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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

February 2, 1979

Docket No. 50-213

Mr. W. G. Counsil, Vice President
Nuclear Engineering and Operations
Connecticut Yankee Atomic Power Company
Post Office Box 270
Hartford, Connecticut 06101

Dear Mr. Counsil:

As recently discussed with you in connection with the NRC Systematic Evaluation Program, a Senior Seismic Review Team will visit Haddam Neck Plant on February 21, 1979, and your engineering office on February 22, 1979. The following staff members and consultants will attend:

NAME	NRC BADGE #
Thomas V. Wambach	C 1282
Michael H. Fletcher	C 1825
Kahtan N. Jabbour	B 1435
Thomas M. Cheng	B 4197
Howard A. Levin	B 1117
William J. Hall (consultant)	
Frank J. Tokarz (consultant)	
John D. Stevenson (consultant)	
Robert P. Kennedy (consultant)	
Robert Murray (consultant)	

The purpose of the visit is to observe site specific features of Haddam Neck relative to the seismic design of the facility, to obtain seismic design information not available to the staff on the Haddam Neck docket, and to discuss and identify seismic design information that the NRC staff and its consultants have reviewed.

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The proposed agenda for the site meeting is provided in Enclosure 1. Enclosure 2 identifies information that is needed for the Senior Seismic Review Team to continue the seismic design review of Haddam Neck. Enclosure 3 identifies the purpose, scope and depth of review of the Senior Seismic Review Team. Enclosure 4 is a summary of a review by the NRC staff and its consultants of the docket and other literature relating to the Haddam Neck Plant seismic analysis and design.

Sincerely,

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Agenda
2. Request for Information
3. SEP Seismic Review
4. Summary

cc w/enclosures:
See next page

cc w/enclosures:
Day, Berry & Howard
Counselors at Law
One Constitution Plaza
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Superintendent
Haddam Neck Plant
RFD #1
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Mr. James R. Himmelwright
Northeast Utilities Service Company
P. O. Box 270
Hartford, Connecticut 06101

Russell Library
119 Broad Street
Middletown, Connecticut 06457

ENCLOSURE 1

AGENDA FOR SITE VISIT

Wednesday, February 21, 1979

8:30 a.m.	Assemble at Haddam Neck Site
9:00 a.m.	Security/Health Physics Lecture
10:30 a.m.	Conference Room - Office Building
	Definition of the scope of the seismic review - (NRC). Identification of Haddam Neck Engineered Safety features, safe shutdown systems and auxiliary systems required for safe shutdown (CYAPCo). (Construction photos, reactor model and PSID's). Discussion of topics identified in Enclosure 2.
12:00 p.m.	Lunch
1:00 p.m.	Tour of Haddam Neck facility
4:00 p.m.	Meeting to discuss questions raised during the tour.
5:00 p.m.	Adjourn

ENCLOSURE 1

AGENDA (Contd.)

<u>Thursday, February 22, 1979</u>

9:00 a.m.	Assemble at NUSCO offices (Berlin, Connecticut)
9:15 a.m.	Engineering discussion with CYAPCo, NUSCO, consultants
12:00 p.m.	Assignments for further review and/or report preparation and interaction among review team, NRC staff, CYAPCo, etc.
1:00 p.m.	Adjourn

ENCLOSURE 2

PERTINENT INFORMATION REQUIRED
FOR REVIEW BY THE
SENIOR SEISMIC REVIEW TEAM

The Senior Seismic Review Team (SSRT) would like to have available for review prior to the February 21, 1979 site visit, simple descriptions, summaries and tabulations of pertinent information relative to the seismic analysis and design of the Haddam Neck facility. In this regard, the NRC staff and its consultants have conducted a review of the docket and other available literature and have summarized the results of this effort in a draft report entitled, "Seismic Design Bases and Criteria for Connecticut Yankee Nuclear Generating Station, Haddam, Connecticut", dated January 1979 (Enclosure 4). The SSRT solicits your comments on this report regarding its accuracy and any other pertinent information that you can provide to make this document more complete and to facilitate the review process. These topics may appropriately be discussed during the engineering discussions the day following the site visit.

The following specific information is desired for review by the SSRT by February 16, 1979. It is recognized that full compliance may be difficult in the short timeframe; therefore, it is requested that you provide all that you can by February 16, 1979, and be prepared to discuss the remaining issues at the time of the site visit.

1. The seismic review for Haddam Neck will focus on an assessment of the integrity of the reactor coolant pressure boundary and the capability of essential systems and components required to safely shutdown the reactor and maintain it in a safe shutdown condition during and after a postulated seismic disturbance. In this regard, the following systems and components have been identified.

   A. Reactor coolant pressure boundary integrity - LOCA prevention

      1. Reactor coolant system, including reactor vessel supports, reactor coolant pump supports, steam generator supports, pressurizer supports and piping supports

      2. All penetrations to the reactor coolant system

      3. Control rod drive housing integrity

   B. Safe shutdown systems (total loss of offsite power, no LOCA)

      1. Steam system ASME code safety valves

      2. Atmospheric dump valves (ADV), steam generator vents and other vent paths

3. Auxiliary feed systems

4. Water sources - Demineralized water storage tank (DWST), primary water storage tank (PWST), and primary water transfer pump

5. Residual heat removal system

6. Service water system

7. Chemical and volume control system

8. Control rod drive system

9. Emergency power systems (AC and DC) for above equipment

10. Instrumentation necessary for operation of the above equipment and the monitoring of core parameters

It is requested that you review this listing, suggesting modifications as appropriate. A more detailed list of specific pieces of mechanical and electrical equipment necessary for the operation of these systems will be requested at a later time. Additionally, please provide examples of equipment design specifications (including the seismic spec or just the seismic spec if separate) and address generically the typical methods of seismic qualification of equipment at Haddam Neck. More detailed and specific information will be requested after the site visit.

2. The docket material is unclear regarding the load combinations used in the design of various safety related or Category I structures and components of the facility. Please summarize the load combinations considered and clarify what is meant by "incident", whether pipe break and seismic have been combined, and how normal operating loads were incorporated.

3. To make a realistic estimate of seismic safety margins, the SSRT requires a more detailed understanding of the basic seismic analysis and design procedures.

A. Describe in detail the procedure for calculating earthquake structural response (Containment Shell and internal structures), including the application of the Rayleigh method and the modeling procedure.

B. Describe in detail how the equivalent static loads were applied in the design of structures (including internal structures), and provide example computations which demonstrate the methods of calculating stresses in the concrete and steel at the base of the containment building for given shears, tension/compression and moments; discussing the actual loads used.

C. Summarize the procedures used for determining the seismic input to piping, and mechanical and electrical equipment.

D. Summarize the design criteria used for the design of cable trays and duct work including any generic evaluations made.

E. Provide complete stress analyses of two piping systems listed in 1 above.

   1. $\geq$ 12 inches in diameter

   2. $\leq$ 6 inches in diameter

   The analyses should show how seismic stresses were included in the total stresses to meet ANSI B31.1 Code requirements.

4. Figures 1.1 and 1.2 of "Additional Information Supplied in Response to December 27, 1963 Request of AEC Division of Licensing and Regulation", dated January 1964 are ground response spectra curves based upon the 1957 Golden State Park record normalized to 0.1 g. Figure 2.5-1 of the FSAR is a generalized envelope ground response spectrum developed by Housner from four California earthquakes normalized to 0.17 g. Please clarify which of these ground spectra were used in the design of specific structures, systems and components. If the Golden State Park spectra were used earlier and the envelope spectra subsequently, specify whether new analyses were completed or whether checks were made to verify the previous analyses.

5. The following structures will be reviewed as Category I structures.

   A. Containment Building
   B. Primary Auxiliary Building
   C. Fuel Storage Building
   D. Service Building
   E. Screen House

   It is requested that you review this listing and verify that the systems and components identified in 1 above are not housed in any other structures, (e.g., the Turbine Building).

SEP SEISMIC REVIEW - EVALUATION OF SELECTED OPERATING PLANTS

SENIOR SEISMIC REVIEW TEAM

CHARTER

A. Purpose

To determine the margins of safety of existing nuclear plants relative to those designed under current standards, criteria, and procedures; and to define the nature and extent of retrofitting to bring these plants to acceptable levels of capability if they are not already at such levels.

B. Scope

To review seismicity and site conditions, structural and equipment capability to resist changed seismic hazards, and margins of safety in relation to current requirements, for all safety-related components elements, and systems. This may be done on the basis of either probability studies, deterministic evaluations, or a combination of these approaches.

C. Depth of Review

The review will consist of two parts, as follows:

1. A preliminary study of the plant considered based on: the data for geology, seismology and site conditions contained in the FSAR & PSAR; the design criteria, load combinations and methods of analysis used in the Operating License review; changes in seismicity evaluations for the region; changes in seismic design and/or design criteria since the OL review; and a site visit to make a spot check visually of selected parts or components of the structure, equipment and engineered safeguards.

2. A detailed review of any items identified in (1) as being possibly questionable, or deficient; a listing of items or criteria or procedures used in design or construction which deviate from current criteria; an engineering analysis of such items in sufficient depth to enable a judgment to be made by the review team as to the acceptability of such deviations; a final judgment of the acceptability of the actual safety margins of all important items in the facility and recommendations, as necessary, for further evaluations.

ENCLOSURE 4

SEISMIC DESIGN BASES AND CRITERIA FOR CONNECTICUT YANKEE NUCLEAR GENERATING STATION, HADDAM, CONNECTICUT

SYSTEMATIC EVALUATION PROGRAM

1. INTRODUCTION

This report presents the results of an evaluation of the seismic design bases used in the design and analysis of the Connecticut Yankee Nuclear Generating Plant located at Haddam Neck, Connecticut. The evaluation was conducted by means of an in-depth review of docket and other available literature. However, a review of the actual analyses was not conducted to assure compliance with the criteria.

The plant is located on the Connecticut River in South-central Connecticut approximately twelve miles from Long Island Sound. The plant is a four loop, Pressurized Water Reactor (PWR) of 575 MWe capacity. The nuclear steam system was supplied by Westinghouse Electric, Inc. and Stone and Webster was the Architect Engineer. Commercial operation was attained in January 1968.

The reactor is housed in a conventional right circular reinforced concrete containment structure. The inside diameter is 135 feet with a 4 foot 6 inch thick cylindrical wall and a 2 foot 6 inch thick hemispherical dome. The liner is 3/8 inch steel on the cylindrical wall and 1/2 inch on the dome. The base slab is founded on granitic gneiss bedrock with the bottom of the slab embedded 30 foot 6 inches below grade. Grade elevation is 21 foot MSL. Figure 1-1 shows the overall configuration of the reactor building and equipment locations. Additional figures are contained in Appendix A.

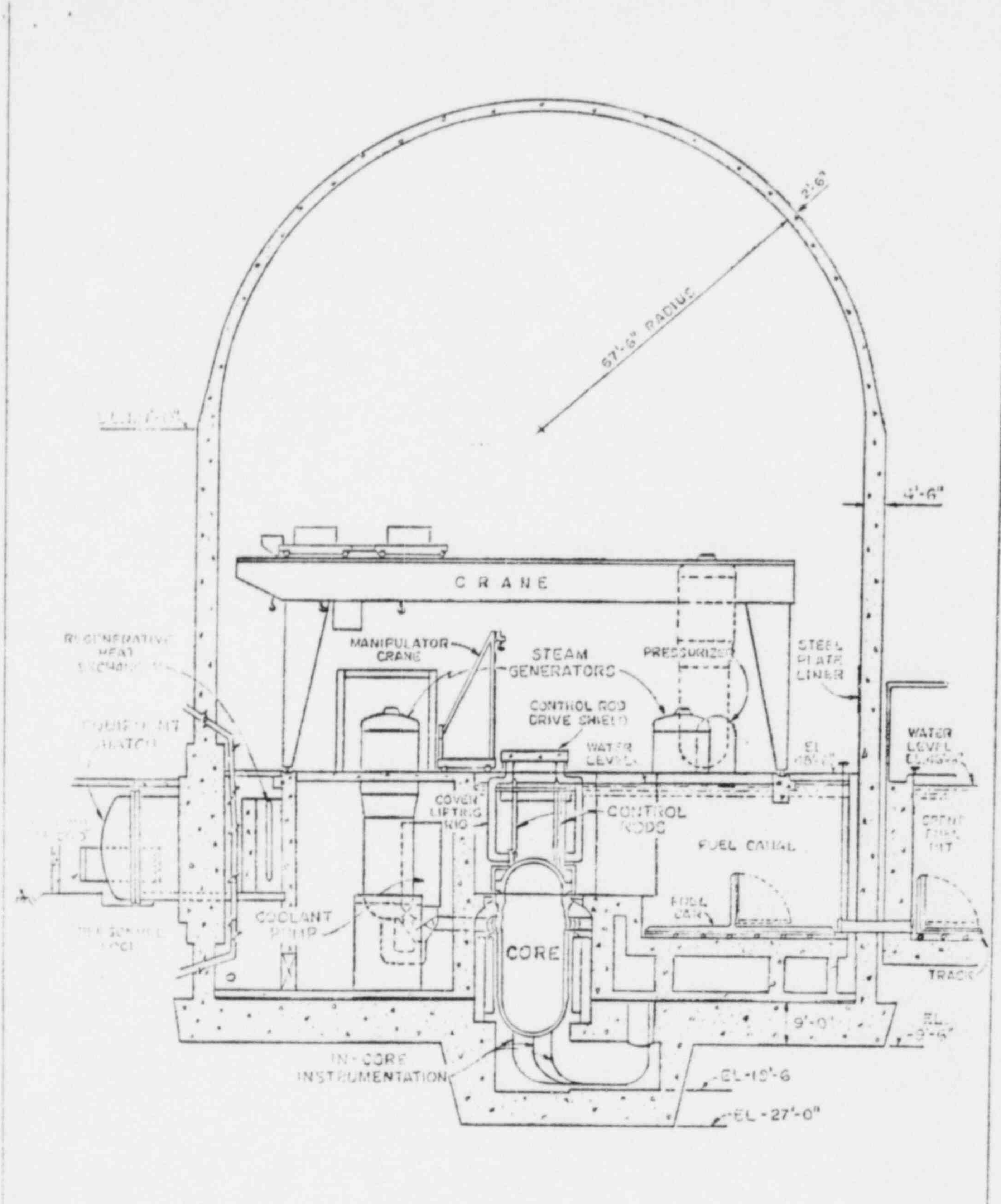

FIGURE 1-1. REACTOR CONTAINMENT BUILDING

2. GEOTECHNICAL

2.1 GEOLOGY

The original site consisted of two principal types of terrain. Above the 10 to 20 foot elevation, the hillsides rise steeply and are underlain by bedrock (Hebron gneiss) partially covered by till and, locally, outwash gravels. Below the 10 to 20 foot elevation, the area was a fairly flat flood plain, locally covered with swamps and tidal marshes, and with natural levees forming elevations of about 10 feet along most of the river and half of the total flood plain area. Figures 2-1 and 2-2 show the site and plot plan.

Organic swampy soils were removed before placing fill in all areas upon which structures were built. The major structures are founded directly on the granitic gneiss bedrock. Minor structures are founded either on rock, on piles driven to the rock, or in a few places on spread footings in compacted granular fill, depending on the character of the structure and its use.

A coordinated program of seismic exploration and borings, carried to and into the rock, was developed. The logs of the borings and their locations are available in the docket (Reference 1). A series of bonds of mica schists was found to run in a general north-south direction across the southern sector of the site. In the northern sector of the site, a broad bank of granitic gneiss substantially covered an area outlined by outcrops. The gneiss is coarsely crystalline. This area presents a comparatively uniform rock stratum upon which the plant is located. The overburden was excavated, thus permitting thorough examination and removal of weathered or excess rock material.

2.2 SEISMOLOGY

The seismic history of the site from Reference 1 at the time of construction is discussed below. The seismic history of the site area is given in Table 2-1 which lists all recorded earthquakes with intensities of 5 or greater (Rossi-Forel Scale) with epicenters in Connecticut.

The most severe recorded earthquake occurred in May, 1791. This earthquake was reported to have caused stone walls to be shaken down, tops to be thrown off chimneys and latched doors to be thrown open. Four other earthquakes, apparently less intense, followed in the relatively short period ending in 1805. These five earthquakes were the most recent to be centered at East Haddam.

The largest fault system in Connecticut bounds the Triassic Basin on the east and is known as the Great Fault or the Triassic Eastern Border Fault. Its locus is approximately from 1 mile west of Rockville near the north end, passing 2 miles east of Middletown to Short Beach on Long Island Sound. This fault had its principal activity about 200 million years ago and is now considered to be a relatively inactive zone geologically. The point on the fault nearest to the site area is 8 miles west-northwest.

A fault trending east-northeast to west-southwest, known as the Honey Hill Fault, passes 5 miles south-southeast of the site area. This displaces pre-Triassic rock and probably had its principal activity no later than that of the Eastern Border Fault.

The map of the United States showing zones of approximately equal seismic probability, as approved by the International Conference of Building Officials in the Twenty-Eighth Annual Meeting for inclusion in the 1961 Edition of the Uniform Building Code, indicates that the area is near the boundary between Zones 1 and 2 (areas having earthquakes with hazards of minor magnitude).

TABLE 2-1

MAJOR RECORDED EARTHQUAKES
WITH EPICENTERS IN CONNECTICUT
FACILITY DESCRIPTION AND SAFETY ANALYSIS
CONNECTICUT YANKEE ATOMIC POWER COMPANY

Date	Location	Approximate Epicenter Latitude North	Approximate Epicenter Longitude West	Approximate Area, Square Miles	Intensity, Ressi-Forel Scale
1791 (May 16)	East Haddam	41.5	72.5	35,000	8
1791 (Aug. 28)	East Haddam	41.5	72.5	—	*
1792	East Haddam	41.5	72.5	—	*
1794	East Haddam	41.5	72.5	—	*
1805	East Haddam	41.5	72.5	—	*
1827	New London	41.4	72.7	—	5
1837	Hartford	41.7	72.7	—	5
1840	South Connecticut	41.5	72.9	7,500	6
1858	New Haven	41.3	73.0	1,000	4-5
1875	Connecticut	41.8	73.2	2,000	6
1908	Housatonic Valley	—	—	—	*
1925	Hartford	41.7	72.7	8,000	5
1935	Stamford	41.1	73.5	—	5-6

* Not Available

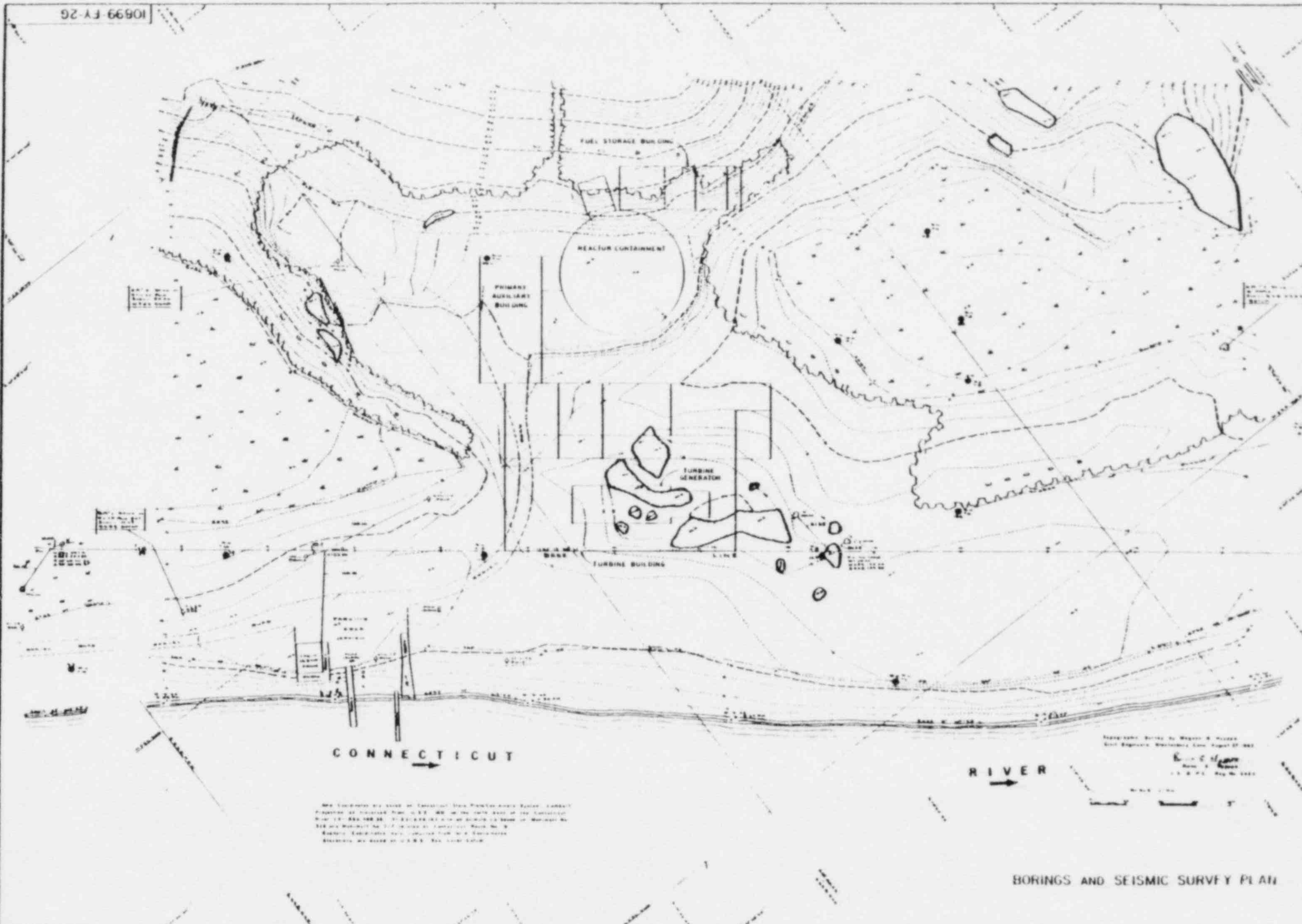

FIGURE 2-1. CONNECTICUT YANKEE SITE

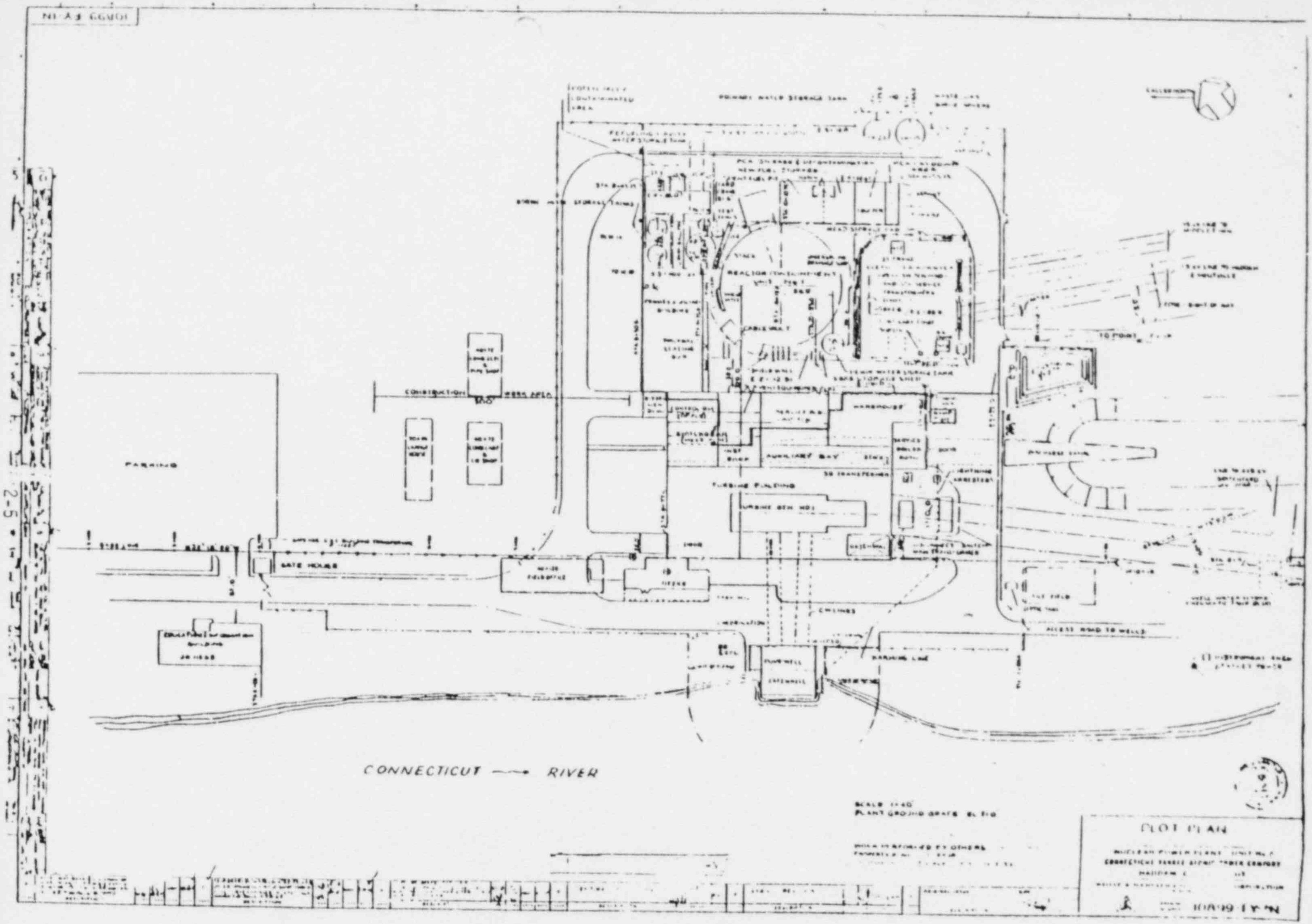

FIGURE 2-2. CONNECTICUT YANKEE PLOT PLAN

3. SEISMIC CRITERIA

On the basis of the U.S. Coast and Geodetic Survey (USCGS) report (Reference 2) it was decided that the maximum ground acceleration for an average earthquake would be 0.03g but that structures and systems that are important to safety be designed for an earthquake with a maximum ground acceleration of 0.17g including a spectrum analysis. The ground response spectra used throughout the analysis were developed by Housner (Reference 3) and are shown in Figure 3-1. Only a single level earthquake was specified with no provision for an OBE and SSE or equivalent. Also, no designation of Seismic Category I Structures or equipment exists for Connecticut Yankee. The ANS Safety Class designation and the basis for this selection is contained in Appendix B.

The following seismic criteria were established for plant design:

1. All structures and elements of the plant were designed to withstand seismic forces corresponding to a ground acceleration of 0.03g in addition to normal loads without damage or loss of function.

2. The main steam piping from the containment up to and including the turbine stop valves was analyzed to demonstrate that stresses resulting from a response spectrum analysis for 0.03g ground acceleration did not exceed the allowable working stress.

3. Components and systems important from the standpoint of nuclear safety were designed so that steady state stresses or stresses resulting from hypothetical accident conditions do not exceed the yield strength of the material when combined with seismic stresses resulting from a response spectrum analysis with a 0.17g ground acceleration. Also, they would not suffer loss or impairment of function because of deflection or distortion.

The following structures and equipment systems were included in the above category:

The reactor containment vessel and its penetrations
The containment air recirculation and filtration system
The containment spray system
The waste gas storage sphere and waste liquid storage tanks
The spent fuel storage pit

4. Two systems were designed so that the stresses resulting from a spectrum analysis with 0.17g ground acceleration were within the allowable working stress range and sufficiently small deflections resulted so that normal operation was not prevented.

These systems were:

The reactor coolant system (piping, reactor vessel and reactor internals)

The safety injection and core deluge systems

In applying the response spectrum to the design of systems or components, exclusive of the reactor internals and control rod drive system, an approximate design was established and the natural period determined. Using appropriate damping factors and this natural period, the average acceleration response was then determined, using Figure 3-1, and the design reviewed to establish whether the stresses and deflections under this acceleration were within acceptable limits. This step was repeated as required until results were satisfactory. For minor systems or for particularly complex systems having a number of degrees of freedom, use was made of the maximum response value, that is, the peak value of the appropriately damped acceleration curve.

The reactor internals were analyzed as part of a complex structure of the interconnection of the masses and stiffnesses of the components involved. The control rod drive system was analyzed separately on the basis that its dynamic behavior does not significantly affect the dynamic response of the reactor structure. A tabulation of damping factors for systems, structures, and components of the plant is given in Table 3-1. A description of the load resisting systems and manner in which the seismic criteria were applied to selected components as abstracted from Reference 1 is contained in Appendix A.

Separate calculations were made for the reactor containment and other structures designed for earthquakes for vertical ground accelerations equal to 2/3 the horizontal ground accelerations and assumed to act nonconcurrently. It was concluded that vertical accelerations would not control the design requirements. No analysis of the pipe stresses resulting from the vertical component was made and no discussion of the effects of vertical excitation on other equipment was found. No time history analysis was conducted and no in-structure response spectra were generated.

TABLE 3-1

EARTHQUAKE DAMPING FACTORS
FACILITY DESCRIPTION AND SAFETY ANALYSIS
CONNECTICUT YANKEE ATOMIC POWER COMPANY

Component or Structure	Per Cent of Critical Damping
Reinforced concrete reactor containment, including foundation mat	7.0
Reinforced concrete framed structures	5.0
Steel framed structures, including supporting structures and foundations	
Bolted	2.5
Welded	1.0
Piping systems	
Carbon steel	0.5
Stainless steel	1.0
Reactor internals and control rod drive	
Welded assemblies	1.0
Bolted assemblies	2.0
Mechanical equipment including pumps and fans	2.0

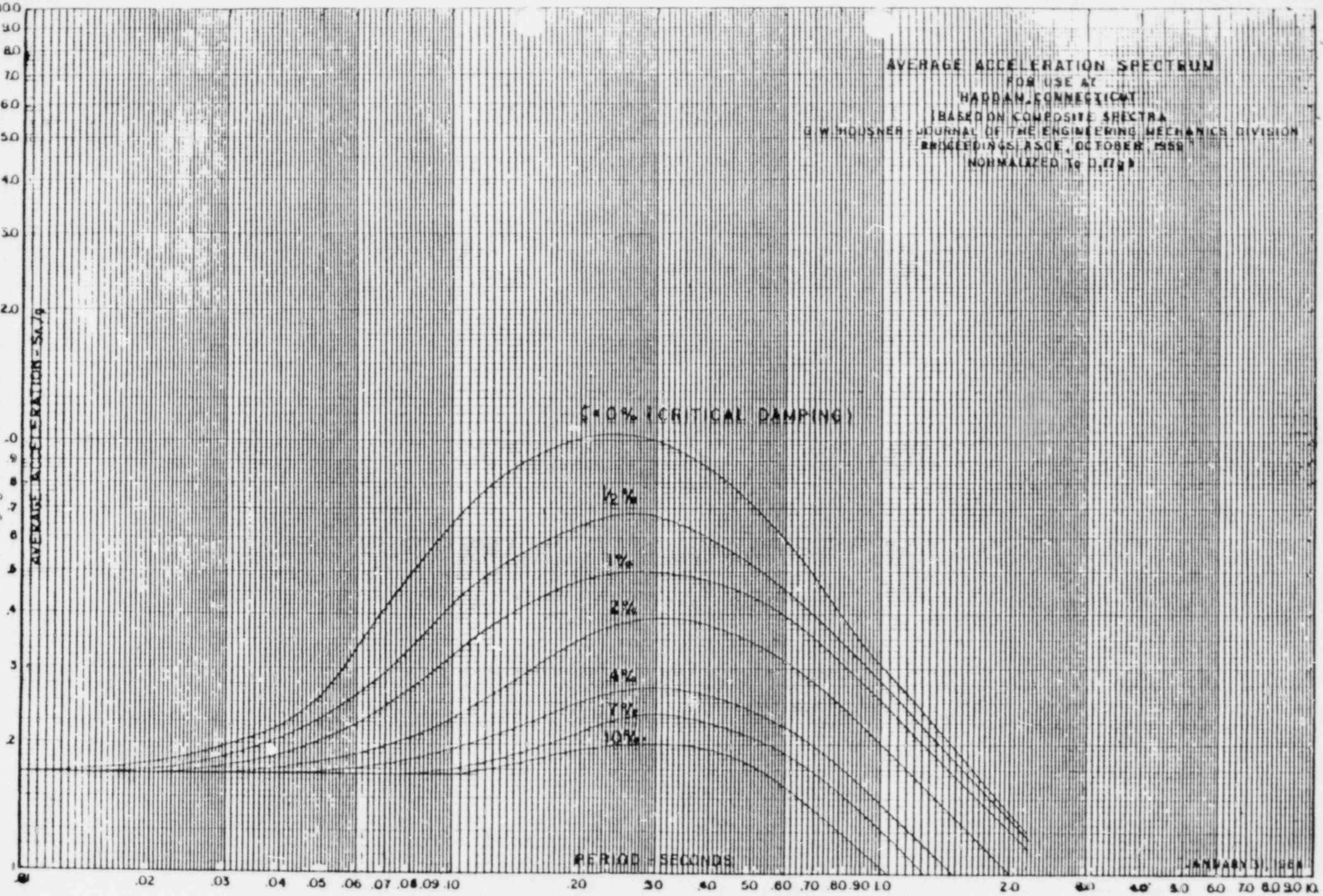

FIGURE 3-1. CONNECTICUT YANKEE GROUND RESPONSE SPECTRA

4. SEISMIC ANALYSIS AND RESULTS

The procedure for calculating the reactor containment earthquake response was based on the Rayleigh method (Reference 4). The containment structure was designed for the equivalent static loads resulting from the above dynamic analysis. For design purposes, only the horizontal accelerations were assumed to be acting.

Values calculated from the Rayleigh analysis used in the containment design are as follows:

Frequency = 30.4 radians per second (4.84 Hz)
Period = 0.21 seconds
Conversion coefficient = 1.64
Acceleration at crown = 0.37g for 7 percent damping, varying linearly to zero at the center of the base
Maximum deflection at crown = 0.0126 ft
Maximum shear = 7,850 kips
Maximum moment = 940,000 ft kips

The maximum shear and moment act at the base of the containment structure. For the calculations, the center line of the mat thickness was used.

Reinforcing steel used in the reactor containment structure conforms to ASTM A408 with a minimum guaranteed yield strength of 50,000 psi. The following table lists the limits on primary and primary-plus-secondary stresses for the reinforcing steel.

Load Conditions	Primary Stress, Psi	Primary % of Yield	Primary-Plus-Secondary Stress, Psi	Primary-Plus-Secondary % of Yield
Operating plus incident	26,700	53-1/3	33,300	66-2/3
Operating plus 0.03g horizontal earthquake	25,000	50	26,700	53-1/3
Operating plus incident plus 0.03g horizontal earthquake	33,300	66-2/3	33,300	66-2/3
Operating plus incident plus 0.17g horizontal earthquake	40,000	80	40,000	80

*Secondary stresses were considered as follows:

Stresses resulting from normal operating temperature gradient inside of containment to outside atmosphere

Stresses resulting from the incident temperature effect

The 3/8 inch steel liner conforms to ASTM A442 with a minimum guaranteed yield strength of 32,000 psi. Under the combination of incident plus earthquake loadings, the internal pressure plus temperature effect causes tensile stresses in the concrete wall. For design purposes, it was assumed that the wall has little or no capacity to resist the tangential shearing forces resulting from the horizontal earthquake loads. The liner alone was designed to resist this tangential shear.

Insulation was added to the lower 17 feet of liner so that the combination of compressive stresses resulting from incident temperature and tangential shearing stresses resulting from horizontal earthquake loading will result in liner stresses not exceeding the yield strength of the material. Table 4-1 contains the specifications for materials used in the containment building.

Table 4-2 lists the design codes used in analysis of piping and mechanical equipment and Table 4-3 contains the type of material and material specifications used for the same components.

TABLE 4-1

REACTOR CONTAINMENT - MATERIALS OF CONSTRUCTION
FACILITY DESCRIPTION AND SAFETY ANALYSIS
CONNECTICUT YANKEE ATOMIC POWER COMPANY

Item	Specification
Liner	
Shell, Bottom, and Dome Plates	ASTM-A442, Gr. 60
Piping Penetration Sleeves	ASTM-A333, Gr. 0
Piping Penetration Reinforcing Rings	ASTM-A442, Gr. 60
Piping Penetration Sleeve Reinforcing Bar Anchoring Rings and Plates	ASTM-A442, Gr. 60
Rolled Shapes	ASTM-A131, Gr. C
Reinforcing Bar Bridging Rings	ASTM-A204, Gr. C, Fbx. normalized
Reinforcing Bar Anchoring Ring and Plates	ASTM-A201, Gr. B, Fbx-A300
Equipment Hatch Insert	ASTM-A201, Gr. B, Fbx-A300
Equipment Hatch Flanges	ASTM-A201, Gr. A, Fbx-A300
Equipment Hatch Head	ASTM-A201, Gr. B, Fbx-A300
Personnel Hatch	ASTM-A201, Gr. B, Fbx-A300
Welding Electrodes	
Carbon Steel to Carbon Steel	ASTM-E7018
Stainless Steel to Stainless Steel	ASTM-E308
Carbon Steel to Stainless Steel	ASTM-E310
Concrete Shell and Interior Structure	
Reinforcing Steel	A408
Cement	ASTM-C150, Type II low alkali
Concrete	Stone & Webster Specification CYS-384 (Mixing and Delivering Concrete) and CYS-614 (Placing Concrete and Reinforced Concrete)
Structural Steel	A36

TABLE 4-2

SYSTEM COMPONENTS

DESIGN CODES

Component	Design Code
Steam Generators	ASME Code Section VIII (1956 ed.)
Reactor Coolant Pumps	ASME Code Section VIII (1956 ed.)
Reactor Coolant Piping	ASA B31.1 (1955 ed.)
Pressurizer	ASME Code Section VIII (1956 ed.) and Code Cases Nos. 1224 and 1234
Safety and Relief Valves	ASME Code Section I (1956 ed.) and Code Cases Nos. 1224 and 1234
Loop Stop Valves	ASA B16.5 (1957 ed.)
Loop Check Valves	ASA B16.5 (1957 ed.)
Pressure Control and Relief System Piping	ASA B31.1 (1955 ed.)
Low Pressure Surge Tank	ASME Code Section VIII (1956 ed.)

TABLE 4-3

SYSTEM COMPONENTS

MATERIAL SPECIFICATION

Component	Material of Construction
Steam Generators	
Coolant Channel Head	Forged Carbon Steel, Clad with Type 304 Stainless Steel
Shell	Carbon Steel
Tubes	Type 304 Stainless Steel
Reactor Coolant Pumps	Type 304 Stainless Steel
Reactor Coolant Piping Fittings	Forged Type 304 (ASTM-A-55T) and Cast Type CF 8 (ASTM-S-351-57T) Stainless Steel
Loop Isolation Valves	Type 304 Cast Stainless Steel
Loop Check Valves	Type 304 Cast Stainless Steel
Pressurizer	Carbon Steel, clad with Type 304 Stainless Steel
Pressurizer Surge Line Piping	Type 316 Stainless Steel
Safety and Relief Valves	Type 304 Stainless Steel
Low Pressure Surge Tank	Type 304 Stainless Steel

5. SUMMARY

The Connecticut Yankee Nuclear Power Plant was one of the earlier facilities for which any dynamic analysis of the structures and equipment was conducted. Consequently, the criteria employed differ in many instances from those currently in use. The seismic design criteria used for Connecticut Yankee are summarized in Table 5-1. Separate criteria were not developed for structures as opposed to equipment or piping. However, differences exist in the load combinations and stress allowables for different systems.

When compared with current state-of-the-art analytical methods and acceptance criteria, numerous examples of both conservative and unconservative assumptions and procedures exist. For instance, stresses of important systems were held below yield or working stress levels for the 0.17g earthquake. This would more closely correspond to current practice for the OBE rather than the maximum level earthquake where yielding and other nonlinear response would be expected. Also, with the exception of the reinforced concrete containment building, damping values which were used were considerably lower than would be considered acceptable today. The Housner spectra used, however, are significantly less conservative than Reg. Guide 1.60 spectra. Even if median centered rock spectra (Reference 5) are compared in the frequency range of interest, somewhat higher levels of response would be expected as compared with those resulting from an analysis based on Housner spectra. In addition, more sophisticated methods of analysis would be required today including consideration of higher mode response and multidirection input effects. Potentially one of the greatest non-conservative assumptions used in the analysis of equipment at Connecticut Yankee was the use of the ground response spectra rather than in-structure response spectra, irrespective of elevation. Finally, no testing or qualification of electrical equipment was done.

TABLE 5-1

NOTES:

(a) All systems designed to withstand horizontal ground acceleration of 0.03g; all systems necessary for safety designed for 0.17g ground motion.

(b) Structures and systems were checked to show that vertical ground motion did not dictate design (loads were not added)

(c) Spectra applied at the centerline of the basemat thickness. Note, however, the statement that the response acceleration varies linearly to zero at the base.

TABLE 5-1

CONNECTICUT YANKEE SEISMIC DESIGN INFORMATION

ITEM	CONNECTICUT YANKEE	CURRENT LICENSING CRITERIA	
1. Type of Plant	PWR	—	
2. Plant Capacity (MWe)	575	—	
3. Architect/Engineer	Stone & Webster	—	
4. Foundation	Bedrock	—	
5. Systems Important for Plant Safety (Equiv. Seismic Category I)	Reactor Containment Containment Spray System Containment Air Recirc. & Filter System Waste Gas Storage Sphere Waste Liquid Storage Tanks Spent Fuel Storage Pit Reactor Coolant System Safety Injection System	Systems necessary to: 1) Maintain Coolant System Pressure Boundary, 2) Shutdown Reactor & Maintain Safe Condition, 3) Prevent or Mitigate Offsite Exposure. Ref. USNRC Reg. Guide 1.29, and S.R.P. 3.2.1	
6. OBE (or Design E)	Not Used	Ref. 10 CFR 100, Appendix A	
7. SSE (or Max. E)	0.03 and 0.17g H[(a)]* 2/3 V[(b)]	Ref. 10 CFR 100, Appendix A SRP 3.7.1	
8. Response Spectra	Housner[(c)]	USNRC Reg. Guide 1.60 or Site Dependent Spectra, S.R.P. 3.7.1	
9. Type of Analysis	Containment Bldg: Rayleigh Equipment: Single D.O.F.	Finite Element or Lumped Mass	
10. Predominant Frequencies	Containment Bldg: 4.84 Hz	—	
11. Material Damping	Containment Bldg. 7%	OBE 4%	SSE 7%
	Rein. Conc. Frame Struct. 5%	OBE 4%	SSE 7%
	Bolted Steel Frame Struct. 2.5%	OBE 4%	SSE 7%
	Welded Steel Frame Struct. 1.0%	OBE 2%	SSE 4%

* see notes

TABLE 5-1 (continued)

ITEM	CONNECTICUT YANKEE	CURRENT LICENSING CRITERIA	
	Piping (Carbon Steel) 0.5%	OBE 1 to 2%	SSE 2 to 3%
	Piping (Stainless Steel) 1.0%	OBE 1 to 2%	SSE 2 to 3%
	Reactor Internals & CRD (Welded) 1.0%	OBE 2%	SSE 4%
	Reactor Internals & CRD (Bolted) 2.0%	OBE 4%	SSE 7%
	Mechanical Equip. 2.0%	OBE 2 to 4%	SSE 4 to 7%
		Ref: USNRC Reg. Guide 1.61, S.R.P. 3.7.1	
12. Modal Combinations	1 DOF or equivalent only	SRSS or Modification, USNRC Reg. Guide 1.92, S.R.P. 3.7.2	
13. Directional Combinations	One Horizontal and Vertical Nonconcurrently(b)	3-Direct. Concurrently (SRSS) Ref. USNRC Reg. Guide 1.92, S.R.P. 3.7.2	
14. Time History Analysis	None	S.R.P. 3.7.1	
15. Floor Response Spectra	Ground Spectra used throughout Structure	Ref. USNRC Reg. Guide 1.122, S.R.P. 3.7.2	
16. Testing of Equipment	None	Ref. IEEE 344	
17. Design Load Combinations	Reactor Coolant and Safety Injection Systems: (E=0.17g earthquake): Oper. Loads + E < Working Stress	ASME B&PV Code Sect. III Div. 2	
	Other Safety Systems (E=0.17g earthquake): Oper. or Accident + E < yield stress	USNRC Reg. Guides 1.10, 1.15, 1.18, 1.19, 1.48, 1.55, S.R.P. 3.8.1, 3.8.3, 3.8.4, 3.8.5	
	Main Steam Piping (E=0.03g earthquake: Oper. Loads + E < Working Stress		
	All Other Structures & Elements (E=0.03g earthquake): no loss of Function		
18. Simplified Design Methods	1 DOF Systems Directly from Ground Response Spectrum	Floor Spectra Req'd. S.R.P. 3.7.2	
	Complex Systems from Peak of Ground Response Spectrum	Peak of Floor Spectrum S.R.P. 3.7.2, 3.7.3	

REFERENCES

1. Docket 50213 Connecticut Yankee Atomic Power Plant Preliminary Safety Analysis Report, May, 1966.

2. U.S. Department of Commerce, Coast and Geodetic Survey "Report on Seismicity of the Haddam, Connecticut Area", January 9, 1964.

3. Housner, G. W., "Behavior of Structures during Earthquakes", Journal of the Engineering Mechanics Div. Proceedings of ASCE, October, 1959.

4. Blume, J. A., Newmark, N. M., Corning, L. H., Design of Multistory Reinforced Concrete Buildings for Earthquake Motions, Portland Cement Associations, Chicago, Illinois, 1961.

5. Nathan M. Newmark Consulting Engineering Services, "A Study of Vertical and Horizontal Earthquake Spectra", U.S. AEC Report WASH-1255, April, 1973

6. Housner, G. W. "Dynamic Pressures on Accelerated Fluid Containers", Bulletin of the Seismological Society of America, 47(1), January, 1957.

APPENDIX A

SEISMIC DESIGN OF SELECTED PLANT COMPONENTS

(From Reference 1)

A. The reactor containment is constructed of reinforced concrete with a 9 foot thick mat, 4-1/2 foot thick walls, and a 2-1/2 foot thick dome. The containment is designed so that the combined forces from operating conditions, incident, and earthquake, using factors derived from Figure 3-1, are within the yield point of the materials used (See Section 4).

B. The reactor containment is penetrated by pipe, either directly or in sleeves, by cartridges in sleeves containing the electrical conductors and by access doors for personnel and equipment. All penetrations are of steel, generally of welded but occasionally of flanged and bolted construction. Forged or wrought material is employed, with no castings. Each penetration is anchored in the containment concrete walls, generally be welded connnections to the reinforcing bars, and is temporarily supported in the pouring forms so that the concrete is poured around it, forming a solid, reinforced, monolithic, and anchored block. All penetrations thus become an integral part of the containment and move with it in response to any ground acceleration. Pipe penetrations are designed so that the combination of operating, incident, and seismic forces does not exceed the yield point of the materials used.

C. For reactor coolant system components, the natural frequency of the components was determined. In general, this computation was made by representing the components with their supports as a single-mass system. A dynamic analysis of the component and its support was made using the response curves (Figure 3-1) to determine the internal stress levels and deflections with their resulting effect on other system components.

The reactor vessel is supported in the neutron shield tank and keyed to it in such a way that relative displacement under horizontally applied forces cannot take place. The neutron shield tank is of heavy wall, welded steel plate construction, filled with water, and anchored to the reactor containment bottom mat. The shield tank, when supporting the reactor vessel, is designed so that the steady state forces, when combined with seismic forces as derived from Figure 3-1, do not exceed the yield point of the materials.

A system of snubbers is employed on the neutron shield tank consisting of fluid pistons placed between the tank and the surrounding reinforced concrete shield wall. The snubbers are sized to dampen vibrtions in the system so that they range from 3 (hot) to 5 (cold) percent of the critical, with a coresponding reduction in the seismic forces on the shield tank, the reactor vessel supported upon it, and the reactor coolant system components attached.

Each steam generator is supported on a rigid steel structure in the form of a cylindrical skirt. This supporting structure is anchored to the reactor containment mat by means of a circular bridging ring which is embedded in the mat and welded to the reinforcing bars. The bridging ring, at the same time, affords a continuous welded seal where it penetrates the liner plate. Each steam generator support is designed to withstand safely the combination of normal operating, incident, and seismic forces without exceeding the yield point of the material or, in the event of a reactor coolant pipeline failure, permitting the steam generator main steam outlet pipe to fail, which would contribute to the extent of the incident.

This support incorporates two systems of keyed and sliding blocks in its design, thus permitting the steam generator to expand radially under operating temperature while the whole reactor coolant system, including piping, expands radially from the center of the reactor.

The pressurizer is provided with a base skirt from which the vessel is suspended from above by tension rods. A system of steel braces allows the pressurizer to expand vertically under operating temperature changes, but prevents the vessel from moving laterally under seismic forces.

The pumps are designed to withstand the acceleration value obtained from Figure 3-1, corresponding to their natural period of vibration at appropriate damping.

Piping in both systems is rigidly anchored when not subject to temperature stress. Where temperature is involved, suitable expansion loops are provided. Large masses in the piping, such as valves, are placed near anchor points. Freely supported pipe runs, including those with provision for expansion, and the connections to the reactor coolant lines have been checked under seismic forces, using the peak of the spectrum curve, Figure 3-1, with appropriate damping. In some areas, analysis indicated additional restraint to motion under earthquake forces was required, and this was provided by installing loose fitting collars around the pipe to limit motion from an earthquake, but not impose steady state loads.

Electric switchgear is solidly bolted to concrete mats or piers so that it cannot overturn in an earthquake. Conduits are rigidly anchored to concrete structures.

Sensory instruments for this system are generally of the pressure-response type employing Bourdon spring tubes or hydraulic bellows, mounted to minimize earthquake shock. Connections are stainless steel or copper tubing.

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Sensory instruments for this system are generally of the pressure-response type employing Bourdon spring tubes or hydraulic bellows, mounted to minimize earthquake shock. Connections are stainless steel or copper tubing.

E. Control rod drive pressure housings are designed as welded structures composed of stainless steel. In designing the internal components of the mechanism, consideration is given to the appropriate damping characteristics of each component; for example 2 percent of critical damping is assumed for bolted components.

The response of the mechanism housing to the earthquake is determined by representing the housing and the reactor vessel as an equivalent system of springs and masses, supported rigidly by the concrete foundation. The natural period was calculated and the peak of the response curve was used to evaluate the stresses and deflections.

F. The reactor containment air filtration and cooling system is in four groups, each with a motor driven fan, moisture separator, cooling coils, and particulate and charcoal filters. Each of these components is separately designed to withstand seismic forces. In each case, the force is determined from the peak value of the appropriate damping curve from Figure 3-1. This analysis extends not only to the components themselves but to the frames and enclosures to which they are attached. All of the components are rigidly bolted to a thick reinforced concrete floor, so that they cannot overturn in an earthquake. This type of equipment consists principally of fabricated and bolted or welded structural steel members and steel plate, flexible pipe coils, and nonmetallic filter media.

The air recirculation system in the fan discharge consists principally of rectangular and circular ducts built of steel plate, reinforced with structural steel members. This is steel plate construction, with walls relatively thin as compared to duct dimensions.

G. The containment spray system consists of a sump supply in the mat of the containment, an auxiliary supply from the same tank described in D. above, pumps, heat exchangers, and piping. The sump is formed in the heavy reinforced concrete mat of the reactor containment and is lined with stainless steel plate. It is designed under the same criteria as employed for the safety injection system under D. above. Spray piping and nozzles are rigidly anchored to the wall of the reinforced concrete reactor containment.

H. The waste gas storage sphere provides an expansion space for waste gas while holding the pressure reasonably constant in the various tanks and vessels in the waste disposal plant which are subjected to variation in liquid level. The average gas pressure is less than 1 inch of water. This welded steel vessel is supported on reinforced concrete piers, and vessel and supports have been designed to withstand seismic forces as determined from Figure 3-1. The vessel welds are spot radiographed and the entire vessel is subjected to a Freon leakage test. Reference 1 indicates that the safety of the public is not endangered, should a failure develop in this vessel for the waste gas accident.

The same design criteria for the waste gas sphere apply to tanks for the storage of waste liquids.

I. The spent fuel storage pit is of reinforced concrete with walls 6 feet thick, lined with welded stainless steel plate and filled with borated water. The pit is founded on bedrock and the lower side walls are embedded in rock and earth. Thus, the structure may be considered to follow the actual ground acceleration without relative displacement. The combined normal stresses and the seismic stress, as derived from Figure 3-1, do not exceed the yield point of the material.

The spent fuel pit is surmounted by a steel frame structure which is rigidly fastened to the top face of the pit walls. This structure is designed so that it does not fail under seismic stresses as derived from Figure 3-1.

Spent fuel assemblies are stored in racks built of fabricated stainless steel. The racks are designed to withstand seismic forces as derived from Figure 3-1, and are so placed in the spent fuel pit that they cannot overturn in an earthquake.

J. Reactor Internals

The core support structure is supported at its upper flange from a ledge in the reactor vessel flange and at its lower end by six lug and clevis joints connected radially between the barrel and the reactor vessel wall. The ractor vessel in turn is supported from the neutron shield tank which is anchored to the reactor containment mat and damped by snubbers connected from the shield tank to the concrete shield wall. Combined seismic and other steady state stresses are within the allowable working stress range of the material for the parts under consideration. Furthermore, the deflections are sufficiently small to permit normal operation and would not impair reactor shutdown.

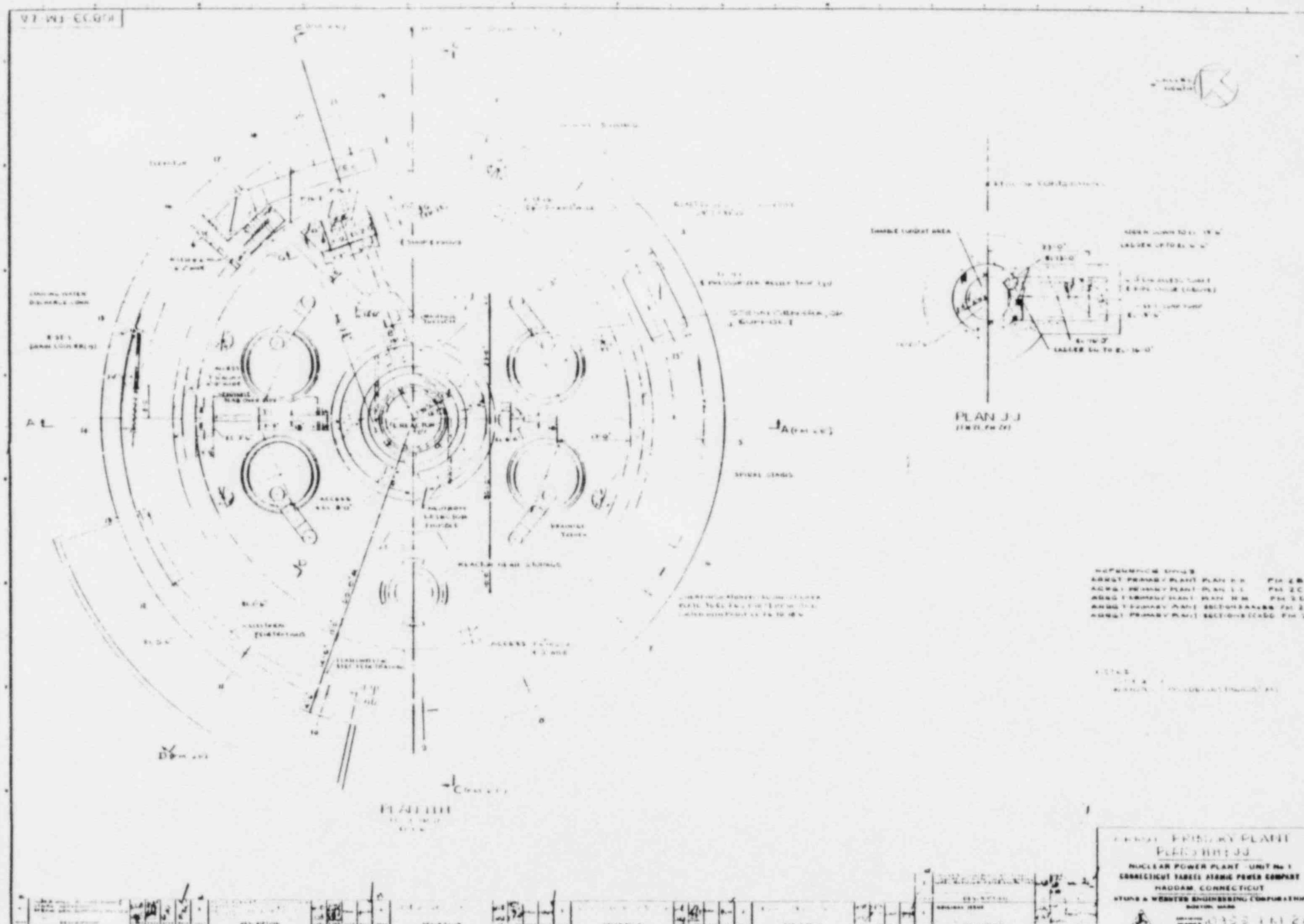

FIGURE A-1. REACTOR CONTAINMENT BUILDING PLAN, EL. 1'-6" (BASE SLAB)

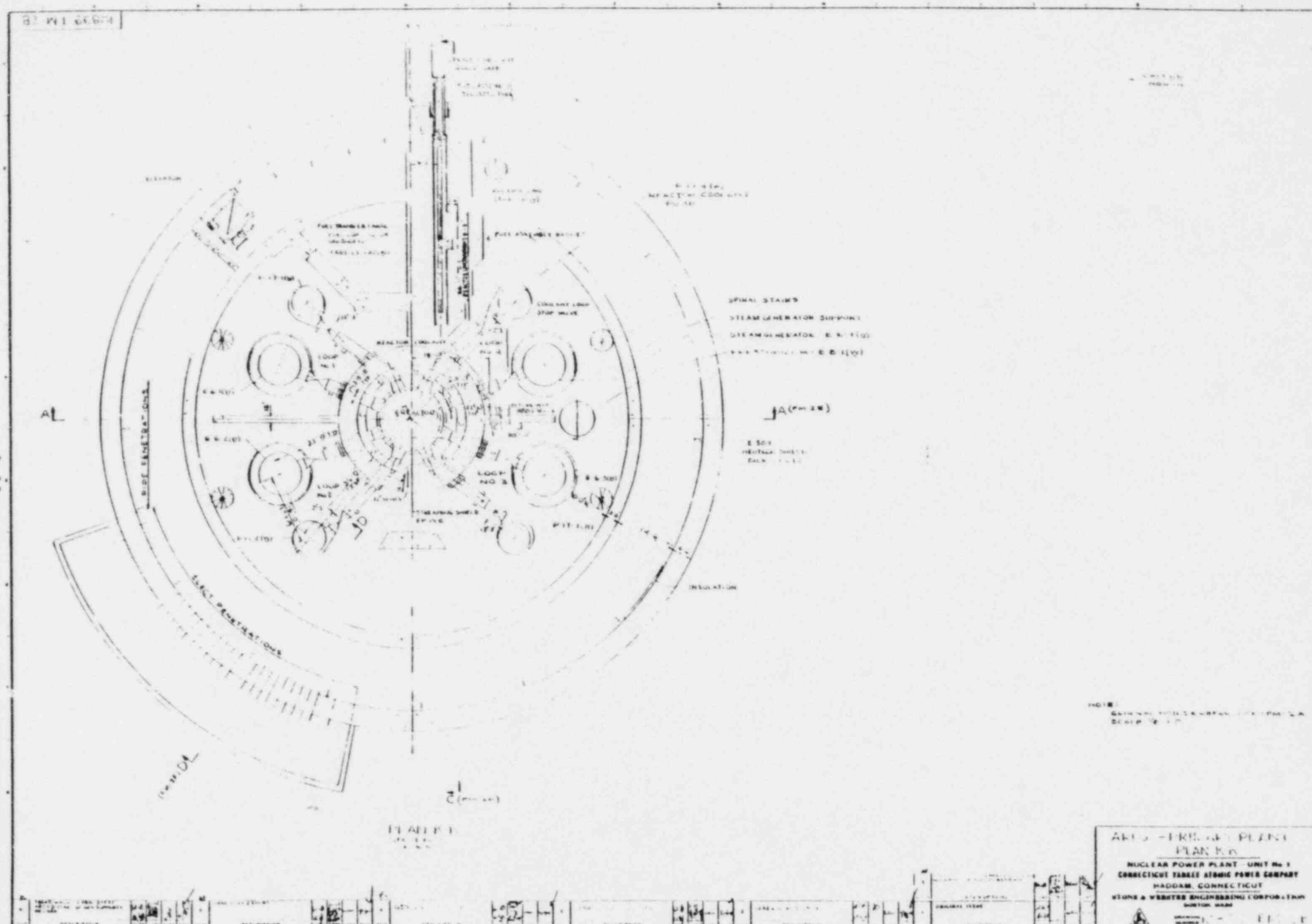

FIGURE A-2. REACTOR CONTAINMENT BUILDING PLAN, EL 16'-0"

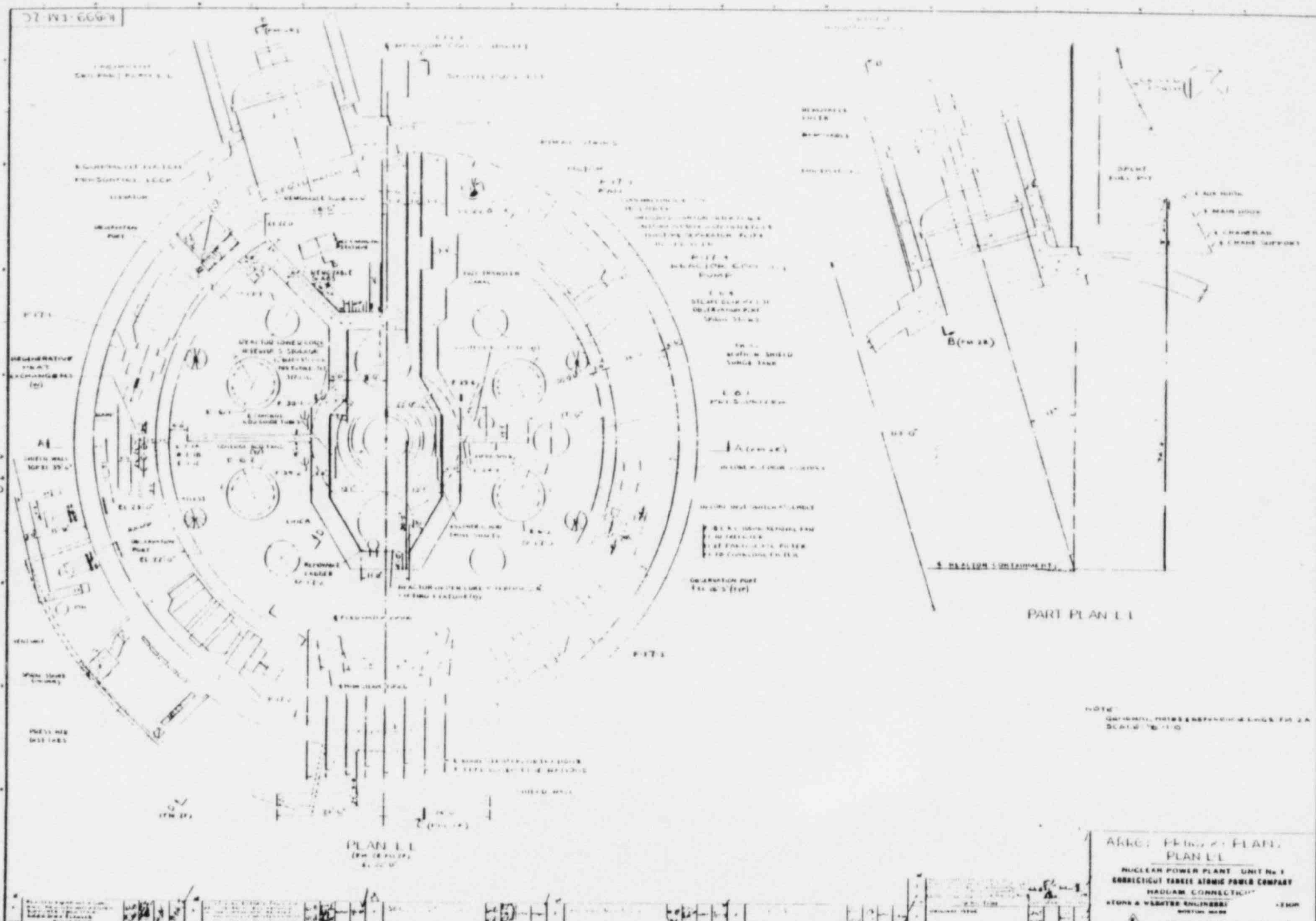

FIGURE A-3. REACTOR CONTAINMENT BUILDING PLAN, EL. 22'-0"

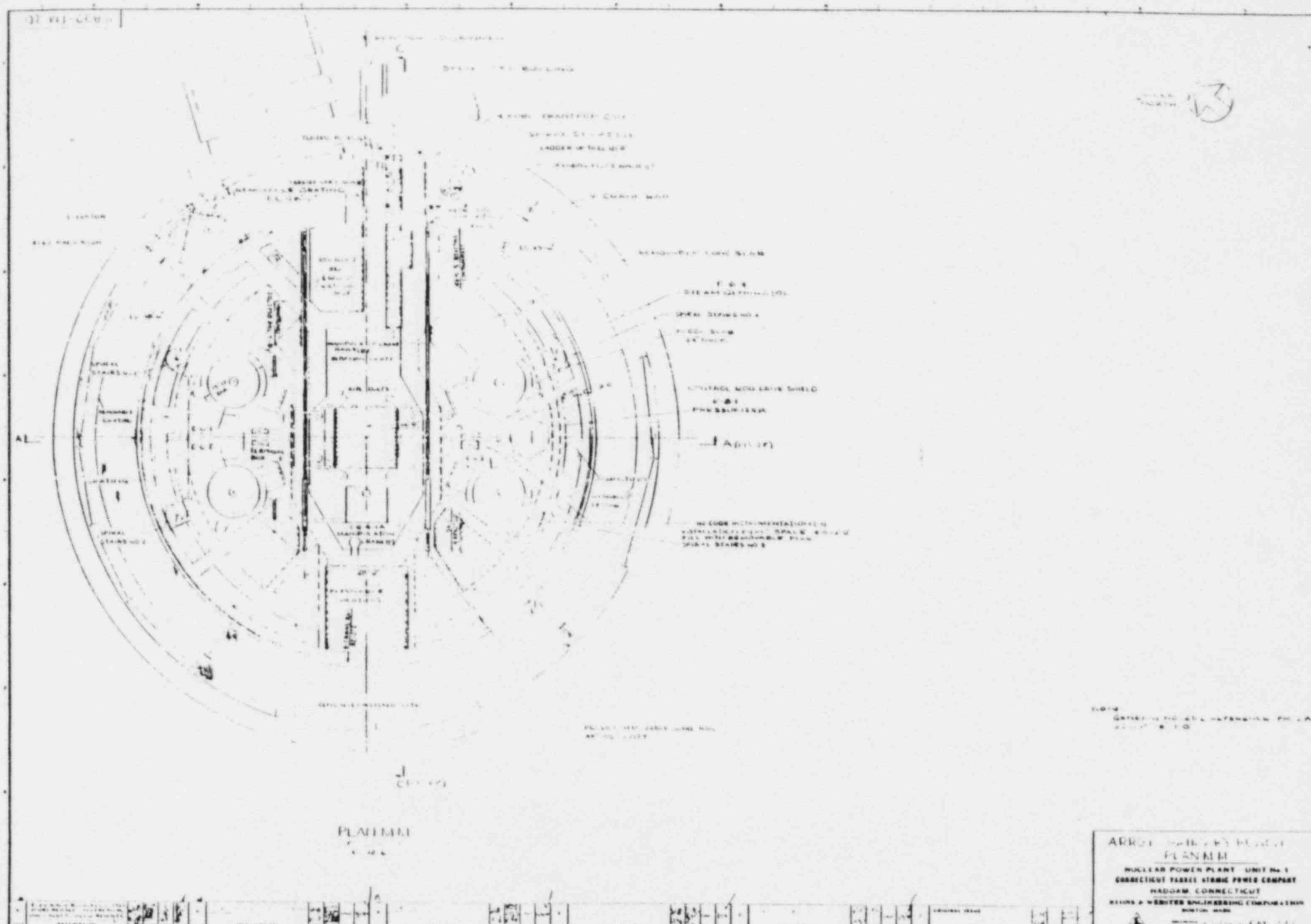

FIGURE A-4. REACTOR CONTAINMENT BUILDING, EL. 48'-6"

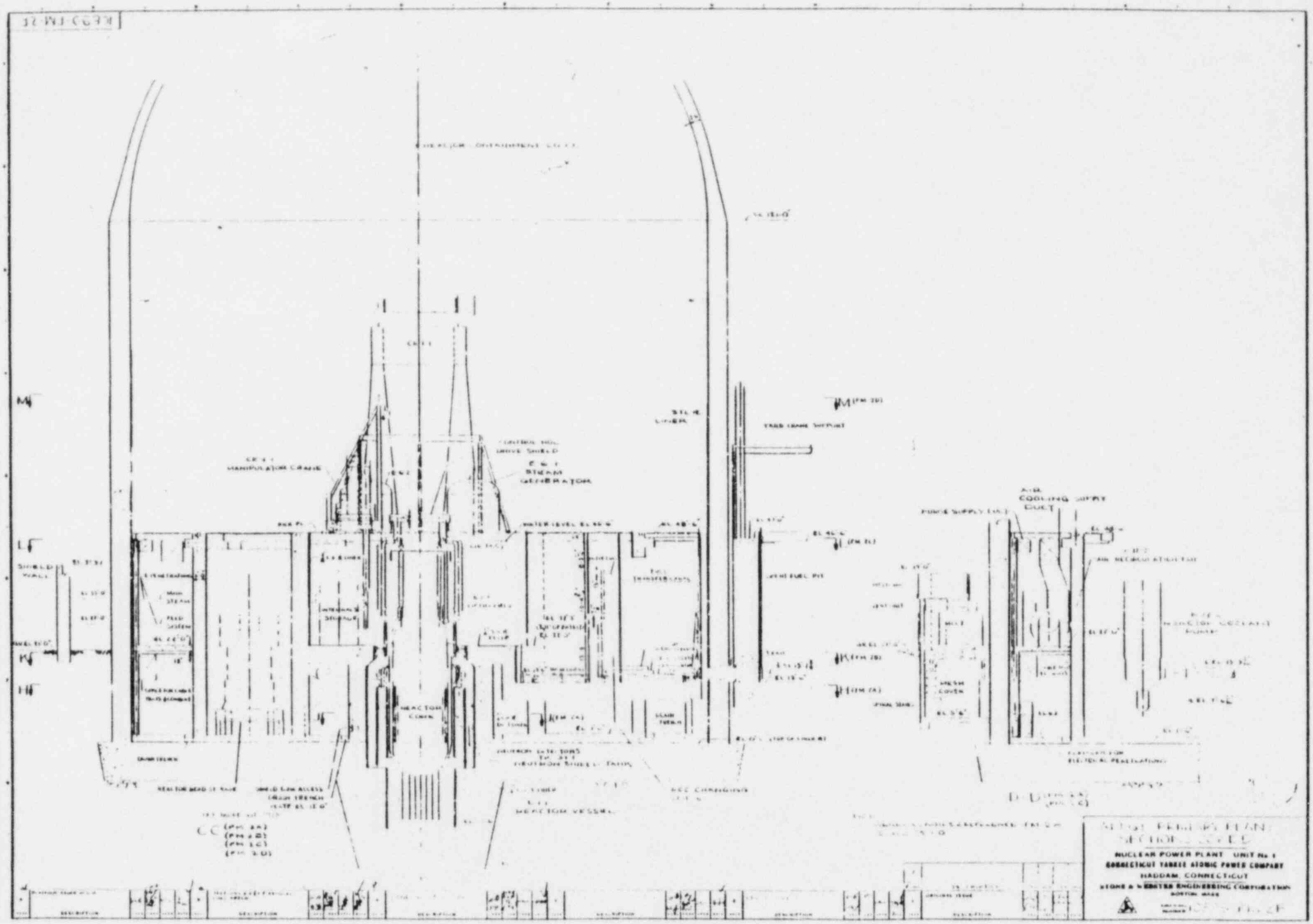

FIGURE A-5. REACTOR CONTAINMENT BUILDING N-S SECTION

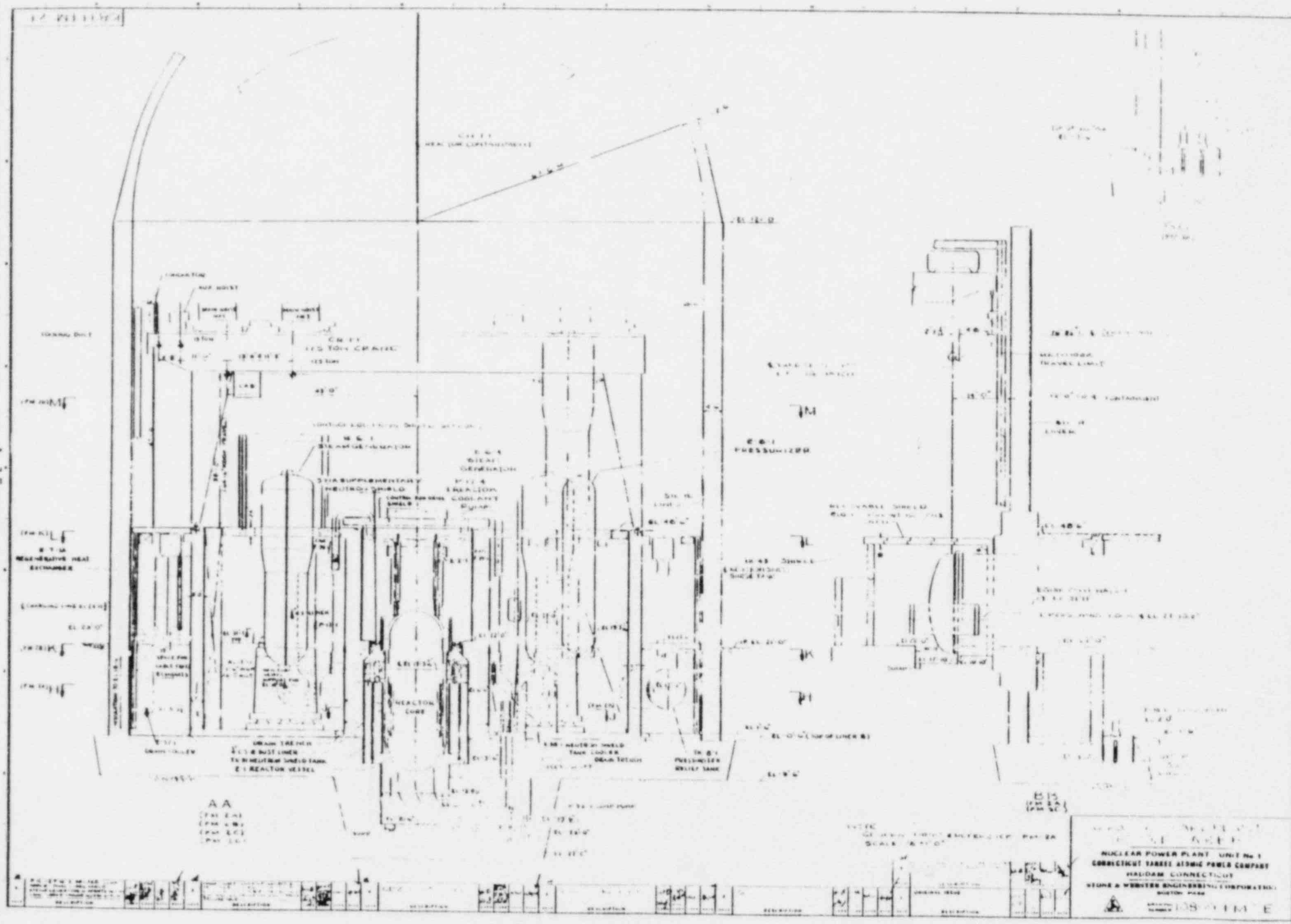

FIGURE A-6. REACTOR CONTAINMENT BUILDING E-W SECTION

APPENDIX B
(From Reference 1)

DESIGN CRITERIA - STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS
Classification of Structures, Components and Systems

B1. Seismic Classifications

The entire plant has been designed using sound engineering practice. The inherent structural characteristics provided by proper deign will enable all plant structures, components and equipment to safely withstand the ground accelerations associated with earthquake intensities expected at the plant site.

B2. System Quality Group Classification

B2.1 Safety Class Definitions

Systems and components are classified as Safety Class 1, Safety Class 2 or Safety Class 3 and non-nuclear safety (NNS) in accordance with their importance to nuclear safety. This importance, as established by class designation, shall be considered in the future in rgard to the design; material aspects, manufacture or fabrication; assembly, erection, and construction; and operation. A single system may have components in more than one class.

Safety Class 1

Safety Class 1, SC-1, applies to reactor coolant system components where failure during normal operations would prevent orderly reactor shutdown and cooldown assuming makeup is provided by normal makeup systems only.

Safety Class 2

Safety Class 2, SC-2, applies to reactor containment and to those components:

1. Of the reactor coolant system not in Safety Class 1,

2. That are necessary to:
   (a) Directly remove residual heat from the reactor,
   (b) Circulate reactor coolant for any safety system purpose*
   (c) Control, within the reactor containment, released radioactivity
   (d) Control hydrogen in the reactor containment, or

3. Of safety systems located inside the reactor containment.

Safety Class 3

Safety Class 3, SC-3, applies to those components not in Safety Class 1 or Safety Class 2:

1. The failure of which would result in release to the environment of radioactive gases normally required to be held for decay or that are necessary to:

2. Provide or support any safety system function.

3. Control, outside the reactor containment, released airborne radioactivity, or

4. Remove decay heat from spent fuel.

B2.2 System and Equipment Classification List

Table B-1, "System and Equipment Classification List", tabulates components by safety class designation.

- * A safety system is any system that functions to shutdown the reactor, cool the core or another safety system or (after an accident) the reactor containment, or that contains, controls, or reduces radioactivity released in an accident. Only those portions of the secondary system are included (a) that are designed primarily to accomplish one of the above functions, or (b) whose failure could prevent accomplishing one of the above functions.

TABLE B-1

SYSTEM AND EQUIPMENT CLASSIFICATION LIST

COMPONENT	ANS SAFETY CLASS
REACTOR COOLANT SYSTEM	
Reactor Vessel	1
Control Rod Drive Mechanism Housing	1
Steam Generator (Tube Side)	1
Steam Generator (Shell Side incl. Feedwater & Steam Relief)	2
Reactor Coolant Loop Isolation Valves	1
Reactor Coolant Loop Check Valves	1
Pressurizer	1
Reactor Coolant Piping	1
Pressurizer Surge Line	1
Loop Bypass Line	1
Safety Valves	1
Relief Valves	1
Valves to Reactor Coolant System Pressure Boundary	1
Low Pressure Surge Tank	2
Reactor Coolant Pump Casing	1
CHARGING AND VOLUME CONTROL SYSTEM	
Feed and Bleed Heat Exchanger (Tube Side)	1
Feed and Bleed Heat Exchanger (Shell Side)	1
Charging Pumps	2
Letdown Orifices	1
CHEMICAL SHUTDOWN SYSTEM	
Boric Acid Mix Tank	3
Boric Acid Transfer Pump	3
PURIFICATION SYSTEM	
Purification Ion Exchangers	3
Purification Pumps	2

TABLE B-1

SYSTEM AND EQUIPMENT CLASSIFICATION LIST
(continued)

SAFETY INJECTION SYSTEM	
Safety Injection Tank	2
Accumulator	2
High Pressure Safety Injection Pumps	2
Low Pressure Safety Injection Pumps	2
SHUTDOWN COOLING SYSTEM	
Shutdown Cooling Pumps	2
Shutdown Cooler (Tube Side)	2
Shutdown Cooler (Shell Side)	3
WASTE PROCESSING SYSTEM	
Primary Drain Collecting Tank	3
Vapor Container Drain Tank	3
Waste Holdup Tank	3
Activity Dilution Tank	3
Gas Stripper	3
Stripper Overhead Condenser (Shell Side)	3
Stripper Overhead Condenser (Tube Side)	NNS
Waste Gas Compressor	3
Waste Gas Compressor Suction Cooler	3
Waste Gas Compressor Discharge Cooler	3
Waste Gas Decay Drums	3

NOTES:

1. Classification of piping and valves between components shall be governed by the component classifications. Consult system flow diagram in applicable FSAR chapter for detailed system safety classification.