October 4, 1982

Docket No. 50-213
LS05-82-10-009

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

Dear Mr. Counsil:

SUBJECT: SEP TOPICS III-6, SEISMIC DESIGN CONSIDERATIONS AND III-11, COMPONENT INTEGRITY - HADDAM NECK PLANT

Enclosed is our draft safety evaluation for the seismic design of the Haddam Neck Plant. The staff's review is based on preliminary analyses and several working-level meetings between CYAPCo and NRC personnel and their consultants. CYAPCo has not yet completed the seismic reevaluation of Haddam Neck. Therefore, the conclusions presented in the evaluation may be revised should new information be presented in the final CYAPCo seismic Safety Analysis Report.

Based on the staff's review of Haddam Neck structures, major piping systems, tanks and equipment, a number of items are open due to a lack of information. These items are identified in the enclosed draft Safety Evaluation Report, Section V, Conclusions, and should be addressed in your final seismic SAR.

SE04

DSU USE(02)

This evaluation will be a basic input to the integrated safety assessment for your facility unless you identify changes needed to reflect the as-built conditions at your facility. This topic assessment may be revised in the future if your facility design is changed or if NRC criteria relating to this topic are modified before the integrated assessment is completed.

ADD: G. Staley

Sincerely,

Original signed by:

Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

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OFFICE	SEPB*	SEPB *	SEPB *	SEPB	ORB#5	ORB#5	AD:SA:DL
SURNAME	TCheng:bl	SBrown	RHermann	WRussell	JLyons	DCrutchfield	FMiraglia
DATE	9/30/82	9/29/82	9/30/82	10/1/82	10/ /82	10/4/82	10/ /82

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NRC FORM 318 (10-80) NRCM 0240

USGPO: 1981—335-960

(E) Information should be provided demonstrating the design adequacy of the auxiliary feedwater pump house.

This evaluation will be a basic input to the integrated safety assessment for your facility unless you identify changes needed to reflect the as-built conditions at your facility. With respect to the potential modifications outlined in the conclusion of this report, a determination of the need to actually implement these changes will be made during the same integrated assessment. This topic assessment may be revised in the future if your facility design is changed or if NRC criteria relating to this topic are modified before the integrated assessment is completed.

Sincerely,

Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

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Haddam Neck
Docket No. 50-213
Revised 3/30/82

Mr. W. G. Counsil

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SYSTEMATIC EVALUATION PROGRAM

TOPICS III-6 AND III-11

HADDAM NECK PLANT

TOPICS: III-6, Seismic Design Consideration
III-11, Component Integrity

I. INTRODUCTION

The nuclear power plant facilities under review in SEP received construction permits between 1956 and 1967. Seismic design procedures evolved significantly during and after this period. The Standard Review Plan (SRP), first issued in 1975, along with the Regulations 10 CFR Part 50 Appendix A and 10 CFR Part 100 Appendix A, constitute current licensing criteria for seismic design reviews. As a result, the original seismic design of the SEP facilities vary in degree from the Uniform Building Code up through and approaching current standards. Recognizing this evolution, the staff found that it is necessary to make a reassessment of the seismic safety of these plants.

Under the SEP seismic reevaluation, these eleven plants were categorized into two groups based upon the original seismic design and the availability of seismic design documentation. Different approaches were used used to review the plant facilities in each group. The approaches were:

Group I: Detailed NRC review of existing seismic design documents with limited reevaluation of the existing facility to confirm judgments on the adequacy of the original design with respect to current requirements.

Group II: Licensees were required to reanalyze their facilities and to upgrade, if necessary, the seismic capacity of their facility. The staff will review the licensee's reanalysis methods, scope and results. A limited independent NRC analysis will be performed to confirm the adequacy of the licensee's method and results.

Based on the staff's assessment of the original seismic design; the Haddam Neck plant was placed in Group II for review.

The Haddam Neck plant, a four loop, pressurized water reactor (PWR) of 575 MWe capacity, is located on the Connecticut River in South central Connecticut approximately twelve miles from Long Island Sound. The Nuclear Steam Supply System (NSSS) was supplied by Westinghouse Electric, Inc. and Stone and Webster Engineering Corporation was the architect-engineering and general contractor. The plant received its construction permit in August 1965, its provisional operating license on June 30, 1967 and its full-term operating license on December 27, 1974.

The Haddam Neck plant was one of the earlier facilities for which dynamic analysis of structures, systems and components was conducted. All safety-related structures and systems were designed for a horizontal peak ground acceleration (PGA) of 0.17g and the balance of the plant was designed for a PGA of 0.03g. Housner ground response spectra scaled to the specific PGA's were used as seismic input for the analysis and design. The structures and systems were checked to show that vertical ground motion did not dictate design, but no vertical seismic loads were added in the final design. For the analysis of most safety-related structures, the buildings were modelled as a single degree of freedom lumped mass-spring systems with fixed bases for calculating the natural frequency of each building; then, the corresponding spectral accelerations were used for performing the equivalent static analyses and design. No floor response spectra were generated for the design of piping systems and components, instead the ground response spectra with lower damping were used. A detailed description about the original seismic design of the Haddam Neck plant is found in a draft summary report, "Seismic Design Bases and Criteria for Connecticut Yankee Nuclear Power Station, Haddam, Connecticut," January 1979 (Attachment 1).

The SEP seismic review of the Haddam Neck facility addressed only the Safe Shutdown Earthquake, since it represents the most severe event that must be considered in the plant design. The scope of the review included three major areas: the integrity of the reactor coolant pressure boundary; the integrity of fluid and electrical distribution systems related to safe shutdown; and the structural integrity of mechanical and electrical equipment and engineered safety features systems (including containment). According to NRC 10 CFR 50.54(f) letters dated August 4, 1980 and April 8, 1981 (Ref. 1 and 2), the licensee, Connecticut Yankee Atomic Power Company was required to seismically reevaluate and upgrade, if necessary, all safety-related structures, systems and components to a level of seismic resistance which is acceptable to the staff. Then, the staff will review the licensee's reanalysis criteria, scope, methods, and results to assess the overall capacity of this facility.

II. REVIEW CRITERIA

Since the SEP plants were not designed to current codes, standards and NRC requirements, it was necessary to perform "more realistic" or "best estimate" assessments of the seismic capacity of the facility and to consider the conservatisms associated with original analysis methods and design criteria. A set of review criteria and guidelines was developed for the SEP plants. These review criteria and guidelines are described in the following documents:

A. NUREG/CR-0098, "Development of Criteria for Seismic Review of Selected Nuclear Power Plants," by N.M. Newmark and W.J. Hall, May 1978.

B. "SEP Guidelines for Soil-Structure Interaction Review," by SEP Senior Seismic Review Team, December 8, 1980.

C. Letter from D.M. Crutchfield, NRC to W.G. Counsil, CYAPCo, "Systematic Evaluation Program Position RE: Consideration of Inelastic Response Using NRC NUREG/CR-0098 Ductility Factor Approach," dated June 23, 1982.

D. Letter from D.M. Crutchfield, NRC to W.G. Counsil, CYAPCo,"Topic III-6, Seismic Design Considerations, Staff Guidelines for Seismic Evaluation Criteria for the SEP Group II Plants," dated July 26, 1982 (Ref. 3).

E. (Revision of Criteria #D above - to be issued.)

For the cases that are not covered by the criteria stated above, the following SRPs and Regulatory Guides were used for the review:

A. Standard Review Plan, Sections 2.5, 3.7, 3.8, 3.9, and 3.10.

B. Regulatory Guides 1.26, 1.29, 1.60, 1.92, 1.100, and 1.122.

Any differences from the criteria or guidelines were justified by the licensee on a case-by-case basis.

III. RELATED TOPICS AND INTERFACES

The related SEP topics to the review of seismic design considerations and component integrity are Topics II-4, II-4.A, II-4.B, and II-4.C. These topics relate to specification of seismic hazard at the site, namely the site specific ground response spectra for the Haddam Neck site. The seismic input spectra proposed by the licensee for the seismic reevaluation of the Haddam Neck facility matches closely with the Haddam Neck site specific ground spectra recommended by the staff as shown in Fig. 1; therefore, the results from these four safety topic evaluations will not affect the review of seismic design considerations and component integrity.

IV. EVALUATION

A. General Approach

The seismic reevaluation of the Haddam Neck plant was initiated by conducting a detailed review of the plant seismic documentation. The results of this review are summarized in the draft docket review report (Attachment 1). Based on the findings from this docket review, two NRC 10 CFR 50.54(f) letters (Ref. 1 and 2) were issued to require the licensee to complete a seismic reevaluation program. The program includes: (1) providing a justification to demonstrate that the plant could continue to operate in the interim until the program was completed, (2) proposing a program plan that addressed the scope, criteria and schedule for completion of the program; and (3) after the staff accepted the proposed program plan, performing seismic analysis and providing final results to the staff for review. The staff's review of results would serve as the basis for seismic safety assessment of the plant facility.

Due to the schedule of SEP, the CYAPCo could not complete its seismic analyses before the staff started its review. Therefore, the seismic review and evaluation of the Haddam Neck plant could not follow the procedure originally planned. Instead, the review was performed in parallel with the licensee's reevaluation effort by conducting a series of working-level review meetings with the NRC staff, NRC consultants, licensee, and licensee's consultants. The meeting summaries and the hand-outs as well as draft analysis reports provided by the licensee were used as the bases of the staff's evaluation.

When a structure was evaluated, it was judged to be adequately designed if:

(1) The analyses are sufficient to adequately determine structural responses consisting of member forces and floor response spectra for the subsystems (piping, equipment and components) evaluations; and

(2) The loads generated from the analyses were less than original loads; or

(3) The seismic stresses from the analyses were low compared to reasonable estimates of the maximum strength of the steel and/or concrete; or

(4) The seismic stresses from the analysis exceeded reasonable estimates of the steel or concrete maximum strengths, but estimated reserved capacity (or ductility) of the structure was such that inelastic deformation without failure or adverse impacts on piping, equipment or component responses would be expected.

If the above criteria were not satisfied, a more comprehensive reanalysis was required to demonstrate its design adequacy. Review Criteria A through C (Section II) provide the basic guidelines for all evaluations in conjunction with the previously referenced SRP and Regulatory Guide guidelines.

For piping reevaluation, the preliminary analysis results presented by the licensee in review meetings were compared with the guidelines for seismic evaluation criteria (Ref. 3) at appropriate service conditions. Piping system is judged to be adequately designed if:

(1) The analyses are sufficient to adequately determine piping system responses; and

(2) The piping responses (stresses) are in conformance with the criteria contained in Review Criteria D and E (Section II); or

(3) The piping responses (stresses) exceed the allowable required in the criteria referenced above, but estimated ductility is such that inelastic deformation could occur without loss of integrity or adverse impacts on the responses of attached piping, equipment or components.

If the above criteria are not satisfied, more comprehensive reanalyses are required to demonstrate its design adequacy. Review Criteria A through E (Section II) provide the basic guidelines for all evaluations, in conjunction with the previously referenced SRP and Regulatory Guide guidelines.

Because limited documentation exists regarding the original specifications applicable to procurement of equipment, as well as for the qualification of the equipment, the seismic review of equipment (electrical and mechanical) was conducted by comparing the results presented in the review meetings with the guidelines for seismic review (Ref. 3). Only the structural integrity of equipment was analyzed and evaluated. The results of this reevaluation served as the basis for the staff to judge if further reanalysis or modification should be undertaken by the licensee.

B. Detailed Evaluation

1. Seismic Input

As a result of NRC Seismic Hazard Analysis (Ref. 4) program conducted by the staff and its consultant, Lawrence Livermore National Laboratory (LLNL), the site specific ground spectra, which are acceptable to the staff as the input for the seismic reevaluation of the Haddam Neck plant, were recommended to the licensee through NRC letters dated August 4, 1980 (Ref. 1) and June 17, 1981 (Ref. 5). In these letters, the staff also encouraged the licensee to propose its own site specific ground response spectra before the final decision about seismic input of this site was made. In Figure 1, a comparison was made for the spectrum recommended by the staff and the spectrum proposed by the licensee (Ref. 6). According to the staff's review, the site specific ground response spectra proposed by the licensee are considered as an appropriate input for the seismic reevaluation of the plant facility.

2. Justification for Continued Operation

Per the requirement of NRC 10 CFR 50.54(f) letters (Ref. 1 and 2), the licensee provided its basis for continued operation of the Haddam Neck plant on September 15, 1980 and June 11, 1981 (Ref. 6 and 7). The NRC safety analysis report (SER) to allow Haddam Neck to continue to operate until the seismic reevaluation program is complete, was issued September 28, 1981 (Ref. 8).

3. Review of Licensee's Seismic Reevaluation Program Plan

A detailed seismic reevaluation program including criteria, scope, analytical procedure, modeling techniques, and schedule for completion, was submitted by the licensee to the staff for review through its letters dated August 5, 1980 (Ref. 9) and September 15, 1980 (Ref. 6). The review of this program plan was performed and discussed with the licensee and its consultants through NRC letter dated January 19, 1982 (Ref. 10).

4. Staff Review of Criteria and Scope

The specific SEP review criteria are documented in NRC NUREG/CR-0098, "SEP Guidelines for Soil-Structure Interaction Review, and Guidelines for Seismic Evaluation Criteria for the SEP Group II Plants." These documents provide guidance for:

(a) selection of the earthquake hazard,
(b) design seismic loadings,
(c) soil-structure interaction,
(d) damping and energy absorption,
(e) methods of dynamic analysis and design procedures,
(f) special topics such as underground piping, tanks and vaults, equipment qualification, etc.; and
(g) allowable stresses and acceptable load combinations.

These criteria are felt to more accurately represent the actual stress level in structures, systems and components during a postulated earthquake event and consider, to certain extent, nonlinear behavior of the systems.

The SEP seismic reevaluation of the Haddam Neck facility was a limited review centering on:

- Assessment of the general integrity of the reactor coolant pressure boundary.

- Evaluation of the capability of essential structures, systems and components required to shutdown the reactor safely and to maintain it in a safe shutdown condition (including the capability for removal of residual heat) during and after a postulated seismic event.

- Evaluation of the capability of structures, systems and components considered as engineered safety features.

All structures, systems and components (structural integrity only) covered by the scope discussed above were reviewed on an audit basis.

5. Review of Reevaluation Criteria and Scope Proposed by the Licensee

The licensee presented its seimsic reevaluation criteria and scope through the letter dated August 5, 1980 (Ref. 9), a series of working-level review meetings, and Vol. I of Seismic Reevaluation reports. As a result of the staff's review and comparison with staff's guidelines, the criteria proposed by the licensee appear reasonable for reevaluation of the plant facility (safety-related structures, systems and components). The detail of the criteria review are found in the staff's contractor reports (Attachments 2 and 3).

As proposed by the licensee, a total of five (5) structures, NSSS Systems, nine (9) piping systems (or sevety-one (71) stress problems), twenty-nine (29) sampled equipment items, and two (2) field erected tanks are to be reanalyzed and evaluated against the acceptance criteria. They are:

structures - containment building (including internal structures), screenwell house, primary auxiliary building, turbine-service building and auxiliary feedwater pump house.

NSSS Systems - Reactor Coolant Loop piping, surge line and major components (reactor pressure vessel, steam generators, RC pumps, pressurizer, and valves).

piping - 9 piping systems (or 71 piping stress problems) are included. All piping systems identified are found in Attachment 2.

equipment - 15 mechanical equipment items and 14 electrical equipment items were reanalyzed. This does not include all equipment necessary for the scope defined by the staff, but is considered representative.

field tanks - demineralized water storage tank (DWST) and primary water storage tank (PWST).

Based on the results of review, the staff agrees that the scope of structures, systems and components covered in the seismic reevaluation program is sufficient. However, if the results of the samples of electrical and/or mechanical equipment show that modifications are required for a particular type of equipment, the balance of equipment in that category should be reanalyzed and upgraded if required.

6. Review of Structures

The structural review of the Haddam Neck plant includes seismic reevaluation draft reports provided by the licensee and discussions at working-level review meetings conducted by the staff and its consultants with the licensee and its consultant. Included in this review were criteria (both analysis criteria and performance criteria), basic assumptions, modelling techniques, analysis methods, and general appropriateness of the results. As a result of this review, the criteria, modelling techniques, assumptions, and analysis methods were found generally acceptable. The review of reevaluation results are briefly discussed below.

(a) Containment Building (including internal structures)

The seismic responses (forces in structural elements and floor response spectra), as discussed in the licensee's draft report, generated from the dynamic models for the containment shell and internal structures are considered to be indicative that integrity of the structures would be maintained under the postulated SSE loadings. In addition, the staff and its consultant also performed a seismic confirmatory analysis of containment shell (Ref. 11) and the results confirmed that the reanalyses generated by the licensee are acceptable. However, some problem areas were identified and additional information should be provided by the licensee to address the following concerns:

(i) A relatively high structural damping ratio (7%) was used in the analyses and very low seismic stresses were reported. As discussed in the staff's guidelines (NUREG/CR-0098), the results might underestimate the seismic input to the subsystems such as piping systems and safety-related equipment items. The licensee should provide further justification for using high structural damping.

(ii) Overstress conditions were predicted at the interface of the pipe gallery between containment shell and primary auxiliary building (PAB) and modifications to these overstressed areas were discussed. The licensee should verify its intent for both in terms of scope and schedule for these overstressed areas.

(iii) Question concerning the adequacy of reinforcing steel in the bottom of the operating floor radial beams. Design information should be provided to show its adequacy or intentions should be provided addressing corrective actions.

(iv) No discussion was given to the percent of modal mass participating in the draft report. This information should be available to the staff for review.

B. Screenwell House

In general, the seismic responses (structural member forces and floor response spectra) generated appear reasonable and the staff believes that this building can withstand the postulated SSE loading without loosing its integrity. The licensee should identify the percent of modal mass participating in the dynamic responses of this structure.

C. Primary Auxiliary Building

The seismic analysis results presented by the licensee appear reasonable and the building should retain its integrity under the postulated SSE loading. The licensee should document the following items in its final safety analysis reports:

(i) Modifications proposed at the pipe gallery connections with the containment and PAB buildings in order to eliminate structural coupling between the buildings.

(ii) Verification that the seismic stresses under the postulated SSE loading in the floor slab are close to yield to justify higher damping ratios used.

(iii) The torsional effects were considered when the in-structure response spectra was generated.

D. Turbine-Service Building

The safety evaluation of this building complex by the licensee was not completed. The staff's review is completly based on discussion at the working-level review meetings. From the results presented by the licensee and its consultant, some structural modifications were identified in the bracing systems of this structural complex. A detailed review of design adequacy of this building should be performed when the final results become available.

E. Auxiliary Feedwater Pump House

The analysis of this structure which is integrally attached to the containment building by steel framing was not completed at the time of review. A detailed review should be performed when the results become available.

The details of the structural review are found in Attachment 2.

7. Review of Primary Reactor Coolant Loop Systems

The seismic reevaluation of primary reactor coolant loop (RCL) is currently being performed by the licensee and its consultant. The staff's review of the RCL major components was based on the preliminary results presented by the licensee and the discussion conducted in the working-level review meetings. Listed below are the findings identified by the staff and its consultant during the review:

(A) The systems were modelled as a three-dimensional lumped mass dynamic model to simulate the as-built condition and the response spectrum analysis method was used for the piping analysis. Major components (reactor vessel, RC pumps, steam generators, valve, etc.) were also included in the model.

(B) A separate analysis with the same modelling techniques and analysis method was performed for the surge line (between RCL hot leg to pressurizer).

(C) A two-step approach was applied for the analysis of most of the major components (reactor vessel, steam generators, pressurizer, and valves). The shell, nozzle and support loads for the evaluation of design adequacy were obtained from the RCL piping analysis. A lumped mass and finite element hybrid model was developed for each of the major components for the evaluation of component internals. Either response spectrum analysis method or time history analysis method was used for the analyses.

(D) Same method used for the first step evaluation of other components was applied for the evaluation of RC pump. The pump internals were analyzed based on the analysis performed for the same type of pump used for SONGS-1 plant. No result was presented by the licensee on this item.

(E) The licensee identified three component supports that need to be upgraded. They are: (1) pressurizer truss support, (2) surge line pipe support, and (3) steam generator hold down bolts.

(F) The criteria for the evaluation of reactor internals have not been provided by the licensee.

(G) The licensee did not consider buckling of component supports (reactor neutron shield tank, pressurizer support truss and surge line pipe supports).

In general, the criteria, modelling techniques, analysis methods and load combinations used for the evaluation of reactor coolant loops and the attached components including supports appear reasonable. However, some open items listed below were identified and additional information should be provided to address the staff's concerns:

(A) Justification for using 4% of critical damping for the RCL piping analyses.

(B) Damping ratios used for pressurizer and steam generator analyses.

(C) Criteria for the evaluation of major component anchorage support systems.

(D) Justification for omitting the impact between the steam generator and its lower supports.

(E) Buckling analyses for supports (reactor neutron shield tank, pressurizer supporting truss, and steam generator skirt supports).

(F) Details of the proposed modifications to the component supports.

The details of review discussed above are found in Attachment 3.

(8) Review of Balance-of-Plant Piping Systems

A total of nine piping systems (main steam, feedwater, auxiliary feedwater, residual heat removal, high pressure safety injection, chemical and volume control, service water, fuel oil, and compressed air lines) are currently being analyzed by the licensee. To date, 57% of piping analysis is complete. The staff's review of these systems as well as their supports was based on the information presented in the working-level review meetings. As discussed in Attachment 3, the licensee's analyses and evaluation of results were performed in accordance with the two documents, "Piping Stress Analysis Procedure For Seismic Qualification of Safety-Related Piping at Connecticut Yankee," and "Connecticut Yankee Atomic Power Company Safety-Related Piping Seismic Qualification Program Criteria Document," which were proposed by the licensee in its program plan and accepted by the staff. From reviewing the results of fifteen (15) piping analyses, the staff concludes that generally the modelling techniques, analysis methods, criteria, and results are acceptable. However, the following items need to be verified by the licensee with either additional design information or justification:

(A) Provide justification for demonstrating the adequacy of modelling techniques applied for the case when the piping penetrate through several walls and floors.

(B) Provide the criteria used for the evaluation of pipe support anchorage systems, e.g., allowable stress for concrete anchor bolts, etc.

(C) If the licensee intends to use chart methods for the analysis of any safety-related piping, provide information to show the validity of the method.

The details of the staff's review are documented in Attachment 3.

(9) Review of Safety-Related Equipment (Mechanical and Electrical)

Due to tight schedule and limitations of man-power, the licensee used a sampling approach for the reevaluation of safety-related equipment (mechanical and electrical equipment including their supports) at the Haddam Neck plant. The selection of samples, as described by the licensee and its consultant, was based on the field inspection and expert's judgement. A total of 15 mechanical and 14 electrical equipment items were selected and are being evaluated by the licensee. Equipment item identifications are found in Attachment 3. The staff's review of these items as well as their supports was based on the preliminary analysis results of four equipment items (diesel exhaust duct, steam driven auxiliary feedwater pump, boric acid tank, and motor control center) presented and discussed in the working-level review meetings. The staff concludes that the criteria (analysis and performance criteria), modelling techniques, and analysis methods as well as results obtained for these four equipment items appear reasonable. However, this condition was based on a very limited sample. Further staff review should be performed when the reevaluation is completed by the licensee. Listed below is a summary of our findings for justification of additional information required by the staff:

(A) The licensee is qualifying equipment required for safe shutdown and ECCS on a sampling basis. This does not meet the specific staff requirements but is deemed reasonable.

(B) Instrument and control, to assure that adequate parameters are available to the operators are not included in the scope of the reevaluation.

(C) Air systems and air operators necessary to insure that necessary safety functions are met are not included in the program scope.

The details of the staff's review are found in Attachment 3.

10. Review of Field Erected Tanks

Two safety-related field erected tanks were identified during the review. They are demineralized water storage tank (DWST) and refueling water storage tank (RWST). The information for reviewing these tanks was presented by the licensee during the review meetings. The criteria, analysis techniques, and results appear reasonable to the staff. The licensee and the staff agreed that modifications are required for the anchorage of RWST. The review of this item is continuing since the DWST analysis was not completed. This item should be evaluated when the final report becomes available.

V. CONCLUSION

As stated previously, the staff's review was based on the preliminary results presented and discussed in the working-level review meetings. Therefore, the conclusions drawn here could be revised upon review of the licensee's final evaluation reports.

Structures

Based on a review that included evaluation of the results presented by the licensee and the results of the staff's confirmatory analysis for the containment shell structure, the staff considers that the safety-related structures and structural elements (containment building and internal structures, screenwell house, and primary auxiliary building) of the Haddam Neck facility are adequately designed for resisting the postulated SSE loadings. Although, no review has been performed for the turbine-service building, based on the fact that this building was originally designed for the same seismic loadings and design criteria, the staff's judgement is that the turbine-service building would withstand the postulated SSE loads. However, the following items are considered as open and require either additional analysis and design information for justifying an analysis method or clarification of the licensee's intended corrective action.

(A) In order to be sure that no underestimation of the seismic input to the subsystems, namely floor response spectra, was made, provide justification of using a high structural damping ratio in the reanalysis.

(B) The pipe gallery connections with containment and PAB buildings were found to be overstressed under the postulated SSE loading. The licensee should state their intended corrective action.

(C) No information was provided as to the participating percentage of modal mass accounted for in the structural analysis.

(D) The information should be provided for demonstrating the adequacy of the reinforcing steel detail in the bottom of the operating floor radial beam.

(E) Information should be provided demonstrating the design adequacy of the auxiliary feedwater pump house.

The staff recommends that the review of safety-related structures be completed when the final reevaluation reports become available.

Primary Reactor Coolant Loop

From evaluations of the results presented by the licensee, the staff concludes that the primary reactor coolant loop piping has sufficient capacity to withstand the postulated SSE loading assuming accceptable clarification is received to address the staff's concerns listed below:

(A) Provide justification for: (1) using 4% of critical damping for RCL piping analysis; and (2) omitting the impact loading between the steam generator and its lower supports.

(B) Provide: (1) criteria for the results of the evaluation of major component anchorage to floors and walls; and (2) damping ratios listed for the seismic analyses of steam generators and pressurizer.

(C) The licensee committed to provide information in its final report showing that buckling will not occur in reactor vessel support (neutron shield tank), steam generator skirt supports and pressurizer truss support.

(D) Provide the details of proposed modifications of major components and their supports.

Final conclusions should be drawn by the stafff when the reevaluaation reports are received.

Balance-of-Plant Piping Systems

Based on our audit review of currently completed analysis results, the staff found out that the stresses in piping systems are within the design allowables and modifications were not required for any of the piping supports. However, the criteria used for the evaluation of pipe support anchorages were not made available to the staff, and, therefore, the criteria for evaluation of support anchors should be provided. In addition, the licensee should provide justification for demonstrating modelling techniques applied for piping that penetrate through several walls and floors are adequate.

Field Erected Tanks

The staff concurs with the licensee results that the anchorage of refueling water storage tanks would be overstressed under the postulated SSE loads. The licensee should complete the analysis of the demineralized water storage tank and upgrade the tanks as required.

Mechanical and Electrical Equipment

Based on the review of the limited results presented for the four (one electrical item and three mechanical items) out of 29 sampled equipment items (15 mechanical items and 14 electrical items) and the assumptions that all equipment items were originally designed for the same criteria and loading conditions, e.g., same allowable stress limits and same level of earthquake loading, the staff believes that limited results for the equipment items infer that the original design was adequate to resist the postulated SSE loads. However, the balance of results from the sampling program should be submitted to the NRC. Further, should any particular type of equipment be shown to be overstressed by the postulated loading, the balance of that type of equipment should be analyzed and upgraded as required.

For the qualification of electrical cable trays, the licensee intended to apply the results of testing through the SEP Owners Group program specifically to their plant and take corrective actions, as required. This program is scheduled for completion by December 1982, and plant specific application will follow.

As far as the operability of equipment is concerned, the staff has initiated a generic program to develop criteria for the seismic qualification of equipment in operating plants as an Unresolved Safety Issue (USI A-46). Under this program, an explicit set of guidelines (or criteria) that should be used to judge the adequacy of the seismic qualifications (both functional capability and structural integrity) of safety-related mechanical and electrical equipment at all operating plants will be developed. The ongoing Owners Group program for equipment qualification will be considered in the development of the USI A-46 criteria.

The staff concludes that following any upgrading identified, the Haddam Neck plant has an adequate capacity to resist the postulated SSE and, therefore, there is reasonable assurance that the operation of the facility will not be inimical to the health and safety of the public.

VI. REFERENCES

1. Letter from D.G. Eisenhut, NRC to W.G. Counsil, NUSCo, dated August 4, 1980.

2. USNRC letter to CYAPCo, dated April 8, 1981.

3. Letter from D.M. Crutchfield, NRC to W.G. Counsil, CYAPCo, dated July 26, 1982.

4. NRC NUREG/CR-1582 Report, "Seismic Hazard Analysis," Vols. 2 - 5, dated October 1981.

5. Letter from D.M. Crutchfield, NRC to All SEP Owners (Except San Onofre Unit 1), dated June 17, 1981.

6. Letter from W.G. Counsil, CYAPCo to D.M. Crutchfield, NRC, dated September 15, 1981.

7. Letter from W.G. Counsil, CYAPCo to D.M. Crutchfield, NRC, dated June 11, 1981.

8. Letter from D.M. Crutchfield, NRC to W.G. Counsil, CYAPCo, dated September 28, 1981.

9. Letter from W.G. Counsil, CYAPCo to D.M. Crutchfield, NRC, dated August 5, 1980.

10. Letter from D.M. Crutchfield, NRC to W.G. Counsil, CYAPCo, dated January 19, 1982.

11. Letter from Ting-Yu Lo, LLNL to W.T. Russell, NRC, dated May 25, 1982.

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ATTACHMENT 2

SMA 12208.21

5160 Birch Street, Newport Beach, Calif. 92660 (714) 833-7552

August 25, 1982

Dr. T-Y Lo (L-90)
Program Manager, SEP Seismic Review
Nuclear Test Engineering Division
Lawrence Livermore National Laboratory
P.O. Box 808
Livermore, California 94550

Subject: Review of Haddam Neck Primary Structures Seismic Evaluation for the Systematic Evaluation Program (SEP)

Dear Dr. Lo:

Enclosed is a revised draft of our review of the Haddam Neck primary structures analysis conducted by URS/J.A. Blume. The revisions include our review of the Primary Auxiliary Building draft report (Volume V) as well as the resolution of several items by telephone with Mr. T. Cheng of the NRC and J. A. Blume staff members. We have still not received the turbine/service building report nor any results for the field-erected tanks or buried utilities.

At Mr. Cheng's request, I am forwarding a copy of this report directly to him. Please don't hesitate to call if you have any questions.

Very truly yours,

STRUCTURAL MECHANICS ASSOCIATES, INC.

Don Wesley

Donald A. Wesley
Vice President

Enclosure

cc: T. M. Cheng (with enclosure)
T. A. Nelson (with enclosure)

1. INTRODUCTION

As part of the Systematic Evaluation Program (SEP), the seismic adequacy of a number of existing nuclear power plants is being reviewed. Most of the plants were designed using different methods and criteria than are in effect today. This does not necessarily imply that the older plants are unsafe, but merely that more sophisticated methods of analysis are in use today and that acceptance criteria are better defined. The major structures of the Connecticut Yankee Atomic Power Plant at Haddam Neck, Connecticut, have recently been reanalyzed by URS/John A. Blume and Associates, Engineers in accordance with current methods (References 1 through 4). The adequacy of the structures has been determined using currently accepted criteria and guidelines. These criteria are summarized in Reference 1. Where modifications have been required, design changes have been developed and will be implemented as part of a scheduled program.

The results of the SEP seismic reanalysis are being documented in a five-volume report which summarizes the analysis and evaluation criteria and describes the analysis results and structural evaluation for the containment building including the internals structure, the auxiliary feedwater pumphouse, the primary auxiliary building, the screen-well house, and the turbine-service building. Also included are the in-structure response spectra developed at critical equipment locations throughout the structures. Although acceptance criteria are presented for the auxiliary feedwater pumphouse, it is understood that the results for this structure will be included in the results presented for the piping systems analysis. Also not included in the URS/Blume evaluation are several items such as field-erected tanks and buried structures which are often included in the civil scope of supply. The field-erected tanks are being evaluated by J. D. Stevenson and Associates in conjunction with the evaluation of the equipment, and the licensee is evaluating the buried utilities. Finally, the adequacy of the masonry walls was not considered in the current SEP review. These walls are being treated under US NRC IE Bulletin 80-11.

A review of the URS/BLUME evaluation has been conducted by members of the US NRC, Lawrence Livermore National Laboratory (LLNL), and Structural Mechanics Associates, Inc. The results of this review are presented herein. The review was based on draft reports for Volumes I, II, III, and V of the five-volume report together with discussions conducted with Northeast Utilities Service Company and URS/Blume personnel. Included in the review were methods and criteria used in the evaluation together with a review of the general appropriateness of the results. The seismic models together with the methods and assumptions used in their development were reviewed. For the turbine/service building, the review was limited to discussions with the URS/Blume staff since written documentation was not available. Summary results based on URS/Blume evaluations were discussed in the April 7, 1982 meeting and subsequently by telephone. The detailed calculations leading to the analytical models or the evaluation of capacities were not reviewed nor were the structural drawings on which the analytical results were based.

SUMMARY AND RECOMMENDATIONS

The seismic evaluation conducted by URS/Blume for the Connecticut Yankee Atomic Power Plant structures is, in general, a thorough and well-executed analysis. For the most part, the evaluation was based on current US NRC regulatory guides and Standard Review Plan (SRP) criteria. The approach adopted was to modify the structures as required in order to assure seismic stresses combined with those from gravity forces remained below yield.

The seismic input used in the analysis was defined by response spectra anchored to 0.17g peak ground acceleration rather than the 0.21g site specific spectra developed by LLNL for the Connecticut Yankee site. The spectra used for analysis exceed the LLNL site specific spectra at all frequencies less than approximately 20 Hz. A comparison of the 5% damped ground response spectrum used for the analysis with the 5% damped LLNL site specific response spectrum is shown in Figure 1. Most of the response of the primary structures results from excitation in the frequency range less than 20 Hz so the response of these structures is expected to be conservatively predicted using the spectra used in the analysis rather than the LLNL site specific spectra. However, the effects of the reduced acceleration levels should be considered for rigid equipment located near the base slabs of the Haddam Neck structures.

The seismic models developed by URS/Blume are based on currently accepted practice and are considered to adequately characterize the seismic response of the structures. Where significant uncertainty as to modeling assumptions existed, parametric studies were conducted to assess the effects of these assumptions. The documentation of the results is generally well-written and complete. Discussions with URS/Blume staff members provided further clarification in some areas. However, not all the analysis was complete and documented at the time of the review.

The status of the structures review is summarized in Table 1. Table 1 presents the overall adequacy of the seismic analysis; however, several further outstanding items listed in Table 2 should be addressed. Finally, a number of additional comments are included, many of which are editorial in nature or request additional information which was discussed in the April 7, 1982 meeting but is not currently documented.

Based on the satisfactory resolution of the items discussed in this report and the implementation of the structural modifications described in the URS/Blume report, it is the judgment of the review team that the seismic evaluation of the Connecticut Yankee plant provides adequate assurance that the plant primary structures will withstand the earthquake excitation expected for the site as characterized by either the response spectra used for the analysis or the LLNL site specific spectra with no loss of function.

TABLE 1

REVIEW SUMMARY OF THE SEISMIC REEVALUATION PROGRAM PLAN

August, 1982

ITEM	ADDRESSED?	ADEQUATE?
I. Soil and Foundation		
A. Rock Site (Mostly)	yes	yes
B. Soil Site (Partly Backfill)	yes	Insufficient Data(1)
• Foundation input	---	---
• Generation of time history	---	---
• Modeling technique	---	---
• Computer codes	---	---
C. Description of Foundation	yes	yes
D. Free-Field Input Spectrum	yes	yes(2)
II. Structural		
A. List and Description of Category I Structures or Structures Affecting Category I Systems or Components	yes	(3)
B. Modeling Techniques		
• Damping	yes	yes(4)
• Stiffness modeling	yes(5)	yes
• Mass modeling	yes	yes
• Consideration of 3-D effects	yes	yes
C. Seismic Analysis Methods		
• Response Spectrum, time history or equivalent static analysis	yes	(6)
• Selection of significant modes	yes(5)	yes
• Relative displacements	yes(5)	yes
• Modal combinations	yes	yes(7)
• Three component input	yes	yes(7)
• Floor spectra generation	yes	yes
• Peak broadening	yes	yes
• Load combination	yes	yes
D. Analytical Criteria		
• Codes and criteria, including AISC, ACI and NUREG/CR-0098	yes	yes

TABLE 1 (Continued)

REVIEW SUMMARY OF THE SEISMIC REEVALUATION PROGRAM PLAN

August, 1982

ITEM	ADDRESSED?	ADEQUATE?
E. Computer Codes		
• Description and verification	yes(4)	yes(8)
III. Field-Erected Tanks and Buried Utilities*		

*No information was available for review. The field-erected tanks are being evaluated by J. D. Stevenson and Associates; the buried utilities are being evaluated by the licensee.

COMMENTS

1. Part of the service building and the demineralized water tank are founded on fill. No soil properties for this fill were presented.

2. A 0.17g SSE was used which does not meet the 0.21g site specific spectrum in the frequency range above 20 Hz but exceeds the site specific spectrum at frequencies below 20 Hz. The response of the primary structures is generally expected to be conservatively predicted using the 0.17g spectra.

3. NRC staff will determine the adequacy of the scope.

4. For reinforced concrete structures with only slight cracking, 7% of critical damping may be slightly nonconservative, thus, the input to equipment may be larger than that predicted. Seven percent of critical damping should be used only for concrete structures with considerable cracking.

5. A discussion of models and programs used for reanalysis was presented at the September 30, 1981 and April 7, 1982 meetings at URS/J. A. Blume.

6. The methods of developing static loads must be justified.

7. General reference is made to RG 1.92.

8. Computer code verification should be provided.

REFERENCES

1. "Haddam Neck Plant, Systematic Evaluation Program, Seismic Reevaluation", Docket No. 50-123, 810051, W. G. Counsil to D. M. Crutchfield, August 5, 1980.

2. "Haddam Neck Plant Seismic Reevaluation Program SEP Topic III-6, Seismic Design Considerations", Connecticut Yankee Atomic Power Company, Document No. A01629, August 11, 1981.

TABLE 2

OPEN ITEMS TO BE RESOLVED FOR CONNECTICUT YANKEE

1. Summarize all proposed and implemented modifications to structures.

2. Identify areas of structures founded on soil and document reasons for not including soil-structure interaction effects.

3. Identify percent of modal mass participating in the dynamic response of each structure. Identify how the inertial loads from any mass not included in the response spectrum modal analysis was accounted for. Document that the inertial loads from any mass not included in the response spectrum analysis was included in the stress and sliding analyses.

4. Quantify the seismic separation between adjacent structures and their calculated relative displacements. Particular attention should be given to the PAB to turbine/service building interface.

5. Justify the level of damping used in the dynamic analysis of each structure on the basis of percent of yield stress experienced by the structure. Justification for the reactor building concrete internal structure and crane support wall, the floor slabs in the screenwell house and the PAB, and the PAB steel superstructure should be provided.

6. Flexure stresses in the concrete shear walls should be presented and the adequacy of these walls to withstand seismic overturning moments should be documented. Stresses at critical locations in the PAB steel superstructure should be documented.

7. Include the buoyancy effects for any affected structures.

8. Document the intended modification for overstressed areas at the pipe gallery connections.

9. Document the stability of the polar crane.

10. Document the results of the auxiliary feedwater building analysis.

11. Document the results of the turbine-service building analysis.

12. Provide the results of the seismic analysis of the new diesel generator building.

13. Document the directional load combination and coefficient of friction assumed in the sliding evaluations.

VOLUME I

SEISMIC ANALYSIS AND EVALUATION CRITERIA

This volume (Reference 1) covers the scope of the analysis and describes the methods and acceptance criteria to be used for the major structures. It is the understanding of the review team that these criteria are applicable to the analyses of all the primary structures conducted by URS/Blume.

Codes and Standards

Codes and standards referenced include:

a. USNRC Standard Review Plan - Sections 3.7.2, 3.8.3 and 3.8.4

b. USNRC Regulatory Guides 1.60, 1.61, 1.92, and 1.122

c. ACI Codes 349-76 and 359-77

d. AISC Specifications for Design, Fabrication and Erection of Structural Steel for Buildings, Eighth Edition

e. Uniform Building Code - 1979 Edition

These codes and standards provide, in general, a conservative basis for the seismic evaluation.

Material Properties

Code design allowables are used in the evaluation increased by a factor for SSE loading as specified in SRP 3.8. Damping in both reinforced concrete and structural steel was assumed at 7% of critical. This is the lower bound damping value recommended in NUREG/CR-0098 for reinforced concrete structures at or just below the yield point. For bolted or riveted steel, the corresponding lower value is 10% so the assumed damping of 7% is conservative for steel structures at or near yield. As discussed in Item 5 of Table 2, the damping levels should be justified for each structure on the basis of percent of yield stress experienced by the structure. Structure stiffnesses were based on gross section properties but the determination of the modulus of elasticity of the concrete is not discussed. Minor variation in the modulus of elasticity is not expected to have a significant effect on the Connecticut Yankee structures.

Analytical Procedure

The analyses were based on linear elastic analyses for all structures. Closely-spaced modes were considered for the response spectrum analyses of the structures and in-structure response spectra were generated using time history methods. Equivalent static analysis were stated to be used where justified. An example of a static analysis is the auxiliary feedwater pumphouse. However, this analysis is not complete and the methods of establishing equivalent static loads need to be documented.

Soil-Structure Interaction

All structures at Connecticut Yankee were analyzed as fixed-base models. This is considered adequate for structures founded on competent rock such as Connecticut Yankee. A small, lightly-loaded region of the service building is founded on fill. During the April 7, 1982 meeting, the basis for neglecting the soil-structure interaction of this area was discussed and was considered acceptable by the review team. It is our understanding that this basis will be documented in the report.

Structural Modeling Including Coupling and Torsion

The criteria presented in Volume I to develop the mass and stiffness properties of the models including the effects of coupling of the NSS system is considered to be generally acceptable. No provision for live loads beyond the installed piping and equipment weights is included. This is expected to have minimal effect on the Connecticut Yankee structures and is usually conservative for input to equipment. It is our understanding that documentation covering the calculations of centers of rigidity of the structures will be included in the final report.

Floor Response Spectra

Floor response spectra were generated by time history methods and the peaks of the spectra were smoothed and broadened ± 15%. This is considered acceptable for the Connecticut Yankee structures.

Allowable Stresses

Allowable stresses for the reinforced concrete portions of the reactor building are in accordance with ACI Code 359-77 and on ACI Code 349-76 for the other concrete structures. Steel portions of the Connecticut Yankee structures are based on elastic design methods of AISC in conjunction with SRP Sections 3.8.3 and 3.8.4. No provision for ductility is required. These criteria are considered to be conservative in general for SEP plants. Applying the 1.6 factor to AISC allowables for stability may, in some cases, be unconservative. If the predicted stress in steel compression members is close to $1.6S$, additional justification should be required.

Structural Foundations

Structural foundation evaluations will be presented that predict factors of safety of 1.1 against overturning and sliding. Overturning for the Connecticut Yankee structures is not expected to be a problem due to the long time required for overturning compared to the earthquake input frequencies and the high bearing capacity of the rock site. However, the directional load combination and coefficient of friction assumed in the sliding evaluations should be documented. Also, the forces from the mass not participating in the modal response should be accounted for in the sliding and overturning calculations.

Block Walls

Block walls are not being considered in this review. Block walls for Haddam Neck are being reviewed as a generic issue according to US NRC IE Bulletin 80-11. However, modifications are being implemented by adding external steel supports for some walls.

Free-Field Acceleration Time History

An artificial earthquake which develops response spectra that essentially envelop the ground spectra used in the analyses was developed based on the S80E component of the 1957 Golden Gate Park earthquake. The earthquake is considered to provide an adequate basis for generating the in-structure response spectra.

Buoyancy Effects

As noted in Item 6 of Table 2, consideration needs to be given to potential buoyancy effects.

VOLUME II

CONTAINMENT STRUCTURE

This volume (Reference 2) covers the analysis and results of the seismic evaluation of the reactor containment building and the concrete internals structure. The auxiliary feedwater pumphouse is integrally attached to the containment building by steel framing. However, the analysis of this structure had not been completed at the time the review of the analyses was conducted.

Modeling Techniques

Separate fixed base models were developed for the reactor containment building and the reinforced concrete internal structure. This is considered adequate since coupling between the two structures on the rock site is expected to be minimal. Bearing capacities for solid rock are listed as 10 tons per square foot and 6 tons per square foot for loose rock. It is the understanding of the review team that no loose rock exists under the Connecticut Yankee reactor building.

The containment building was modeled using axisymmetrical finite elements. Parametric studies were conducted to show the effect of equipment hatch opening did not invalidate the assumption of axisymmetry. Also, a parametric analysis was conducted to show the coupling of the containment building to the Primary Auxiliary Building (PAB) through the pipe gallery did not substantially affect the dynamic characteristics of the structures. Finally, an independent analysis (Reference 5) of the containment building was conducted which verified the dynamic characteristics of the structure. It was concluded by the review team that the structural modeling of the reactor building can be relied on to provide appropriate results. A value of seven percent of critical damping was assumed.

The interior concrete structure was modeled as a three-dimensional finite element model which included the NSSS and polar crane. The cavity walls were modeled as a lumped mass cantilever structure combined with a thin shell finite element representation of the crane wall. The details of the NSSS model were not evaluated by the review team. This review is being conducted by the NRC piping and equipment contractor for Haddam Neck. However, the coupled NSSS and concrete internals model is assumed to provide appropriate seismic response for the concrete structure evaluation.

The auxiliary feedwater pumphouse model was not included in the review since the analysis of this structure was not complete. The model should include provision for the pipe bridge loads including any relative motion between the pumphouse and the service building.

Analysis and Evaluation

The seismic responses developed from the models for the containment shell and the concrete internals structure are generally considered to provide valid results on which the capacity evaluation may be based. The seven percent damping assumed for the concrete is considered realistic based on the reported shear stress of 105 psi. Mass proportional composite modal damping ratios based on seven percent of critical for concrete and four percent of critical for the NSSS and crane were computed for the internals structure. However, the maximum shear stress in the crane wall is reported as only 29 psi. A damping value of seven percent for the concrete is considered to be too high on the basis of this stress. This is not a problem for the concrete structure since higher stresses with expected higher damping must occur before damage to the concrete is expected. However, the effects of lower concrete damping on the input to equipment should be evaluated or further justification of the seven percent assumed damping should be provided.

Overstress conditions are predicted at the interfaces of the pipe gallery with the containment shell and the PAB. This condition was predicted based on a relative horizontal displacement of 0.02 inches. Discussions with URS/Blume personnel indicate this was determined from the modal analysis on an absolute sum basis. This is considered conservative. Furthermore, any minor cracking can be expected to provide a significant reduction in stress, and this damage mode is not considered to pose a major hazard to the ability to achieve safe shutdown. However, as noted in Item 8 of Table 2, the proposed resolution of this problem should be documented.

URS/Blume staff members further identified a question concerning the adequacy of the reinforcing steel detail in the bottom of the operating floor radial beams. This detail was not available to the review team nor was the calculated stress in this location or whether the stress of concern results primarily from dead weight or seismic loads. However, this potential problem should also be addressed if it is judged to be a safety concern.

In Tables 2.1 and 2.9, the percent of the mass participating should be presented. In Tables 2.2 and 2.3, the notation "Input" is confusing since these are apparently the spectral responses at the given locations. Also, the base input accelerations are not 0 as noted in these tables. Discussions with URS/J. A. Blume personnel indicated the values in Tables 2-2 and 2-3 include only the response of the modes with frequencies below 33 Hz. The participation of the remaining mass was not included. It is the understanding of the review team that the forces from this mass will be included and the tables revised. It is not clear why a slight decrease in shear and overturning moment are calculated for the base slab compared to the next higher node as noted in Table 2.8.

In Table 2.10, modal damping values are listed as a percent of critical damping rather than as a decimal fraction as appears to be the correct notation. In Table 2.11, relative displacements at the annulus slab are reported to be an order of magnitude less than at the operating floor. It is not apparent why this should be the case. Also, for N-S response, the annulus slab is reported as having higher response accelerations than the operating floor. Discussion with URS/J. A. Blume personnel indicated that the response reported for the annulus slab is actually for one of the beams which is only attached at the two ends and hence has higher response at a mid-node. The locations of the nodes for which the response is predicted will be indicated in Table 2-11, and the text will be clarified.

Figure 2-16 does not show a lateral support near the top of the pressurizer. This implies the guides located on the slab at Elevation 48'-6" have no function during an earthquake. Based on clearances from drawings available to J. A. Blume, this is apparently the case since the relative displacements of the pressurizer at this location do not exceed the clearance (assuming the pressurizer is centered with respect to the guides). J. A. Blume personnel will check with the NSSS vendor to assure the clearances in the guides are correct for the hot condition.

Figures 2.C.35 and 2.C.36 show significantly different response spectra for the crane support for the E-W and N-S directions. It is assumed this applies to the crane support rail at approximately Elevation 48'-6". Since the crane support wall is essentially a cylinder which is stiffened by the operating floor, this is considered unrealistic. Apparently the figure shown in 2.C.35, is actually the vertical response and Figure 2.C.37 labeled vertical response is actually the E-W horizontal response.

VOLUME III

SCREENWELL HOUSE

This volume (Reference 3) covers the analysis and results of the seismic evaluation of the screenwell house. No details of the steel superstructure were reviewed by the review team and no calculations for the development of the model, the stress analysis, or the recommended modifications to the structure were checked by the review team.

Modeling Techniques

A three-dimensional, combined lumped-mass, finite element fixed-base model was developed for the screenwell house. A separate two-dimensional finite element model was developed to calculate the vertical response of the floor slab at Elevation 21'-6".

Details were not available to evaluate the design of the roof of the screenwell house which consists of a ten-inch concrete slab over a braced steel frame. Depending on whether the steel frame or the slab resists the diaphragm loads, considerable variation in stiffness may be expected, and the assumption of a lumped mass at the roof should be verified. Discussion with URS/J. A. Blume personnel indicated this portion of the analysis will be verified and documented in the final report.

The modeling of the screenwell house as a fixed base system is considered acceptable. The discussion of the soil backfill on the east, west, and north sides of the structure is confusing since the Connecticut River borders the west side. The results of the parametric study indicate modeling the stiffness of a 37 degree wedge of soil results in an increase in frequency of only 0.02 percent. The details of how the soil stiffness was computed were not available to the review team but the change in frequency appears surprisingly low. The details of how the soil and hydrodynamic masses were lumped in the model is not well documented. If the soil and water masses were distributed throughout the structure for both horizontal and vertical response, the results are expected to be conservative. An approximation of the total mass of the soil and water lumped at the base for vertical response is considered more accurate. Also, it is not clear whether the soil and water masses were assumed to act in tension. If so, this again is expected to result in conservative structure loads. These effects are expected to be minor. However, the text will be clarified in this area.

The evaluation of floor diaphram rigidity and decoupling of the slab at Elevation 21'-6" from the wall stiffness effects appears correct. Also, the combined lumped-mass, finite element model of the overall structure is considered adequate to characterize the seismic response of the structure. Details of how the shear stiffnesses of the structure and how the location of the centers of rigidity were computed would improve the report.

The modeling of the floor slab at Elevation 21'-6" includes representations of both the slab and beams. The walls may not provide total fixity against rotation as was assumed in the analysis, although this is expected to cause only minor changes in the input to equipment. Verification that the slab seismic stress is near the yield level should be provided in order to justify the assumed seven percent of critical damping for the slab, however.

Analysis and Evaluation

In general, the seismic response appears consistent with the site conditions and the Haddam Neck spectra input. As stated in Item 3 of Table 2, the percent of modal mass participating in the dynamic response should be documented.

The results of the analysis for E-W response at Elevation 8'-0" appear inconsistent. A response acceleration of only 0.15g is reported compared to the 0.17g input. This compares with 0.25g at these locations for N-S response and a zero period acceleration (ZPA) of over 0.25g from the E-W in-structure response spectra generated by time history analysis. Discussions with URS/J. A. Blume indicate that apparently the response presented in Table III.4 only includes the modal response for modes with frequencies less than 33 Hz. It is the understanding of the review team that the rigid body response will be included in the results and the stress analysis and sliding analysis checked to assure adequacy for the increased response.

The in-structure response spectra were broadened ± 15 percent and appear reasonable. Discussions with the URS/Blume staff indicates the locations are for a worst-case location including the effects of torsional response in the structure. This should be documented.

VOLUME IV

TURBINE-SERVICE BUILDING

The documentation covering the results of the analyses and evaluation of this structure was not available at the time this review was completed. Discussions with the URS/Blume staff indicate that a detailed, finite element model has been developed, and that some structural modifications are required in the bracing systems of this structure. The structure was modeled as a fixed-base system although there is some fill under the southeast corner of the service building. This appears to be under an unbraced column line and hence lightly loaded. Therefore, the omission of soil flexibility in the model is not expected to result in substantial variation in the calculated response of this structure. It is the understanding of the review team that further description and justification will be provided in the report including any relative motion effects which could affect the analysis of the auxiliary feedwater pumphouse and pipe systems connecting the service building and containment building.

VOLUME V

PRIMARY AUXILIARY BUILDING

This volume (Reference 4) covers the analysis and results of the seismic evaluation of the Primary Auxiliary Building (PAB). The analysis and evaluation were based on the PAB being separated from the turbine/service building. Also, additional steel framing in the roof truss and the addition of pipe support columns for the roof are required. The results are based on the assumption that these modifications will be implemented.

Modeling Techniques

A three-dimensional finite element model consisting of plate, shell, and beam elements was developed for the horizontal response analysis. Based on a parametric study, an uncoupled model of the PAB was used although the PAB and the reactor building are connected through the pipe gallery. The uncoupled building approximation is considered acceptable for these two structures. The analysis also assumed no interaction with the turbine/service building based on modifications to introduce a gap between these structures. The amount of clearance between these structures together with the computed relative displacements as a function of elevation should be documented. The horizontal model was developed using fixed-base assumptions since the structure is founded on rock.

Separate finite element models of the floor slabs were developed for the slabs at Elevations 15'-6", 21'-6", and 35'-6". The vertical rigidity of the walls was determined to be sufficiently high so that they could be neglected in the analysis of the vertical response of the floor slabs.

The mathematical models developed for both the horizontal structure analysis and the vertical response of the floor slabs are considered to adequately characterize the seismic response. However, neither the calculations used to develop the models nor the structural drawings were checked by the SEP review team.

A parametric study conducted to determine the degree of coupling between the PAB and the reactor building was based on simplified lumped mass models. The fundamental frequency of the simplified PAB model used in the coupling analysis was approximately 15.35 Hz compared to approximately 9.76 and 10.5 Hz for the finite element model of the PAB. Possibly some additional coupling could be expected with the 5.55 Hz reactor building if the PAB were modeled more exactly for the coupling analysis. However, the increase in coupling is expected to be small, and the use of uncoupled models for the PAB and reactor building is considered acceptable. The recommended modification to eliminate the overstress conditions at the pipe gallery connections with the PAB and reactor building would eliminate any structural coupling between the buildings.

Analysis and Evaluation

The seismic response calculated with both the horizontal model and the floor slab models appears consistent with the Haddam Neck site conditions and seismic input. As stated in Item 3 of Table 2, the percent of modal mass participating in the dynamic response should be documented, and provision made for any rigid body response in the stress and sliding analyses.

Shear stresses reported in Reference 4 for the PAB for the concrete walls are below allowables but are considered high enough to justify the 7% of critical damping used in the analysis. However, verification that the seismic stress in the floor slabs is near yield should be provided in order to justify the damping in the slab. The adequacy of the shear walls in flexure should also be documented. Stress levels in critical locations in the steel superstructure should be documented together with modal damping assumptions for the superstructure.

A recommendation was made in Reference 4 that the PAB be cut loose from the pipe gallery in order to relieve the overstress conditions occuring in both the PAB and the containment building shell. A commitment should be obtained from the licensee that this recommendation will be implemented, or another resolution to the problem should be presented.

The in-structure response spectra were broadened ± 15% and appear reasonable. It should be documented that the spectra include the effects of torsional response or are applicable for a worst-case location.

BURIED UTILITIES AND FIELD-ERECTED TANKS

No documentation of any analysis for the buried utilities or the field-erected tanks was available for Connecticut Yankee at the time this review was completed. Because of the rock site, peak effective ground velocity will be high and rock strains are expected to be low for the Connecticut Yankee site. Consequently, it is not anticipated that the buried utilities will develop significant problems for the 0.17g earthquake. The results of these evaluations should be reviewed when available however.

The Demineralized Water Storage Tank (DWST) and the Primary Water Storage Tank (PWST) have both been identified as important to safety. The DWST was designed for a 0.17g earthquake. However, the PWST was originally designed for 0.03g and is expected to require upgrading. The DWST is founded on approximately 20 ft of fill. Since the response of this type of tank is often greater for softer foundation conditions, inclusion of soil-structure interaction effects and amplification of the earthquake excitation through the soil layer should be included in the analysis or justification should be provided for neglecting these effects.

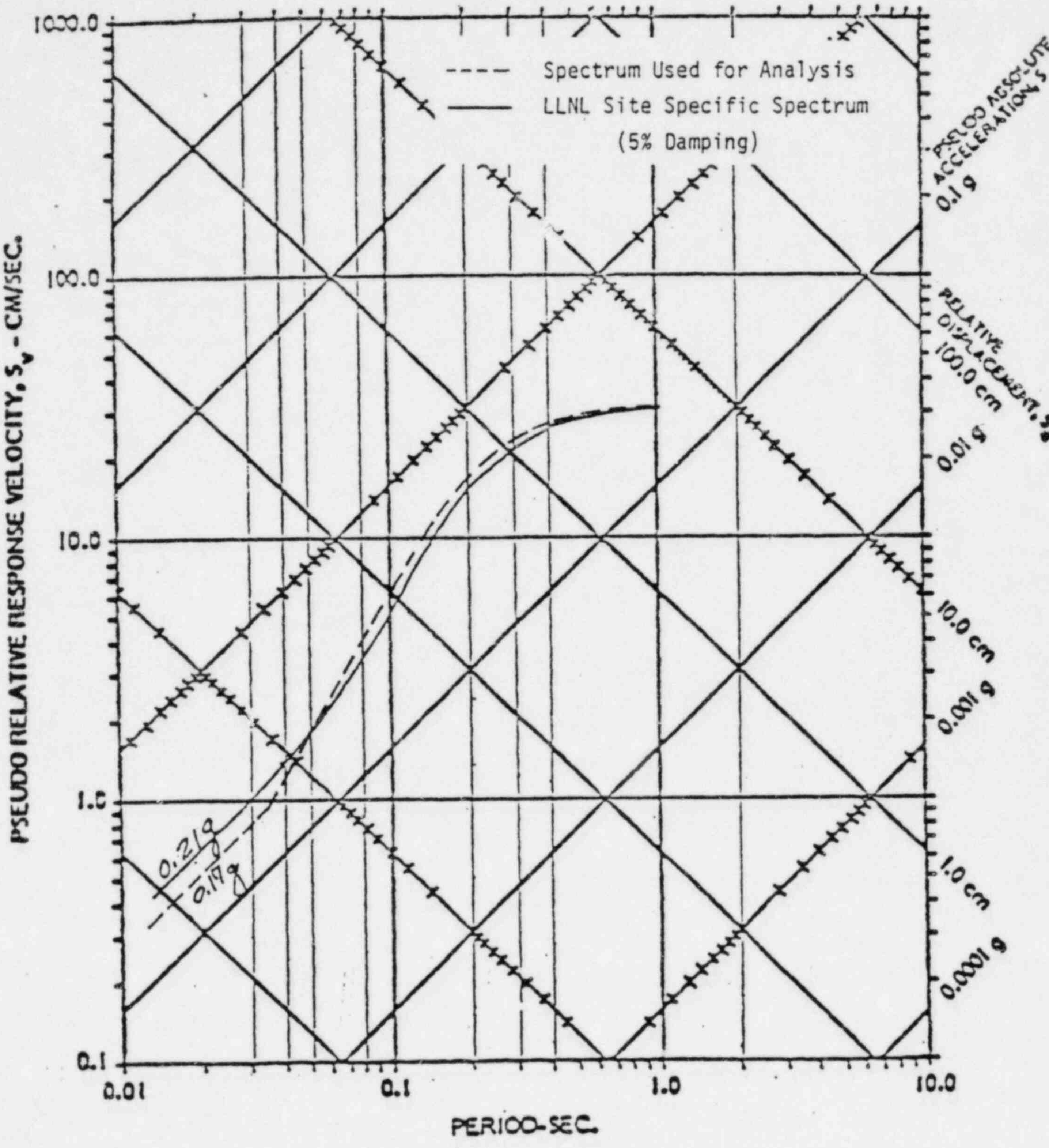

FIGURE 1. COMPARISON OF HADDAM NECK SPECTRUM USED FOR ANALYSIS WITH LLNL SITE SPECIFIC SPECTRUM

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SEPB

ATTACHMENT 1

EDAC 175-130.01

DRAFT

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

SYSTEMATIC EVALUATION PROGRAM

prepared for

Nuclear Test Engineering Division
Lawrence Livermore Laboratory
Livermore, California

January, 1979

ANO
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1. INTRODUCTION

This report presents the results of an evaluacion 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 <u>the bottom of the slab embedded 30 foot 6 inches below grade.</u> 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.

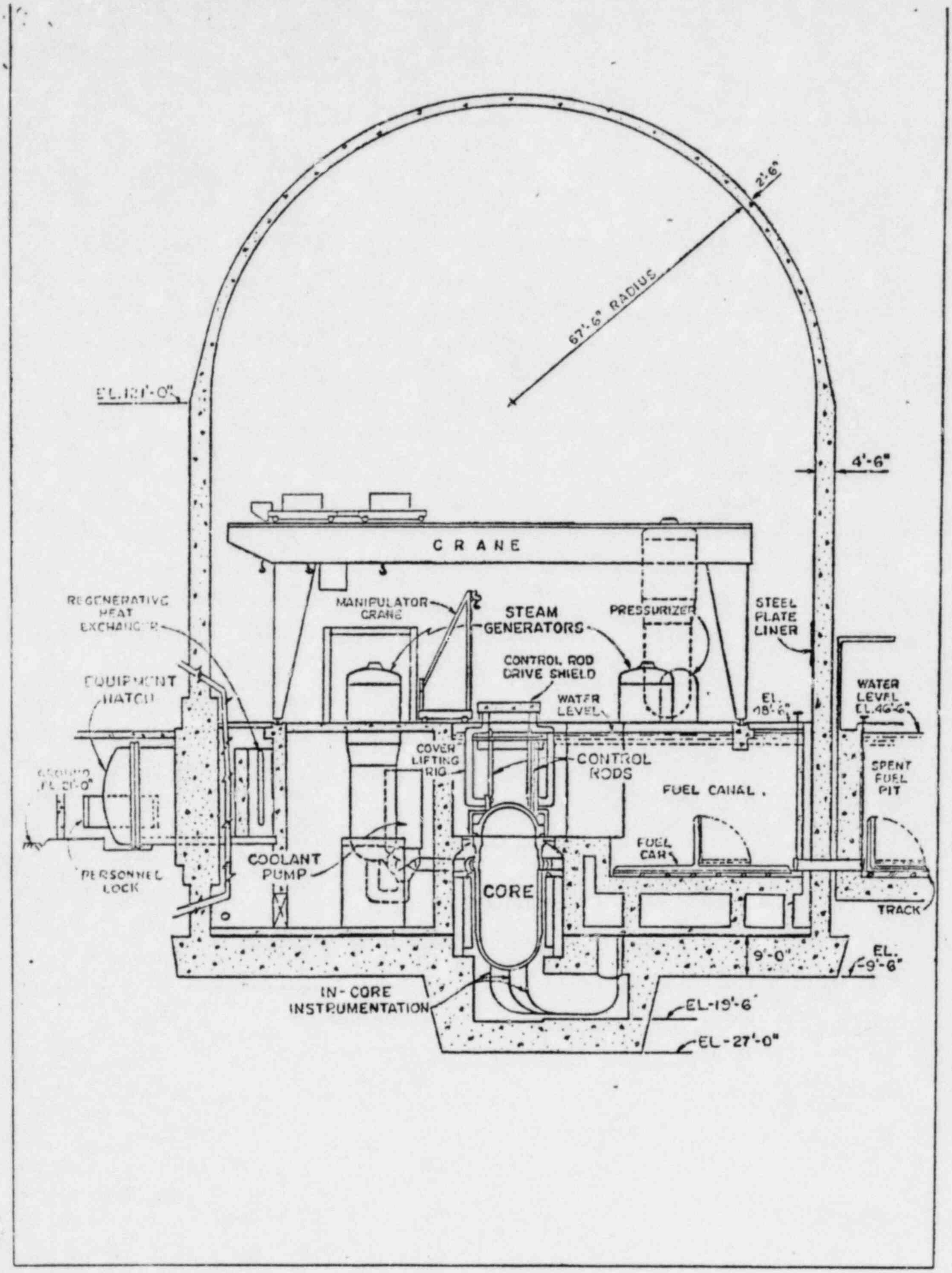

FIGURE 1-1. REACTOR CONTAINMENT BUILDING

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. <u>The point on the fault nearest to the site area is 8 miles west-northwest.</u>

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	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

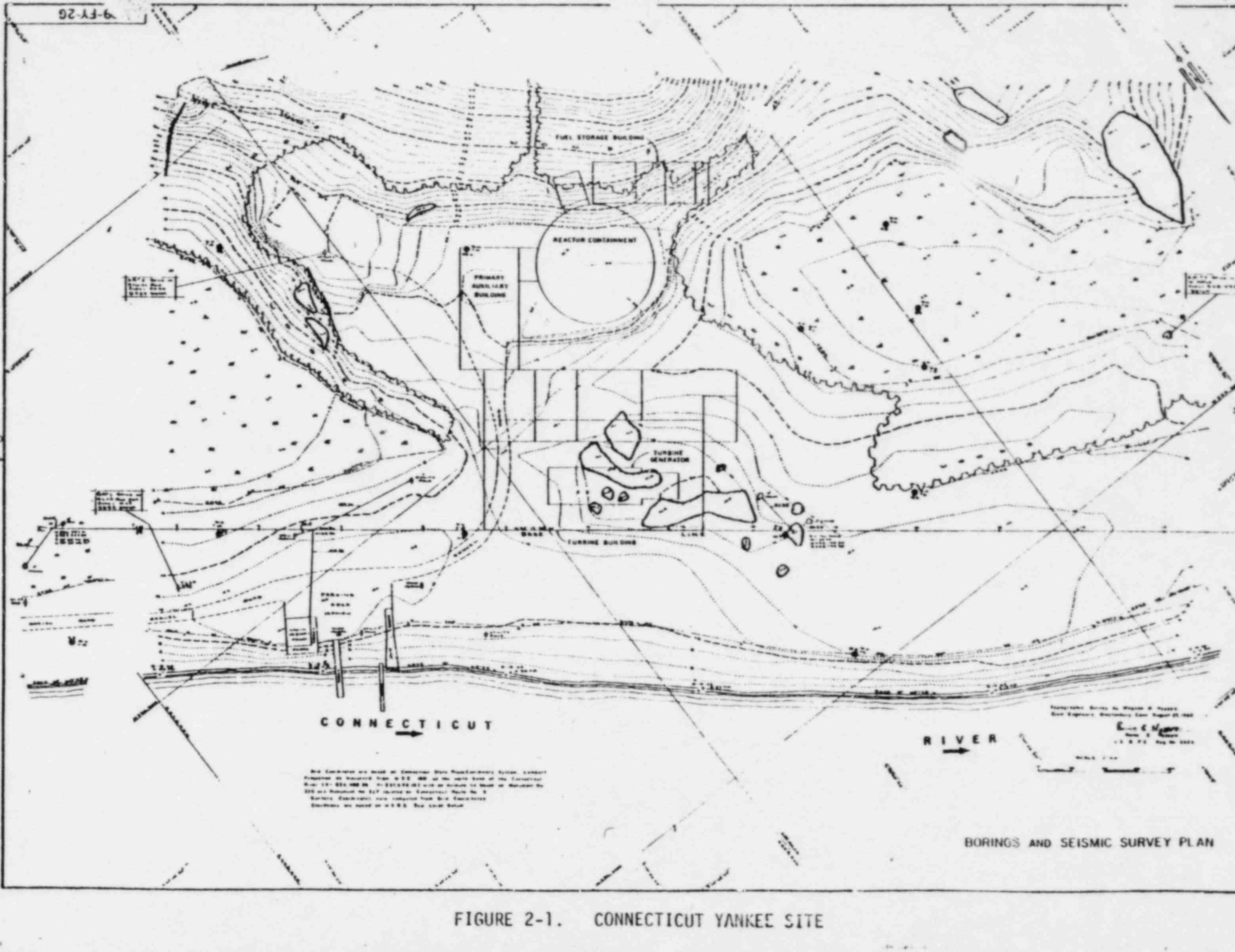

FIGURE 2-1. CONNECTICUT YANKEE SITE

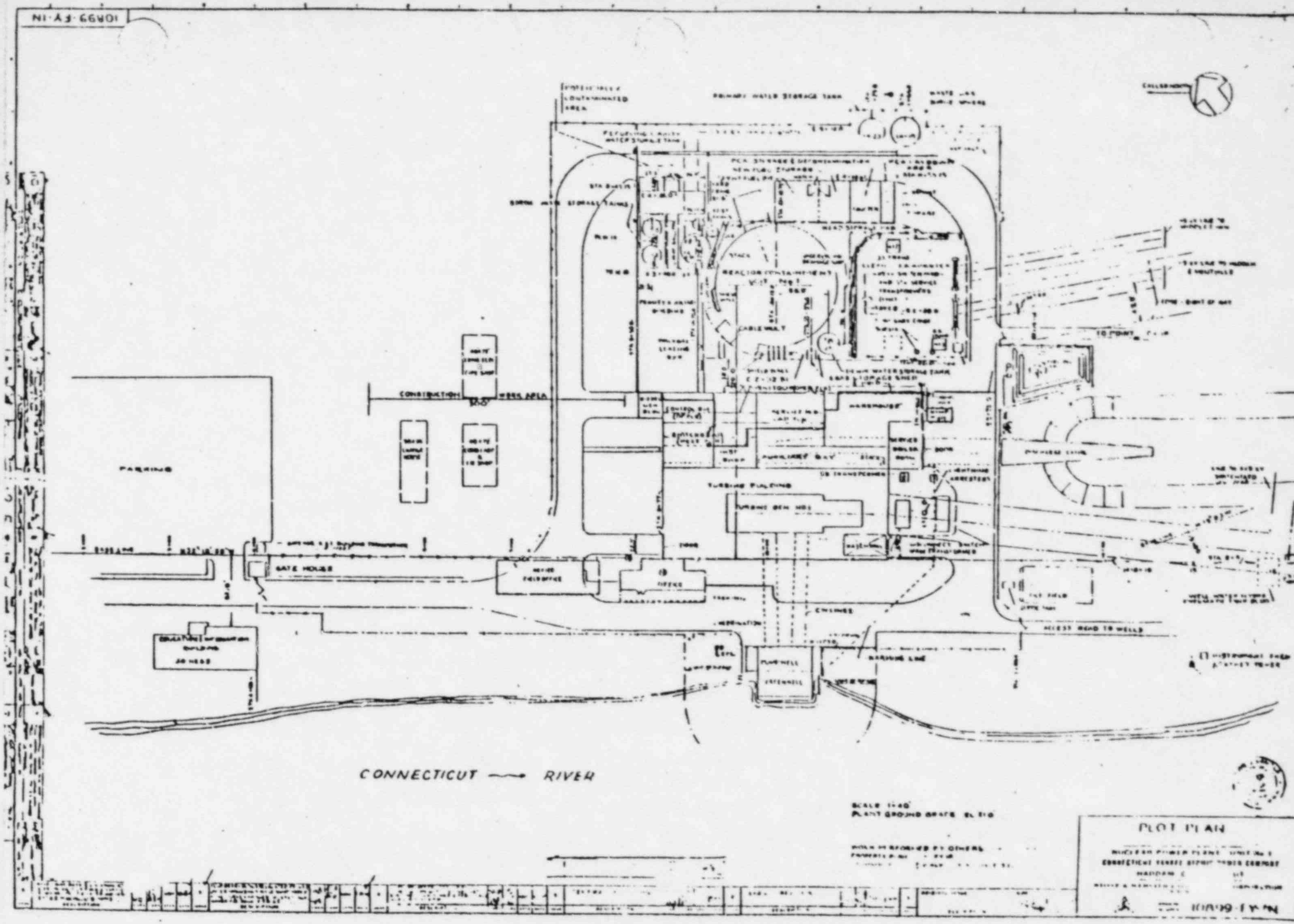

FIGURE 2-2. CONNECTICUT YANKEE PLOT PLAN

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

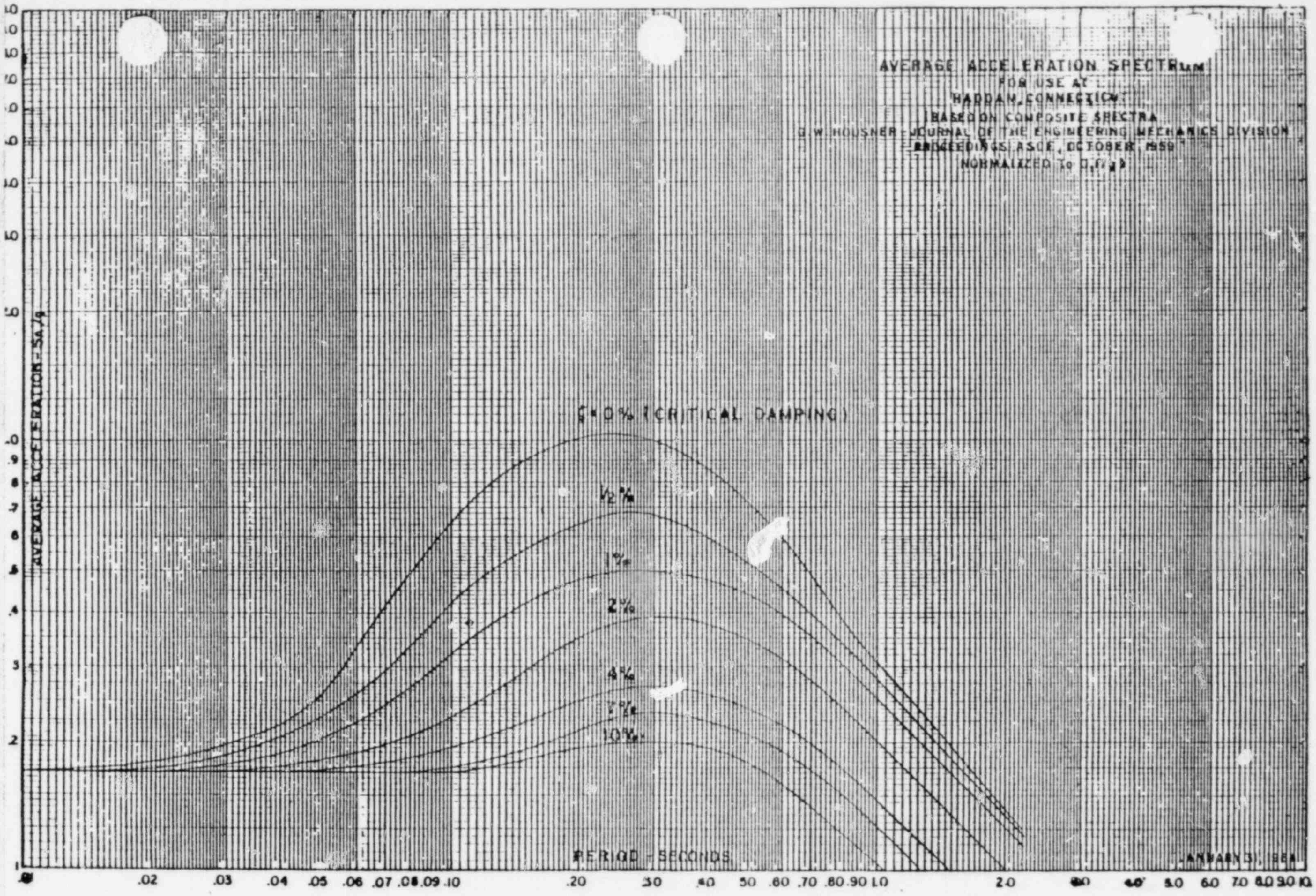

FIGURE 3-1. CONNECTICUT YANKEE GROUND RESPONSE SPECTRA

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

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Final ?

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

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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.

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

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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.

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 structual 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.

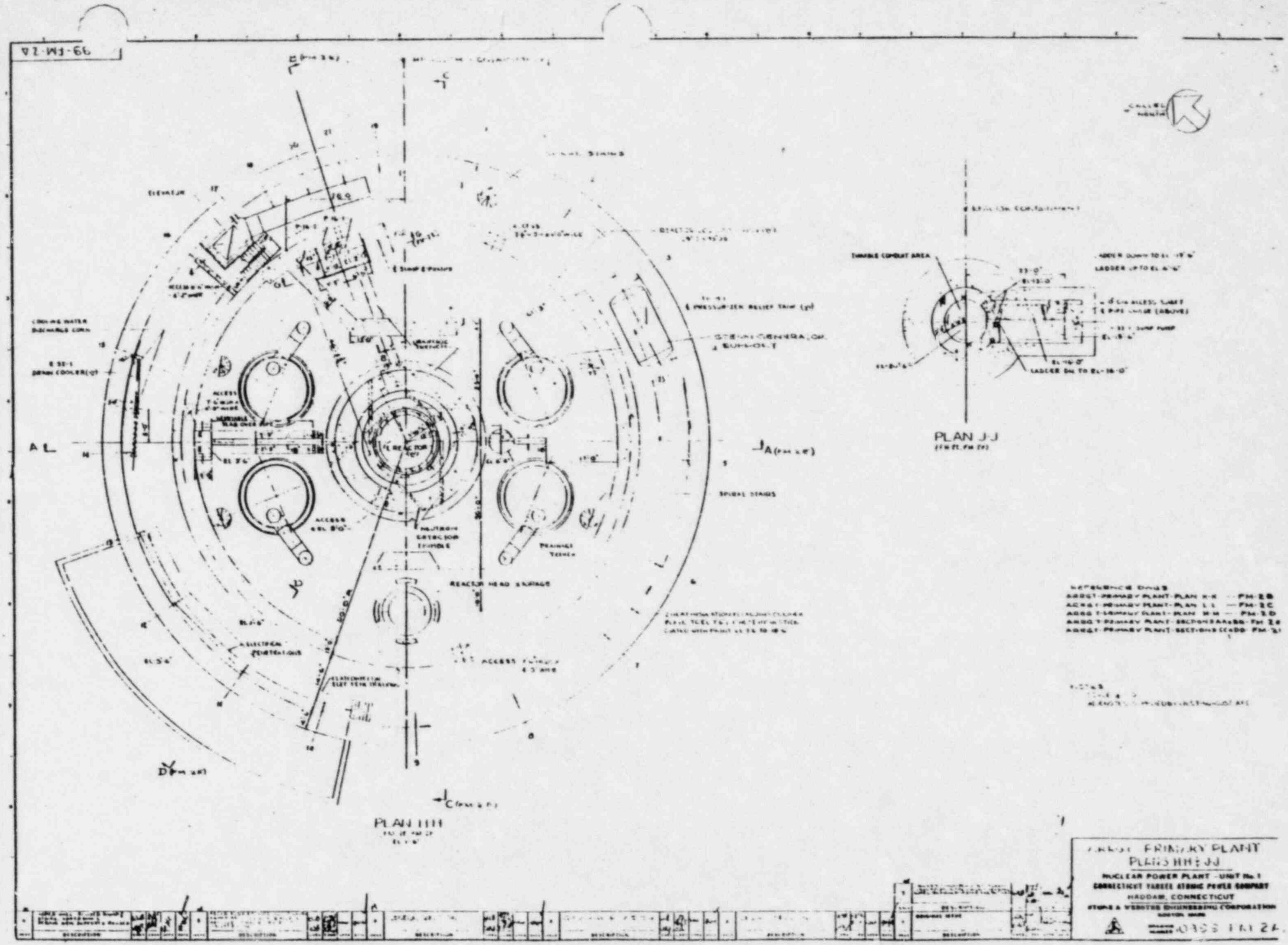

FIGURE A-1. REACTOR CONTAINMENT BUILDING PLAN, EL. 1'-6" (BASE SLAB)

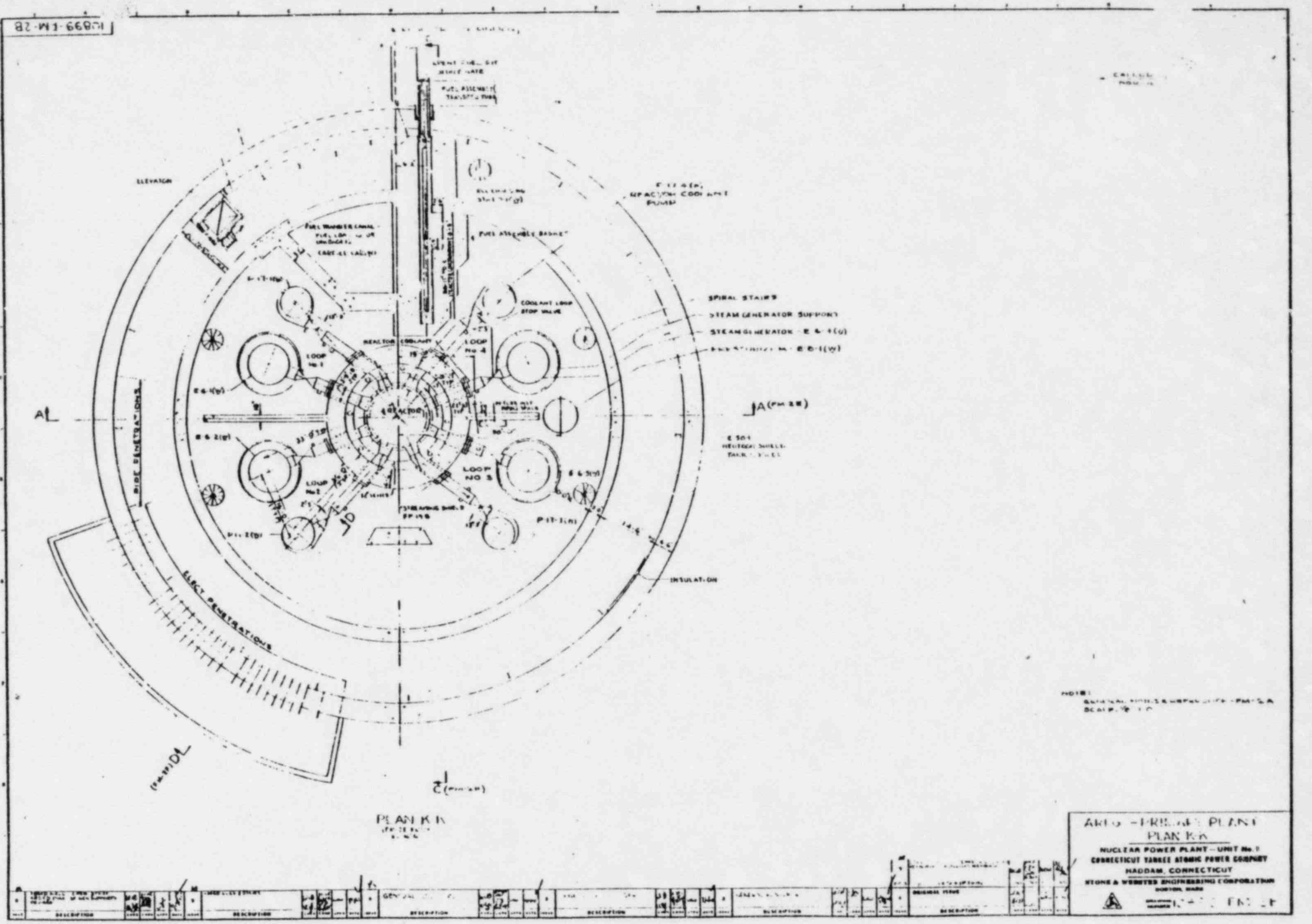

FIGURE A-2. REACTOR CONTAINMENT BUILDING PLAN, EL 16'-0"

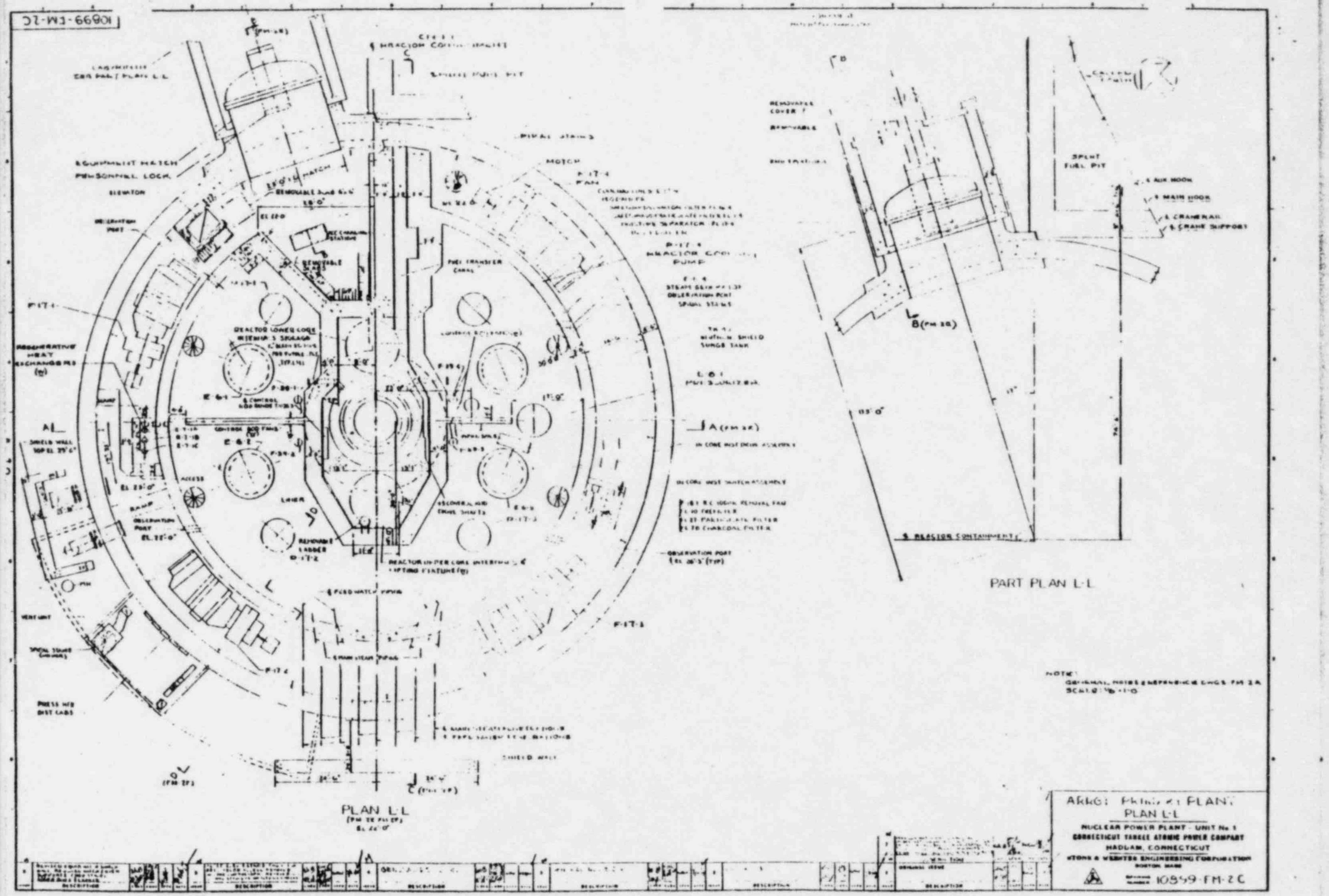

FIGURE A-3. REACTOR CONTAINMENT BUILDING PLAN, EL. 22'-0"

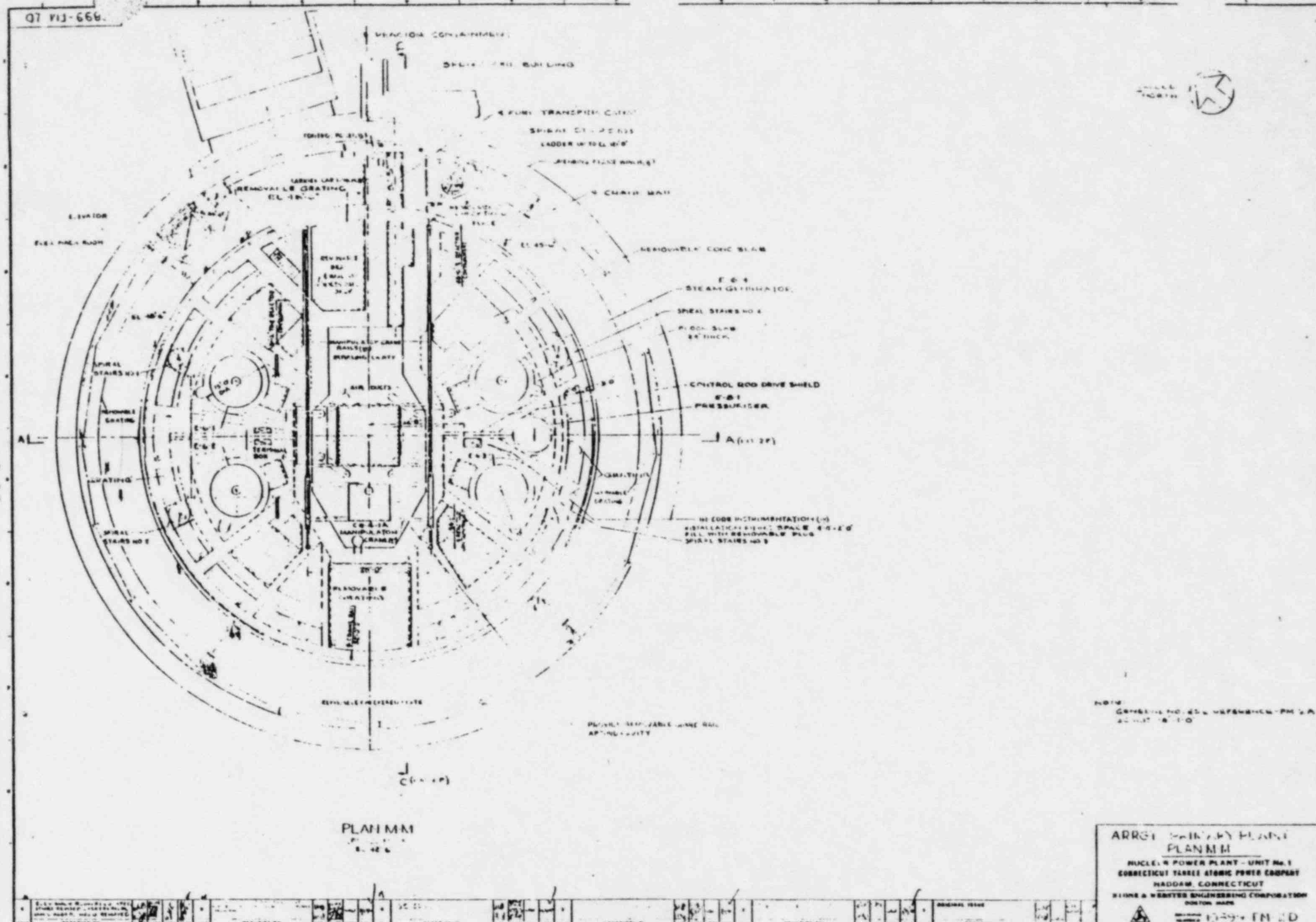

FIGURE A-4. REACTOR CONTAINMENT BUILDING, EL. 48'-6"

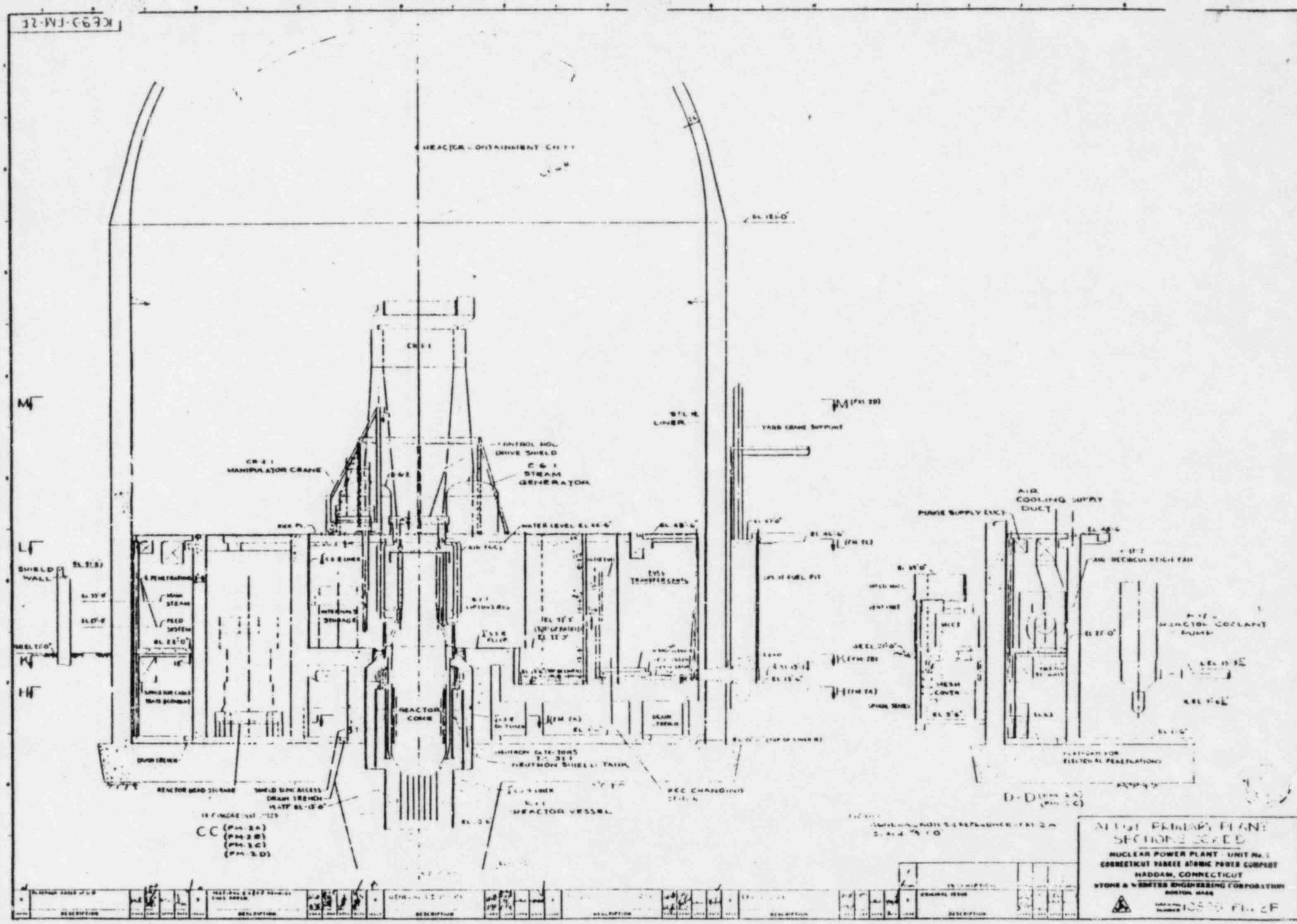

FIGURE A-5. REACTOR CONTAINMENT BUILDING N-S SECTION

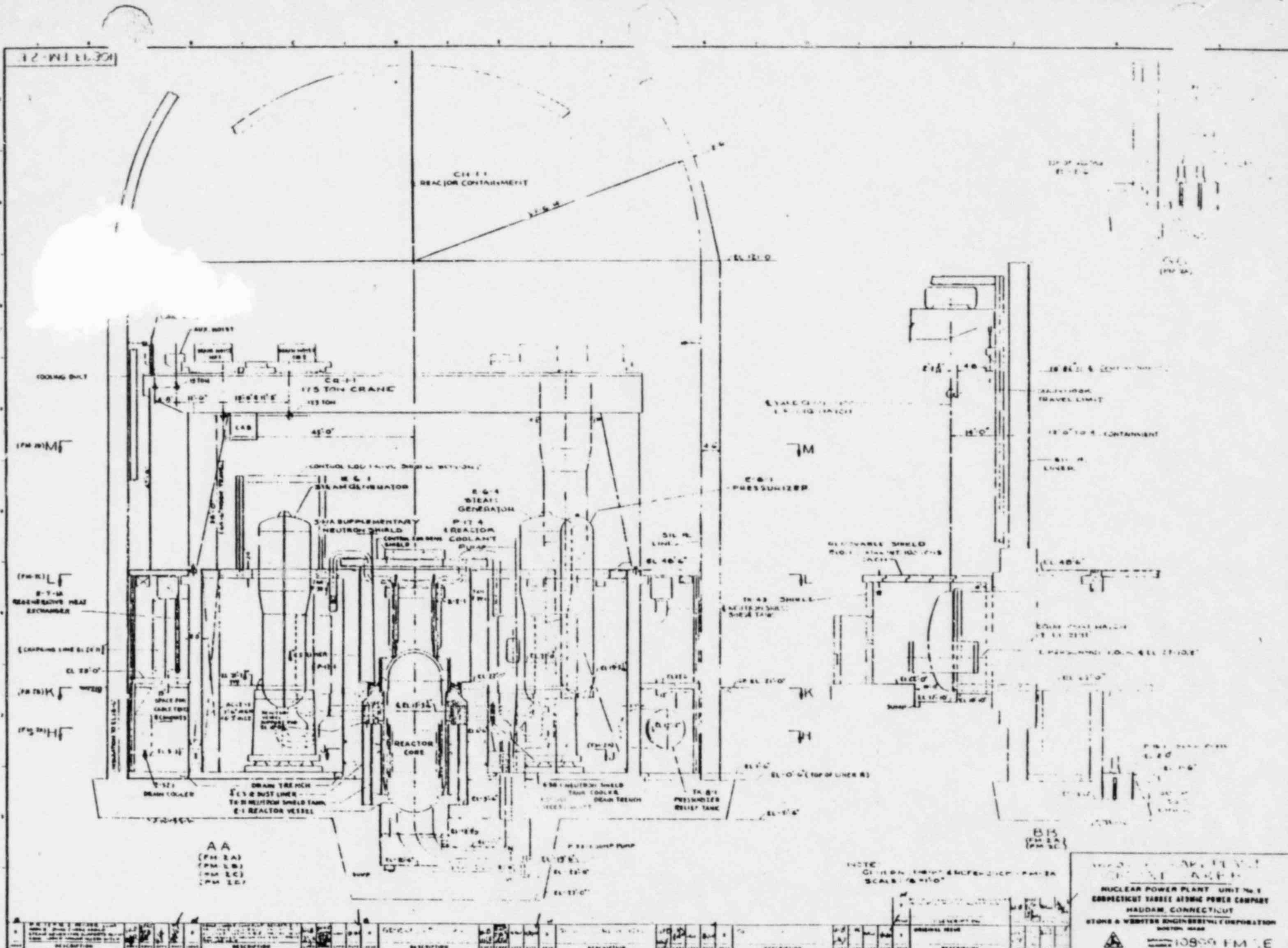

FIGURE A-6. REACTOR CONTAINMENT BUILDING E-W SECTION

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.