February 25, 1983

Docket No. 50-213
LS05-83- 02-049

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

Dear Mr. Counsil:

SUBJECT: SEP TOPIC III-6, SEISMIC DESIGN CONSIDERATIONS
HADDAM NECK PLANT

Enclosed is a copy of the staff's safety evaluation report for the seismic design of the Haddam Neck Nuclear Power Plant. Since the staff has not yet received the final seismic reanalysis reports from you, this safety evaluation was based on preliminary analysis results, several working-level meetings between the staff and CYAPCo and their consultants, and CYAPCo's comments on the staff's draft safety evaluation report which was issued on October 4, 1982. Therefore, the conclusions presented in this report may be revised should new information be provided in the final CYAPCo analysis reports.

In the enclosed safety evaluation, the staff concluded that Haddam Neck plant was generally capable of withstanding postulated seismic loads, except for certain portions of structures, piping systems and mechanical and electrical equipment (including supports). Additional information and analyses from CYAPCo are necessary to enable the staff to complete the seismic design review. These open issues will be evaluated and resolved during the integrated assessment.

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

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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,

Original signed by:

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

Enclosure:
As stated

cc w/enclosure:
See next page

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Docket No. 50-213
Haddam Neck

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SYSTEMATIC EVALUATION PROGRAM

TOPICS III-6 AND III-11

HADDAM NECK PLANT

TOPICS: III-6, Seismic Design Considerations
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 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 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 (References 1 and 2), the licensee, Connecticut Yankee Atomic Power Company was requested 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.

A draft Safety Evaluation Report (SER) was issued on October 4, 1982 (Reference 13). The licensee provided its comments (Reference 14) on the draft SER. The staff's evaluation of those comments have been incorporated into this report. Open items identified in the conclusion (Section V) will be addressed in the integrated assessment to determine any follow-up actions.

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 (Reference 3).

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

III. RELATED TOPICS AND INTERFACES

The related SEP topics to the review of seismic design considerations and component integrity are Topic 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 (References 1 and 2) were issued to request 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.

The staff's 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 analyses 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 (Reference 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 reanalysis 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 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 (Reference 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 (Reference 4) program conducted by the staff and its consultant, Lawrence Livermore National Laboratory (LLNL), the site specific ground spectrum, which is acceptable to the staff as the input for the seismic reevaluation of the Haddam Neck plant, was developed. This spectrum and its supporting documentation was forwarded to the licensee through NRC letters dated August 4, 1980 (Reference 1) and June 17, 1981 (Reference 5). In these letters, the staff also encouraged the licensee to develope its own site specific ground response spectrum. In Figure 1, a comparison was made for the spectrum developed by the staff and the spectrum developed by the licensee (Reference 6). According to the staff's review, the site specific ground response spectrum developed by the licensee is 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 (References 1 and 2), the licensee provided its basis for continued operation of the Haddam Neck plant on September 15, 1980 and June 11, 1981 (References 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 (Reference 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 (Reference 9) and September 15, 1980 (Reference 6). The review of this program plan was performed and discussed with the licensee and its consultants through NRC letter dated January 19, 1982 (Reference 10).

4. Staff's Review 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, non-linear 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 seismic reevaluation criteria and scope through the letter dated August 5, 1980 (Reference 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 are acceptable for reevaluation of the plant facility (safety-related structures, systems and components). The detail of the criteria review and bases for acceptance of differences from the criteria listed in Section II, 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 seventy-one (71) stress problems), twenty-nine (29) sampled equipment items, and two (2) field erected tanks were 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 - 18 mechanical equipment items and 13 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 refueling water storage tank (RWST).

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 (Reference 11) and the results confirmed that the analysis results 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) 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.

(iii) No discussion was given to the percent of modal mass participating in the draft report. This information should be provided 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) Verification that the seismic stresses under the postulated SSE loading in the floor slab are close to yield to justify higher damping ratios used.

(ii) 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 completely 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.

(f) Pipe Gallery

Overstress conditions were predicted at the interfaces of the pipe gallery between containment shell and the primary auxiliary building (PAB). The licensee has not proposed corrective actions for the modifications.

The details of structural review are found in Attachment 2. According to the licensee (Reference 12), all structural open items, except Auxiliary Feedwater Pump House, discussed above will be provided in its final structural analysis reports. These final analysis reports are scheduled for completion in February 1983. As far as the auxiliary feedwater pump house, the reevaluation is scheduled for completion by August 31, 1983.

7. Review of Primary Reactor Coolant Loop Systems

The staff's review of primary reactor coolant loop (RCL) systems and attached major components was based upon 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 consultants 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 supports, (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 are acceptable. However, some open items listed below were identifed 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.

(g) Complete the analyses of Reactor Coolant Pump and Reactor Internals and upgrade, if necessary.

The details of review discussed above are found in Attachment 3. As stated in the licensee's January 27, 1983 letter (Reference 12), the open items identified above will be addressed in the final analysis report. This final report is scheduled for completion in February 1983.

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. The staff's review of these systems as well as their supports was based on the information presented in the working-level review meetings, the licensee's responses to the open items identified in the review meetings (Reference 12), and the licensee's comments on the draft SER (Reference 13). As discussed in Attachment 3, the licensee's analysis 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. Some pipe supports were identified to be upgraded. As indicated by the licensee (Reference 12), the support modifications in accessible areas were scheduled to begin around October 15, 1982. However, the completion of this effort is not expected before June 30, 1983. For areas considered inaccessible during normal plant operation, support modifications are expected to be completed during the refueling outage scheduled for the first quarter of 1983. In addition, 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)

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 judgment. A total of 18 mechanical and 13 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 during discussions in the working-level review meetings and licensee's response to the open items identified in the review meetings. The staff concludes that the criteria (analysis and performance criteria), modelling techniques, analysis methods and approach as well as results obtained for these four equipment items are acceptable. However, this condition was based on a very limited sample. Further staff review should be performed when the reevaluation is completed by the licensee. The details of the staff's review are found in Attachment 3.

Instrument and control, to assure that adequate parameters are available to the operators are not included in the scope of the reevaluation. They will be covered by NRC Unresolved Safety Issue (USI) A-46, "Seismic Qualification of Equipment in Operating Plants."

Air systems and valve operators necessary to insure that necessary safety functions are met were not included in the scope of the licensee's program. It is the staff's position that the licensee should demonstrate by appropriate analysis and modify, where necessary, the air systems and valve operators to assure structural integrity.

For the qualification of electrical cable trays, the licensee proposed to quatify the evaluation method by testing through the SEP Owners Group Program and then apply the results to their plant.

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 will 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 plant will be developed. The ongoing Owners Group program for equipment qualification will be considered in the development of the USI A-46 criteria.

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 meeting. The criteria, analysis technicques, and results are acceptable to the staff. However, modifications are required for the anchorage systems of both tanks. The licensee agreed to complete the installations of these modifications by June 30, 1983 (Reference 12).

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

With the exception of the following items which 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, all safety-related structures are considered to be capable of withstanding the postulated SSE loads:

A. Interfaces of the pipe gallery between containment shell and primary auxiliary building.

B. Radial beams at the operating floor of containment internal structures.

C. Steel bracings in Turbine-Service Building.

D. Complete the analyses of Auxiliary Feedwater Pump House by August 31, 1983 and upgrade, if necessary.

E. Justification of using high structural damping ratios when very low stress level in structures was identified.

F. Identification of the percent of modal mass participating in the dynamic responses of structures (Screenwell house and primary auxiliary building).

G. Inclusion of torsional effects when the primary auxiliary building instructure response spectra were generated.

Primary Reactor Coolant Loop Systems

As a result of the staff's review, the criteria, modelling techniques, analysis methods and load combinations used for the reevaluation are considered to be acceptable. However, open items listed below were identified and additional information should be provided:

A. Justification of high damping ratios used for piping and component analyses.

B. Criteria for the evaluation of component support systems.

C. Justification for omitting the impact between the steam generator and its lower supports.

D. Buckling evaluation of component supports (reactor neutron shield tank, pressurizer supporting truss, and steam generator support skirt).

E. Complete the analyses of Reactor Coolant Pump and Reactor Internals and upgrade, if necessary.

Balance of Plant Piping Systems

Based on a detailed audit review of fifteen (15) piping analyses, the staff concludes that generally the criteria, modelling techniques, analysis methods, and results are acceptable. Some supports were identified to be upgraded. In addition, information should be provided for the following items:

A. Justification of modelling techniques applied for the case when the piping penetrating through several walls and floors.

B. Criteria used for the evaluation of pipe supports.

C. Information to demonstrate the validation of "Chart" methods if it has been used for piping analyses.

Safety-Related Equipment (Mechanical and Electrical)

Based on a detailed review of four (4) sample evaluations of equipment, the staff concluded that criteria, approach (sampling approach), modelling techniques, analysis methods, and results are acceptable.

This staff conclusion does not apply to air systems and valve operators which were not included in the original program scope. The licensee should complete the analysis of the structural integrity of air systems and air operated valves. A sampling technique is acceptable.

Qualification of electrical cable trays will be evaluated by applying the criteria developed through the SEP Owners Group Testing Program.

The qualification of equipment functional capability (including instrumentation and control systems) will be performed generically as part of the NRC USI A-46 program when guidelines are completed.

Field Erected Tanks

The criteria, analysis techniques and results are acceptable. However, modifications are required for the anchorage systems of refueling water storage tank and demineralized water storage tank.

VI. REFERENCES

1. Letter from D. G. Eisenhut, NRC to W. G. Counsil, NUCo, 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 Tin-Yu Lo, LLNL to W. T. Russell, NRC, dated May 25, 1982.
12. Letter from W. G. Counsil, CYAPCo to D. M. Crutchfield, NRC, dated September 27, 1982.
13. Letter from D. M. Crutchfield, NRC to W. G. Counsil, CYAPCo, dated October 4, 1982.
14. Letter from W. G. Counsil, CYAPCo to D. M. Crutchfield, NRC, dated January 28, 1983.

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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

~~8005281148~~
PDR

EDAC

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1. INTRODUCTION

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

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

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

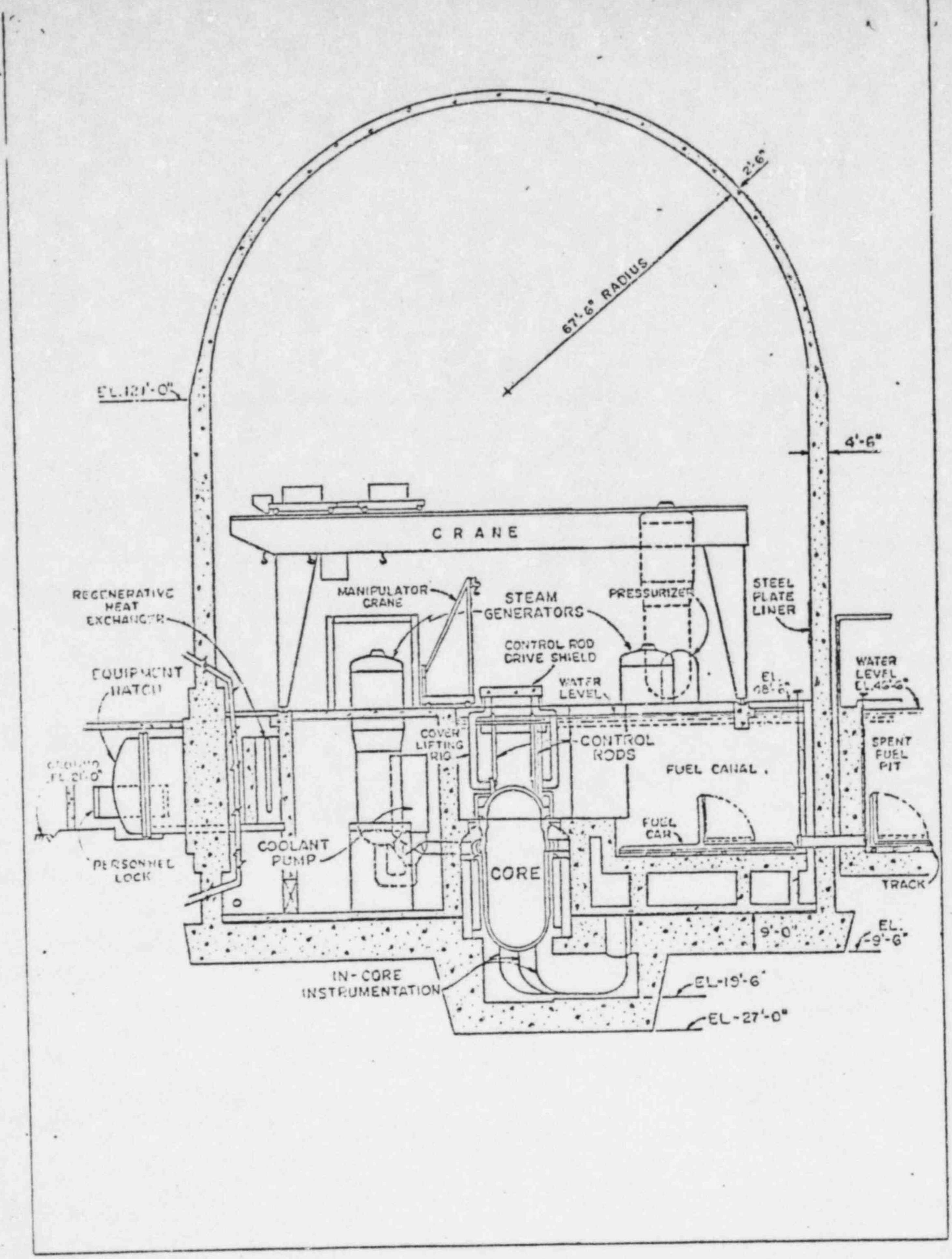

FIGURE 1-1. REACTOR CONTAINMENT BUILDING

2. GEOTECHNICAL

2.1 GEOLOGY

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

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

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

2.2 SEISMOLOGY

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

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

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

A fault trending east-northeast to west-southwest, known a 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 approximatel equal seismic probability, as approved by the International Conference of Building Officials in the Twenty-Eighth Annual Meetir for inclusion in the 1961 Edition of the Uniform Building Code, indicates that the area is near the boundary between Zones 1 and 2 (areas having earthquakes with hazards of minor magnitude).

TABLE 2-1

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

Date	Location	Approximate Epicenter Latitude North	Approximate Epicenter Longitude West	Approximate Area, Square Miles	Intensity, Ressi-Forel Scale
1791 (May 16)	East Haddam	41.5	72.5	35,000	8
1791 (Aug. 28)	East Haddam	41.5	72.5	—	*
1792	East Haddam	41.5	72.5	—	*
1794	East Haddam	41.5	72.5	—	*
1805	East Haddam	41.5	72.5	—	*
1827	New London	41.4	72.7	—	5
1837	Hartford	41.7	72.7	—	5
1840	South Connecticut	41.5	72.9	7,500	6
1858	New Haven	41.3	73.0	3,000	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

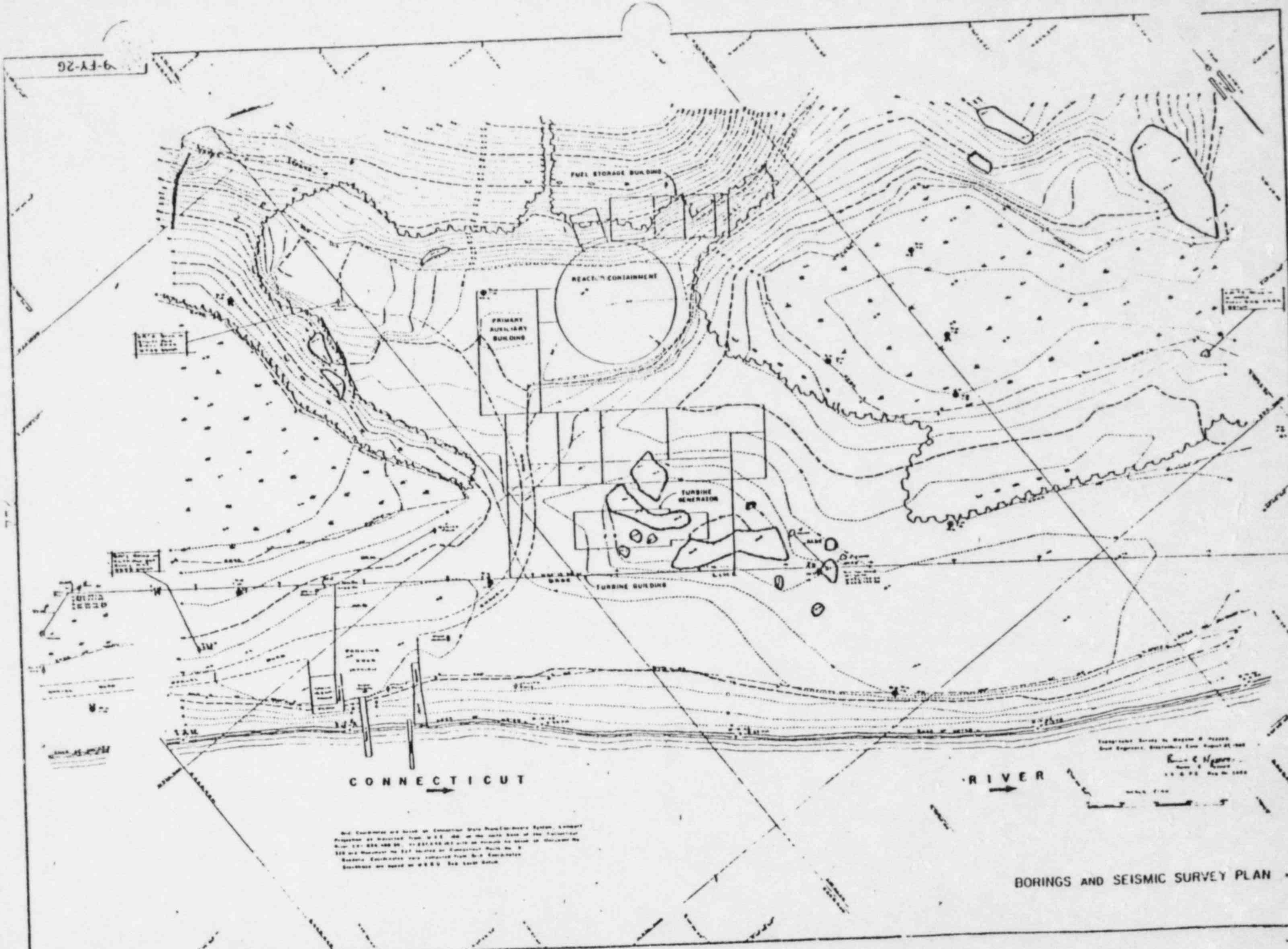

FIGURE 2.1 CONNECTICUT YANKEE SITE

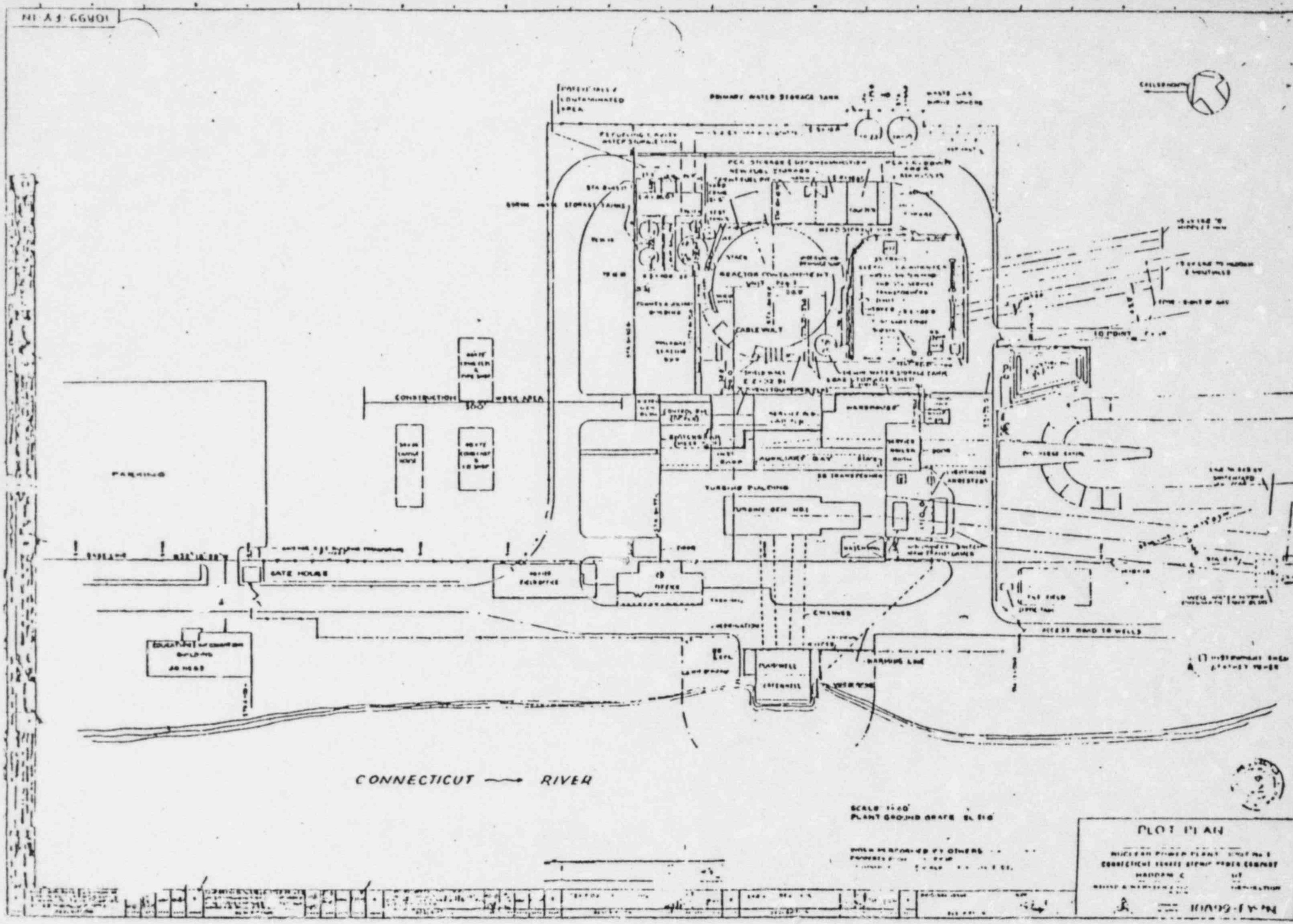

FIGURE 2-2. CONNECTICUT YANKEE PLOT PLAN

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

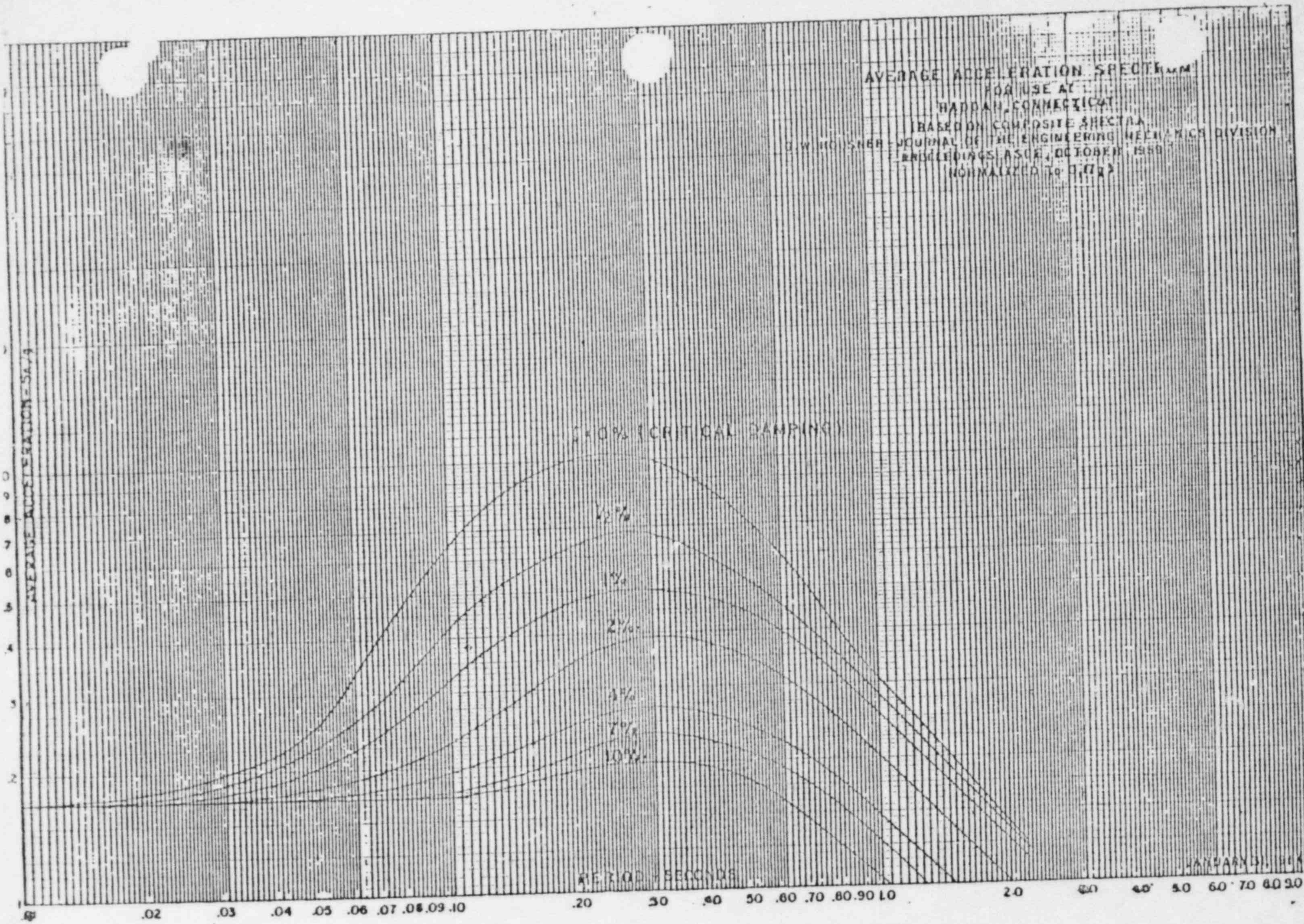

FIGURE 3-1. CONNECTICUT YANKEE GROUND RESPONSE SPECTRA

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 Yiel
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% Rein. Conc. Frame Struct. 5% Bolted Steel Frame Struct. 2.5% Welded Steel Frame Struct. 1.0%	OBE 4% SSE 7% OBE 4% SSE 7% OBE 4% SSE 7% 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 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
	Other Safety Systems (E=0.17g earthquake): Oper. or Accident + E < yield stress	
	Main Steam Piping (E=0.03g earthquake: Oper. Loads + E < Working Stress	
	All Other Structures & Elements (E=0.03g earthquake): no loss of Function	
18. Simplified Design Methods	1 DOF Systems Directly from Ground Response Spectrum	Floor Spectra Req'd. S.R.P. 3.7.2
	Complex Systems from Peak of Ground Response Spectrum	Peak of Floor Spectrum S.R.P. 3.7.2, 3.7.3

REFERENCES

Journal ?

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

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

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

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

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

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

APPENDIX A

SEISMIC DESIGN OF SELECTED PLANT COMPONENTS

(From Reference 1)

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

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

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

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

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

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

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

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

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

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

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

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

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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.

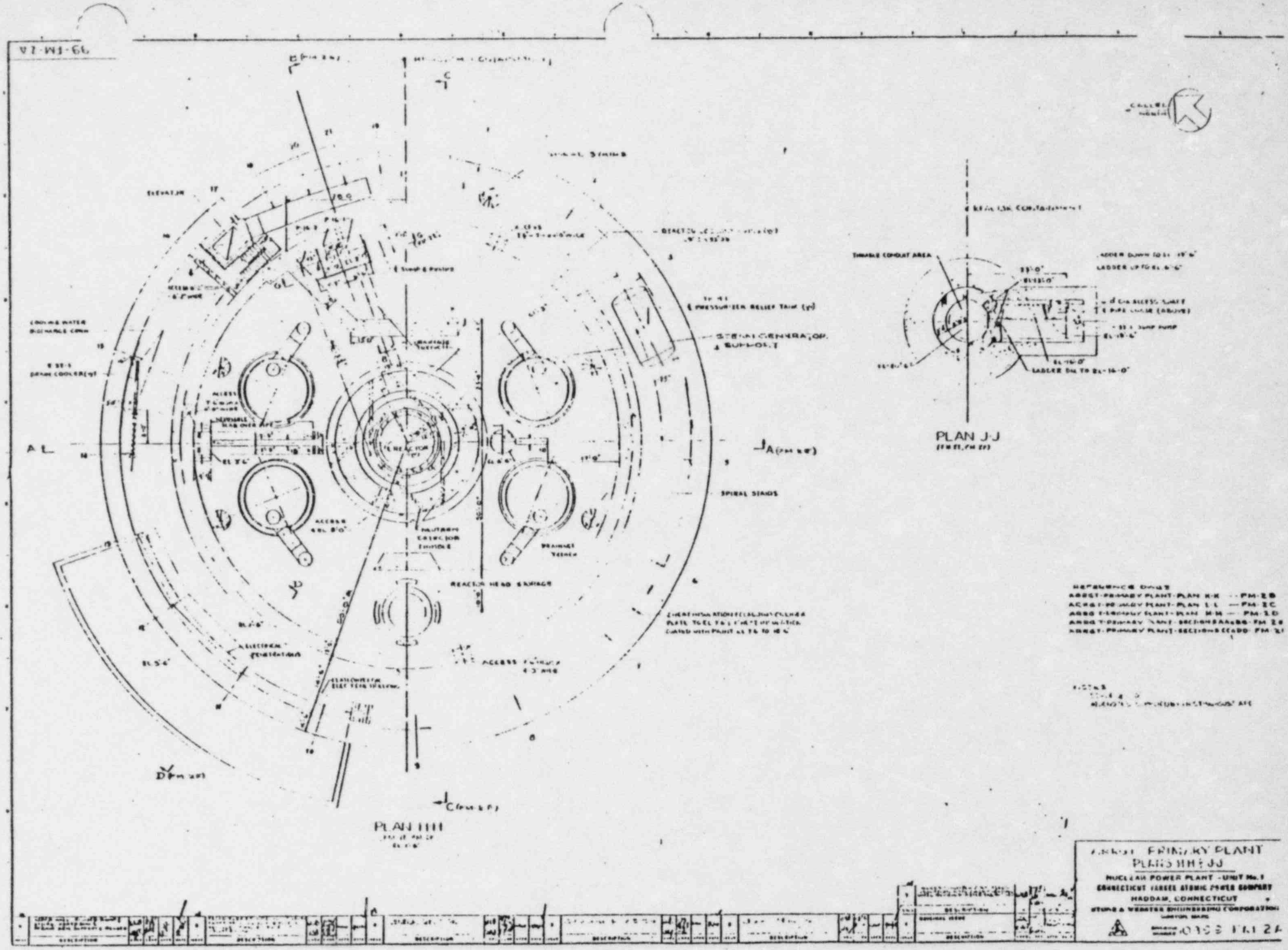

FIGURE A-1. REACTOR CONTAINMENT BUILDING PLAN, EL. 1'-6" (BASE SLAB)

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.

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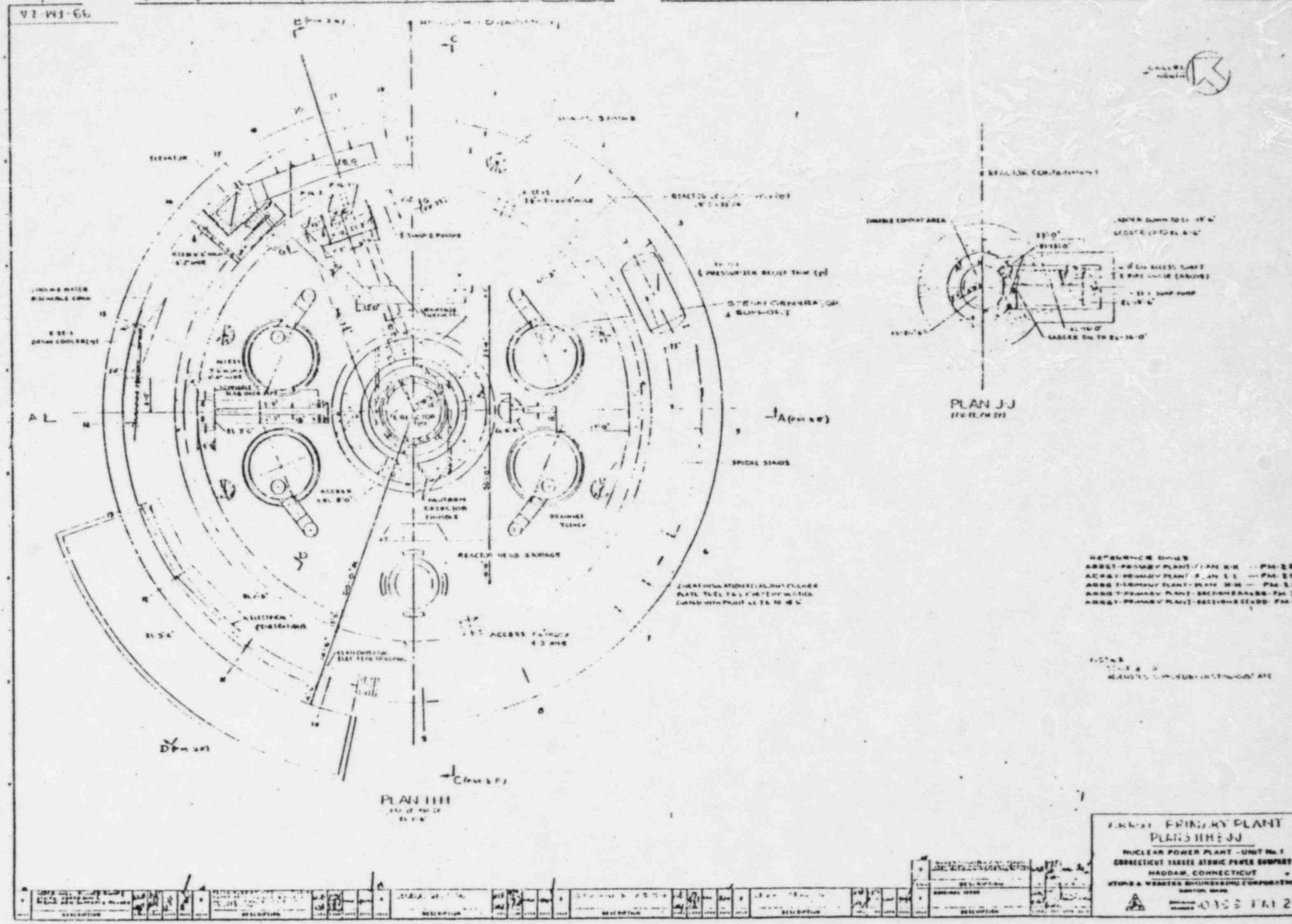

FIGURE A-1. REACTOR CONTAINMENT BUILDING PLAN, EL. 1'-6" (BASE SLAB)

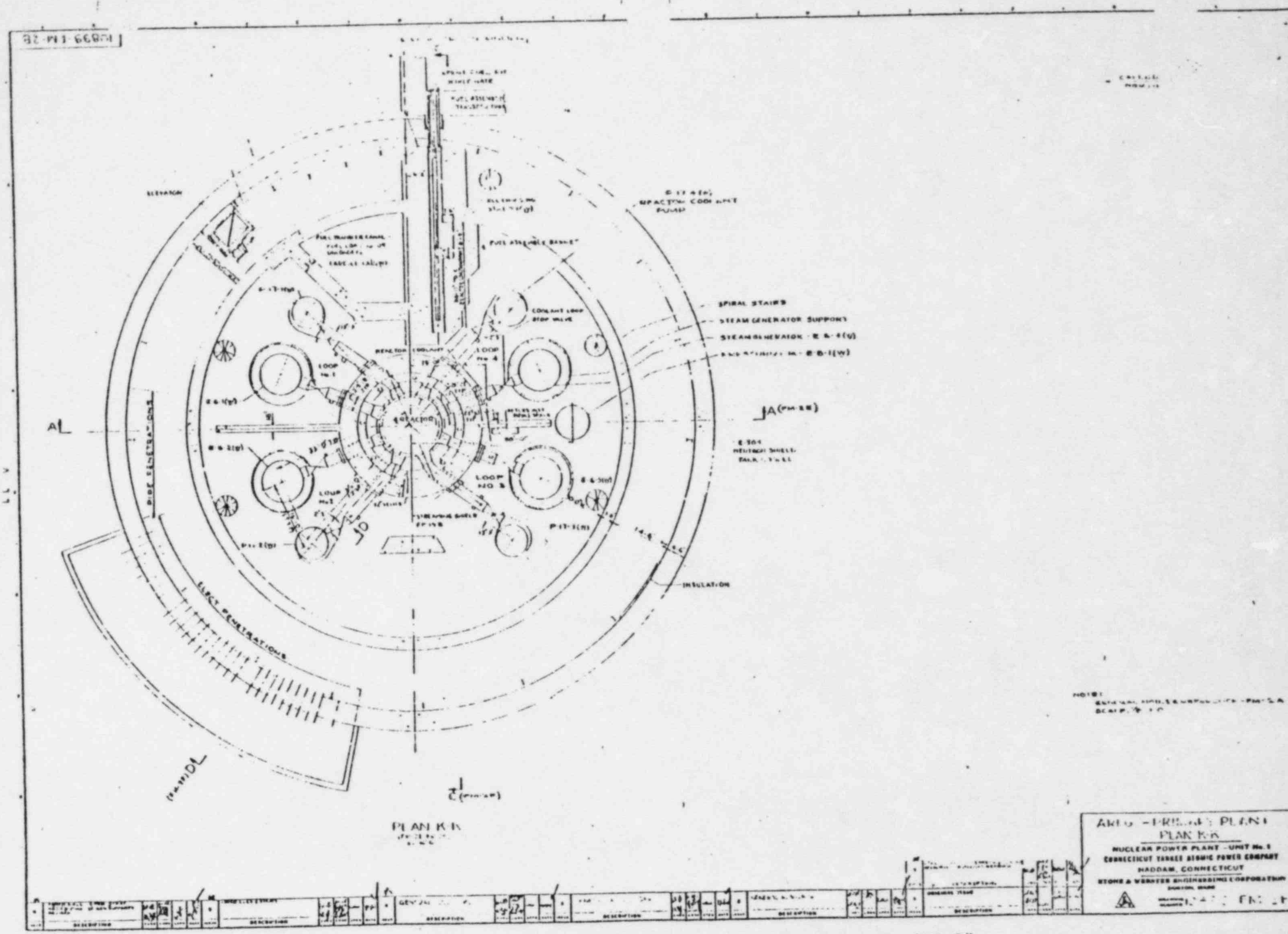

FIGURE A-2. REACTOR CONTAINMENT BUILDING PLAN, EL 16'-0"

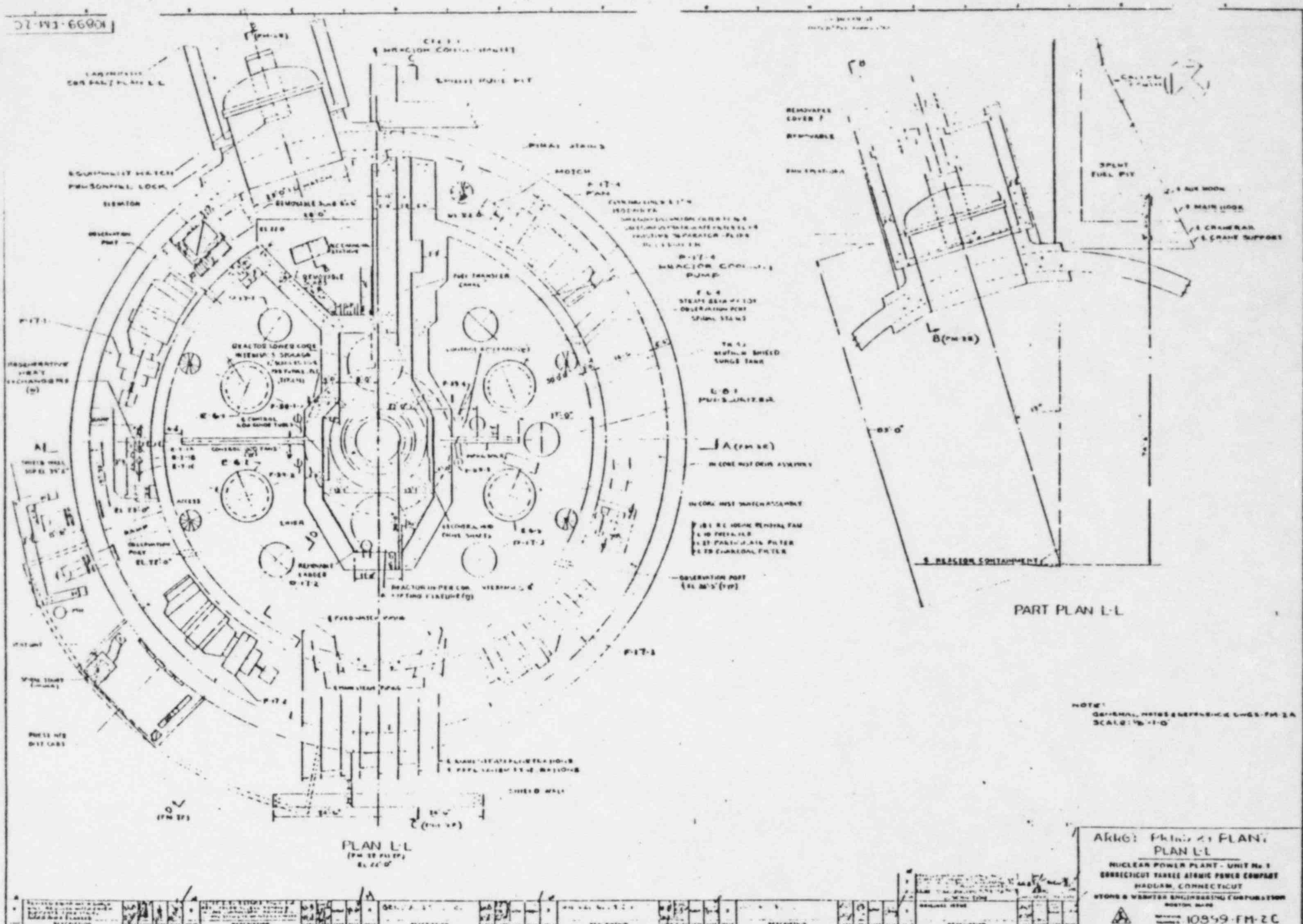

FIGURE A-3. REACTOR CONTAINMENT BUILDING PLAN, EL. 22'-0"

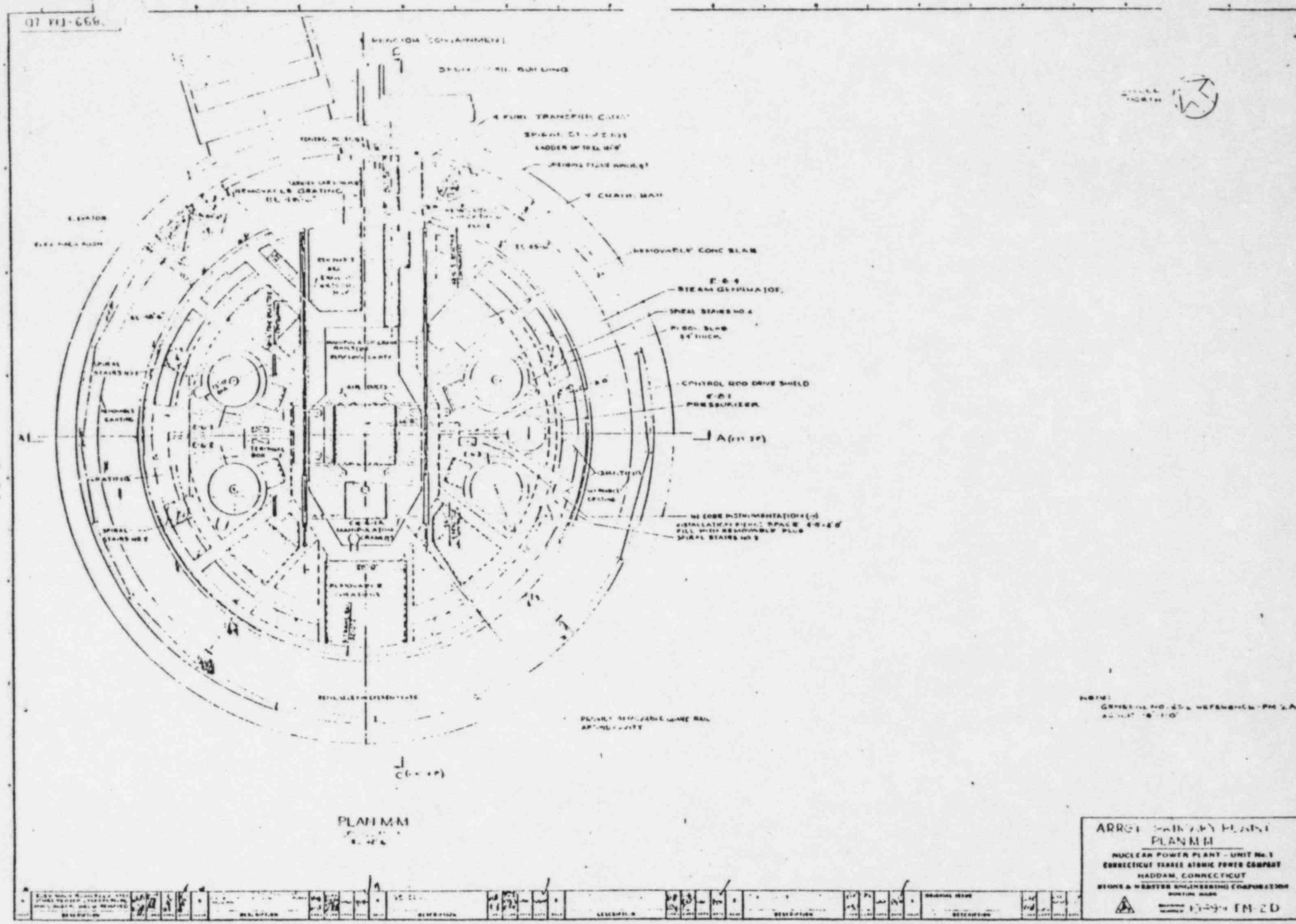

FIGURE A-4. REACTOR CONTAINMENT BUILDING, EL. 48'-6"

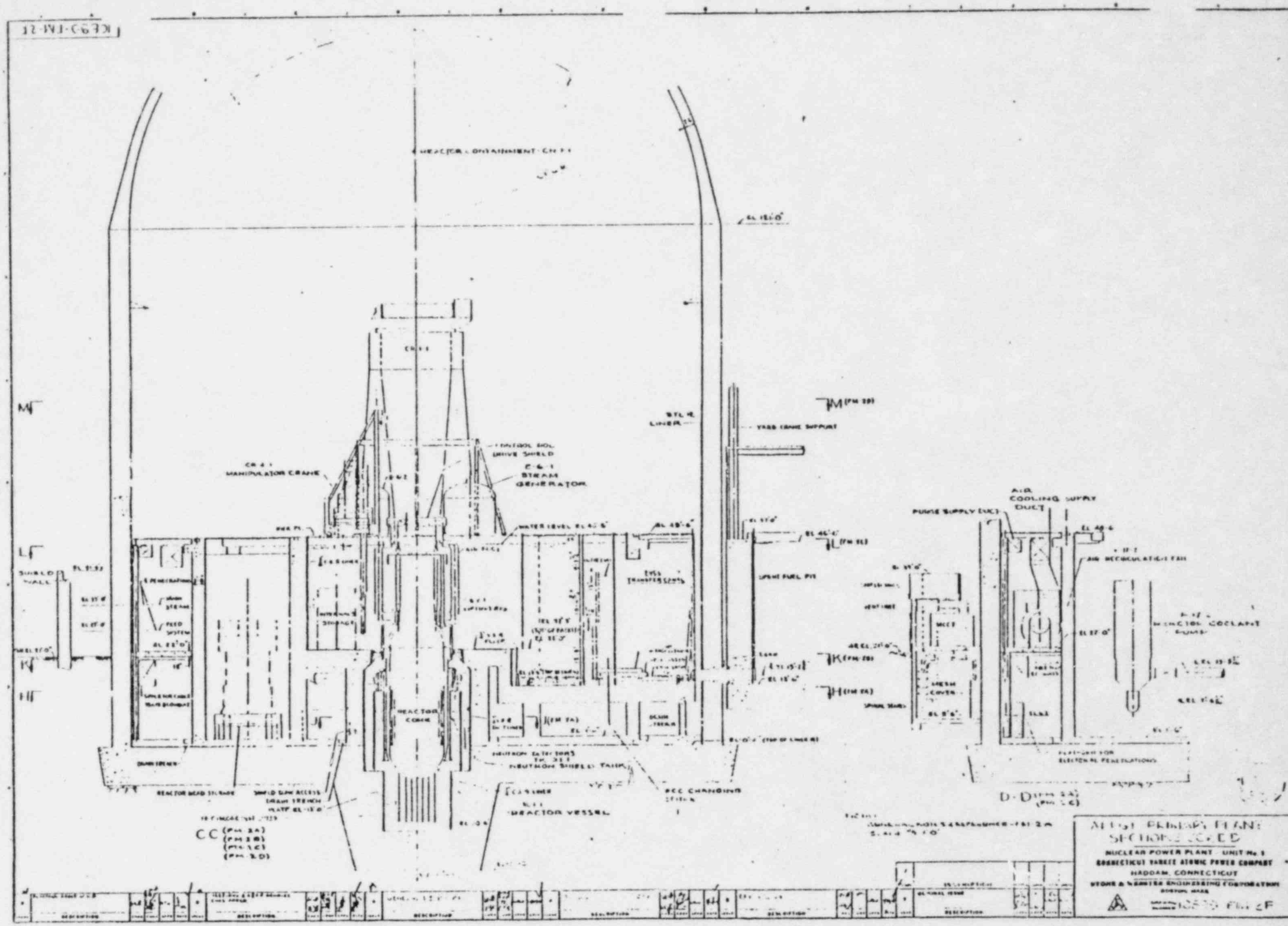

FIGURE A-5. REACTOR CONTAINMENT BUILDING N-S SECTION

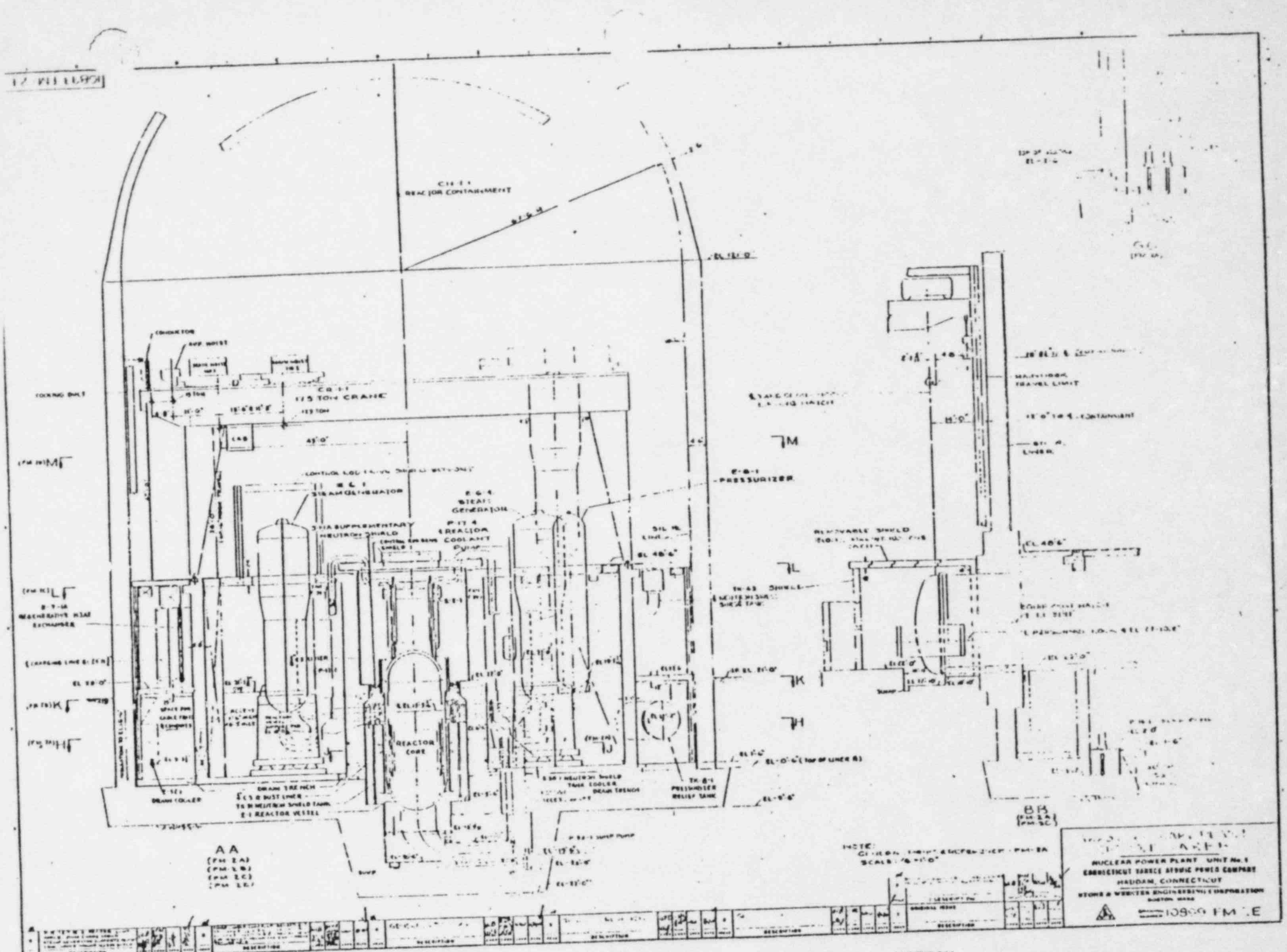

FIGURE A.6 REACTOR CONTAINMENT BUILDING E-W SECTION

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.

ATTACHMENT 3

EGG-EA-6016

August 1982

TECHNICAL EVALUATION REPORT

HADDAM NECK PLANT SEISMIC DESIGN

T. L. Bridges
S. L. Busch

D. K. Morton
T. R. Thompson

Prepared for the
U.S. NUCLEAR REGULATORY COMMISSION
Under DOE Contract No. DE-AC07-76ID01570
FIN No. A-6426

EGG-EA-6016

August 1982

Technical Evaluation Report
Haddam Neck Plant Seismic Design

Applied Mechanics Branch
Engineering Analysis Division

T. L. Bridges/S. L. Busch/
D. K. Morton/T. R. Thompson

RGRahl/

RCGuenzler/

8210120300
PDR

APPENDIX A

HADDAM NECK AUDIT PLAN FOR SEP SEISMIC QUALIFICATION
OF PIPING, MECHANICAL, AND ELECTRICAL EQUIPMENT

HADDAM NECK AUDIT PLAN FOR SEP SEISMIC QUALIFICATION OF PIPING, MECHANICAL, AND ELECTRICAL EQUIPMENT

I. Background

In October, 1977, the office of Nuclear Reactor Regulation (NRR) initiated Phase I of the Systematic Evaluation Program (SEP) to determine the margin of safety relative to current standards for eleven selected operating nuclear power plants and to define the nature and extent of retrofitting required to bring these plants to acceptable levels of safety if they are not already at these levels. Phase I of SEP involved Group I plants, where Phase II involves Group II plants, consisting of San Onofre 1, La Crosse, Big Rock Point, Yankee Rowe, and Haddam Neck. The review for seismic requalification of SEP Group II plants will be performed by two teams. One team consisting of NRC staff personnel and NRC consultants from Lawrence Livermore National Laboratory (LLNL) will evaluate the Group II plants' structures. A second team consisting of NRC staff personnel and NRC consultants from EG&G Idaho, Inc., will evaluate the Group II plants' piping, mechanical, and electrical equipment important to safety. This audit plan provides a description of how the SEP seismic requalification of Haddam Neck piping, mechanical, and electrical equipment important to safety will be reviewed.

II. Scope

The scope of review for the SEP seismic re-evaluation program will include the systems and components (including emergency power supply and distribution, instrumentation, and actuation systems) with the following functions:

1. The reactor coolant pressure boundary as well as the core and vessel internals. This should also include those portions of the steam and feedwater system extending from and including the secondary side of the steam generator up to and including the outermost containment isolation valve and connected piping for

all safety related systems up to and including the first valve that is either normally closed or is capable of automatic closure during all modes of normal reactor operation.

2. Systems or portions of systems that are required for safe shutdown as identified in the SEP safe shutdown review (SEP Topic VII-3). The system boundary includes those portions of the system required to perform the safety function and connected piping up to and including the first valve that is either normally closed or capable of automatic closure when the safety function is required.

3. Systems or portions of systems that are required to mitigate design basis events, i.e., accidents and transients (SEP Topics XV-1 to XV-24). The functions to be provided include emergency core cooling, post-accident containment heat removal, post-accident containment atmosphere cleanup, as well as support systems, such as cooling water, needed for proper functioning of these systems.

4. Systems and structures required for fuel storage (SEP Topic IX-1). Integrity of the spent fuel pool structure including the racks is nee . Failure of the liner plate due to the safe shutdown earthquake must not result in significant radiological releases, or in loss of ability to keep the fuel covered. Failure of cooling water systems or other systems connected to the pool should not permit draining of the fuel pool. Means to supply makeup water to the pool as needed must be provided.

For the Haddam Neck plant, the following systems, and components should be addressed:

1. Reactor Coolant System (RCS)

2. Portions of Main Steam System

3. Portions of Main Feedwater System

4. Portions of systems directly connected to the RCS up to and including isolation valves

5. Control Rod Drives

6. Auxiliary Feedwater System

7. Residual Heat Removal System (including ECCS recirculation mode)

8. Portions of Chemical and Volume Control System

9. Portions of Service Water System

10. High Pressure Safety Injection System

11. Low Pressure Safety Injection System

12. Containment Cooler System

13. Spent Fuel Pool and Makeup

As discussed previously, a "system" also includes the power supply, instrumentation and actuation systems.

III. <u>General Criteria and References</u>

The criteria contained in the following documents will be the bases used to evaluate the SEP seismic re-evaluation of Haddam Neck Plant piping, mechanical, and electrical equipment important to the plant's ability to safely withstand the effects of a postulated safe shutdown earthquake event.

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

2. Standard Review Plan, Sections 3.2, 3.7, 3.8, 3.9, 3.10.

3. Regulatory Guides, 1.29, 1.48, 1.60, 1.61, 1.89, 1.92, 1.100, 1.124, 1.130.

4. ANSI/IEEE Standard 344-1975.

5. ASME Boiler and Pressure Vessel Code Section III, 1980 Edition or subsequent.

6. AISC, "Manual of Steel Construction," Eighth Edition.

The intent of Phase II of SEP is to demonstrate that the structural integrity of the systems and components being re-evaluated will not be impaired when subjected to a postulated Safe Shutdown Earthquake (SSE) in combination with other normal design loadings. As a minimum, component primary stresses must be evaluated using current criteria provided in the above standards for Level D (faulted) service limits.

IV. Review Procedures

A. General

The review team (NRC and NRC consultants) will perform the review effort parallel with the licensee's seismic re-evaluation efforts. A minimum of three working level meetings among the review team, licensee, and licensee's consultants are anticipated. This method of review has been selected in order to expedite the review. The working level meetings will permit an exchange of information which will minimize formal written communication, thus expediting the program. One of the meetings will be conducted at the plant so the review team can perform a field inspection of the equipment being re-evaluated.

The review process will be accomplished in three steps. The first step will consist of the review team reviewing the details of the seismic re-evaluation program plan submitted by the licensee. A substantial portion of this review will be performed prior to the first working meeting. Any concerns the review team has with the program plan will be discussed and preferably resolved at the first working meeting.

The next step of the review will consist of review of analyses performed by the licensee or licensee's consultants. This review will be performed by one or more of the following methods: (a) The review team will perform a review of seismic re-evaluation analyses at the working meetings. (b) The review team will perform review of seismic re-evaluation analyses at their offices. These analyses will either be given to the review team at the working meetings or transmitted by mail to the review team upon completion. (c) The review team will perform independent analyses for some components and systems. Information necessary to perform these analyses will be supplied by the licensee at the working meetings or transmitted later. The depth of review of analyses will vary depending on the complexity of the item being evaluated. The analysis review guidelines are contained in Appendix A.

The third and final step of the review process will consist of the review team preparing and submitting a technical evaluation report (TER) which identifies the results of the seismic re-evaluation review.

B. Audit Meeting Agenda

As previously mentioned, the SEP will require working level meetings among the review team members, licensee, and licensee consultants to be held either at the plant or at licensee's engineering offices. For the meetings at the engineering offices, the following agenda is anticipated:

1. Detailed presentation of seismic re-evaluation program plan by licensee or licensee's consultants.[a]

2. Discussion and resolution of concerns which the review team has with the program plan.[a]

3. Presentation of licensee's progress towards completion of seismic re-evaluation program by licensee.

4. Presentation of anticipated schedule for completing program by licensee.

5. Summary presentation of seismic re-evaluation analyses results (include identification of systems and components which require retrofitting) by licensee.

6. Detailed review of completed seismic re-evaluation analyses for selected systems and equipment (include detailed review of required retrofits).

7. Exit briefing identifying acceptable areas of review and areas of concern requiring additional information to resolve by review team.

For the meeting at the plant, the following agenda is anticipated:

1. Presentation of licensee's progress towards completion of seismic re-evaluation program by licensee.

2. Presentation of anticipated schedule for completing program by licensee.

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a. Required at initial meeting only.

3. Summary presentation of seismic re-evaluation analyses results (include identification of systems and components which require retrofitting) by licensee.

4. Field inspection of selected equipment being re-evaluated by review team and licensee.

5. Detailed review of newly completed seismic re-evaluation analyses, by review team (include detailed review of required retrofits).

6. Exit briefing identifying acceptable areas of review and areas of concern requiring additional information to resolve, by review team.

V. Review Team Members

The SEP review team for Haddam Neck nuclear powr plant will consist of the following NRC and EG&G Idaho, Inc., personnel.

NRC

Thomas M. Cheng

EG&G Idaho, Inc.

Tom L. Bridges
Sheryl L. Busch
D. Keith Morton[a]
Tommie R. Thompson

---

a. First working meeting only.

VI. Review Schedule

The anticipated schedule for completing Phase II of SEP for Haddam Neck nuclear power plant is as follows:

1. First working meeting — Week of 04-19-82
2. Plant visit — Not Scheduled
3. Final working meeting — Not Scheduled
4. Complete TER — 08-31-82

APPENDIX A ANALYSIS REVIEW GUIDELINES

The following is a list of guidelines to be used in reviewing analyses for the SEP Group II Plants. Although the list may not be all inclusive, it does provide the areas of interest pertaining to the SEP review.

I. Analysis Audit Format (Piping)

1. What computer codes were used in the analyses?

    a. How were the above computer codes verified?

2. Is the proper input forcing function being utilized?

    a. If response spectra method is used:

        (1) Is correct spectra and damping utilized?

        (2) Have sufficient modes been used to adequately describe system response?

        (3) Is spectra properly broadened?

        (4) Do system frequencies straddle any peaks?

    b. If time history method is used:

        (1) Is sufficient system response achieved?

        (2) Is an adequate time step utilized?

        (3) Is proper damping utilized?

c. If static equivalent method is used:

(1) Is justification provided for performing a static equivalent analysis?

(2) How was required level of input determined?

3. Has the piping system been properly modeled?

a. Have valves been properly modeled including any eccentricity?

b. Has adequate mass point spacing been utilized?

c. Are adjacent element length ratios reasonable?

d. Have all significant branch piping systems been included?

e. Have all supports been specified with correct imposed loads (if any), direction and stiffness?

f. Have supports with significant nonlinear characteristics been properly handled?

g. Have correct pipe sizes, geometry, thicknesses, and uniform weights been specified?

h. Have correct design and operating pressure and temperature data been specified?

4. Has the piping system been evaluated against proper criteria?

a. Has a proper minimum thickness check been performed?

b. Have excessive deflections been considered?

c. Have proper stress intensification factors been utilized?

d. Have proper load combinations been analyzed?

e. Have proper allowable stress limits been selected in order to assure the required operation of the piping?

f. Were standard or nonstandard components used?

g. What criteria were used in evaluating adequacy of supports?

II. Analysis Audit Format (Mechanical Equipment)

1. Is the equipment rigid or flexible?

   a. How were the natural frequencies determined?

   b. If flexible, is its response single-directional or multi-directional?

   c. If flexible, is its response at one predominant frequency or at several frequencies?

2. What type of analysis was performed?

   a. Static g level

      (1) How was required level of input determined?

   b. If response spectra method is used:

      (1) Is correct spectra and damping utilized?

      (2) Is sufficient system response achieved?

(3) Is spectra properly broadened?

(4) Do system frequencies straddle any peaks?

(5) How were directional components of input applied (combined)?

c. If time history method is used:

(1) Is sufficient system response achieved?

(2) Is an adequate time step utilized?

(3) Is proper damping utilized?

(4) How were directional components of input applied (combined)?

d. If testing was used for requalification:

(1) What type of test was performed?

(2) What justification is provided for the type of test used?

(3) How were system natural frequencies determined?

(4) How was the required response spectra (RRS) determined?

(5) How does the test response spectra (TRS) compare to the RRS?

(6) What g level was used in the test?

(7) Were support and boundary conditions, including anchor bolts, properly simulated in the test?

(8) How was functional operability verified during the test?

(9) What criteria were used in evaluating the adequacy of the test results?

3. What computer codes were used in the analyses?

   a. How were the above computer codes verified?

4. Has the system been properly modeled?

   a. Has adequate mass point spacing and distribution been used?

   b. Have all supports and boundary conditions, including anchor bolts, been properly modeled?

   c. Have significant nonlinear effects been properly handled?

5. Has the system been evaluated against proper criteria?

   a. Have the proper load combinations been analyzed?

   b. Have proper stress intensities been evaluated?

   c. Have deflections been considered?

   d. Have proper allowable stress limits been selected?

   e. How were computer output responses combined (directional and modal)?

III. Analysis Audit Format (Electrical Equipment)

1. Is the equipment rigid or flexible?

   a. How were the natural frequencies determined?

   b. If flexible, is its response single-directional or multi-directional?

   c. If flexible, is its response at one predominant frequency or at several frequencies?

2. What type of analysis was performed?

   a. Static g level

      (1) How was required level of input determined?

   b. If response spectra method is used:

      (1) Is correct spectra and damping utilized?

      (2) Is sufficient system response achieved?

      (3) Is spectra properly broadened?

      (4) Do system frequencies straddle any peaks?

      (5) How were directional components of input applied (combined)?

   c. If time history method is used:

      (1) Is sufficient system response achieved?

(2) Is an adequate time step utilized?

(3) Is proper damping utilized?

(4) How were directional components of input applied (combined)?

d. If testing was used for requalification:

(1) What type of test was performed?

(2) What justification is provided for the type of test used?

(3) How were system natural frequencies determined?

(4) How was the required response spectra (RRS) determined?

(5) How does the test response spectra (TRS) compare to the RRS?

(6) What g level was used in the test?

(7) Were support and boundary conditions, including anchor bolts, properly simulated in the test?

(8) How was functional operability verified during the test?

(9) What criteria were used in evaluating the adequacy of the test results?

3. What computer codes were used in the analyses?

a. How were the above computer codes verified?

4. Has the system been properly modeled?

    a. Has adequate mass point spacing and distribution been used?

    b. Have all supports and boundary conditions, including anchor bolts, been properly modeled?

    c. Have significant nonlinear effects been properly handled?

5. Has the system been evaluated against proper criteria?

    a. Have the proper load combinations been analyzed?

    b. Have proper stress intensities been evaluated?

    c. Have deflections been considered?

    d. Have proper allowable stress limits been selected?

    e. How were computer output responses combined (directional and modal)?

APPENDIX B HADDAM NECK SEP PROGRAM PLAN REVIEW SUMMARY

At the first SEP audit meeting for the Haddam Neck plant, the licensee and its engineering consultants provided a detailed presentation of the Haddam Neck plant SEP program plan for seismic re-evaluation of piping, mechanical, and electrical equipment important to safety. In general, the program plan presented satisfies the SEP requirements reasonably well with a few exceptions. For mechanical and electrical equipment, the following open items must be addressed by the licensee:

1. Provide sample calculations to justify the damping compatible with the stress levels used in the equipment evaluation.

2. The NRC position is that all equipment on the safe shutdown list should be qualified. NUSCO needs to clarify their intent on the remaining equipment not specified by Stevenson and Associates.

3. What are the capabilities to cool the spent fuel pool?

4. What safety related equipment is Westinghouse covering and what are their criteria and methods? What is the schedule for completion?

5. Provide soil properties for the evaluation of the field erected tanks, buried tanks, and buried piping. Justify the modeling in conjunction with these soil properties.

For the piping analyses being performed, the following items require additional attention:

1. The piping stress allowables are currently undecided. The NRC and NRC consultants will make a decision on this issue and will transmit the decision with the trip report.*

2. The licensee is requested to provide assurance that unsymmetrical bending will be addressed for piping supports where applicable.

3. The licensee is requested to provide clarification of support load combinations. It appears that the algebraic sum of weight plus thermal plus seismic is not always consistent with acceptabl criteria.

4. What items is Westinghouse covering with regard to safety relatec piping? What is the schedule for completion?

In addition, the scope of SEP includes seismic re-evaluation of emergency power supply and distribution, instrumentation, and actuation systems. Clarification of the licensee's schedule for completion of this effort is requested.

---

* NRC and its consultants have made a decision as to what the piping allowable stresses should be. The allowables should correspond to ASME Code service level D (faulted) allowables. If the piping system is a Class 1 system and a Class 1 analysis is being performed, then the primary allowable stress should be 3.0 S_m. If the piping system is a Class 2 system and a Class 2 or ANSI 31.1 analysis is being performed, then the primary allowable stress should be 2.4 S_h. If the piping system is a Class 1 system and a Class 2 or ANSI 31.1 analysis is being performed, then the primary allowable stress should be 1.8 S_h to account for the difference in stress indices between the two types of analyses where:

S_m = ASME Code allowable stress intensity

S_h = ASME Class 2 or ANSI 31.1 allowable stress.

APPENDIX B

PIPING STRESS ANALYSIS PROCEDURE FOR SEISMIC QUALIFICATION OF SAFETY RELATED PIPING AT CONNECTICUT YANKEE

B-1

NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
THE HARTFORD ELECTRIC LIGHT COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

short sections
8 & 9

PIPING STRESS ANALYSIS PROCEDURE

FOR

SEISMIC QUALIFICATION OF SAFETY RELATED PIPING

AT

CONNECTICUT YANKEE

PRELIMINARY

PREPARED BY: ______________________ DATE __________

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PSE PROCEDURE FOR CY

SEISMIC QUALIFICATION OF SAFETY RELATED PIPI

1.0 PURPOSE

This procedure provides an outline of the criteria and methodology to be employed by Generation Piping Systems Engineering in the seismic qualification of Category I safety related piping at Connecticut Yankee.

2.0 SCOPE

This procedure applies to all aspects of the system engineering evaluation including the following.

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2.1 Piping analysis.

2.2 Evaluation of the adequacy of existing supports and design of subsequent modifications.

2.3 Design of new support functions as determined by the piping analysis.

2.4 Review of equipment nozzle loads.

2.5 Review of fabricated branch connections.

3.0 REFERENCES

3.1 ANSI B31.1 Power Piping Code, 1973 Edition, Summer 1973 Addenda.

3.2 AISC Specification, Manual for Steel Construction, 7th Edition, 1970.

3.3 ACI Standard 318-77, "Building Code Requirements for reinforced Concrete".

3.4 CYS-579, Revised December 10, 1965, "Specification for Shop Fabricated Piping for Secondary Plant and Primary Plant Waste Disposal and Other Miscellaneous Systems".

3.5 CYS-1550, Revised July 21, 1965, "Specification for Shop Fabricated Nuclear Piping".

3.6 CYS-579A, Revised January 7, 1966, "Supplement to Piping Specifications CYS-579 and CYS-1550 Covering Field Erection".

3.7 Welding Research Counsil (WRC) Bulletin 107, "Local Stresses in Spherical and Cylindrical Shells Due to External Loadings".

3.8 American Petroleum Institute (API) Standard 610, "Centrifugal Pumps for General Refinery Services".

3.9 W. G. Counsil letter to D. M. Crutchfield, dated August 11, 1981.

3.10 D. M. Crutchfield letter to W. G. Counsil, dated September 28, 1981.

3.11 CYS-500, Revised May 20, 1966, "Summary of Design Conditions".

3.12 NUSCO Procedure GE&C 4.04, "Preparation, Review, Approval, and Control of Design Analyses, Technical Evaluations, and Manual and Computer Calculations".

3.13 American National Standard ANSI N45.2.11, Section 4.2.

3.14 Northeast Utiltiies Quality Assurance Program Topical Report, QAP 6.0.

3.15 Connecticut Yankee - Inservice Inspection Boundary Diagrams.

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4.0 NOMENCLATURE

4.1 Line Designations

Piping is identified on the various isometrics, plans, and sections by a unique line designation. The line designation is comprised of a line size and a line number in conjunction with a fluid designation and pipe class. Examples are given below.

Size	Fluid Designation	Pipe Class	Line Number
10"	AC	601R	56
2"	CH	151N	186
12"	WFPD	601	10

Once the pipe class is identified, the pipe schedule and material can be determined from References 3.4 or 3.5.

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4.2 Support Function

Type	Isometric Abbreviation
Anchor	Anc
Lateral Restraint	Lat
Axial Restraint	Axial
Vertical Restraint	Vert
Spring Hanger	S.H.
Rod Hanger	R.H.
Sliding Support	S.S.
Lateral Shock Suppressor	LSS
Axial Shock Suppressor	ASS
Vertical Shock Suppressor	VSS
Vertical Support	VS

Multiple support functions are represented by a combination of the above symbols; e.g., VERT-LAT.

5.0 GENERAL

Safety related piping systems will be divided into several individual stress problems based on analytical terminal points, such as structural anchors and equipment nozzles. For each stress problem the piping geometry will be based on as-built isometric piping drawings developed under the I&E Bulletin 79-14 program.

Support information shall be derived from the I&E Bulletin 79-02 and I&E Bulletin 79-14 hanger inspection packages.

6.0 MODELING/TECHNICAL CONSIDERATIONS

6.1 Single Acting Restraints

Single acting restraints, such as rod hangers and sliding supports, shall be evaluated on an individual basis applying the following method.

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A deadweight and "free" thermal analysis shall be run with all single acting restraints removed to determine pipe movement under both load cases. This unrestrained movement will then be combined at each restraint location. The restraint shall be considered active only if the calculated net displacement is negative (i.e., downward).

Single acting restraints will not be considered in the seismic analysis. However, during a seismic event an additional load could be imposed on these restraints. The evaluation of single acting restraints, therefore, will include a seismic load equal to a dynamic load factor of 1.5 times the deadweight reaction times the peak value of acceleration from the appropriate floor response spectra.

6.2 Wall/Floor Penetrations

Thimble drawings shall be reviewed to determine gaps around the pipe. Thermal movement at this location will be compared with the gap to determine if the penetration will impede the total thermal movement. If so, then the penetration shall be incorporated into the thermal analysis as a restraint in the appropriate direction once the gap is closed. DISPLACE CARD FOR MOVEMENT

~~SAME THING FOR SE~~ NOT CONSIDERED FOR SEISM

Where the pipe is embedded in the concrete, the penetration will be considered to act as one of the following restraints for all load cases. 3/32

6.2.1 Full anchor: for embedded lines with anchor rings or lugs welded to the pipe.

6.2.2 Four-way restraint with no axial and torsional restraint; for embedded lines without welded rings or lugs.

6.3 Valves

Dimensions, weight, and center of gravity of valves shall be taken from the original valve drawing. In many instances, however, individual valve drawings are not available. In lieu of original drawings, valve properties may be obtained from other appropriate sources, such as a standard valve drawing, the manufacturer's catalog, or other plant records.

Where the valve wall thickness cannot be determined, then an assumed value of four (4) times the nominal pipe wall thickness shall be used.

For piping two inches (2") nominal and larger, the valve center of gravity will be modeled to consider the effects of the eccentric mass. If the center of gravity is not given, then one will be calculated using the methods shown in Figure 1.

6.4 Non-Standard Fittings

Stress intensification factors for non-standard fittings such as weldolets shall be obtained from the manufacturer. If manufactorer's data is not available, then the stress intensification factor may be determined by engineering judgement.

Reinforcement area of fabricated branch connections shall be reviewed in accordance with ANSI B31.1, Paragraph 104.3.1.

6.5 Flanged Joints

Flanged joints shall be evaluated to include the effect of moments and forces acting on the joint as a result of load conditions other than internal pressure.

The methods of ASME Section III, Subsection NC, 1980 Edition, Winter 1980 Addenda, Paragraph NC-3658, shall be used for this review.

6.6 Branch Lines

Where the moment of inertia of the run pipe is a minimum factor of ten (10) times greater than than of the branch line, the branch line may be analzyed separately. For these cases the run will then be considered to act as an anchor with respect to the branch line.

6.7 Anchors

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7.0 LOAD CONDITIONS

7.1 Primary Loads

7.1.1 Pressure

The longitudinal pressure stress shall be calculated using the system design pressure.

7.1.2 Weight

The weight of piping and components, insulation, and the contents under normal operating conditions, shall be considered. Using the line designation, this information can be determined from the pipe fabrication specifications and a table of standard pipe properties.

NOTE: Hydrostatic test loads will ~~now~~ not? be considered in this evaluation.

Spring hangers shall be represented in the analysis as an external force equal to the "HOT" load. The "HOT" load will be determined from the IEB 79-14 hanger inspection information. Where the inspection package specifies a "COLD" load, then the "HOT" load shall be calcualted as follows.

HOT Load = COLD Load - (thermal displacement)
x (spring constant)

This information shall be recorded on the Spring Hanger Summary Sheet (Attachment 1) and the results evaluated to determine if the load range is within the load carrying capability of the spring. If the calculated load range falls outside the spring range, then adjustment or replacement of the spring shall be considered.

7.1.3 Seismic

The structural integrity of safety related piping under a safe shutdown earthquake will be evaluated using one of the following methods.

7.1.3.1 Dynamic

A dynamic analysis may be performed using lumped mass dynamic models with the appropriate amplified floor response spectra as input. Zero period acceleration (ZPA); i.e., missing mass effects shall be considered as a static load case in conjunction with the inertial response of the system.

Where the stress problem involves piping that is supported at different floor elevations or runs between two separate structures, the response spectra in each direction shall be an envelope of the applicable floor spectra.

The analysis shall consider a simultaneous input of the two horizontal components and the one vertical component of the earthquake. Results for each mode shall be obtained by the square root sum of the squares (SRSS) method.

Output from the dynamic analysis shall be reviewed to determine the cutoff frequency for each direction. This will coincide with the mode for which the deflection in the given direction is less than .001 inches. The corresponding modal effective mass fraction can now be established.

To account for the effects of missing mass a static load case shall be performed for each of the two horizontal and the one vertical direction. The equivalent "g" value for this load case will be equal to the quantity one minus the modal effective mass fraction times the "g" value corresponding to the cutoff frequency from the appropriate floor response spectra. The results shall then be combined by the SRSS method.

7.1.3.2 Static

In lieu of dynamic analysis an equivalent static seismic analysis may be performed. Each of the two horizontal and the vertical shall be addressed in a separate static analysis. The results shall then be combined by the SRSS method.

The equivalent static "g" loading shall be calculated by multiplying the maximum value of acceleration from the appropriate floor response spectra by a dynamic load factor of 1.5.

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7.2 Secondary Loads

7.2.1 Thermal

Forces and moments resulting from thermal expansion or contraction including the thermal displacements of nozzles and anchors shall be evaluated. The analysis shall take into account the complete range of system and plant operation.

Clear~~r~~ances between the pipe and a rigid restraint or building penetration shall be compared to the unrestrained thermal movement of the piping to determine if the pipe movement is restricted. Where the "free" thermal movement is less than the clearance the restraint or penetration need not be considered in the thermal analysis.

7.2.2 Seismic Anchor Movement

The effects of relative seismic anchor displacements shall be considered in the evaluation. Movements will be assumed to occur out-of-phase between anchor points.

The results of this load case shall be combined with the results of the thermal analysis.

PRELIMINARY

APPENDIX C

CONNECTICUT YANKEE ATOMIC POWER COMPANY SAFETY RELATED PIPING SEISMIC QUALIFICATION PROGRAM CRITERIA