPSES. Accountable for all phases of electrical maintenance including developing and implementing the Meter and Relay program. 60

1985 - Assigned as Manager, Start-up, TUGCO. 60

1987 - Assigned as Manager, Startup and Test, CPSES. 68

68 | Results Engineering Manager - To be Announced

Table 13.1-1

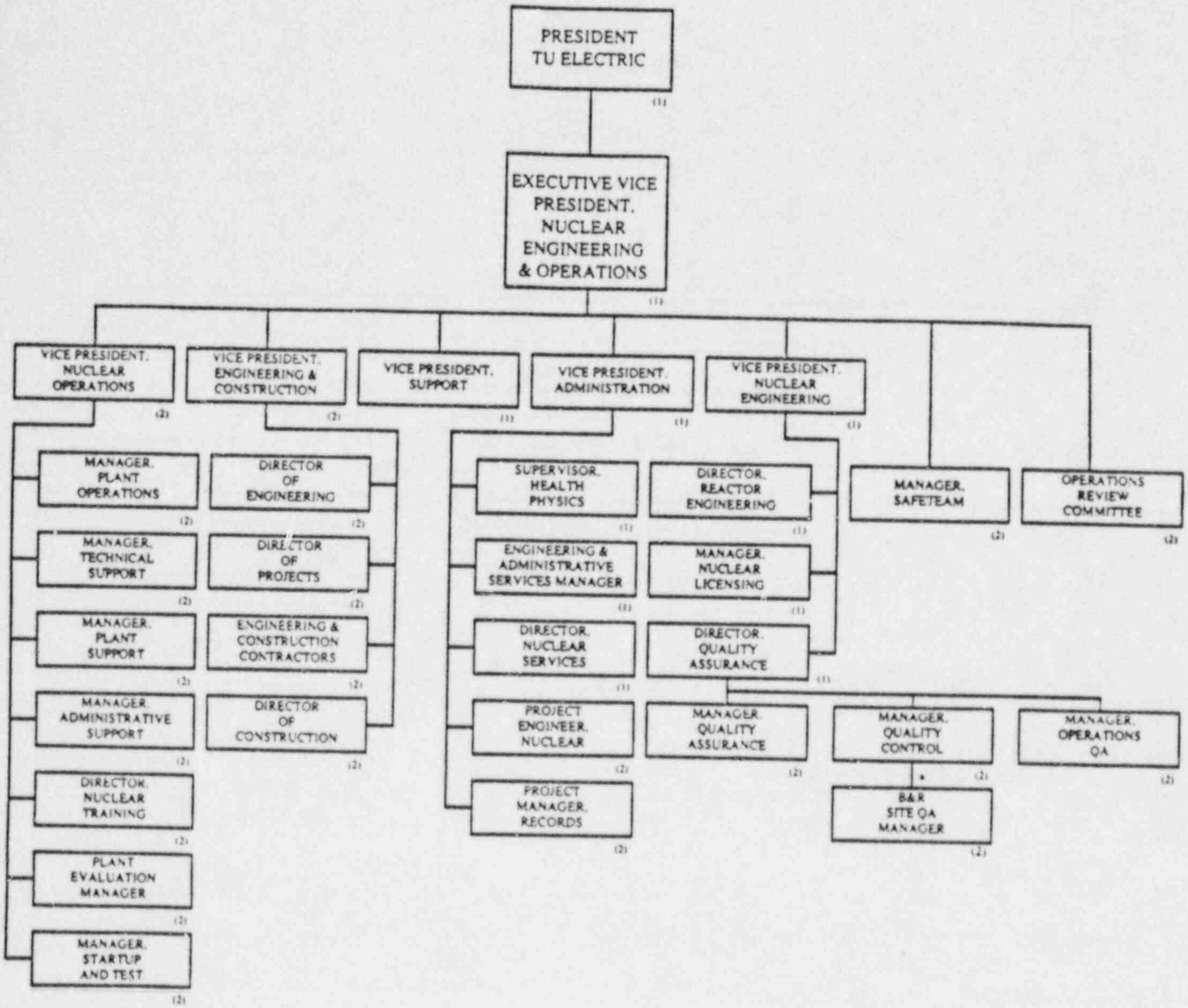
(Sheet 2)

	HIGH SCHOOL DIPLOMA OR EQUIVALENT	B.S. IN ENGINEERING OR SCIENCE OR EQUIVALENT	TOTAL POWER PLANT EXPERIENCE	NUCLEAR POWER PLANT EXPERIENCE	OTHER EXP. IN SPECIALTY CRAFT OR DISCIPLINE	TECHNICAL OR ACADEMIC TRAINING ALLOWANCE FOR EXPERIENCE		NRC REACTOR OPERATOR LICENSE	NRC SENIOR REACTOR OPERATOR LICENSE	
						MIN.	MAX.			
INSTRUMENT AND CONTROL TECHNICIAN					2		1(2)			| 62
MANAGER, TECHNICAL SUPPORT		X	8	1			4			| 62
MANAGER, PLANT SUPPORT		X(1)	5	1			2			| 62
RESULTS ENGINEERING MANAGER		X	6	2			4			| 62
RADIATION PROTECTION MANAGER		X	5	2		2	4			| 62
CHEMISTRY AND ENVIRONMENTAL MANAGER		X	5	1		2	4			| 62
CHEMISTRY AND ENVIRONMENTAL SUPERVISOR	X				4	2	4			| 62
CHEMISTRY AND ENVIRONMENTAL STAFF CHEMIST		X								| 62 | 62
RADIATION PROTECTION TECHNICIAN					2		1(2)			| 62
CHEMISTRY AND ENVIRONMENTAL TECHNICIAN					2		1(2)			| 62
MANAGER, OPERATIONS QA	X			2	6	1	4			| 62
MANAGER, STARTUP AND TEST		X	8	2						| 68

NOTE: (1) Desirable, but not mandatory

(2) Recommended in addition to experience requirement

(3) Source Regulatory Guide 1.8



PRIMARY LOCATION
(1) Corporate Office
(2) CPSES

* For other than
ASME Sec. III, Div.1
activities only.

COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

NUCLEAR ENGINEERING
AND OPERATIONS (NEO) GROUP

FIGURE 13.1-2

65 Operations

65 Maintenance

65 Instrumentation & Control

65 Technical Support

65 Radiation Protection

65 Quality Assurance

65 Emergency Planning

65 Security

68 Test

65 The members and their alternates, including the alternate for the chairman shall be designated, in writing, by the Vice President, Nuclear Operations.

13.4.1.1.2 SORC Membership Qualifications

The qualifications of the regular members of the SORC will be as stated in Section 13.1.3.1, Minimum Qualification Requirements.

13.4.1.1.3 Conduct of SORC Meetings

Meetings of the SORC will be held on a routinely scheduled basis at intervals of a least once per calendar month. Special meetings may be called as needed by the SORC Chairman or by the SORC Vice Chairman in
55 the absence of the SORC Chairman. A quorum of SORC members will consist of the Chairman, or in his absence, the Vice Chairman, and four of the remaining regular members or their designated alternates. An agenda whose items will be submitted by the committee members will be prepared prior to each regular SORC meeting. Minutes of each SORC meeting will be recorded and these minutes along with other pertinent documentation will become a part of the permanent plant records.

6. Abnormal occurrences and unusual events.

7. Emergency Plan activities.

55 8. Safety evaluations for: procedures, changes to procedures, equipment, systems, or facilities, and tests or experiments completed under the provisions of 10CFR50.59 to verify that such actions did not constitute an unreviewed safety question.

55 9. Indications of deficiency in design or operation of nuclear-safety-related systems.

68 10. Review Test program activities.

55 11. Any other nuclear-safety-related matter as defined appropriate by the Manager, Plant Operations.

15 13.4.2 INDEPENDENT REVIEW

15 Activities affecting station safety occurring during the CPSES operational phase will be independently reviewed by the Independent Safety Engineering Group (ISEG) and the Operations Review Committee (ORC).

15 13.4.2.1 <u>Independent Safety Engineering Group (ISEG)</u>

15 The Independent Safety Engineering Group is the organization whose
65 function is to perform independent reviews of plant operations. It consists of a group of offsite engineers who report to the Plant Evaluation Manager. This group will have experience and training in
59 nuclear power plant operations and engineering.

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| Q442.2

1. Definition of the specific area where the reactor operator who is at the controls of the unit must remain. This area is shown on Figure 13.5-1 and defines the area of applicability of position C.1.n of Regulatory Guide 1.29 (See Appendix 1A(B)). | 3 | 59

2. Measures to control access to the Control Room.

3. Procedures for proper shift relief and turnover.

4. Procedures and instructions for the control of log and record keeping. | 53

During station operation, the Shift Supervisor shall be responsible for ensuring that equipment control procedures are followed and properly implemented. These procedures will provide control of equipment, as necessary, to maintain personnel safety and reactor safety, and to avoid unauthorized operation of equipment. To secure and identify equipment in a controlled status, measures such as temporary bypass lines, electrical jumpers, lifted electrical leads, and temporary trip point settings shall be controlled by approved procedures which shall include requirements for independent verification where appropriate. A log will be maintained of the current status of temporary modifications which are not specified and controlled by approved procedures or instructions. | 55

| 68

The plant operations, Test, and technical support departments shall be responsible for developing and implementing procedures, instructions and schedules to describe and control a surveillance inspection | 68

53 Maintenance of safety-related equipment will be properly pre-planned and performed in accordance with written procedures, written instructions, or drawings appropriate to the circumstances (for example, skills normally possessed by qualified maintenance personnel may not require detailed step-by-step delineation in a written
65 procedure). It is the responsibility of the Maintenance Manager to implement a maintenance program for safety-related mechanical and
68 electrical equipment. It is the responsibility of the Manager, Start-up and Test to assist in performance testing of safety-related mechanical and electrical equipment to assure adequate levels of performance. It is the responsibility of the Instrument and Controls
65 Manager to implement a maintenance program for safety-related instruments and controls.

General rules for the control and administration of the maintenance
53 program will be written before fuel loading. These general rules will form the basis for developing the repair or replacement procedures and instructions at the time of failure.

53 Procedures and instructions will be written early in plant life for maintenance of safety-related equipment expected to require recurring maintenance. When failure of safety-related equipment occurs, the cause will be evaluated; however, since the probability of failure is usually unknown and the time and mode of failure are usually unpredictable, procedures and instructions will not generally be written for repair of most equipment prior to failure. As experience is gained in operation of the plant, routine maintenance will be altered to improve equipment performance and repair procedures and instructions will be written and improved as required.

A preventive maintenance schedule will be developed which will describe the frequency and type of maintenance to be performed. A preliminary schedule will be developed early in plant life and will be refined and changed as experience with the equipment is gained.

13.5.2.2.7 Security Procedures

65 It is the responsibility of the Security Manager to prepare and maintain detailed, written and approved procedures to implement the security plan. These procedures will supplement the physical barriers and other features designed to control access to the station and, as appropriate, to vital areas within the station. Information concerning specific design features and administrative provisions of the security plan is accorded limited distribution on a need-to-know basis. The security procedures are discussed in greater detail in Section 13.6, Industrial Security.

13.5.2.2.8 Fuel Handling Procedures

65 The Manager, Technical Support is responsible for developing fuel handling procedures to include receipt and receipt inspection, fueling/ refueling and fuel handling, storage of fuel, and fuel shipment.

13.5.2.2.9 Fire Protection Procedures

Q422.3

65 The Fire Protection Supervisor is responsible for the preparation and implementation of procedures for fire protection. These procedures
3 are described in more detail in FSAR Section 13.3 Appendix B, Section 4.0.

68 13.5.2.2.10 Test Procedures

68 The Manager, Startup and Test is responsible for the preparation and implementation of procedures for prerequisite, preoperational acceptance, design modification acceptance, station performance, initial startup tests, and those surveillance tests assigned to the Test Department.

14.2 INITIAL TEST PROGRAM

14.2.1 SUMMARY OF TEST PROGRAM AND OBJECTIVES

The purpose of the startup program for Comanche Peak Steam Electric Station (CPSES) is to assure that the installed station structures, systems and components will be subjected to tests as required to verify that the plant has been properly designed and constructed and is ready to operate in a manner that will not endanger the health and safety of the public, and to provide assurance of total plant reliability for operation. The startup program will also ensure that the procedures for operating the plant safely have been evaluated and demonstrated, and that the operating organization is knowledgeable about the plant and procedures and is prepared to safely operate the facility.

The necessary procedures to control, implement, and document the test 68
program are established by the Test Department Administration Manual and the Station Administration Manual, and summarized in the following sections. TU Electric, as the applicant, has responsibility for overall direction and management of the startup program.

The testing activities to be performed on safety-related systems at 11
the CPSES are divided into three major phases: prerequisite testing, preoperational testing and initial startup testing. Prerequisite testing will be conducted in order to verify the integrity, proper installation, cleanliness, and functional operability of the system components.

For systems and components which have no nuclear safety-related requirements, acceptance testing will be performed to verify proper system performance and to ensure reliable and efficient operation of the plant. There are no unique or "first of a kind" design features of CPSES which require special testing provisions.

Preoperational and initial startup tests will be conducted in accordance with approved tests procedures. Review, approval, and revision of test procedures and the evaluation and disposition of test results will be accomplished by methods specified in the Test Department Administrative procedures and summarized in Section 14.2.3. | 68

The startup program will utilize, to the extent practical, operations personnel and operating procedures to provide familiarization with the plant installation and demonstrate the adequacy of operating procedures.

Testing activities will be coordinated through a Joint Test Group (JTG), as described in Section 14.2.2.5. | 68

| 68

14.2.2 ORGANIZATION AND STAFFING

14.2.2.1 <u>General Description</u>

TU Electric is responsible for the overall administration and | 37
technical direction of the CPSES startup program. The Manager, Startup and Test is responsible for the coordination, direction, and implementation of the startup program. | 68

68 The Test Department organization is shown in Figure 14.2-1. The JTG
37 organization is shown in Figure 14.2-2. The SORC membership is
described in Section 13.4.1.1.1.

68 14.2.2.2 Test Department Organizational Responsibilities

68 The duties and responsibilities of the Test Department under the
direction of the Manager, Startup and Test are as follows:

38 1. Preparation of testing schedules;

38 2. Coordination with appropriate construction organizations for release of completed components/systems to facilitate testing;

38 3. Coordination with the appropriate engineering organizations to resolve component/system design and operating problems;

38 4. Preparation of test procedures;

5. Conduct tests to demonstrate adequate and safe component and system performance.

68 6. Perform the initial startup test sequence to ensure a safe and orderly power ascension program.

Support personnel including craft labor and technicians who will be engaged in testing activities may be supplied by TU Electric or contractors. The number of such support personnel will vary according to test program requirements. 68

14.2.2.2.1 Manager, Startup and Test 68

The Manager, Startup and Test will supervise all testing personnel assigned to the Test Department. The responsibilities of the Manager, Startup and Test include: 68

1. Chairman of the JTG; 68

2. Membership on the SORC for test activities. 68

3. Administration of the development of plans and schedules regarding the status of the startup program; 68

4. Administration of the development of individual test procedures; 38

5. Continuing analysis of construction and equipment installation schedules for compatability with testing schedules and implementation of corrective actions to minimize conflicts; 38

6. Review test procedures and results or assure that such reviews are conducted by qualified personnel within the Test Department; 68

68 7. Assure proper and timely notifications and reports pertaining to testing activities are submitted to the Nuclear Regulatory Commission and other regulatory agencies.

38 14.2.2.2.2 Lead Startup Engineer

38 The Lead Startup Engineer will provide technical direction to system test engineers and others assigned to the preoperational test group and will have the following duties and responsibilities:

68 1. Review of Test Department administrative and prerequisite test procedures and ensure that the procedures and test results have been approved at the appropriate level;

68 2. Review and submittal of design change requests originated by the Test Department in accordance with the appropriate Test Department Administrative Procedures;

68 3. Assure testing activities are conducted in accordance with the Quality Assurance Program and applicable procedures.

68 14.2.2.2.3 Unit 1 Test Manager

68 The Unit 1 Test Manager is responsible for the completion of the Unit 1 test program through commercial operation. This responsibility commences with the Unit 1 Prestart Test Program and includes the following:

68 1. Development and implementation of the Unit 1 Prestart Test program;

68 2. Development and implementation of the Test Department surveillance tests;

3. Development and implementation of the initial startup program; and | 68

4. Coordination of the Unit 1 testing with those activities of engineering, construction, and operations required to support the startup program. | 68

14.2.2.2.4 System Test Engineers | 68

The duties and responsibilities of the system test engineers include:

1. Preparation of assigned test procedures to direct and guide specific tests in accordance with a standard format;

2. Direction to support personnel and others during performance of tests including appropriate interface with station operators;

3. Ensuring the safety of personnel and plant equipment during testing;

4. Familiarization of support personnel with specific tests;

5. Identification of deficiencies that could adversely affect test performance;

6. Assembly of test data and preparation of test reports for evaluation of test results by others;

7. Implementation of tagging procedures; and

8. Responsibility to disallow or terminate testing due to conditions which could endanger personnel or equipment.

14.2.2.3 Operating Staff

The CPSES operating staff will be involved in the startup program in several capacities throughout preoperational and initial startup testing. This involvement will include review of test procedures and results and the direct participation of operating personnel in test
38 activities. TU Electric maintenance personnel will assume the responsibility for performing routine preventative and corrective maintenance activities on station components when they are released from construction to the preoperational testing group. Operating staff technicians will be assigned to assist system test engineers in performing tests. Station operators will assist system test engineers in performing tests and will take over the routine operations of systems for which prerequisite-type testing has been completed. The operating staff will direct the fuel loading and will be responsible for the operation of the plant during initial startup testing.

68 The technical operating staff is divided into four departments headed by the Operations Manager; Maintenance Manager; Manager, Startup & Test; and Manager, Technical Support; respectively. The Operations
15 and Maintenance Managers report administratively and technically to the Manager, Plant Operations. The duties and responsibilities of the operating staff during plant operation are described in Chapter 13. The duties of key operating personnel with regard to the startup program are summarized below.

60 14.2.2.3.1 Vice President, Nuclear Operations

60 The Vice President, Nuclear Operations has overall responsibility for station operations.

14.2.2.3.2 Manager, Plant Operations

The Manager, Plant Operations will provide a continuing analysis of 68
testing schedules, operator training schedules, and plant operating
staff workloads in order to minimize conflicts with the startup
program, and he also will provide coordination between the operating
staff and the Test Department. The Manager, Plant Operations will
coordinate any required changes to station operating procedures based 38
upon test results.

68

14.2.2.3.3 Operations Department

The Operations Manager is a member of the JTG. He will be 68
responsible for the proper operation of all equipment in the custody
of the operations department and for ensuring that the conduct of the
startup program does not place the plant in an unsafe condition. He
will provide personnel from the operating staff as required to support 33
the conduct of testing activities. In addition, he will direct the
development of station operating procedures.

The shift supervisors report to the Operations Manager and are 62
responsible for the safe operation of the plant during assigned
shifts. They also are responsible for the implementation of
appropriate safety and custody tagging procedures and have the responsibility to disallow or terminate testing due to conditions which could endanger personnel or equipment or violate technical specifications.

14.2.2.3.4 Maintenance Department

62 The Maintenance Manager will be responsible for performing preventive and corrective maintenance when required on components and systems that have been turned over from construction to TU Electric. He will
33 provide personnel from the maintenance department to support testing activities as required.

62 14.2.2.3.5 Technical Support Department

62 The Manager, Technical Support is a member and serves as the Vice Chairman of the JTG. He will coordinate the review of all
60 preoperational test procedures and results by the appropriate
62 operating staff personnel. The Manager, Technical Support will provide personnel from the technical support department as required to support testing activities.

68

56 14.2.2.3.6 Director, Quality Assurance

68 The Director, Quality Assurance is responsible for approving and verifying implementation of the TU Electric CPSES QA Program

14.2.2.4 <u>Major Participating Organizations</u>

60 14.2.2.4.1 Engineering and Construction

60 Engineering and Construction will be responsible for assisting to the extent required in ensuring that tests sufficiently verify the adequacy of system design and in the formulation of changes required to correct detected design deficiencies. This responsibility includes:

46 1. Review of preoperational test objectives and acceptance criteria to verify implementation of operating license commitments and consistency with the station safety analyses;

2. Review and approval of design change requests originated during the course of the startup program.

3. Management of Contractors (e.g. Gibbs & Hill, Brown & Root, Westinghouse). | 15

4. Technical review and comments of Unit 2 preoperational test procedures objectives and acceptance criteria, prior to approval. | 46

14.2.2.4.2 Brown & Root, Inc. (B&R) | 68

B&R, as the constructor for CPSES, is responsible for the construction completion, performance of associated constructions tests, and orderly release of components and systems to TU Electric consistent with the startup program schedules. This responsibility includes:

1. Completion of construction and construction testing activities;

2. Provision of craft technical manpower support as required for performance of the startup program. | 11

14.2.2.4.3 Westinghouse Electric Corporation | 68

Westinghouse, as the Nuclear Steam Supply System (NSSS) supplier, is responsible for providing technical direction to TU Electric during preoperational and initial startup testing performed on the NSSS and associated auxiliary equipment. Technical direction is defined as technical guidance, advice and counsel based on current engineering, installation, and testing practices. This responsibility includes: | 60

1. Assignment of personnel to provide advice and assistance to TU Electric for test and operation of all equipment and systems in the Westinghouse area of responsibility.

2. Assignment of an operational physicist to the site organization during fuel loading and power testing; and

3. Provision of test procedure outlines and technical assistance for tests of Westinghouse furnished components and systems.

68 | 14.2.2.4.4 Allis-Chalmers Power Systems, Inc. (ACPSI)

ACPSI, as supplier of the main turbine generator set, is responsible for providing technical direction to TU Electric during preoperational and initial startup testing performed on the turbine generator and related auxiliary equipment.

37 |

14.2.2.5 Joint Test Group (JTG)

68 | The Joint Test Group (JTG) is comprised of certain station supervisory and technical personnel as described in Section 14.2.2.5.1. The JTG functions as a subcommittee of the Station Operations Review Committee (SORC) for testing matters. The JTG is charged with reviewing testing activities described in 14.2.2.5.2 and advising the SORC on the disposition of those items reviewed.

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14.2.2.5.1 JTG Membership 68

As a minimum, the Joint Test Group (JTG) shall be composed of managers, supervisors, or engineers from the organizations listed below. The Manager, Startup and Test serves as the chairman of the JTG. 68

Test 68
Operations 68
Technical Support 68
Engineering and Construction 68
Westinghouse (for all matters concerning preoperational testing performed on the NSSS and associated auxiliary systems) 68

The members and their alternates, including the alternate for the chairman shall be designated in writing by their cognizant Vice President. 68

The primary function of the JTG is the review and approval of all preoperational and initial startup program test procedures, procedure revisions, and test results. 68

68

In addition to the above, representatives of other organizations will participate, as requested by the group chairman.

14.2.2.6 <u>Quality Assurance</u>

The TU Electric Quality Assurance Department is responsible for assuring the quality of construction, plant testing, and operations activities as described in Chapter 17. 68

14.2.2.7 Qualifications of Key Personnel

Q423.1
Q423.7
Q423.21
68 The minimum qualifications of individuals responsible for review and approval of preoperational and initial startup test procedures and results shall be; (1) Bachelor's degree in engineering or the physical sciences and four years of applicable power plant experience; or (2) high school diploma or equivalent and eight years of applicable power plant experience. Credit for up to two years for related technical
40 training may be substituted on a one-for-one time basis. A minimum of two years shall be applicable nuclear power plant experience.

68

37 The minimum qualifications of individual that direct or supervise the conduct of preoperational test at the time of performance of the duties, shall be; (1) Bachelor's degree in engineering or the physical sciences and one year of experience in power plant testing or operation. Included in the one year shall be three months of familiarization of system and component operation unique to the design of similar nuclear power plants at which the individual will be employed, or (2) a high school diploma or equivalent and four years of experience in power plant testing or operation or credit for up to two years for related technical training may be substituted on a one-for-one time basis. Included in the four years shall be three months of familiarization of system and component operation unique to the design of similar nuclear power plants at which the individual will be employed.

37 The minimum qualifications of individuals that direct or supervise the conduct of initial startup test at the time of performance of the duties, shall be: (1) Bachelor's degree in engineering or the physical sciences and two years of experience in power plant testing or operation. Included in the two years shall be a minimum of one year of nuclear power plant testing, operation or training on a nuclear facility, or (2) a high school diploma or equivalent and five years of experience in power plant testing of which two years will

be nuclear power plant experience. Credit for up to two years for 37
related technical training may be substituted on a one-for-one time basis.

All test personnel will be indoctrinated in the use of Test Department 68
and applicable station administrative control procedures, test procedures and familiarized with all applicable quality assurance requirements.

14.2.3 TEST PROCEDURES

14.2.3.1 General

All preoperational and initial startup tests will be performed in 37
accordance with written approved test procedures. The following sections describe the general methods employed to control procedure development and review, and they also describe the responsibilities of the various organizational units participating in this process.

The detailed controls and methods will be prescribed in the startup 37
administrative procedures and station administrative procedures, as applicable.

14.2.3.2 Development of Test Procedures 37

The Manager, Startup and Test is responsible for the development of 68
all preoperational and initial startup procedures and maintenance of the test procedure indexes and preparation schedules.

68

Technical information required for the preparation of the test procedures will be provided by the appropriate engineering organizations. This information will consist of system descriptions, technical specifications, design drawings and other technical documents which define the functional requirements and performance objectives for the various systems and components. Additional technical data may also be obtained from the various component vendors and other contractors as required.

The functional requirements provided by the system designers will be incorporated into the acceptance criteria for each test procedure. This information will also be used by the system test engineer in developing the detailed test methods which ensure that the capability of systems and components to function properly within design specifications is adequately demonstrated.

14.2.3.3 Review and Approval of Test Procedures

Preoperational and initial startup test procedures will be reviewed by members of the plant operating staff and appropriate design organizations.

68 Preoperational and initial startup test procedures will be forwarded to appropriate members of the Joint Test Group for review and comment.

68 The responsibility for final approval of all preoperational and initial startup test procedures rests with the Chairman of the Joint Test Group (JTG).

11 ### 14.2.3.4 Format of Test Procedures

68 All preoperational and initial startup tests will be prepared based on formats specified by Test Department administrative procedures. This standard format will help ensure that each procedure contains all information and instructions required to satisfactorily perform and document the test.

The procedures format and content will reflect the guidance provided in Regulatory Guide 1.68 as discussed in Appendix 1A(B). The standard format will include, as a minimum, the following: | 11

1. Test Objectives:
   A detailed statement of the test objectives and method of system operation to be demonstrated.

2. References:
   References to technical specifications, supporting procedures, vendor's manual, or other technical documents will be included as required.

3. Prerequisites and Initial Conditions:
   This section will specify prerequisites to ensure that all relevant initial conditions are satisfied and that the prerequisite tests and construction activities have been satisfactorily completed.

4. Precautions and Limitations:
   Precautions and limitations relating to personnel safety, equipment integrity, and overall plant safety will be specified.

5. Detailed Test Method:
The test method will contain detailed step by step instructions for operating the system in the test configuration, performing actual test manipulations, and for use of off-normal procedures such as jumper cables or mechanical bypasses and restoration of the system to normal status following test. The detailed test instructions will utilize normal and emergency station operating procedures to the extent practical.

6. Acceptance Criteria:
The performance objectives and functional requirements for system operations will be specified. The criteria used to judge the success or failure of the test may be qualitative or quantitative.

7. Data Collection:
Provisions will be made for recording all pertinent test data regarding system conditions and performance. Records will identify the observer, type of instrumentation, acceptability of results, and any deficiencies, and will become a part of the permanent station records.

14.2.3.5 Revisions to Procedures

68 | Revisions to preoperational and initial startup test procedures will be reviewed and approved by appropriate members of the JTG.

38 | Preoperational and Initial Startup Test procedure modifications required during conduct of test will be approved as follows;

38 | 1. Preoperational Test Procedures

a. Test procedure modifications that do not change the intent | 38
of the test will be approved by the system test engineer.

b. Test procedure modifications that change the intent of the | 38
test will be approved by the system test engineer and
another member of the startup organization qualified to
review preoperational test procedures.

c. All procedure modifications will be included with the | 38
completed procedure and be subject to review and approval
with the test results as described by paragraph 14.2.5.

2. Initial Startup Test Procedures | 38

a. Modifications to initial startup tests that do not change | 38
the intent of the test will have the shift supervisor's
concurrence and approval of the person in charge of the
test.

b. Modifications to initial startup tests that change the | 68
intent of the test will be reviewed and approved by the
JTG.

14.2.4 CONDUCT OF TEST PROGRAM

14.2.4.1 Administrative Procedures

The conduct of the startup program will be controlled by the CPSES | 68
Test Department Administrative Procedures Manual. These procedures
will prescribe controls for startup activities such as the following: | 37

1. Preparation, review and approval, of test procedures

37 2. Turnover of systems

3. Format and content of test procedures

11 4. Safety and custody tagging procedures

11 5. Temporary system modifications

11 6. Design change processing

11 7. Test deficiency processing

11 8. Review of Reactor Operating/Startup experiences.

40 14.2.4.2 Prerequisite Testing

40 Startup Administrative procedures will be established to ensure that applicable prerequisites are met before testing is initiated. Upon completion of construction phase activities, custody of the component/system will be transferred to TU Electric for conduct of prerequisite testing and preoperational testing. During the turnover process, systems and components will be reviewed for completeness, installation damage and conformance with appropriate installation and/or design documents. Outstanding construction, document and test deficiencies will be identified and controlled prior to fuel load.

14.2.4.3 Preoperational and Initial Startup Testing

68 Technical direction and administration, including test procedure preparation, test execution, and data recording, of the preoperational and initial startup testing is the responsibility of the

Test Department, with the operating staff retaining responsibility for 68
performing actual equipment operations and maintenance.

The Manager, Startup and Test is responsible for the administration 68
and implementation of all preoperational and initial startup testing activities during the startup program.

68

The system test engineers will direct support personnel in the performance of tests and will provide appropriate interface with station operators. The shift supervisors will be responsible for insuring that the conduct of testing does not place the plant in an unsafe condition at any time. Additionally, the shift supervisors and system test engineers have the authority to terminate or disallow testing at any time.

14.2.4.4 Test Prerequisites

Each test procedure will contain a set of prerequisites and initial 68
conditions as prescribed by the Test Department Administrative procedures. The system test engineer will ensure that all specified prerequisites are met prior to performing the test. The format for test procedures is described in Section 14.2.3.4.

14.2.4.5 Phase Evaluation

Between each major phase of the test program the test results for all 68
tests that have been performed will be reviewed by the JTG.

11 This review ensures that all required systems have been tested satisfactorily and that test results are approved before proceeding to the next stage of testing. This review is described in Section 14.2.5.

14.2.4.6 Design Modifications

Modifications to the design of safety-related equipment during the test program may be initiated in order to correct deficiencies discovered as a result of testing. Any such modification will be referred to the appropriate engineering organization for approval. Modifications made to safety-related components or systems after completion of preoperational or initial startup testing will be reviewed for retesting requirements on affected portions of the system.

11 14.2.5 REVIEW, EVALUATION, AND APPROVAL OF TEST RESULTS

11 Following completion of a particular test, the responsible system test
68 engineer will assemble the test data package for evaluation. Test results will be reviewed by appropriate members of the JTG.

11 Each test data package will be reviewed to ensure that the test has been performed in accordance with the written approved procedure and that all required data, checks, and signatures have been properly recorded and that system performance meets the approved acceptance criteria.

Deficiencies identified in the review process will be resolved to the 68
satisfaction of the JTG. If the evaluation indicates that
deficiencies in the test method are responsible for unsatisfactory 11
test results, the test procedure will be revised accordingly before
retesting is initiated. Whenever an evaluation of test results
indicates deficiencies in system performance, the problem will be
referred to the appropriate engineering organization for evaluation.

The responsibility for final approval of test results rests with the 68
Chairman of the JTG.

Q423.3

Following each major phase of the test program test results and/or 11
test status will be reviewed to ensure that all required tests have
been performed and that the test results have been approved. This
review will ensure that all required systems are operating properly
and that testing for the next major phase will be conducted in a safe
and efficient manner. This type of review will be performed to the 40
extent required before major test phases such as fuel load, initial
criticality, and power escalation. During the power escalation
phase, review and approval of initial startup test procedure results 2
will be completed for each of these plateaus (30 percent, 50 percent,
and 75 percent) prior to proceeding with power ascension testing to
the next plateau.

14.2.6 TEST RECORDS

Test procedures and test data relating to preoperational and initial startup testing will be retained in accordance with the measure described in Section 17.2.17, "Quality Assurance Records."

The following conditions will be met prior to core loading:

1. The reactor Containment structure will be complete and Containment integrity established and maintained during fuel loading.

2. Fuel handling tools and equipment will have been checked out and operators familiarized in the use and operation of equipment. Inspections of fuel assemblies, rod cluster control assemblies, and reactor vessel will be satisfactorily completed.

3. The reactor vessel and associated components will be in a state of readiness to receive fuel. Water level will be maintained above the bottom of the nozzles and recirculation maintained to assure the required boron concentration. Boron concentration can be increased via the recirculation path or directly to the open vessel.

Criteria for safe loading require that loading operations stop immediately if any of the following conditions occur.

1. An unanticipated increase in the neutron count rates by a factor of two occurs on all responding nuclear channels during any single loading step after the initial nucleus of eight fuel assemblies is loaded.

2. An unanticipated increase in the count rate by a factor of five on any individual responding nuclear channel during any single loading step after the initial nucleus of eight fuel assemblies is loaded.

3. An anticipated decrease in boron concentration greater than 20 ppm is determined from two successive samples of the reactor coolant.

| Q423.12

on-site and from alternate off-site to on-site sources | 6

| Q423.12

2. Manual transfer, live, from any one of the three sources - preferred off-site, alternate off-site and on-site - to the other (six combinations). | 6

| Q423.12

B. For Non-Class 1E 6.9kV Buses - | 6

1. Automatic transfer fast and slow from unit auxiliary transformer to off-site source. | 6

2. Manual transfer from off-site source to unit auxiliary transformer and vice versa. | 6

| Q423.12

5. Verify proper load shedding and/or undervoltage tripping of all 6.9kV and 480V switchgear. | 6

| Q423.34
| 68

| Q423.12

6. Verify load-carrying capability of Class 1E and non-Class 1E transformers, cables, switchgear and feeder breakers. | 68

ACCEPTANCE CRITERIA

| Q423.11
| Q423.34

The AC Power Distribution System provides power to safety-related loads when supplied from normal and emergency power sources. No interaction occurs between redundant trains. Automatic transfer of power supplies to vital buses occurs properly and meets design requirements. Interlocks, alarms, controls and tripping devices function properly in response to normal or simulated input signals. Emergency bus voltages are maintained within design limits under load. | 13 | 68

OPERATIONAL VIBRATION TESTING
TEST SUMMARY

OBJECTIVE

To verify that the vibration level of selected (1) Class 1, 2, and 3 piping (2) other high-energy piping systems inside Seismic Category I Structures, (3) high-energy portions of systems whose failure could reduce the functioning of any Seismic Category I Plant Feature to an unacceptable level is within acceptable levels. | 66

PREREQUISITES

1. Systems are operational as required.
2. Instrumentation is in place for testing as required. | 66

TEST METHOD

1. Subject the specified piping systems to various flow modes and transients such as pump trips and valve closures as required.

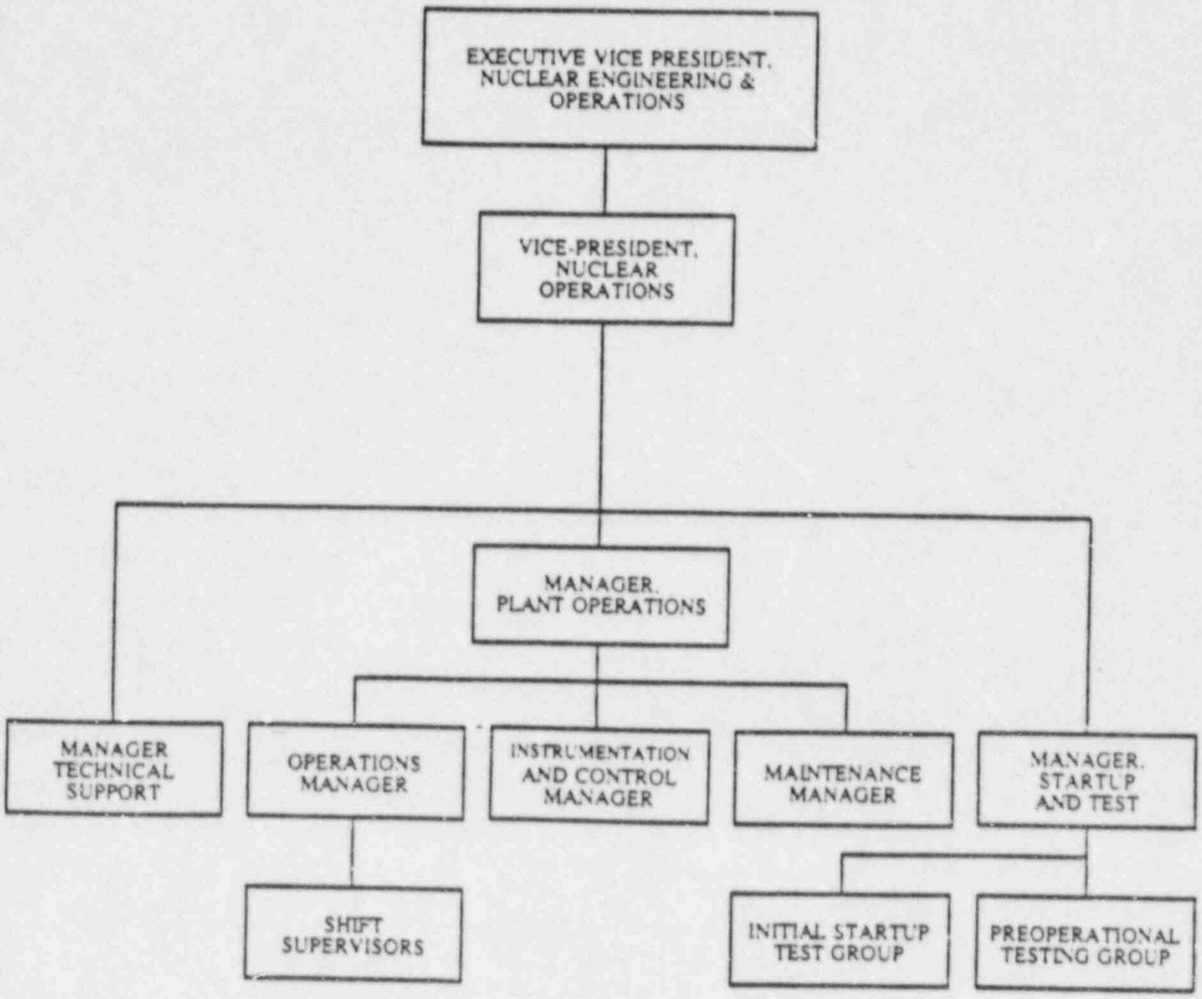
2. Visually inspect and/or measure the vibration level of the piping and components at the specified locations. | 66

3. Following completion of the system test, visually inspect the piping and supports including snubbers for damage, looseness of parts etc. | 66

ACCEPTANCE CRITERIA

| Q423.11

The vibration level for piping and components are within acceptable limits. For acceptance criteria basis see FSAR Sections 3.9B.2.1.2 and 3.9B.2.1.3. | 68

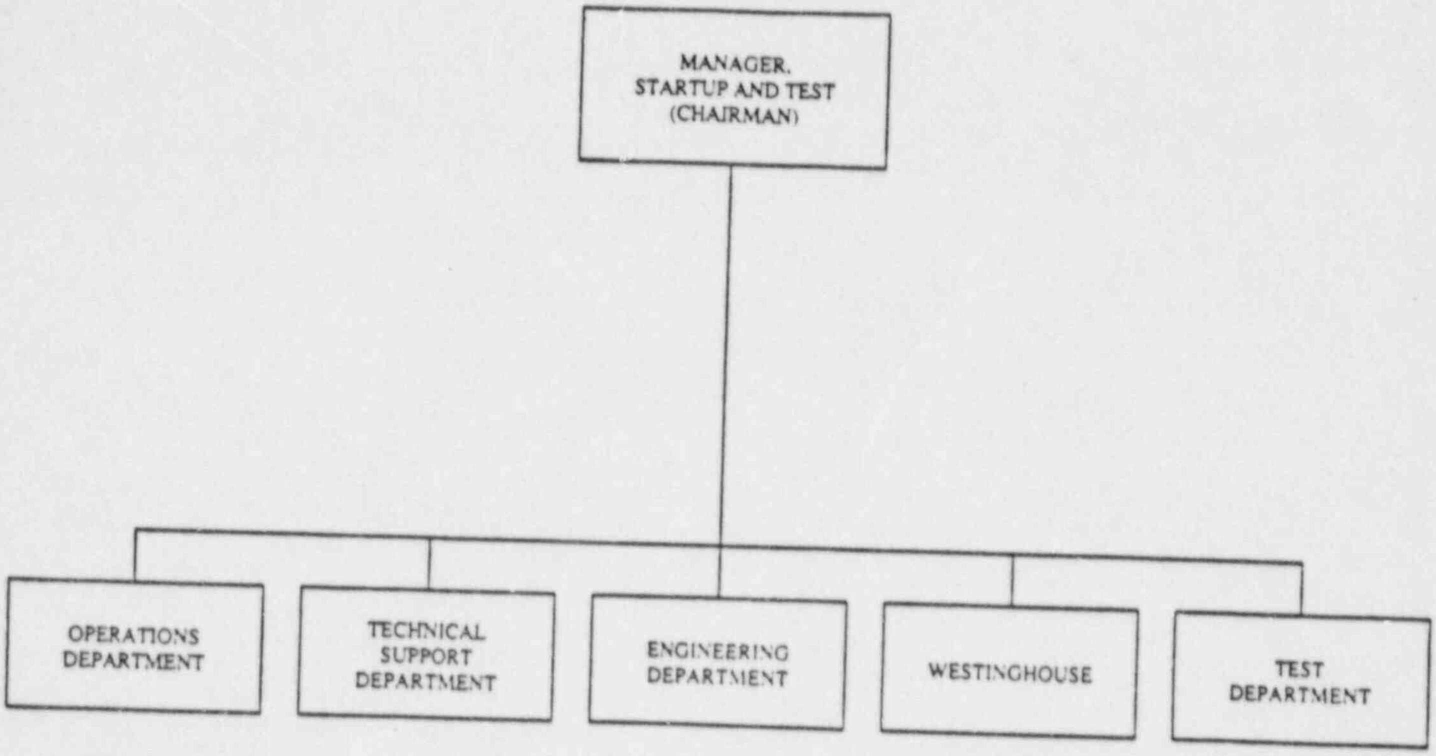


ADVANCE COPY

COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

STARTUP ORGANIZATION
AND OPERATING STAFF

FIGURE 14.2-1



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

JOINT TEST GROUP

FIGURE 14 2-2

15.4.8.3 Environmental Consequences

66

The conservative analysis doses resulting from a rod ejection accident are based on a conservative fission product release to the reactor coolant of the gap activity from 10 percent of the fuel rods in the core plus the core activity from the assumed 0.25 percent core melt.
66 The method of analysis complies with the requirments of Appendix B of Regulatory Guide 1.77 except as noted in Appendix 1AB.

66 Following a postulated rod ejection accident, two activity release paths contribute to the total radiological consequences of the accident. The first release path is via Containment leakage resulting from release of activity from the primary coolant to the Containment. The second path is the contribution of contaminated steam in the secondary system dumped through the relief valves, since offsite power is assumed to be lost.

68 Prior to the accident, it is assumed that the plant has been operating with simultaneous fuel defects and steam generator tube leakage for a sufficient period of time to establish equilibrium levels of activity in the primary and secondary coolants equal to the Table 15.1-4 limits.

66 Following a postulated rod ejection accident, the activity released from the pellet-clad gap due to failure of a portion of the fuel rods and the melted fuel is assumed to be instantaneously released to the primary coolant. The activity released to the primary coolant is assumed to be uniformly mixed throughout the coolant instantaneously. Thus, the total activity of the primary coolant is assumed to be immediately available for release from the RCS. Of the activity released to the Containment

TABLE 17A-1

(Sheet 1 of 48)

LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
1. Reactor Coolant System (RCS)								
Reactor vessel	1	ASME III	1	I	Note 3,A	5.3		; 68
CRDM housings	1	ASME III	1	I	Note 6,A			; 68
CRDM head adapter plugs	1	ASME III	1	I	Note 4,A			; 68
Reactor vessel supports	1	ASME III	1	I	Note 3,A	5.4.14		; 68
Steam generator								
Tube side	1	ASME III	1	I	Note 6,A	5.4.2		; 68
Shell side	2	ASME III	1	I	Note 6,A		Note 8	; 68
Steam generators supports	1	ASME III	1	I	Note 6,A	5.4.14		; 68
Steam generator restraints	1	ASME III	1	I	Note 3,A			; 68
Pressurizer	1	ASME III	1	I	Note 6,A	5.4.10	Note 1b	; 68
Pressurizer support skirt	1	ASME III	1	I	Note 3,A	5.4.14		; 68
Reactor coolant hot- and cold-leg piping and fittings, and fabrication	1	ASME III	1	I	Note 3,A	5.4.3		; 68 ; 68
Surge pipe & fittings and fabrication	1	ASME III	1	I	Note 3,A	5.4.10		; 68
Piping and Valves	2	ASME III	2	I	Note 26,A			; 68
Piping and valves	1	ASME III	1	I	Note 3,A	3.9N		; 68
Crossover leg piping & fittings and fabrication	1	ASME III	1	I	Note 3,A	5.4.3		; 68
Pressurizer safety valves	1	ASME III	1	I	Note 3,A	5.4.13		; 68
Power-operated relief valves	1	ASME III	1	I	Note 3,A	5.4.13		; 68
PORV accumulators	3	ASME III	3	I	Note 26,A			; 68
Check valves for PORV Accumulators	3	ASME III	3	I	Note 32,A	R312.32		; 68
Tubing and Supports (between check valves upstream of air accumulator and AOV)	3	ASME III	3	I	Note 32,A	3.9B	Notes 41	; 68 ; 68 ; 68
Pressurizer PORV Block Valves	1	ASME III	1	I	Note 3,A	5.4.12		; 68
PORV and Safety Valves Limit Switches	1E	IEEE-323	-	I	Note 3,A	II.D.3		; 68 ; 68

TABLE 17A-1

(Sheet 2)

LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Valves of Safety Class 1 to Safety	1	ASME III	1	I	Note 3,A	5.4.12		68
Class 2 interface								68
High Point vent valves	2	ASME III	2	I	Note 3,A	5.1		68
Pressurizer relief tank	NNS	ASME VIII	-	NONE	Note 4,28	5.4.11		68
Reactor Coolant Pump								
Casing	1	ASME III	1	I	Note 6,A	5.4.1		68
Main flange	1	ASME III	1	I	Note 6,A			68
Thermal barrier	1	ASME III	1	I	Note 6,A			68
Thermal barrier heat exchanger	1	ASME III	1	I	Note 6,A			68
No. 1 seal housing	1	ASME III	1	I	Note 6,A			68
No. 2 seal housing	2	ASME III	1	I	Note 6,A		Note 8	68
Pump Shaft	2	-	-	I	Note 6,A	5.4.1		68
Pump Impeller	2	-	-	I	Note 6,A	5.4.1		68
Pressure-retaining bolting	1	ASME III	1	I	Note 6,A			68
Vertical and lateral supports	1	ASME III	1	I	Note 3,A	5.4.14		68
Reactor Coolant Pump Motor								
Motor rotor	2	-	-	I	Note 6,9,A			68
Motor shaft	2	-	-	I	Note 6,9,A			68
Shaft coupling	2	-	-	I	Note 6,9,A			68
Spool piece	2	-	-	I	Note 6,9,A			68
Flywheel	2	-	-	I	Note 6,9,A			68
Bearing (Motor upper thrust)	2	-	-	I	Note 6,9,A			68
Motor bolting	2	-	-	I	Note 6,9,A		Note 30	68
Motor stand	2	-	-	I	Note 6,9,A			68
Motor frame	2	-	-	I	Note 6,9,A			68
Upper oil reservoir	3	Mfrs Stds	-	I	Note 6,A			68
Cooling coil	3	ASME III	3	I	Note 6,A			68

TABLE 17A-1

(Sheet 3)

LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Lower oil reservoir	3	Mfrs Stds	-	I	Note 6,A			68
Cooling coil	3	ASME III	3	I	Note 6,A			68
Lube-oil piping	3	Mfrs Stds	-	I	Note 6,A			68
Motor-air coolers	3	ASME III	3	I	Note 6,A		Note 2	68
Reflective insulation assemblies (except for the portions installed on the RCS cold leg and hot leg pipes which are located inside the biological shield tunnels)	NNS	Mfrs Stds	-	II	Note 27,B	6.1B, 6.2.2		68
Reactor vessel nozzles non-crushable insulation	1	Mfrs Stds	-	I	Note [illegible],A	3.9N.1.4.6	Note 64	68
Reflective insulation assemblies on RCS cold leg and hot leg pipes and located in biological shield tunnels	NNS	Mfrs Stds	-	NONE	NONE	6.1b		68
Supports for Class 1 piping	1	ASME III	1	I	Note 27,A	3.9N		68
Supports for Class 2 piping	2	ASME III	2	I	Note 27,A	3.9B		68
Class 5 piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 41,B	3.7B		68
2. Chemical and Volume Control System (CVCS)								
Regenerative heat exchanger	2	ASME III	2	I	Note 3,A	9.3.4		68
Letdown Heat Exchanger								
Tube side	2	ASME III	2	I	Note 3,A	9.3.4		68
Shell side	3	ASME III	3	I	Note 3,A	9.3.4	Note 1c, 2	68
Mixed-bed demineralizer	3	ASME III	3	NONE	Note 4,A	9.3.4	Note 11	68
Cation-bed demineralizer	3	ASME III	3	NONE	Note 4,A	9.3.4	Note 11	68
Reactor coolant filter	2	ASME III	2	I	Note 3,A	9.3.4		68

TABLE 17A-1

(Sheet 4)

LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Volume control tank	2	ASME III	2	I	Note 3,A	9.3.4		| 68
Centrifugal charging pump	2	ASME III	2	I	Note 3,A	9.3.4	Note 1a and 1d, 2	| 68
Positive displacement pump	2	ASME III	2	I	Note 3,A	9.3.4	Note 1a and 1c, 2	| 68
Positive displacement pump discharge dampener	2	ASME III	2	I	Note 3,A	9.3.4		| 68
Seal water injection filter	2	ASME III	2	I	Note 3,A	9.3.4		| 68
Letdown orifices	2	ASME III	2	I	Note 4,A	9.3.4		| 68
Excess letdown heat exchanger								
Tube side	2	ASME III	2	I	Note 3,A	9.3.4		| 68
Shell side	2	ASME III	2	I	Note 3,A	9.3.4	Note 1c	| 68
Seal water return filter	2	ASME III	2	I	Note 3,A	9.3.4		| 68
Seal Water Heat Exchanger								
Tube side	2	ASME III	2	I	Note 3,A	9.3.4		| 68
Shell side	3	ASME III	3	I	Note 3,A	9.3.4	Note 1c, 2	| 68
Boric acid transfer pump	3	ASME III	3	I	Note 3,A	9.3.4	Note 1b	| 68
Boric acid filter	3	ASME III	3	I	Note 4,A	9.3.4		| 68
Boric acid batching tank	NNS	ASME VIII	-	NONE	Note 4, 26	9.3.4		| 68
Boron Concentration Measurement System	NNS	ANSI B31.1	-	NONE	Note 5	9.3.4	Note 29	| 68
Chemical mixing tank	NNS	ASME VIII	-	NONE	Note 5	9.3.4		| 68
Reactor coolant pump seal bypass orifice	1	ASME III	1	I	Note 4,A	9.3.4		| 68
R.C. Pump standpipe	NNS	ASME VIII	-	NONE	Note 5			| 68
Boric acid tanks	3	ASME III	3	I	Note 26,A	9.3.4		| 68
Boric acid blender	3	ASME III	3	I	NOTE 4,A	9.3.4		| 68
Piping and valves	2	ASME III	2	I	Note 26,34N,A	9.3.4		| 68
Piping and valves	3	ASME III	3	I	Note 26,34N, 34B,A	9.3.4		| 68

TABLE 17A-1

(Sheet 5)

LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Supports for Class 2 piping	2	ASME III	2	I	Note 27,A	3.9B		68
Supports for Class 3 piping	3	ASME III	3	I	Note 27,A	3.9B		68
Class 5 piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	3.7B		68
Reflective insulation assemblies	NNS	Mfrs Stds	-	II	Note 27,B	6.1, 6.2.2		68
3. Boron Thermal Regeneration Subsystem (BTRS)								
Moderating Heat Exchanger								
Tube side	3	ASME III	3	NONE	Note 4,A	9.3.4	Note 11	68
Shell side	3	ASME III	3	NONE	Note 4,A	9.3.4	Note 11	68
Letdown Chiller Heat Exchanger								
Tube side	3	ASME III	3	NONE	Note 4,A	9.3.4	Note 11	68
Shell side	NNS	ASME VIII	-	NONE	Note 4, 28	9.3.4		68
Letdown Reheat Heat Exchanger								
Tube side	2	ASME III	2	I	Note 3,A	9.3.4		68
Shell side	3	ASME III	3	NONE	Note 3,A	9.3.4	Note 11	68
Thermal regeneration demineralizer	3	ASME III	3	NONE	Note 4,A	9.3.4	Note 11	68
Chiller pump	NNS	Mfrs Stds	-	NONE	Note 4, 28	9.3.4		68
Chiller surge tank	NNS	ASME VIII	-	NONE	Note 5	9.3.4		68
Chiller Unit								
Evaporator	NNS	ASME VIII	-	NONE	Note 4, 28	9.3.4		68
Condenser	NNS	ASME VIII	-	NONE	Note 4, 28	9.3.4		68
Compressor	NNS	Mfrs Stds	-	NONE	Note 4, 28	9.3.4		68
Piping and valves	2	ASME III	2	I	Note 26,A	9.3.4		68
Piping and valves	3	ASME III	3	I	Note 26,A	9.3.4		68
Supports for Class 2 piping	2	ASME III	2	I	Note 27,A	3.9B		68
Supports for Class 3 piping	3	ASME III	3	I	Note 27,A	3.9B		68

LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Class 5 piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	3.7B		| 68
4. Safety Injection System (SIS)								
Accumulators	2	ASME III	2	I	Note 3,A	6.3		| 68
High-head SIS pumps	2	ASME III	2	I	Note 3,A	6.3	Note 1a, 1d, 2	| 68
Valve isolation tanks	N/A	ASME III	2	I	Note 26,A	6.2, 6.3		| 68
Valve isolation tank expansion joints	N/A	ASME III	MC	I	Note 26,A	6.2, 6.3		| 68
Valve isolation tank piping and valve	3	ASME III	2	I	Note 26,A	6.3		| 68
Piping and valves	2	ASME III	2	I	Note 26,A	6.3		| 68
Piping and valves	1	ASME III	1	I	Note 3,A	6.3		| 68
Supports for Class 2 piping	2	ASME III	2	I	Note 27,A	3.9B		| 68
Supports for Class 1 piping	1	ASME III	1	I	Note 27,A	3.9B		| 68
Class 5 piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	3.7B		| 68
Reflective insulation assemblies	NNS	Mfrs Stds	-	II	Note 27,B	6.1, 6.2.2		| 68
5. Residual Heat Removal (RHR) System								
RHR pump	2	ASME III	2	I	Note 3,A	5.4.7	Note 1a, 1c, 2	| 68
Residual Heat Exchanger								
Tube side	2	ASME III	2	I	Note 3,A	5.4.7		| 68
Shell side	3	ASME III	3	I	Note 3,A	5.4.7	Note 1c	| 68
Piping and valves	1	ASME III	1	I	Note 26,A	5.4.7		| 68
Piping and valves	2	ASME III	2	I	Note 26,A	5.4.7		| 68
Supports for Class 1 piping	1	ASME III	2	I	Note 27,A	3.9B		| 68
Supports for Class 2 piping	2	ASME III	2	I	Note 27,A	3.9B		| 68
Class 5 piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	3.7B		| 68
Reflective insulation assemblies	NNS	Mfrs Stds	-	II	Note 27,B	6.1, 6.2.2		| 68
Relief Valves	2	ASME III	2	I	Note 3,A	5.4.7		| 68

TABLE 17A-1

(Sheet 7)

LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
6. Boron Recycle System (BRS)								
Recycle evaporator feed pump	3	ASME III	3	NONE	Note 3,A	9.3.4	Note 11	; 68
Recycle evaporator feed demineralizer	3	ASME III	3	NONE	Note 4,A	9.3.4	Note 11	; 68
Recycle evaporator feed filter	3	ASME III	3	NONE	Note 4,A	9.3.4	Note 11	; 68
Recycle evaporator condensate	NNS	ASME VIII	-	NONE	Note 4, 28	9.3.4		; 68
demineralizer								; 68
Recycle evaporator condensate filter	NNS	ASME VIII	-	NONE	Note 4, 28	9.3.4		; 68
Recycle holdup tank	3	ASME III	3	I	Note 4,A	9.3.4		
Recycle evaporator concentrate filter	NNS	ASME VIII	-	NONE	Note 4, 28	9.3.4		
Recycle evaporator reagent tank	NNS	ASME VIII	-	NONE	Note 5	9.3.4		
Recycle Evaporator Package								
Feed preheater								
a. feed side	3	ASME III	3	NONE	Note 4, 6,A	9.3.4	Note 11	; 68
b. steam side	NNS	ASME VIII	-	NONE	None	9.3.4		
Gas stripper	3	ASME III	3	NONE	Note 4, 6,A	9.3.4	Note 11	
Submerged tube Evaporator								
a. feed side	3	ASME III	3	NONE	Note 4, 6,A	9.3.4	Note 11	; 68
b. steam side	NNS	ASME VIII	-	NONE	None	9.3.4		
Evaporator condenser								
a. Distillate water side	3	ASME III	3	NONE	Note 4,A	9.3.4	Note 11	; 68
b. Cooling water side	3	ASME III	3	I	Note 4,A	9.3.4		
Distillate Cooler								
a. Distillate side		ASME III	3	NONE	Note 4,A	9.3.4	Note 11	; 68
b. Cooling water side	3	ASME III	3	I	Note 4,A	9.3.4		; 68
Absorption tower	3	ASME III	3	NONE	Note 4,A	9.3.4	Note 11	; 68

TABLE 17A-1

(Sheet 8)

LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Recycle Holdup tank vent ejector	3	ASME III	3	I	Note 34N	9.3.4		; 68
Vent Condenser								
a. Gas side	3	ASME III	3	NONE	Note 4,A	9.3.4	Note 11	; 66
b. Cooling water side	3	ASME III	3	I	Note 4,A	9.3.4		; 68
Distillate pump	3	ASME III	3	NONE	Note 4,A	9.3.4	Note 11	; 68
Concentrate pump	3	ASME III	3	NONE	Note 4,A	9.3.4	Note 11	; 68
Piping and valves								
a. Feed	3	ASME III	3	NONE	Note 4,A	9.3.4	Note 11	; 68
b. Distillate	NNS	ASME III	3	NONE	Note 4,A	9.3.4	Note 11	; 68
c. Concentrate	NNS	ASME III	3	NONE	Note 4,A	9.3.4	Note 11	; 66
d. Cooling	3	ASME III	3	I	Note 4,A	9.3.4		; 66
Piping and valves	3	ASME III	3	I	Note 26, 34N	9.3.4		; 68
					34B, A			; 42
Valves	NNS	ANSI B31.1	-	NONE	Note 34B, 34N, C			; 68
Supports for Class 3 piping	3	ASME III	3	I	Note 27,A	9.3.4		; 66
Class 5 piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	3.7B		; 68
7. Containment Spray System (CSS)								
Chemical additive tank	3	ASME III	3	I	Note 26,A	6.2.2 and 6.5		; 66
Containment spray pumps	2	ASME III	2	I	Note 26,A	6.2.2	Note 1a, 1c, 2	; 68
Chemical Eductor	2	ASME III	2	I	note 26,A	6.5		; 68
Containment Spray Heat Exchanger								
Tube side	2	ASME III	2	I	Note 26,A	6.2.2		; 68
Shell side	3	ASME III	3	I	Note 26,A	6.2.2	Note 1c	; 68
Spray nozzles	2	Mfrs Stds	-	I	Note 26,A	6.2.2		; 68
Ring header line orifices	2	ASME III	2	I	Note 26,A	6.2.2		; 66

TABLE 17A-1

(Sheet 9)

LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Piping and Valves	2	ASME III	2	I	Note 26,A	6.2.2		; 68
Piping and Valves	3	ASME III	3	I	Note 26,A	6.2.5, 6.5		; 68
Valve isolation tanks	N/A	ASME III	2	I	Note 26,A	6.2		; 68
Valve isolation tank expansion joints	N/A	ASME III	MC	I	Note 26,A	6.2		; 68
Valve isolation tank piping and valves	3	ASME III	2	I	Note 26,A	6.2		; 68
Supports for Class 2 piping	2	ASME III	2	I	Note 27,A	3.9B		; 68
Supports for Class 3 piping	3	ASME III	3	I	Note 27,A	3.9B		; 68
Class 5 piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	3.7B		; 68
Refueling water storage tank	2	ACI 318-71	-	I	Note 32,A	3.8.4, 6.2.2	Note 55	; 68
8. Containment Isolation System								
Piping and valves of all systems	2	ASME III	2	I	Note 26,A	6.2.4		; 68
penetrating Containment;								; 68
from isolation inside Containment								; 68
to isolation outside Containment								; 68
Penetration assemblies (Mechanical and	2	ASME III	2 & MC	I	Note 26,A	6.2.4,		; 68
electrical)						8.3.1		; 68
9. Combustible Gas Control System								
a. Hydrogen Recombiner System								
Electric hydrogen	1E	IEEE 323	-	I	Note 3,A	6.2.5		; 68
recombiner								; 68
b. Hydrogen Purge System								; 41
Exhaust filter units	3	Mfrs Stds	-	I	Note 26,A			; 68
Exhaust ductwork, supports &	3	Mfrs Stds	-	I	Note 32,A			; 68
dampers (outside Containment)								; 68
Exhaust Fans	3	Mfrs Stds	-	I	Note 26,A			; 68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Supply ductwork, supports & dampers (outside Containment)	NNS	Mfrs Stds	-	II	Note 32,B			68
Supply blowers	NNS	Mfrs Stds	-	II	Note 26,B			68
Piping & valves	2	ASME III	2	I	Note 26,A			68
Piping and valves	3	ASME III	3	I	Note 26,A			68
Supports for Class 2 piping	2	ASME III	2	I	Note 27,A			68
Supports for Class 3 piping	3	ASME III	3	I	Note 27,A			68
10. Component Cooling Water System (CCWS)								
Heat exchangers	3	ASME III	3	I	Note 26,A	9.2.2	Note 1d	68
Pumps	3	ASME III	3	I	Note 26,A	9.2.2	Note 1a	68
Surge tank	3	ASME III	3	I	Note 26,A	9.2.2		68
Piping and valves	2	ASME III	2	I	Note 26,A	9.2.2		68
Piping and valves	3	ASME III	3	I	Note 26,A	9.2.2		68
Piping and valves (Rad. Monitor Sample lines, safeguards loops)	NNS	ANSI B31.1	-	II	Note 44,B	9.2.2	Note 57	68
Recirculation loop orifice	3	ASME III	3	I	Note 26,A	9.2.2		68
Supports for Class 2 piping	2	ASME III	2	I	Note 27,A	3.9B		68
Supports for Class 3 piping	3	ASME III	3	I	Note 27,A	3.9B		68
Class 5 piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	3.7B		68
Air Accumulator Tanks	3	ASME III	3	I	Note 27,A	9.2.2	Note 1c	66
11. Station Service Water System (SSWS)								
Service Water Pumps	3	ASME III	3	I	Note 26,A	9.2.1	Note 1a	68
Piping and valves	3	ASME III	3	I	Note 26,A	9.2.1		68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Screen wash booster pumps	NNS	ASME III	3	II	Note 26,B	9.2.1		; 68
Traveling screens	N/A	Mfrs Stds	–	I	NOTE 26, 23A	9.2.1	Note 46	; 68
								; 66
Recirculation loop orifice	3	ASME III	3	I	Note 26,A	9.2.1		; 68
Supports for Class 3 piping	3	ASME III	3	I	Note 27,A	3.9B		
Class 5 piping and supports	NNS	ANSI B31.1	–	II/NONE	Note 44,B	3.7b		; 68
12. Main Steam, Reheat and Steam Dump System								
Main steam piping	2	ASME III	2	I	Note 26,A	10.3		; 68
Piping, valves, and drain pots	2	ASME III	2	I	Note 26,A	10.3		
Piping, valves, and drain pots	3	ASME III	3	I	Note 26,A	10.3		
Main steam safety valves	2	ASME III	2	I	Note 26,A	10.3		
Main steam relief valves	2	ASME III	2	I	Note 26,A	10.3		
Steam generator PORV air accumulator	3	ASME III	3	I	Note 26	R212.32		
Turbine driven auxiliary feedwater pump steam supply isolation valve accumulator tanks	3	ASME III	3	I	Note 26,A	10.3		; 68
Check valves for accumulator tanks	3	ASME III	3	I	Note 32,A	10.3		; 68
Tubing and supports (between check valves upstream of air accumulator and AOV)	3	ASME III	3	I	Note 32,A	3.9B	Note 41	; 68
Steam generator blowdown system piping	2	ASME III	2	I	Note 26,A	10.3		; 68
Steam flow restrictor (integral to steam generator)	2	ASME III	2	I	Note 4,A	10.3, 5.4.4		; 68
Main steam isolation valves	2	ASME III	1	I	Note 26,A	10.3	Note 8	; 68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Main steam isolation bypass valves	2	ASME III	1	I	Note 26,A	10.3	Note 8	: 68
and bypass valve piping								: 68
Reflective insulation assemblies	NNS	Mfrs Stds	-	II	Note 27,B	6.2.2		: 68
Piping and valves	2	ASME III	2	I	Note 26,A	10.3		: 68
Piping and valves	3	ASME III	3	I	Note 26,A	10.3		: 68
								: 68
Orifices	3	ASME III	3	I	Note 26,A	10.3		: 68
Supports for Class 2 Piping	2	ASME III	2	I	Note 27,A	3.9B		: 68
Supports for Class 3 Piping	3	ASME III	3	I	Note 27,A	3.9B		: 68
Class 5 Piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	3.7B		: 68
13. Auxiliary Feedwater System								
Motor-driven auxiliary feedwater pumps	3	ASME III	3	I	Note 26,A	10.4.9	Note 1a	: 68
Turbine-driven auxiliary feedwater pump	3	ASME III	3	I	Note 26,A	10.4.9	Note 1e	: 68
Auxiliary feedwater pump turbine driver	3	Mfrs Stds	-	I	Note 26,A	10.4.9, 10.3		: 68
and associated equipment								: 68
Turbine Driven Pump Control Panel	N/A	IEEE-344	-	I	Note 26,A	10.4.9	Associated Class 1E	: 51
Piping and valves	2	ASME III	2	I	Note 26,A	10.4.9		: 68
Piping and valves	3	ASME III	3	I	Note 26,A	10.4.9		: 68
Air accumulators (AFW Flow and	3	ASME III	3	I	Note 26,A	9.3.1		: 68
Miniflow Control Valves)								: 68
Check valves for accumulators	3	ASME III	3	I	Note 32,A	9.3.1		: 68
Tubing and supports (between check	3	ASME III	3	I	Note 32,A	9.3.1	Note 41	: 68
valves upstream of accumulator and								: 68
AOV)								: 68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Supports for Class 2 piping	2	ASME III	2	I	Note 27,A	3.9B		; 68
Supports for Class 3 piping	3	ASME III	3	I	Note 27,A	3.9B		; 68
Class 5 piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	3.7B		; 68
Condensate storage tank	3	ACI 318-71	-	I	Note 32,A	3.8.4, 10.4.9	Note 55	; 68
14. Steam Generator Feedwater System								
Piping and valves	2	ASME III	2	I	Note 26,A	10.4.7		; 68
Feedwater isolation valves	2	ASME III	2	I	Note 26,A	10.4.7		; 68
Orifices	2	ASME III	2	I	Note 26,A	10.4.7		; 68
Reflective insulation assemblies	NNS	Mfrs Stds	-	II	Note 27,B	6.2.2		; 68
Supports for Class 2 Piping	2	ASME III	2	I	Note 26,A	3.9B		; 68
Class 5 Piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	3.7B		; 68
15. Diesel Generator, Fuel Oil, and Auxiliary Systems						8.3.1 and 9.5.4 thru 9.5.8		
Diesel Generator Set (including voltage regulator and excitation system)	1E	Mfrs Stds	-	I	Note 26,A		Note 22	; 68
Diesel generator Fuel Oil Storage & Transfer System						Fig. 9.5.52		
Fuel Oil Storage Tank	3	ASME III	3	I	Note 26,A			; 68
Fuel Oil Transfer Pumps	3	Mfrs Stds	-	I	Note 26,A		Note 21	; 68
Strainers	3	ASME III	3	I	Note 26,A			; 68
Fuel Oil Day Tank	3	ASME III	3	I	Note 26,A			; 68
Duplex Strainers w/change valves	N/A	Mfrs Stds	-	I	Note 26,A		Note 22	; 68
Fuel Oil Booster (Feed) Pump	N/A	Mfrs Stds	-	I	Note 26,A		Note 22	; 68
Duplex Filter w/change valves	N/A	Mfrs Stds	-	I	Note 26,A		Note 22	; 68
Injection Pump	N/A	Mfrs Stds	-	I	Note 26,A		Note 22	; 68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Diesel Generator Control Panels (local)	1E	IEEE 323	-	I	Note 26,A	8.3.1		; 68
Diesel Generator Starting System						Fig. 9.5-55		
Compressors	N/A	Mfrs Stds	-	I	Note 26		Note 22	; 68
After Coolers	N/A	Mfrs Stds	-	I	Note 26		Note 22	; 68
Air Dryers	N/A	Mfrs Stds	-	I	Note 26		Note 22	; 68
Starting Air Receivers	3	ASME III	3	I	Note 26,A			; 68
Diesel Generator Jacket Water System						Fig. 9.5-54		
Jacket Water Cooler	3	ASME III	3	I	Note 26,A			; 68
Standpipe	3	ASME III	3	I	Note 26,A			; 68
Engine Jacket Water Pump	N/A	Mfrs Stds	-	I	Note 26,A		Note 22	; 68
Auxiliary Jacket Water Pumps	3	ASME III	3	I	Note 26,A			; 68
Keep Warm Pump	3	ASME III	3	I	Note 26,A			; 68
Thermostatically Controlled Valve	3	ASME III	3	I	Note 26,A			; 68
Diesel Generator Lube Oil System						Fig. 9.5-56		
Lube Oil Sump Tank	3	ASME III	3	I	Note 26,A			; 68
Engine Lube Oil Pump	N/A	Mfrs Stds	-	I	Note 26,A		Note 2	; 68
Auxiliary Lube Oil Pump	N/A	Mfrs Stds	-	I	Note 26,A		Note 22	; 68
Lube Oil Cooler	3	ASME III	3	I	Note 26,A			; 68
Duplex Filter	3	ASME III	3	I	Note 26,A			; 68
Strainer	N/A	Mfrs Stds	-	I	Note 26,A		Note 22	; 68
Prelube Heater	3	ASME III	3	I	Note 26,A			; 68
Keep Warm Prelube Pump	N/A	Mfrs Stds	-	I	Note 26,A		Note 22	; 68
Keep Warm Filter	3	ASME III	3	I	Note 26,A			; 68
Diesel Generator Combustion Air Intake & Exhaust System						Fig. 9.5-57		
Intake Air Filter & Silencers	N/A	Mfrs Stds	-	I	Note 27,A		Note 22	; 68
Intake Air Flexible Connectors	3	Mfrs Stds	-	I	Note 50,A			; 68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Exhaust Air Silencers	N/A	Mfrs Stds	-	I	Note 26,A		Note 22	¦ 68
Piping & Valves outside Scope of	3	ASME III	3	I	Note 26,A			¦ 68
Diesel Generator Mfr.								¦ 68
Supports for Class 3 Piping	3	ASME III	-	I	Note 27,A		Note 25	¦ 68
		& AISC						¦ 68
Class 5 Piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	3.7B		¦ 68
16. Spent Fuel Pool Cooling and Cleanup System								
								¦ 66
Spent fuel pool cooling water pumps	3	ASME III	3	I	Note 26,A	9.1.3	Note 1b	¦ 68
Spent fuel pool heat exchangers	3	ASME III	3	I	Note 26,A	9.1.3	Note 1c	¦ 68
Spent fuel pool demineralizers	3	ASME III	3	I	Note 26,a	9.1.3		¦ 68
Spent fuel pool suction screens	3	Mfrs Stds	-	I	Note 26,A	9.1.3	Note 21	
Resin traps	3	ASME III	3	I	Note 26,A	9.1.3		¦ 68
Piping and valves	2	ASME III	2	I	Note 26,A	9.1.3		¦ 68
Piping and valves	3	ASME III	3	I	Note 26, 34B,A			¦ 68
Flow restricting orifice	3	ASME III	3	I	Note 26,A	9.1.3		¦ 68
Supports for Class 2 Piping	2	ASME III	2	I	Note 27,A	3.9B		¦ 68
Supports for Class 3 Piping	3	ASME III	3	I	Note 27,A	3.9B		¦ 68
Class 5 Piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	3.7B		¦ 68
17. Liquid Waste Processing System (LWPS)								
Reactor coolant drain tank	NNS	ASME VIII	-	NONE	Note 4, 28	11.2		¦ 68
Reactor coolant drain tank pump	NNS	Mfrs Stds	-	NONE	Note 4, 28	11.2		¦ 68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Reactor coolant drain tank heat exchanger								
Tube side	NNS	ASME VIII	-	NONE	Note 4, 28	11.2		: 68
Shell side	2	ASME III	2	I	Note 3,A	11.2		: 68
Waste holdup tank	3	ASME III	3	I	Note 4,A	11.2		: 59
Waste evaporator feed pump	3	ASME III	3	I	Note 4,A	11.2		: 59
Waste evaporator feed filter	3	ASME III	3	I	Note 4, 34N,A	11.2		: 59
Waste evaporator package								: 59
a. Feed preheater								: 42
Feed side	3	ASME III	3	II	Note 4, 34N,A	11.2	Note 11	: 68
Steam side	NNS	ASME VIII	-	II	None	11.2		: 68
b. Gas Stripper	NNS	ASME III	3	II	Note 4, 34N,	11.2	Note 11	: 68
c. Submerged Tube evaporator								: 42
Feed side	3	ASME III	3	II	Note 4, 34N,A	11.2	Note 11	: 68
Steam side	NNS	ASME VIII	-	II	None	11.2		: 68
d. Evaporator Condenser								: 42
Distillate side	3	ASME III	3	II	Note 4, 34N,A	11.2	Note 11	: 68
Cooling water side	3	ASME III	3	I	Note 4,A	11.2		: 68
e. Distillate Cooler								: 42
Distillate side	3	ASME III	3	II	Note 4, 34N,A	11.2	Note 11	: 68
Cooling water side	3	ASME III	3	I	Note 4,A	11.2		: 68
f. Absorption Tower	3	ASME III	3	II	Note 4, 34N,A	11.2	Note 11	: 68
g. Vent Condenser								: 42
Gas side	3	ASME III	3	II	Note 4, 34N,A	11.2	Note 11	: 68
Cooling water side	3	ASME III	3	I	Note 4,A	11.2		: 68
h. Distillate Pump	3	ASME III	3	II	Note 4, 34N,A	11.2	Note 11	: 68
i. Concentrate Pump	3	ASME III	3	II	Note 4, 34N,A	11.2	Note 11	: 68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
3. Piping and Valves								; 42
Feed	3	ASME III	3	II	Note 4, 34N,A	11.2	Note 11	; 68
Distillate	NNS	ASME III	3	II	Note 4, 34N	11.2		; 68
Concentrate	NNS	ASME III	3	II	Note 4, 34N	11.2		; 68
Cooling	3	ASME III	3	I	Note 4,A	11.2		; 68
Vent	NNS	ASME III	3	II	Note 4	11.2		; 68
Waste evaporator condensate demineralizer	NNS	ASME VIII	–	NONE	Note 4, 34N	11.2		; 68
Waste evaporator condensate filter	NNS	ASME VIII	–	NONE	Note 4, 34N	11.2		; 68
Waste evaporator condensate tank	NNS	ASME VIII	–	NONE	Note 4, 34N	11.2		; 68
Waste evaporator condensate tank pump	NNS	Mfrs Stds	–	NONE	Note 4, 34N	11.2		; 68
Chemical drain tank pump	NNS	Mfrs Stds	–	NONE	Note 4, 28			; 68
Chemical drain tank	NNS	ASME VIII	–	NONE	Note 5	11.2		; 68
Spent resin storage tank	3	ASME VIII	–	NONE	Note 4, 34N,A	11.2		; 68
Spent resin sluice pump	3	Mfrs Stds	–	NONE	Note 4, 34N,A	11.2		; 68
Spent resin sluice filter	3	ASME VIII	–	NONE	Note 4, 34N,A	11.2		; 68
Laundry and hot shower tank	NNS	ASME VIII	–	NONE	Note 4, 28	11.2		; 68
Laundry and hot shower tank pump	NNS	Mfrs Stds	–	NONE	Note 4, 28	11.2		; 68
Laundry and hot shower filter	NNS	ASME VIII	–	NONE	Note 5, 34N	11.2		; 68
Floor drain tanks I and II	NNS	ASME VIII	–	NONE	Note 4, 28	11.2		; 68
Floor drain tank III	NNS	API 620	–	NONE	None	11.2		; 68
Floor drain tank pumps	NNS	Mfrs Stds	–	NONE	Note 4, 28	11.2		; 68
Waste monitor tanks	NNS	ASME VIII	–	NONE	Note 4, 34N	11.2		; 68
Waste Monitor Tank Pumps	NNS	Mfrs Stds	–	NONE	Note 34N	11.2		; 68
Waste monitor tank demineralizer	NNS	ASME VIII	–	NONE	Note 4, 34N	11.2		; 68
Waste monitor tank filter	NNS	ASME VIII	–	NONE	Note 4, 34N	11.2		; 68
Floor drain tank filter	NNS	ASME VIII	–	NONE	Note 4, 34N	11.2		; 68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Floor drain tank strainer	NNS	Mfrs Stds	–	NONE	Note 4, 34N	11.2		68
Floor Drain Waste Evaporator Package						11.2		
a. Feed preheater								
Feed side	NNS	ASME III	3	NONE	Note 4, 34N			68
Steam side	NNS	ASME VIII	–	NONE	None			68
b. Gas stripper	NNS	ASME VIII	–	NONE	Note 4, 34N			68
c. Submerged tube evaporator								
Feed side	NNS	ASME III	3	NONE	Note 4, 34N			68
Steam side	NNS	ASME VIII	–	NONE	None			68
d. Evaporator condenser								
Distillate side	NNS	ASME III	3	NONE	Note 4, 34N			68
Cooling water side	3	ASME III	3	I	Note 4,A			68
e. Distillate cooler								68
Distillate side	NNS	ASME III	3	NONE	Note 4, 34N			68
Cooling water side	3	ASME III	3	I	Note 4,A			68
f. Absorption tower	NNS	ASME III	3	NONE	Note 4, 34N			68
g. Vent condenser								68
Gas side	NNS	ASME III	3	NONE	Note 4, 34N			68
Cooling water side	3	ASME III	3	I	Note 4,A			68
h. Distillate pump	NNS	ASME III	3	NONE	Note 4, 34N			68
i. Concentrate pump	NNS	ASME III	3	NONE	Note 4, 34N			68
j. Piping and valves								
Feed	NNS	ASME III	3	NONE	Note 4, 34N			68
Distillate	NNS	ASME III	3	NONE	Note 4, 34N			68
Concentrate	NNS	ASME III	3	NONE	Note 4, 34N			68
Cooling	3	ASME III	3	I	Note 4,A			68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Laundry and hot shower tank strainers	NNS	Mfrs Stds	-	NONE	Note 4, 34N	11.2		; 68
Waste evaporator reagent tank	NNS	ASME VIII	-	NONE	Note 5	11.2		; 68
Laundry holdup and monitor tanks	NNS	API 620	-	NONE	Note 34B	11.2		; 68
Laundry holdup and monitor tank pump	NNS	Mfrs Stds	-	NONE	Note 34B	11.2		; 68
Laundry reverse osmosis (RO) system	NNS	Mfrs Stds	-	NONE	Note 34B	11.2		; 68
RO concentrate tank	NNS	ASME VIII	-	NONE	Note 34B	11.2		; 68
RO concentrate tank pump	NNS	Mfrs Stds	-	NONE	Note 34B	11.2		; 68
Laundry water head tank	NNS	API 620	-	NONE	None	11.2		; 68
Piping and valves	3	ASME III	3	I	Note 26, 34B, 34N,A	11.2		; 68 ; 42
Valves	NNS	ANSI B31.1	-	NONE	Note 34B, 34N, C			; 68
Supports for Class 3 piping	3	ASME III	3	I	Note 27,A	3.9B		
Class 5 piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	3.7B		; 68
18. Gaseous Waste Processing System (GWPS)								
Gas compressors	3	ASME III	3	NONE	Note 3, 34N,A	11.3	Note 11	; 68
Gas decay tank	3	ASME III	3	I	Note 3, 34N,A	11.3		; 68
Hydrogen recombiner (catalytic)	3	ASME III	3	NONE	Note 3, 4, 34N,A	11.3	Note 11	; 68
Waste gas drain filter	NNS	Mfrs Stds	-	NONE	Note 5, 34N	11.3		; 68
Gas decay tank drain pump	NNS	ASME III	3	NONE	Note 3, 34N,A	11.3		; 68
Piping and valves	3	ASME III	3	I	Note 26, 34N, 34B,A	11.3		; 68
Valves	NNS	ANSI B31.1	-	NONE	Note 34B,C			; 68
Supports for Class 3 piping	3	ASME III	3	I	Note 27,A	3.9B		; 68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Class 5 piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	3.7B		; 68
Gas Trap	NNS	Mfrs Stds	-	II	Note 5, 34N	11.3		; 68
19. Solid Waste Processing System (SWPS)								
Solidification System	NNS	Mfrs Stds	-	NONE	Note 34B	11.4		; 68
Valves	NNS	ANSI B31.1	-	NONE	Note 34B,C			; 68
Class 5 piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	3.7B		; 68
Handling Equipment								; 42
-Filter transfer cask	NNS	Mfrs Stds	-	NONE	Note 26	11.4		; 68
-Flat bed trailer	NNS	Mfrs Stds	-	NONE	Note 26	11.4		; 68
-Remote handling tools	NNS	Mfrs Stds	-	NONE	Note 26	11.4		; 68
20. Demineralized and Reactor Makeup Water System								
Reactor makeup water pumps	3	ASME III	3	I	Note 26,A	9.2.3		; 68
Reactor makeup water storage tank	3	ACI 318-71	-	I	Note 32,A	3.8.4, 9.2.3	Note 55	; 68
Piping and valves	2	ASME III	2	I	Note 26,A	3.9B		; 68
Piping and valves	3	ASME III	3	I	Note 26,A	3.9B		; 68
Supports for Class 2 piping	2	ASME III	2	I	Note 27,A	3.9B		; 68
Supports for Class 3 piping	3	ASME III	3	I	Note 27,A	3.9B		; 68
Class 5 piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	3.7B		; 68
21. Vents and Drains System								
Safeguards building sump pumps	3	ASME III	3	I	Note 26,A	9.3.3		; 68
Piping and valves	2	ASME III	2	I	Note 26,A	9.3.3		; 68
Piping and valves (pump discharge)	3	ASME III	3	I	Note 26,A	9.3.3		; 68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Supports for Class 2 piping	2	ASME III	2	I	Note 27,A	3.9B		: 68
Supports for Class 3 piping	3	ASME III	3	I	Note 27,A	3.9B		: 68
Class 5 piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	3.7B		: 68
Containment Sump Flow Monitor	1E	IEEE-323	-	I	Note 26,A	5.2.5		: 68
22. Containment Ventilation Systems								
a. Containment recirculation								: 68
ductwork and supports	NNS	Mfrs Stds	-	II	Note 32,B	9.4A		: 68
dampers and supports	NNS	Mfrs Stds	-	II	Note 26, 32,B	9.4A		: 68
b. CRDM cooling system								
ductwork and supports	NNS	Mfrs Stds	-	II	Note 32,B	9.4A		
dampers and supports	NNS	Mfrs Stds	-	II	Note 26, 32,B	9.4A		
c. Neutron detector well cooling								
ductwork and supports	NNS	Mfrs Stds	-	II	Note 32,B	9.4A		
dampers and supports	NNS	Mfrs Stds	-	II	Note 26, 32,B	9.4A		
d. Reactor coolant sleeve cooling								
ductwork and supports	NNS	Mfrs Stds	-	II	Note 32,B	9.4A		
dampers and supports	NNS	Mfrs Stds	-	II	Note 26, 32,B	9.4A		
e. Containment preaccess filtration								
ductwork and supports	NNS	Mfrs Stds	-	II	Note 32,B	9.4A		: 31
dampers and supports	NNS	Mfrs Stds	-	II	Note 26, 32,B	9.4A		: 31
filtration unit	NNS	Mfrs Stds	-	II	Note 26,A	9.4A		: 31
f. Containment purge supply and exhaust								: 56
ductwork and supports	3/NNS	Mfrs Stds	-	I/II	Note 32,B	9.4A		: 68
dampers and supports	3/NNS	Mfrs Stds	-	I/II	Note 26, 32,B	9.4A		: 56
containment isolation valves and piping	2	ASME III	2	I	Note 26A	9.4A		: 56

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
g. Containment pressure relief								: 56
ductwork and supports	3	Mfrs Stds	-	I	Note 32,B	9.4A		: 56
containment isolation valves								: 56
and piping	2	ASME III	2	I	Note 26,A	9.4A		: 56
suction pipe	NNS	ANSI B31.1	N/A	II	Note 50,B	9.4A	Note 62	: 68
debris screen	NNS	Mfrs Stds	N/A	II	Note 27,B	9.4A	Note 62	: 68
HVAC panels (control room)	1E	IEEE-323	-	I	Note 26,A	9.4		: 68
21. Control Room Air-Conditioning System								
Air-conditioning units								
a. Roughing filter, fan, heating	3	Mfrs Stds	-	I	Note 26,A	9.4.1	Note 21	: 68
and direct-expansion refrigerant								: 68
type coils								: 68
b. Refrigerant piping & tubes	3	ASME B42 or	-	I	Note 26,A	9.4.1	Note 21	: 68
		ASTM B88			27, 50			: 60
c. Water side	3	ASME III	3	I	Note 26,A	9.4.1		: 68
Fans	3	Mfrs Stds	-	I	Note 26,A	9.4.1		: 68
Emergency pressurization and	3	Mfrs Stds	-	I	Note 26,A	9.4.1	Note 21	: 68
filtration units (roughing,								: 68
charcoal, and HEPA filters and								: 68
fans)								: 68
Piping and valves	3	ASME III	3	I	Note 26,A	9.4.1		: 68
Dampers and supports	3	Mfrs Stds	-	I	Note 26, 32,A	9.4.1	Note 21	
Ductwork and supports	3	Mfrs Stds	-	I	Note 26, 32,A	9.4.1	Note 21	: 68
Air accumulators (intake dampers)	3	ASME III	3	I	Note 26,A	9.3.1		: 68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Tubing and supports (between check	3	-	-	I	Note 26,A	9.3.1	Note 41	; 68
valves upstream of accumulators and								; 68
intake dampers)								; 68
Check valves for accumulators	3	ASME III	3	I	Note 32,A	9.3.1		; 68
HVAC panels (control room)	1E	IEEE-323	-	I	Note 26,A	9.4		; 68
24. Safeguard Building HVAC System								
a. Safeguards area								
dampers and supports (exhaust)	3/NNS	Mfrs Stds	-	I/II	Note 26, 32,A	9.4.5	Note 21	; 68
ductwork and supports	3/NNS	Mfrs Stds	-	I/II	Note 32,A	9.4.5	Note 21	; 68
(exhaust)								; 68
dampers and supports	NNS	Mfrs Stds	-	II	Note 26, 32,B	9.4.5		; 68
(supply)								; 68
ductwork and supports	NNS	Mfrs Stds	-	I/II	Note 32,B	9.4.5		; 68
(supply)								; 68
Emergency Fan Coil Units								
fans and housing	3	Mfrs Stds	-	I	Note 26,A	9.4.5	Note 21	; 68
cooling coils	3	ASME III	3	I	Note 26,A	9.4.5		; 68
b. Electrical area								
dampers and supports	3/NNS	Mfrs Stds	-	I/II	Note 26, 32,A	9.4C.3	Note 21	; 68
ductwork and supports	3/NNS	Mfrs Stds	-	I/II	Note 32,A	9.4C.3	Note 21	; 68
thermostats	1E	IEEE-323	-	I	Note 26,A	9.4C.3		; 56
supply units/fans	NNS	Mfrs Stds	-	II	Note 26,B	9.4C.3		; 68
Emergency Fan Coil Units								
fans and housing	3	Mfrs Stds	-	I	Note 26,A	9.4C.3	Note 21	; 68
cooling coils	3	ASME III	3	I	Note 26,A	9.4C.3		; 68
c. HVAC panels (control room)	1E	IEEE-323	-	I	Note 26,A	9.4		; 68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
25. Fuel Building Ventilation System								
Dampers and supports (exhaust)	3/NNS	Mfrs Stds	-	I/II	Note 26, 32,A	9.4.2	Note 21	68
Ductwork and supports (exhaust)	3/NNS	Mfrs Stds	-	I/II	Note 32,A	9.4.2	Note 21	68
Dampers and supports (supply)	NNS	Mfrs Stds	-	II	Note 26, 32,B	9.4.2		68
Ductwork and supports (supply)	NNS	Mfrs Stds	-	II	Note 32,B	9.4.2		68
Demister in fuel pool exhaust	NNS	Mfrs Stds	-	None	Note 26,B	9.4.2	Note 21	68
Emergency Fan Coil Units								
Cooling coils	3	ASME III	3	I	Note 26,A	9.4.2		68
Fans and housing	3	Mfrs Stds	-	I	Note 26,A	9.4.2		68
HVAC panels (control room)	1E	IEEE-323	-	I	Note 26,A	9.4		68
26. Diesel Generator Building Ventilation System								
Fans	3	Mfrs Stds	-	I	Note 26,A	9.4C.1	Note 21	68
Dampers and supports	3	Mfrs Stds	-	I	Note 26, 32,A	9.4C.1	Note 21	68
Ductwork and supports	3	Mfrs Stds	-	I	Note 32,A	9.4C.1	Note 21	68
HVAC panels (control room)	1E	IEEE-323	-	I	Note 26,A	9.4		68
27. Uncontrolled Access Area Ventilation System								
a. Battery room exhaust system								
Fans	3	Mfrs Stds	-	I	Note 26,A	9.4C.4	Note 21	68
Ductwork and supports	3	Mfrs Stds	-	I	Note 32,A	9.4C.4	Note 21	68
Dampers and supports	3	Mfrs Stds	-	I	Note 26, 32,A	9.4C.4	Note 21	68
b. Balance of system								
Supply unit/fan	NNS	Mfrs Stds	-	II	Note 26,B	9.4.3		68
Ductwork and supports	3/NNS/NNS	Mfrs Stds	-	I/II/None	Note 32,B	9.4.3		68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Dampers and supports	NNS	Mfrs Stds	-	II	Note 26, 32,B	9.4.3		: 53
Electric Unit Heaters								: 66
For Train A & B Battery								: 66
Rooms	1E	IEEE-323	-	I	Note 26,A	9.4C.4		: 66
c. HVAC panels (control room)	1E	IEEE-323	-	I	Note 26,A	9.4		: 68
28. Primary Plant Ventilation System								
a. Exhaust								
Modular exhaust filtration	3	Mfrs Stds	-	I	Note 26,A	9.4.3	Note 21	: 68
units (ESF)								: 68
Modular exhaust filtration	3	Mfrs Stds	-	I	Note 26,B	9.4.3		: 68
units (non-ESF)								: 68
Fans	3	Mfrs Stds	-	I	Note 26,A	9.4.3	Note 21	: 68
Ductwork and supports	3	Mfrs Stds	-	I	Note 32,A	9.4.3	Note 21	: 60
Dampers	3	Mfrs Stds	-	I	Note 26, 32,A	9.4.3	Note 21	: 68
Supports	3	Mfrs Stds	-	I	Note 32,A	9.4.3		: 68
Plant Vent Stack	NNS	Mfrs Stds	-	II	Note 32,B	9.4.D		: 68
b. Supply								: 56
Units/fans	NNS	Mfrs Stds	-	II	Note 26,B	9.4.3		: 68
Ductwork and supports	NNS	Mfrs Stds	-	II	Note 32,B	9.4.3		: 68
Dampers and supports	NNS	Mfrs Stds	-	II	Note 26, 32,B	9.4.3		: 68
c. HVAC panels (control room)	1E	IEEE-323	-	I	Note 26,A	9.4		: 68
29. Auxiliary Building HVAC System								: 56
Dampers and supports (exhaust)	3	Mfrs Stds	-	I	Note 26, 32,A	9.4.3	Note 21	: 56
Dampers and supports for	3	Mfrs Stds	-	I	Note 26, 32,A	9.4.3		: 66
Boron Injection Surge								: 66
Tank Room Emergency System.								: 56

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Dampers and supports (supply)	NNS	Mfrs Stds	-	II	Note 26, 32,B	9.4.3		; 68
Ductwork and supports (exhaust)	3/NNS	Mfrs Stds	-	I/II	Note 32,A	9.4.3	Note 21	; 68
Ductwork and supports for	3	Mfrs Stds	-	I	Note 32,A	9.4.3		; 66
Boron Injection Surge								; 66
Tank Room Emergency System.								; 66
Ductwork and supports (supply)	NNS	Mfrs Stds	-	II	Note 32,B	9.4.3		; 68
Emergency fan coil units								
Fans and housing	3	Mfrs Stds	-	I	Note 26,A	9.4.3	Note 21	; 68
Cooling coils	3	ASME III	3	I	Note 26,A	9.4.3		; 68
Emergency Fans for Boron	3	Mfrs Stds	-	I	Note 26,A	9.4.3		; 66
Injection Surge Tank Room								; 66
Emergency System								; 66
HVAC panels (control room)	1E	IEEE-323	-	I	Note 26,A	9.4		; 68
30. Service Water Intake Structure Ventilation System								
Fans (pump room exhaust)	3	Mfrs Stds	-	I	Note 26,A	9.4B		; 68
Ductwork and supports	3	Mfrs Stds	-	I	Note 32,A	9.4B		; 68
Dampers and supports	3	Mfrs Stds	-	I	Note 26, 32,A	9.4B		; 68
31. Chilled Water Systems								
A. Plant Ventilation chilled water system								
Class 5 Piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	9.4E, 3.6B		; 68
B. Safety chilled water system								
Chillers	1E	Mfrs Stds	-	I	Note 26,A	9.4F		; 68
Water side of chillers	3	ASME III	3	I	Note 26,A	9.4		; 68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Piping and valves	3	ASME III	3	I	Note 26,A	9.4F		| 68
Surge Tank	3	ASME III	3	I	Note 26,A	9.4F		| 68
Pumps	3	ASME III	3	I	Note 26,A	9.4F		| 68
Supports for Class 3 Piping	3	ASME III	3	I	Note 27,A	3.9B		| 68
32. Process Sampling System								
Sample heat exchanger	NNS	Mfrs Stds	-	II	Note B	9.3.2		
Delay coil	2	ASME III	2	I	Note 26,A	9.3.2		
Piping and valves	2	ASME III	2	I	Note 26,A	9.3.2		
Reflective insulation	NNS	Mfrs Stds	-	II	Note 27,B	6.1B		
Supports for Class 2 Piping	2	ASME III	2	I	Note 27,A	3.9B		
Class 5 Piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	3.7B		| 68
32a. Post Accident Sample System								
Sample heat exchanger	NNS	Mfrs Stds	-	NONE	Note C	II.B.3		| 68
Sample panel	NNS	Mfrs Stds	-	NONE	Note C	II.B.3		| 68
Piping and Valve (containment isolation portion)	2	ASME III	2	I	Note 26,A	9.3.2, II.B.3		| 68
Supports for Class 2 Piping	2	ASME III	2	I	Note 27,A	3.9B		| 68
Class 5 Piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	3.7B		| 68
Isolation valve Control Panel	1E	IEEE-323	-	I	Note 26,A	9.3.2, II.B.3		| 68
33. Fuel handling Equipment								
Refueling machine	N/A	Mfrs Stds	-	II	Note 4,B	9.1	Note 16	| 68
Containment Fuel Handling Bridge Crane	N/A	CMAA 74	-	II	Note 26,B		Note 23, 24	| 68
Fuel Handling Bridge Crane	N/A	CMAA 70	-	II	Note 4, 9,B		Note 23, 24	| 68
Rod cluster control changing fixture	N/A	Mfrs Stds	-	NONE	Note 4, 28		Note 16	| 68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Reactor vessel stud tensioner	N/A	Mfrs Stds	-	NONE	Note 5		Note 15	68
Spent fuel handling tool	N/A	Mfrs Stds	-	NONE	Note 4		Note 10	68
Fuel Transfer System:								41
-Fuel Transfer Tube & Flange	2	ASME III	MC	I	Note 4,A		(also evaluated as part of containment structure)	68
-Conveyer System & Controls	N/A	Mfrs Stds	-	NONE	Note 4, 28			68
-Remainder of System	N/A	Mfrs Stds	-	NONE	Note 4, 28			68
Refueling gates	N/A	AISC	-	I	Note 27,A			68
Fuel transfer tube expansion joint	2	ASME III	MC	I	Note 26,A			68
Stud hole plug handling fixture	N/A	Mfrs Stds	-	NONE	Note 4, 28		Note 10	68
Stud hole plugs	N/A	Mfrs Stds	-	NONE	Note 4, 28		Note 10	68
Lower internals storage stand	N/A	Mfrs Stds	-	NONE	Note 4, 28		Note 10	68
Upper internals storage stand	N/A	Mfrs Stds	-	NONE	Note 4, 28		Note 10	68
Rod cluster control thimble plug tool	N/A	Mfrs Stds	-	NONE	Note 5		Note 10	68
Source installation guide	N/A	Mfrs Stds	-	NONE	Note 5		Note 10	68
Crane scales	N/A	Mfrs Stds	-	NONE	Note 5		Note 10	68
Control rod drive shaft handling fixture	N/A	Mfrs Stds	-	NONE	Note 4, 28		Note 10	68
Stud tensioner Handling Device	N/A	Mfrs Stds	-	NONE	Note 5		Note 10	68
Irradiation tube end plug seat jack	N/A	Mfrs Stds	-	NONE	Note 5		Note 10	68
New fuel elevator	N/A	Mfrs Stds	-	NONE	Note 4, 28		Note 10	68
Portable underwater lights	N/A	Mfrs Stds	-	NONE	Note 5		Note 10	68
Load cells	N/A	Mfrs Stds	-	NONE	Note 4, 28			68
Damaged fuel container	N/A	ASME VIII	-	NONE	Note 4, 28		Note 10	68
Burnable Poison Rod handling tool	N/A	Mfrs Stds	-	NONE	Note 5		Note 10	68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Control rod drive shaft unlatching	N/A	Mfrs Stds	-	NONE	Note 5		Note 10	; 68
tool, full length								; 68
								; 68
New fuel storage racks	3	Mfr. Stds.	-	I	Note 4,A	9.1	Note 14	; 68
Spent fuel storage racks	3	Mfrs Stds	-	I	Note 4,A	9.1	Note 14	; 68
Refueling cavity seal ring	N/A	Mfrs Stds	-	NONE	Note 5		Note 10	; 41
New fuel handling tool	N/A	Mfrs Stds	-	NONE	Note 4, 28		Note 10	; 41
Rod control cluster assembly	N/A	Mfrs Stds	-	NONE	Note 5		Note 10	; 41
handling fixture								; 41
Fuel Assemblies	2	N/A	-	I	Note 6,A	4.2		; 68
34. Containment Building Miscellaneous Equipment								
Containment polar crane	N/A	CMAA 70	-	I	Note 26,A	9.1, 3.8	Note 53	; 68
Reactor vessel head lifting device	N/A	Mfrs Stds	-	I	Note 4,A	9.1	Note 48	; 68
Upper internals lifting device	N/A	Mfrs Stds	-	NONE	Note 4, 28	9.1		; 68
Containment auxiliary upper crane	N/A	CMAA 70	-	II	Note 26,B		Note 23, 24	; 41
Containment dome access rotating	N/A	CMAA 70	-	II	Note 26,B		Note 23, 24	; 41
platform								; 68
Neutron detector positioner	2	Mfrs Stds	-	I	Note 4,A		Note 23	; 68
Reactor vessel or core related								
components:								
Reactor vessel shoes and shims	1	Mfrs Stds	-	I	Note 4,A			; 68
Irradiation Sample holder	2	Mfrs Stds	-	I	Note 4,A		Note 18	; 68
Irradiation Samples	N/A	Mfrs Stds	-	NONE	Note 3			; 41
Full length control rod cluster	2	Mfrs Stds	-	I	Note 3,A			; 68
Control rod drive mechanism (CRDM)	NNS	Mfrs Stds	-	NONE	Note 5		Note 10	; 68
dummy can assemblies								; 68

LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
CRDM air cool baffle assemblies	NNS	Mfrs Stds	-	NONE	Note 5		Note 10	; 68
CRDM seismic support platform	2	Mfrs Stds	-	I	Note 3,A		Note 19	; 68
CRDM seismic support spacer plate	2	Mfrs Stds	-	I	Note 3,A		Note 19	; 68
CRDM seismic support tie rod assemblies	2	Mfrs Stds	-	I	Note 3,A		Note 19	; 68
Burnable poison rod assemblies	N/A	Mfrs Stds	-	NONE	Note 3		Note 10	
Reactor vessel insulation	NNS	Mfrs Stds	-	NONE	Note 4, 28		Note 10	; 68
Reactor vessel internals	2	Mfrs Stds	-	I	Note 6,A			; 68
Primary source rods	N/A	Mfrs Stds	-	NONE	Note 3			; 68
Incore Instrumentation								
-Seal table assembly	1	Mfrs Stds	-	I	Note 3,A		Provides support	; 68
							for safety class 1 pressure	; 68
							boundary conduit.	; 68
-Flux thimble tubing	2	ASME III	2	I	Note 3,A		Note 20	; 68
-Flux thimble tubing fittings	2	ASME III	2	I	Note 3,A		Note 20	; 68
Roll-away missile shield:								
-critical parts	2	CMAA 70	-	I	Note 26,B		Note 23	; 68
-non-critical parts	NNS	CMAA 70	-	II	Note 26,B		Note 23	
Sump Screen	2	AISC Code	-	I	Note 32,A	6.2.2	Frame structure	; 68
							supports recirculation	; 68
							piping moment restraint	; 68
Sump vortex suppression device	2	AISC Code	-	I	Note 32,A			
35. Miscellaneous Handling Equipment								
Drumming storage area crane	N/A	CMAA 70	-	II	Note 26,B	9.1	Note 23, 24	; 41
Fuel Building Overhead Crane	N/A	CMAA 70	-	I	Note 26,A	9.1, 3.8	Note 53	; 66
SWIS Crane	N/A	CMAA 70	-	II	Note 26,B	9.1	Note 23, 24	; 41
Auxiliary Filter Hoist	N/A	CMAA 70	-	I	Note 26,A		Note 53	; 41
Safety Chiller Hoist	N/A	CMAA 70	-	I	Note 26,A		Note 53	; 41

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
36. Structures								
Containment Building	N/A	Note 36	-	I	Note 27,A	3.8.1		68
Containment internal structure	N/A	ACI 318-71 &	-	I	Note 27,A	3.8.3		68
(including containment sump)		AISC Code						68
Safeguards buildings (including diesel	N/A	ACI 318-71 &	-	I	Note 27,A	3.8.4		68
generator room and emergency SWGR room		AISC Code						68
Auxiliary Building	N/A	ACI 318-71 &	-	I	Note 27,A	3.8.4		68
		AISC Code						68
Electrical and Control Bldg.	N/A	ACI 318-71 &	-	I	Note 27,A	3.8.4		68
		AISC Code						68
Fuel Building	N/A	ACI 318-71 &	-	I	Note 27,A	3.8.4		68
		AISC Code						68
Service Water Intake Structure	N/A	ACI 318-71 &	-	I	Note 27,A	3.8.4		68
		AISC Code						68
Safe Shutdown Impoundment Dam	N/A	-	-	I	Note 27,A	2.4 & 2.5		68
Containment Personnel Airlock	2	ASME III	MC	I	Note 27,A	3.8.1.1.6		68
Containment Equipment Hatch	2	ASME III	MC	I	Note 27,A	3.8.1.1.6		68
Containment Emergency Airlock	2	ASME III	MC	I	Note 27,A	3.8.1.1.6		68
Masonry Walls	N/A	-	-	None	Note 52	Q130.36		64
Removable Precast Block Walls	N/A	-	-	II	Note 63,B	Q130.36		64
Gypsum Walls	N/A	-	-	II	Note 27,B		Note 63	68
Missile Barriers	N/A	-	-	I	Note 27,A			68
Missile Resisting Doors	N/A	Mfrs Stds	-	II	Note 26,B			68
Watertight Doors	N/A	Mfrs Stds	-	I	Note 26,A			68
Handrails in Seismic	N/A	ACI 318-71&	-	II	Note 27,B			68
Category I Buildings		AISC Specification						68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
37. Electrical Equipment								
6900-V Switchgear (safety related)	1E	IEEE-323	-	I	Note 26,A	8.3.1		68
6900-V to 480-VAC transformers	1E	IEEE-323	-	I	Note 26,A	8.3.1		68
(safety related)								68
480-VAC switchgear (safety related)	1E	IEEE-323	-	I	Note 26,A	8.3.1		68
(safety related)								68
480-VAC motor local control stations	1E	IEEE-323	-	I	Note 26,A	8.3.1		68
(safety related)								68
Low voltage AC distribution panels	1E	IEEE-323	-	I	Note 26,A	8.3.1		68
(safety related)								68
118-V uninterruptible AC instrument	1E	IEEE-323	-	I	Note 26,A	8.3.1		68
distribution panels and subpanels								68
(safety related)								68
480 to 208/120-VAC transformers	1E	IEEE-323	-	I	Note 26,A	8.3.1		68
(safety related)								68
125-VDC station batteries	1E	IEEE-323	-	I	Note 26,A	8.3.2		68
(safety related)								68
125-VDC switchboards and distribution	1E	IEEE-323	-	I	Note 26,A	8.3.2		68
panels (safety related)								68
125-VDC Battery Chargers	1E	IEEE-323	-	I	Note 26,A	8.3.2		68
(safety related)								68
118-VAC static uninterruptible	1E	IEEE-323	-	I	Note 26,A	8.3.1		68
power systems (BOP safety related)								68
125-V station battery racks and supports	1E	IEEE-323	-	I	Note 26,A	8.3.2		68
(safety related)								68
Auxiliary relay racks safety	1E	IEEE-323	-	I	Note 26,A	8.3.1		68
related (BOP)								68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Solid-state safeguard sequencer	1E	IEEE-323	-	I	Note 26,A	8.3.1		; 68
Solid-state isolation equipment	1E	IEEE-323	-	I	Note 26,A	8.3.1		; 68
Containment electrical penetration	1E	IEEE-323	-	I	Note 26,A	8.3.1		; 68
assemblies								; 68
Electrical equipment supports	N/A	AISC Code	-	I	Note 27,A	8.3.1		; 68
(associated with safety related								; 68
equipment)								; 68
Motors (safety related)	1E	IEEE-323	-	I	Note 26,A	8.3.1		; 68
Power cables (associated with safety	1E	IEEE-323	-	N/A	Note 26,A	8.3.1		; 68
elated equipment)								; 68
Instrumentation and	1E	IEEE-323	-	N/A	Note 26,A	8.3.1		; 68
control cable (associated								; 68
with safety related equipment)								; 68
Wire and Cable Raceway System								
Cable trays (Containing Class 1E	N/A	-	-	I	Note 26,A	8.3		; 68
wires or cables)								; 68
Cable trays (Not containing Class	N/A	-	-	II	Note 26,B			; 68
1E wires or cables)								; 68
Conduit (Containing Class 1E	N/A	-	-	I	Note 49,A			; 68
wires or cables)								; 68
Conduit (Not containing Class 1E	N/A	-	-	II	Note 49,B			; 68
wires or cables)								; 68
								; 66
Supports (For cable trays or	N/A	AISC	-	I	Note 27,A			; 68
conduit that contain Class 1E								; 68
wires or cables)								; 68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Supports (For cable trays or	N/A	AISC	-	II	Note 27,B			| 68
conduit that do not contain								| 68
Class 1E wires or cables)								| 68
								| 66
480/120V Bypass transformers	1E	IEEE-323	-	I	Note 26,A	8.3.1		| 68
480/208-120V isolation	1E	IEEE-323	-	I	Note 26,A	8.3.1		| 68
transformers								| 68
Priority Panels	N/A	-	-	I	Note 27A	8.3.1		
Prefabricated Cables and	1E	IEEE-323	-	I	Note 26,A	8.3.1		| 68
Connectors (associated with								| 68
Safety Related equipment)								| 68
8KV Cable termination and	1E	IEEE-323	-	N/A	Note 32,A	8.3.1		
transition joints								
Terminal blocks	1E	IEEE-323	-	I	Note 32,A	8.3.1		| 68
Conduit Seals	1E	IEEE-323	-	I	Note 26,A	8.3.1		| 68
Heat Shrinkable cable	1E	IEEE-323	-	N/A	Note 32,A	8.3.1		
insulation sleeves								
38. Radiation Monitoring System								
Containment High Range Radiation	1E	IEEE-323	-	I	Note 27,A	12.3, 7.5		| 68
Monitors								| 68
Containment Air Monitors	N/A	Mfrs Stds	-	I	Note 27,A	11.5, 5.2.5		| 68
(including Particulate and								| 68
Gas Channels)								| 68
Plant Vent Stack Monitors	N/A	Mfrs Stds	-	NONE	Note 27,C	11.5, 7.5	Note 56	| 68
(including Particulate and								| 68
Iodine Samplers)								| 68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Main Steam Line Monitors	N/A	Mfrs Stds	-	II	Note 27,C	11.5, 7.5	Note 56	; 68
High Range Area Monitors	N/A	Mfrs Stds	-	NONE	Note 27,C	12.3, 7.5	Note 56	; 68
Low Range Area Monitors	N/A	Mfrs Stds	-	NONE	Note 27,C	12.3, 7.5	Note 56	; 68
Component cooling Water Monitors	N/A	Mfrs Stds	-	II	Note 27,B	11.5, 9.2		; 68
Waste Gas Monitor	N/A	Mfrs Stds	-	II	Note 27,B	11.5, 11.3		; 68
								; 66
Liquid Waste Effluent Monitors	N/A	Mfrs Stds	-	NONE	Note 27, 34B,C	11.5, 7.5		; 68
								; 66
Control Room Equipment Rack	N/A	Mfrs Stds	-	I	Note 27,A	11.5		; 68
(seismic)								; 68
Control Room Equipment Rack (1E)	1E	IEEE-323	-	I	Note 27,A	11.5		; 68
Control Room Ventilation Monitors	1E	IEEE-323	-	I	Note 27,A	11.5, 9.4		; 68
39. Fire Protection System								
Fire Suppression Systems	NNS	NFPA	-	II/None	Note 42,D	9.5	Note 58	; 55
Class 5 Piping and supports	NNS	NFPA	-	II/NONE	Note 44,B	3.6B		; 68
Portable Fire Extinguishers	N/A	NFPA	-	II/None	Note 42,D	9.5	Note 59	; 55
Fire Stops and Seals	N/A	ASTM E119	NONE	NONE	Note 42,D	9.5.1.5		; 66
Fire Rated Coating Systems	N/A	ASTM E119	-	II/None	Note 42,D	9.5	Note 60	; 68
RCP Lube Oil Collection System		9.5.1.3.6 (1)						
RCP Cowlings	NNS	Mfrs Stds	-	II	Note 32,B			; 68
Piping and Valves	NNS	ANSI B31.1	-	II	Note 44,B		Note 57	; 68
Tanks	NNS	ASME III	3	II	Note 27,D			; 68
Fire Detection System	N/A	NFPA-72D	-	II	Note 42,D	9.5.1.4.2 (1)		; 41
Fire Dampers	N/A	Mfrs Stds	-	II/None	Note 42,D	9.5	Note 54	
Fire Doors	N/A	Mfrs Stds	-	NONE	Note 42,D	9.5	Note 50	; 55
Fire Rated Barriers	N/A	ASTM E119	-	II/None	Note 42,D	3.8, 9.5	Note 50	; 66

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
40. Plant Gas System								
a) Nitrogen system								| 42
piping and valves	2	ASME III	2	I	Note 26,A	3.9B		| 68
Valves	NNS	ANSI B31.1	-	NONE	Note 34B, C			| 68
supports for Class 2 piping	2	ASME III	2	I	Note 27,A	3.9B		| 68
Class 5 piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	3.7B		| 68
b) Hydrogen system								
piping and valves	2	ASME III	2	I	Note 26,A	3.9B		| 68
supports for Class 2 piping	2	ASME III	2	I	Note 27,A	3.7B		| 68
Class 5 piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	3.6B		| 68
41. Instrumentation and Control								
Electronic transmitters (pressure and	1E	IEEE-323	-	I	Note 26, 3,A	7.0		| 68
differential pressure)								| 68
Electronic transmitters (flow)	1E	IEEE-323	-	I	Note 26, 3,A	7.0		| 68
Pressure switches	1E	IEEE-323	-	I	Note 26,A	7.0	Note 39	| 68
Level Switches	2,3	ASME III	2,3	I	Note 26,A	7.0	Pressure integrity only	| 68
Level Transmitters	1E	IEEE-323	-	I	Note 26,a	7.0	Functional Integrity only	| 68
Thermowells	2,3	ASME III	2,3	I	Note 25,A	3.2	Pressure Integrity only	| 68
Resistance Temperature Detectors	1E	IEEE-323	-	I	Note 26,A	7.0		| 68
Source and Intermediate Range Neutron	1E	IEEE-323	-	I	Note 3,A	7.2		| 68
Power Range neutron detectors	1E	IEEE-323	-	I	Note 3,A	7.2		| 68
Thermowells (RWMS)	NNS	ANSI B31.1	-	-	Note 34B			| 68
Neutron Flux Monitors	1E	IEEE-323	-	I	Note 27	7.0		| 68
Flow elements (RWMS)	NNS	ANSI B31.1	-	-	Note 34B, 34N			| 68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Containment hydrogen analyzers	1E	IEEE-323	-	I	Note 26,A	7.5		; 68
Differential Pressure Switches	1E	IEEE-323	-	I	Note 26,A	7.0		; 68
Rotameters	3	ASME III	3	I	Note 26, 4,	7.0	Pressure Integrity only	; 68
Rotameters	NNS	Mfrs Stds	-	II	Note 34N	11.3		; 68
Orifice Plates (Flow Metering)	2, 3	Mfrs Stds	-	I	Note 26,A	7.0		; 68
Chlorine detectors (control room intake)	N/A	Mfrs Stds	-	I	Note 26,A	9.4		; 68 ; 68
I & C impulse tubing, fittings and valves	2, 3	ASME III	2, 3	I	Note 32,A	7.0	Note 41	; 68 ; 68
I & C impulse tubing, fittings, valves and supports	2, 3	-	-	I	Note 32,A	7.0	Note 65	; 68 ; 68
I & C supports for impulse tubing, fittings and valves	NNS	-	-	II	Note 32,A	7.0	Note 51	; 68 ; 68
Instrument supports (seismic Category I instruments)	N/A	-	-	I	Note 32,A	7.0	Note 38	; 68 ; 68
Nuclear instrument racks (NIS)	1E	IEEE-323	-	I	Note 3,A	7.0		; 68
Process instrumentation and control racks (NSSS)	N/A	IEEE-344	-	I	Note 4,A	7.0		; 68 ; 68
Rod control equipment	N/A	-	-	NONE	Note 4	7.0		; 68
Rod position indication containment cabinets	N/A	-	-	NONE	Note 4	7.0		; 68 ; 68
I & C Power supply inverters (NSSS)	1E	IEEE-323	-	I	Note 3,A	7.0		; 68
Solid-state protection system cabinet	1E	IEEE-323	-	I	Note 3,A	7.0		; 68 ; 68
Control board demultiplexer	N/A	-	-	NONE	Note 4	7.0		
Hot shutdown panel	1E	IEEE-323	-	I	Note 26,A	7.0		; 68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Component	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
Shutdown Transfer Panel	1E	IEEE-323	-	I	Note 26,A	7.0		; 68
Process instrumentation protection racks	1E	IEEE-323	-	I	Note 3,A	7.0		; 68
Reactor trip switchgear	1E	IEEE-323	-	I	Note 3,A	7.0		; 68
Protective relay rack (6.9-kv bus UV/UF)	1E	IEEE-323	-	I	Note 26,A	7.0		; 68
Cable termination racks	1E	IEEE-323	-	I	Note 26,A	7.0		; 68
Local instrument racks	N/A	IEEE-344	-	I	Note 32,A	7.0		; 68
Local instrument racks	N/A	-	-	I	Note 32,A	7.0	Note 31	; 68
Control Room Benchboards, Vertical	1E	IEEE-323	-	I	Note 3, 26	7.0		; 68
Panels or Hot Shutdown Panel Mounted								; 68
Equipment Classified 1E								; 68
Control Room Benchboards, Vertical	NNS	Mfrs Stds	-	II	Note 50	7.0		; 68
Panels or Hot Shutdown Panel Mounted								; 68
Equipment Classified as Non-1E								; 68
Analog Instrumentation	1E	IEEE-323	-	I	Note 26,A	7.0		; 68
Cabinets (BOP Safety Related)								; 68
Auxiliary Relay Rack (NSSS)	1E	IEEE-323	-	I	Note 3,A	7.0		; 68
Upgrade Protection and	1E	IEEE-323	-	I	Note 3,A	7.0		; 68
Surveillance Cabinet (NSSS)								; 68
42. Tornado Venting Components						3.3.2		
Dampers	NNS	Mfrs Stds	-	II	Note 27,B		Note 24	; 68
Blowout Panels (Control Room)	NNS	Mfrs Stds	-	II	Note 27,B		Note 24	; 68
43. Compressed Air Systems						9.3.1		
Class 5 piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44, B			[illegible]

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
44. Protective Coatings						6.1B		
Coatings inside the Containment	NNS	NONE	-	-	None		Note 33	: 55
Coatings outside the Containment	NNS	NONE	-	-	None		Note 33	: 55
45. Potable and Sanitary Water System								
Domestic water storage tank	NNS	ASME III	3	II	Note 27,B	9.2.4		: 68
46. Condensate System								
Class 5 piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	10.4.7		: 68
47. Auxiliary Steam System								
Class 5 piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	10.4.13		: 68
48. Steam Generator Blowdown & Cleanup System								
Steam Generator blowdown Heat Exchanger	NNS	ASME VIII	-	NONE	Note 34B	10.4.8		: 68
Blowdown Cation Demineralizers	NNS	ASME VIII	-	NONE	Note 34B	10.4.8		: 68
Blowdown Mixed Bed Demineralizers	NNS	ASME VIII	-	NONE	Note 34B	10.4.8		: 68
Mixed Bed Resin Trap	NNS	ANSI B31.1	-	NONE	Note 34B	10.4.8		: 68
Cation Resin Trap	NNS	ANSI B31.1	-	NONE	Note 34B	10.4.8		: 68
Steam Generator Spent Resin Sluice Pump	NNS	Mfrs Stds	-	NONE	Note 34B	10.4.8		: 68
Filter	NNS	ANSI B31.1	-	NONE	Note 34B	10.4.8		: 68
Steam Generator Blowdown Spent Resin Storage Tank	NNS	ASME VIII	-	NONE	Note 34B	10.4.8		: 68
Valves	NNS	ANSI B31.1	-	NONE	Note 34B, C	10.4.8		: 68
Class 5 piping and supports	NNS	ANSI B31.1	-	II/NONE	Note 44,B	3.7B		: 68
Steam generator blowdown filter	NNS	ANSI B31.1		NONE	Note 34B	10.4.8		: 68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

System and Components	Safety Class (7)	Applicable Code or Standard (12)	Code Class	Seismic Category	Quality Assurance	Reference Section	Remarks	
49. Pipe Whip Restraints	N/A	Mfrs Stds	-	I	Note 26, 43,B	3.6B		| 68
50. Meteorological Instrumentation								
Meteorological Tower	N/A	Mfrs Stds	-	NONE	Note C	2.3		| 68
Sensors	N/A	Mfrs Stds	-	NONE	Note C	2.3		| 68
Signal Conditioners	N/A	Mfrs Stds	-	NONE	Note C	2.3		| 68
Power Supplies	N/A	Mfrs Stds	-	NONE	Note C	2.3		| 68
Data Logging and Computational Devices	N/A	Mfrs Stds	-	NONE	Note C	2.3		| 68
RM-21 Report Processor	N/A	Mfrs Stds	-	NONE	Note 26,C	11.5		| 68
51. Uninterruptible Power Supply (UPS) and Distribution Rooms System								| 66
Air-conditioning units								| 66
a. Roughing filter, fan, and direct-expansion refrigerant coils	3	Mfrs Stds	-	I	Note 26,A	9.4C.8	Note 21	| 68
b. Refrigerant piping & tubes	3	ASTM B42 ASTM B88	-	I	Note 26,A Note 27, 50	Fig. 9.4-15	Note 21	| 68 | 66
c. Water Side	3	ASME III	3	I	Note 26,A	Fig. 9.4-15		| 68
Dampers and supports	3	Mfrs Stds	-	I	Note 26, 32,A	Fig. 9.4-15	Note 21	| 66
Ductwork and supports	3	Mfrs Stds	-	I	Note 26, 32,A	Fig. 9.4-15	Note 21	| 68
Booster Return Fans	3	Mfrs Stds	-	I	Note 26,A	Fig. 9.4-15	Note 21	| 68

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LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

Notes

1. Services provided to support a safety or other necessary function:

   a. Emergency power - automatic loading

   b. Emergency power - manual loading

   c. Component cooling water

   d. Service water

   e. Steam

2. Portions of equipment containing component cooling or service water are Safety Class 3, Code Class 3.

3. Meets Quality Control System Requirements, Westinghouse QCS-1, which satisfies requirements of 10 CFR Part 50, Appendix B, Quality Assurance Criteria.

4. Meets Quality Requirements for Manufacture of Nuclear Plant Equipment, Westinghouse QCS-2, which satisfies requirements of 10 CFR Part 50, Appendix B.

5. Access for inspection and test is required by Westinghouse; however, no formal quality program approval is required.

6. Meets the quality assurance program of one of the Westinghouse NES manufacturing divisions, and is in accordance with 10 CFR Part 50, Appendix B.

7. Safety classes for fluid system components are defined by the engineering flow diagrams and are in accordance with ANSI N18.2, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants. Safety classes for electrical, instrumentation and control components are defined by the one-line diagrams, electrical wiring diagrams, and instrumentation and control diagrams. Safety classes for reactor containment pressure boundary components are in accordance with ANSI N18.2. | 57|D68

8. Represents code class upgrading as permitted by paragraph NA-2134 of the ASME Code, Section III, this component is upgraded from the minimum required Code Class 2 to Code Class 1.

9. Parts are mechanically of safety class and must meet the structural integrity requirements of the specification and quality assurance requirements of 10 CFR Part 50, Appendix B.

LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

10. Failure can cause no nuclear safety problem, although an economic loss may result.

11. This component is Safety Class 3 under the definition 2.2.3(1), (3), or (4) of ANSI N18.2-1973 and qualifies for no special seismic design by meeting the four following conditions. Portions of systems in which this component is located that perform the same safety function likewise qualify for no special seismic design.

Conditions to be met for exemption are the following:

a. Failure would not directly cause a Condition III of IV event (as defined in ANSI N18.2-1973, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants).

b. There is not safety function to mitigate nor could failure prevent mitigation of the consequences of a Condition III or IV event.

c. Failure during or following any Condition II event would result in consequences no more severe than allowed for a Condition III event.

d. Routine post seismic procedures would disclose loss of the safety function.

12. The applicable code or standard provided is the primary ASME code or industry standard which applies to the given component. Additional information about the design requirements is provided in the referenced FSAR Sections.

ASME: American Society of Mechanical Engineers. III stands for Section III of the ASME Boiler and Pressure Vessel (B&PV) Code. VIII stands for Section VIII of the ASME B&PV Code. Pressure vessels which are part of the RCPB meet the requirements of 1971 Version, with application of all Addenda through to and including the Summer 1972 Addenda. Pumps, valves, and piping which are part of the RCPB meet the requirements of the 1971 Version, with application of all Addenda through to and including the Winter 1972 Addenda. Later Code versions may be used optionally.

13. Not Used | 57|D68

14. Must maintain fuel array to prevent criticality under adverse conditions including occurrences of the SSE.

15. To be safety classified, failure of the tool must be directly a nuclear safety problem. If a nuclear safety problem arises from tool failures combined with a procedural failure thereafter, the tool is Non-Nuclear Safety.

16. Failure inside isolable reactor Containment prevents substantial release to the environment of radioactive gases from damaged spent fuel.

LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

17. Meets the quality assurance requirements as set forth in Westinghouse Quality Procurement Specification QPS-105-1, Rev. 0, which satisfies the pertinent requirements of 10CFR50, Appendix B.

18. Any reactor vessel internal, the single failure of which could cause release of a mechanical piece having potential for direct damage (as to the vessel cladding) or flow blockage, shall be classified to a minimum of Safety Class 2.

19. These items are not required as mechanical supports for CRDM housings, but are required to ensure functioning of the control rods.

20. Failure could cause a LOCA, but less than a Condition III loss-of-coolant.

21. This equipment is not commercially available as ASME Code, Section III, Class 3.

22. This equipment is included in the scope of IEEE 387.

23. Critical parts required to maintain structural integrity during a seismic event are subjected to QA program.

24. These devices will remain in place following a SSE.

25. Component supports are designed in accordance with ASME B&PV Code, Section III, Class 3 but fabricated to AISC -1970.

26. Meets quality assurance requirements as set forth in the applicable specification, which satisfy requirements of 10 CFR Part 50, Appendix B. 57[D68

27. Meets pertinent portions of the quality assurance criteria set forth in 10 CFR Part 50, Appendix B as defined by specification.

28. Procurement QA applied for non-safety considerations and may be discontinued after shipment from vendor or after installation.

29. Classified NNS on basis that flow restriction is provided in the piping.

30. Applies only to that bolting involved with coastdown function.

31. Pressure boundary parts for instruments connected to ANS Class 2 and 3 systems are procured to ASME Section III Class 2 and 3 requirements. Instruments and racks are not covered by ASME Code.

LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

32. Meets quality assurance provisions of Brown & Roots' Quality Assurance program (see FSAR Section 17.1.1.3), which satisfies requirements of 10 CFR Part 50, Appendix B.

33. The requirements for coating systems during all phases of installation is defined by G&H Specification 2323-AS-31. The requirements of ANSI N101.2, ANSI N512, and the coating manufacturers' recommendations will be used for guidance in the development of construction and maintenance procedures. These procedures will establish the requirements necessary to ensure a good quality coating system. | 55

34. The quality requirements of Branch Technical Position ETSB 11-1 are satisfied. Note: this is not within the scope of FSAR Section 17.1. Note 34N indicates NSSS scope of supply; Note 34B indicates BOP scope of supply. Where note 34 is used for safety-related equipment (Safety Class 1, 2, or 3, or Seismic Category I), the QA provisions of Appendix B envelope and satisfy the requirements of ETSB 11-1. The scope of ESTB 11-1 is defined by the engineering flow diagrams by "RWMS" boundary demarkation.

35. ATCOR topical report no. 132A gives a breakdown of system components and applicable design codes.

36. The applicable code for the Containment is the proposed Standard Code for Concrete Reactor Vessels and Containment (April 1973) issued for trial use and comments. It was developed by the Joint ACI-ASME Technical Committee on Concrete Pressure Components for Nuclear Service (see Section 3.8.1.2.1).

37. Deleted | 41

38. This also applies to NNS instruments which are connected to piping or ducting with seismic Category I or II tubing and supports. | 41

39. All pressure switches are differential pressure switches with the low side open to atmosphere.

40. Deleted | 41

41. Impulse tubing, valves, and fittings are supported as seismic Category I, but do not comply with ASME III, subsection NF. Therefore, this | 30

ASME material will not have Third Party Inspection, Code Stamping, and Code Data reports as specified in ASME subsection NA 5000 and NA 8000. | 66
Site fabrication and installation of this ASME material will be in accordance with NRC approved QA program governing non-ASME work which meets the | 30
requirements of Appendix B to 10 CFR Part 50 (ASME III subsection NA 4000 excluded). The leak testing of the instrument tubing between the | 50
instrument isolation valve and the instrument will be accomplished by completion of normal instrument calibration. |

TABLE 17A-1

(Sheet 45)

LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

42. The quality assurance requirements of Appendix A of the Branch Technical Position APCSB 9.5-1 of Standard Review Plan, Section 9.5.1, Revision 1, are satisfied.

43. The quality assurance requirements of Branch Technical Position MEB 3-1 and APCSB 3-1 are satisfied.

44. Meets quality assurance requirements as set forth in mechanical specification 2323-MS-46B, "Non-Nuclear Pipe Hanger and Supports" and mechanical specification 2323-MS-100, "Piping Erection".

Class 5 piping and its supports are designated Seismic Category II or None. Seismic Category II is utilized for piping included in ASME Section III stress analyses, high energy lines, and other special cases indicated by engineering evaluation. Seismic Category None is normally utilized for 2 inch and under piping, and 4 inch and under air filled copper tubing/pipe, unless otherwise indicated by engineering evaluation. | 68

Seismic Category II piping has Seismic Category II supports. Seismic Category None piping has Seismic Category II supports if engineering evaluation shows that a seismic failure would adversely affect the safety function of nuclear safety related items. | 68

45. NOT USED | 68

46. Safety function is achieved by remaining in place. Component safety function is non-active.

47. Applicable codes and standards for concrete radiation shielding are listed in Section 3.8.4.2.

48. Portions that furnish support to CRDMs only.

49. Structural support elements such as strut channels and strut fittings, procured as raw material; quality verified by sampling, testing, and certification. Conduit purchased commercially with certification by Vendor and Receipt Inspection. Conduit fittings purchased commercially with certification by Vendor and/or Receipt Inspection. | 68

50. Purchased as commercial grade with certification by vendor.

51. This applies only to instruments connected to Containment Hydrogen Purge Supply ducts.

LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

| Q130.36

52. Masonry walls have been evaluated for seismic interactions (remained non-seismic), removed, or replaced with seismically designed walls of other material. | 59

53. This load handling equipment satisfies NUREG-0554 single-failure-proof requirements. Seismic Category I denotes the capability to retain control of the lifted load and to maintain structural integrity during and after a seismic event. The loading combinations include OBE and SSE with lifted load except for the polar crane due to the low probability of SSE with lifted load (less than 1×10^{-7} per year). The polar crane criteria is discussed in FSAR Section 9.1.4.3.2. | 66

54. Fire dampers are functionally part of fire barriers but are shown on ventilation flow diagrams because they are physically located in ventilation ducts and penetrations. The safety class designation on ventilation diagrams does not apply to the fire dampers. Fire dampers which are required to remain open after an SSE are designated seismic Category II by the specification and are qualified to remain open. | 41

55. The tank boundary extends to the first weld connecting the penetration nozzles to system piping outside the tank. The tank and associated piping inside the tank is not N-stamped. | 46

56. The application of IEEE-323 is applied to those channels identified in FSAR Section 7.5 . | 57|D68

57. The Class 5 piping and valves in these 2" and under seismic Category II lines are seismically analysed and are supported by seismic Category II supports. Installation requirements are the same as other Class 5, seismic Category II pipes. | 46

58. Excluding existing buried yard piping. QA requirements of Appendix A to BTP APCSB 9.5-1 of Standard Review Plan Section 9.5.1 Revision 1 are to be satisfied for future activities associated with buried yard piping that supplies fire protection water to safety related buildings. | 66

59. QA for water extinguishers limited to UL listing. | 53

60. Deleted. | 66

61. The control panel and associated electrical components of the turbine driver are associated Class 1E located in mild environment. | 53

62. This component has been qualified as seismic category I by analysis. | 55

63. Applicable where wall is supported by structural steel specified seismically. | 59

64. Application of modified GDC-4 (leak-before-break) removes the need for this insulation. The insulation may be removed. | 61

65. This applies only to instruments connected to HVAC Ducts. | 66

TABLE 17A-1

(Sheet 48)

LIST OF QUALITY ASSURED STRUCTURES, SYSTEMS AND COMPONENTS

Operations Quality Assurance Notes:

A. An Operations QA Program will be implemented which satisfies applicable requirements of Regulatory Guide 1.33, Rev. 2, "Quality Assurance Program Requirements (Operations).

B. Critical parts required to maintain structural integrity during a seismic event are subjected to an Operations QA Program which satisfies applicable requirements of Regulatory Guide 1.33, Rev. 2.

C. Full QA requirements were not imposed for manufacture and/or installation; however, a QA program will be instituted during operation of equipment.

D. An Operations QA Program will be implemented which satisfies applicable requirements of Appendix A of the Branch Technical Position APCSB 9.5-1 of Standard Review Plan, Section 9.5.1, Rev. 1. 53

TABLE 17A-2

QUALITY ASSURANCE SUMMARY

SAFETY CLASS	SEISMIC CATEGORY	ENGINEERING, DESIGN, AND FABRICATION (NOTE 1)															SAFETY RELATED	CONSTRUCTION AND OPERATIONS QA APPLIED	
		NSSS SCOPE						BOP SCOPE											
		3	4	5	6	34N	17	26	27	32	33	34B	42	44	49	50			
1	I	X	X		X		X	X		X							YES	YES	
2	I	X	X		X			X		X		X					YES	YES	
3	I	X	X			X		X		X		X				X	YES	YES	68
3	II/NONE	X	X		X	X											NO	NOTE 2	68
1E	I (or N/A)	X						X		X							YES	YES	
N/A	I	X	X			X		X	X	X					X		YES	YES	68
NNS (or N/S)	II		X			X		X	X	X		X		X		X	NO	NOTE 2	
NNS (or N/A)	NONE	X	X	X		X		X			X	X	X				NO	NOTE 3	

NOTE 1: Numbers correspond to QA note in Table 17A-1.

NOTE 2: Construction and Operations QA are applied to the appropriate level defined by the QA notes.

NOTE 3: Construction and Operations QA are applied to the appropriate level on a case by case basis.

Q032.6 Describe the environmental qualification procedures and the limits of qualification that each of the following class 1E components located inside of the containment were qualified to.

1) Splices.

(2) Terminal blocks.

(3) Termination cabinets, and

(4) Connectors.

R032.6 Westinghouse scope of supply does not include distinct and separate splices, terminal blocks, connectors or termination cabinets located inside containment. For certain equipment, such as valve motor operators, terminal blocks (strips) are supplied as an integral part of the equipment and are included in the equipment qualification program. These programs are described in Revision 1 and Supplement 1 to WCAP-8587. Table 3.11N-3 relates the Westinghouse supplied equipment and its applicable Equipment Qualification Data Package.

BOP Scope of supply components will be qualified as follows:

1. Class 1E splices inside the containment are environmentally qualified to meet the requirements of IEEE 323-1974 and IEEE 383-1974. These splices are listed and their use justified in Appendix 8A, "Analysis to Justify Cable Splices in Raceway." 68

2. Class 1E terminal blocks inside containment will be required to meet the requirements of IEEE 323-1974.

3. There are no Class 1E termination cabinets inside the containment.

4. Class 1E cable connectors inside containment will be required to meet the requirements of IEEE 323-1974 and IEEE 383-1974.

Provide the details of your evaluation, the results and conclusions, and a tabulation of the following information:

(a) all conditions that render the diesel generator incapable of responding to an automatic emergency start signal for each operating mode as discussed above;

(b) the wording on the annunciator window in the control room that is alarmed for each of the conditions identified in (a);

(c) any other alarm signals not included in (a) above that also cause the same annunciator to alarm;

(d) any condition that renders the diesel generator incapable of responding to an automatic emergency start signal which is not alarmed in the control room; and

(e) any proposed modifications resulting from this evaluation.

R040.9 Each Diesel Generator (D-G) contains approximately 55 annunciator windows on the local control panel. Each window identifies the origin of any malfunction in the subject D-G, and includes conditions that would render the D-G inoperable for auto start in an emergency.

The control room annunciators contains a total of three (3) alarms for both D-G's: One alarm is to advise if the control switch of either of the two D-G's Remote-Local Maintenance Switch is not in "auto," Two (2) windows (one per unit) alarm if any trouble develops. The trouble alarm does not differentiate if the trouble renders the D-G inoperative for auto start in emergency.

Additionally, the Diesel Generator Power Window on the 66
Safety System Inoperable Indication Panel (SSII-see Section 7.1.2.6) is activated by those conditions that render the D-G inoperable for auto start during emergency conditions. The same SSII window is also activated if the following conditions exist:

1. Loss of Station Service Water
2. 6.9KV generator breaker control switch in lock-out position
3. Any D-G room cooler switch in the lock-off position

Although none of the three conditions listed above are 20
electrically interlocked with the D-G, it is recognized that they may render the D-G inoperable immediately after a short period of operation, or are otherwise important enough to advise the operator of its existence.

a. The condition that renders the D-G incapable of responding to an automatic emergency start signal are:

    1. 125V DC not available
    2. Overspeed trip not reset
    3. Differential lock-out relay not reset
    4. Remote-Local-Maintenance (R-L-M) switch in local or maintenance mode
    5. Starting air pressure low
    6. Diesel generator breaker CR-HSP Selector Switch in HSP position 68

Q040.109 The response to Question Q040.53 is not complete.

a) Provide out of tolerance values for voltages on Safeguard buses.

b) Provide identification of relays used in the coincidence logic scheme.

c) Provide time delays assumed in the SAR accident analysis and describe how the time duration of degraded voltage condition shall not result in failure of safety systems or components.

d) Provide copy of meter and test relay procedure noted in your response which shows trip set points and time values of these voltage relays. Also state which breaker will be affected by each relay.

R040.109 a) The preferred and alternate offsite power sources 68
to the safety-related electrical systems have been re-aligned to feed only safety-related 6900-V buses. This re-alignment has improved the available voltages at the safety-related buses. The preliminary calculations have indicated that minimum and maximum voltages at the safety-related buses will ensure adequate voltages at all safety-related equipment. A revised response to this question will be provided after finalization of voltage profile calculations.

b) Relays used in the 2nd level undervoltage 12
coincidence logic are identified in Table 040.109-1 "2nd Level Undervoltage Protective Relay Identification and Setting."

Q130.17 Identify any limitations in boundary conditions and the basic assumptions applicable to the computer programs identified in this section of the FSAR.

R130.17 The computer programs used to analyze the reactor containment building are ANSYS (for the foundation mat and large openings), and Shell-1 (for the containment shell). These two programs are particularly suitable for the analysis needs of nuclear containment structures. The ANSYS program is a general three-dimensional finite element computer program, capable of taking practically any type of loading conditions and boundary conditions. The Shell-1 program for shells of revelation is applicable only for axisymmetric structures such as containment shells. However, the program has the capacity of taking non-symmetric loads and boundary conditions through the technique of Fourier harmonic decomposition. | 68

Q130.25 In your answers to Q-130.5, Q130.16 and Q130.18, you changed the load combinations for the Containment Building to agree with the requirements of ACI 359 Code (1973) with certain exceptions as identified in the applicable sections of SRP 3.8.1. For the internal structures and for other Category I structures, you stated compliance with the respective requirements identified in SRP 3.8.3 and 3.8.4. In view of these changes, identify in detail how these changes in the design criteria have affected the final design of the Containment and other structures, if any. Specifically, state if they have resulted in any changes in the physical sizes of the structural components, rebar placement, properties, design stress levels, etc...

R130.25 The various Seismic Category I structures were designed to conform to the loading combinations and their related acceptance criteria which are specified by U.S. NRC Standard Review Plans 3.8.3 and 3.8.4. | 68

Q130.27 State the criteria used to account for accidental torsional effect of all Category I structures, including the Containment building. It is our position that a minimum of 5% accidental eccentricity should be considered due to the fact that both construction tolerances and the internal structures would introduce some degree of eccentricity effect. This can be accomplished by evaluating the structure considering 5% of the largest base mat dimension as an accidental eccentricity. That is the distance between the actual modified by the 5% eccentricity.

Provide information to demonstrate the extent to which the containment structure and the components located within the structure are capable of withstanding the largest load resulting from this criteria together with other applicable loads. In addition, provide the same information for all other Category I structures.

R130.27 The seismic analysis of all the seismic Category I structures, including the Containment Building is based on three dimensional modeling in which the actual center of mass at each floor elevation has been determined individually, and six degrees of freedom have been considered at each mass point with the corresponding stiffness properties obtained from a three dimensional finite element representation. Therefore, the approach used in the seismic analysis accounts for all torsional effects.

In the design of the Seismic Category I strucutures, shears resulting from torsion were computed using the larger of either the actual computed eccentricity at each sotry elevation or 5% of the maximum building dimension (normal to the direction of excitation). | 68

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Q130.28 In your answers to Q130.5, 16, 18, and 25, you stated how you considered the design and acceptance criteria identified in ACI-359 and SRP 3.8.1, 3.8.3 and 3.8.4 to validate the actual structural design of the Category I structures of the Comanche Peak NPP. In your conclusions, you stated that the actual design meets the requirements of ACI-359 and SRP 3.8.1, 3.8.3, and 3.8.4. Provide a detailed description of the specific controlling sections and components investigated in your re-evaluation, including pertinent sketches and results.

R130.28 The design of Seismic Category I structures conformed to the loading combinations which are specified by U.S. NRC Standard Review Plans 3.8.1, 3.8.3 and 3.8.4. | 68

| 68

Q212.20 Regulatory Guide 1.45 states that all three methods used in the detection of unidentified leakage should meet the sensitivity and response time requirements of RG 1.45. The containment air particulate monitors and radioactive gas monitors are dependent on background radiation levels for detection of a leak. The containment background level is in turn dependent on coolant activity and normal (expected) unidentified leakage. Show how the radiation monitors satisfy the requirements of RG 1.45 considering a range of containment activity levels stemming from change in coolant activity and normal unidentified leak rates.

R212.20 The sensitivity of the containment airborne particulate 68
and gas monitors for detection of 1 gpm primary coolant leakage is dependent on both the primary coolant activity level and the background radiation level in containment.

Conservative analysis indicates that the maximum pre- 68
existing containment background levels that will not prevent reliable leak detection by the particulate monitor (without spurious alarms) will vary as the square of the primary coolant activity for levels which are above concentrations corresponding to approximately 0.84% failed fuel and will vary linearly to concentrations below that level. This relationship is derived from the general sensitivity equation in ANSI 13.10 and the time constants specified for the monitor used in this service.

Analysis also shows that the maximum pre-existing 68
containment background levels for which reliable detection is possible will vary directly with the activity levels in the primary coolant until

concentrations corresponding to approximately 0.0046% failed fuel. With primary coolant concentrations less than this value, the increase in detector count rate due to leakage will be partially masked by the statistical variation of the minimum detector background count rate, rendering reliable detection of a 1 gpm leak uncertain. 68

In conclusion, reliable leak detection is possible, provided that the equilibrium activity of the containment atmosphere is below the level that would mask the change in activity corresponding to a 1gpm leak in one hour. Given the above limitations, the intent of the leak detection requirements of Regulatory Guide 1.45 is met in the following manner. The monitors are seismically qualified as required in Section C of Regulatory Guide 1.45. The minimum sensitivities of the containment air particulate and the radioactive gas monitors are 5×10^{-11} Ci/ml (reference nuclide Cs-137) and 1×10^{-6} Ci/ml (reference nuclide Xe-133), respectively. These are the minimum detectable activities when situated in a 2.5 mR/hr, of 1 Mev gamma background field, which is the normal maximum anticipated at the location of the monitors. These sensitivities meet or exceed the sensitivities required of these monitors by Section B of Regulatory Guide 1.45. 68

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Q421.18 Identify the organizational position with the authority and responsibility to approve changes to the plant Q-List and describe those provisions for controlling the distribution of the plant Q-List, including signature approval and revision numbers and/or dates. Describe the involvement of QA and/or QC personnel in this area.

R421.18 The plant Q-List (Table 17A-1) identifies major safety-related items within the scope of the nuclear quality assurance (QA) programs. This list is extended to include other items of the plant not classified as safety-related but for which some degree of quality assurance was applied during design and construction. Changes to FSAR Table 17A-1 are reviewed by and accepted by the Director,Quality Assurance (or his designee). | 68 | 66

The FSAR amendment that promulgates the change is approved and submitted by the Executive Vice President, Nuclear Engineering and Operations. | 66

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Q423.12 We could not conclude from our review of the preoperational test phase description and the test summaries provided in Table 14.2-2 that comprehensive testing is scheduled for several systems and components. Therefore, clarify or expand the description of the preoperational test phase to address the following:

1. Station Service Water System Test - State your plans to demonstrate proper operation of the strainers.

2. Component Cooling Water System Test - State your plans to demonstrate the system's capability to supply adequate cooling water flows to the "non-safety-related" loop loads.

3. Vents and Drains System Test - State your plans to include testing which demonstrates the operability of all vents and drains which can affect the capability of any equipment important to safety, even though those vents or drains may not handle potentially radioactive substances.

4. Fire Protection System Test - State your plans to verify that installation of features designed to contain fires (e.g., fire stops, fire doors, penetration seals) has been completed as a prerequisite to this test.

5. Spent Fuel Pool Cooling and Cleanup System Test - State your plan to verify correct flows entering and leaving the system for all modes of operation. (e.g., flow to purification loop from the refueling water storage tanks).

6. Residual Heat Removal System Test - Expand the test summary to describe how each mode of system operation will be demonstrated.

7. CVCS Chemical Control, Purification, and Makeup Subsystem Test - State your plans to demonstrate boric acid bathing and transfer capabilities and operation in the emergency borate mode.

8. Safety Injection System Hydraulic Performance Test - Describe the tests that will demonstrate each mode of ECCS operation. If any mode of operation will not be demonstrated by preoperational testing, provide technical justification for it omission. Also describe the testing that will demonstrate proper sequencing and operation of components on automatic switchover from injection to recirculation mode.

9. Safety Injection Accumulators Test - Modify the test summary to clarify which valves are opened in test method #3.

10. Gaseous Waste Processing System Test - State your plans to demonstrate the capability of the hydrogen recombiner.

11. Control Room Ventilation System Test; Auxiliary, Fuel and Safeguards Building Ventilation Test; Combustible Gas Control Systems Test - Revise Section 14.2.7 to state that testing of atmosphere cleanup system, air filtration and absorption units to be consistent with FSAR Appendix 1A (which refers to Regulatory Guide

1.52, Rev. 1, 7-76). Modify each of the test summaries to show that preoperational testing of each ESF air filtration and absorption unit will be performed in accordance with the regulatory guide.

13. Diesel Generator Compartment Ventilation Systems Test - The test method described by the test summary does not describe tests that will satisfy the stated test objective. Expand the test method to demonstrate the capability of the ventilation systems to provide adequate ventilation for the diesel generator compartments.

14. Diesel Generator Test - Expand the text summary to show that your test conforms to regulatory positions 2.a and 2.b of Regulatory Guide 1.108 (Rev. 1, 7-77) or provide technical justification for all exceptions to these positions.

15. AC Power Distribution System Test - Expand the test summary to show that this test will include normal power supply buses as well as ESF buses and vital buses. Also state your plans to perform full load tests using all sources or power supplies to each bus.

| 68

Q423.19 Provide a description of the electrical lineup for Unit No. 2 during preoperational tests that will be conducted to satisfy regulatory positions in Regulatory Guide 1.41 for Unit No. 1. Provide a description of the lineup for both plants during similar preoperational testing on Unit No. 2 subsequent to initial criticality of Unit No. 1. The descriptions should address both normal and emergency power distribution systems. Provide assurance that crossties will not exist which could cause loss of emergency bus power to one unit due to testing of the other unit.

R423.19 For Unit No. 2 equipment required to be energized prior to completion of Unit 1 preoperational testing, the Unit No. 2 electrical lineup will be as follows: | 68

6.9kV Buses 2EA1 & 2EA2 preferred source - Startup Transformer XST1

6.9kV Buses 2EA1 & 2EA2 alternate source - Startup Transformer XST2 | 68

6.9kV Buses 2A1, 2A2, 2A3, & 2A4 source - Startup Transformer 2ST, or backfeed through 2MT1, 2MT2 and 2UT | 68

The electrical lineup for both units during Unit No. 2 preoperational testing will be as follows:

Unit No. 1

6.9 kV Buses 1EA1 & 1EA2 preferred source -
Startup Transformer XST2

6.9 kV Buses 1EA1 & 1EA2 alternate source - | 68
Startup Transformer XST1 |

6.9kV Buses 1A1, 1A2, 1A3, & 1A4 sources - | 68
Startup Transformer 1ST or Unit Aux. |
Transformer 1UT

Unit No. 2

6.9 kV Buses 2EA1 & 2EA2 preferred source -
Startup Transformer XST1

6.9 kV Buses 2EA1 & 2EA2 alternate source - | 68
Startup Transformer XST2 |

6.9 kV Buses 2A1, 2A2, 2A3, & 2A4 source - | 68
Startup Transformer 2ST | 68

For preoperational testing on Unit No. 2, Unit No. 1 6.9 kV Class 1E buses will be electrically lined up in the designed plant operational mode with two offsite power sources (preferred and alternate) and one onsite power source (diesel) available to each Class 1E bus. Plant electrical distribution circuits will be lined up in only the designed operational modes which precludes not only the loss of all sources to Unit No. 1 emergency buses, but also eliminates the possibility of distribution system crossties.

Q423.34 Your response to item 423.18 is not acceptable. On FSAR page 8.2-6 you state that "in the event one startup transformer (e.g., XST1, a preferred source) becomes unavailable to its normally fed Class 1E buses, power is made available from the other startup transformer (e.g., XST2, an alternate source) by an automatic transfer (fast or slow transfer)." Our concern is that this transfer might overload a startup transformer resulting in a degraded voltage situation on the ESF buses and failure of ESF equipment. (See Power Systems Branch item 040.44). It is our position that you demonstrate the capability of each startup transformer to carry the maximum load that it is postulated to carry during any mode of plant operation (including configurations where other transformers have failed and loads are automatically transferred to it). Provide a summary of this test.

R423.34 Preferred and alternate power sources to the safety-related electrical systems have been realigned to feed only safety-related buses. This re-alignment has eliminated the possibility of overloads on a startup transformer. 68

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B. LOCATION 27

The onsite TSC at CPSES is in the observation area above the control room, on elevation 840'6" of the Control Building (refer to Figure III.A.1.2-1). The TSC and control room are connected by a common stairwell, which can provide face-to-face interaction between control room personnel and the TSC Manager. TSC personnel will have access to information in the control room that is not available through the TSC Data System. 27

The stairwell between the TSC and control room is located outside of the control room environment; therefore, provisions will be made for safe and timely movement of personnel under emergency conditions. These provisions will include considerations of effects of direct radiation and airborne radioactivity from inplace sources. 27

There will be no major security barriers between the TSC and control room, other than access stations at each facility. 27

C. STAFFING AND TRAINING 27

The staffing requirements of the TSC are in accordance with Figure 1.4 of the CPSES/Emergency Plan. Figure 1.4 staffing applies to all emergency action levels for which the TSC is activated. 36

Upon declaration of an Alert or higher classification of Emergency, the TSC should be activated within sixty (60) minutes. 68 Activation of the TSC will ensure only designated operating personnel are in the control room and that needed 49 technical support will be provided without obstructing plant manipulations or overcrowding the control room.

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27 The data set available to the TSC Data System will be complete
enough to permit accurate assessment of an accident without
interference with the control room emergency operation.

68 The data set available to the ERF Computer System includes the
Type A, B, C and E Accident Monitoring variables (as described
in FSAR Section 7.5) and the Type D Accident Monitoring
variables which are also identified as Type D variables in
Regulatory Guide 1.97, Rev. 2, Table 2.

42 All sensor data and calculated variables used in the data set
for SPDS, EOF, or for transmission to offsite locations will be
available for display. The accuracy of the data displayed will
27 be substantially the same as the accuracy of comparable data
36 displayed in the control room. The time resolution of data
acquisition will be sufficient to provide data without loss of
information during transient conditions where that data is
necessary for operator action. The time resolution of each
sensor signal will respond on the potential transient behavior
of the variable being measured.

27 Disk and tape storage and recall capability will be provided for
the TSC data set. Two hours of pre-event and 12 hours of post
event data will be recorded.

27 The sample frequency will be consistent with the use of the
data. Capacity to record two weeks of additional post event
data will be provided. Archival data storage will be provided
automatically, and retrieval will be accomplished without
36 interrupting the TSC data acquisition function. The ERF
displays (3 of which will be provided in the CPSES TSC) if used
for data retrieval, will not be available for real time
parameters, but can be returned to on line display service very
quickly.