October 4, 1982

Docket No. 50-29
LS05-82-10-007

Mr. James A. Kay
Senior Engineer - Licensing
Yankee Atomic Power Company
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Kay:

SUBJECT: SEP TOPICS III-6, SEISMIC DESIGN CONSIDERATIONS
AND III-11, COMPONENT INTEGRITY
YANKEE NUCLEAR POWER STATION

Enclosed is our draft safety evaluation for the seismic design of the Yankee Nuclear Power Station. As stated in the evaluation, since YAEC has not yet completed the seismic reevaluation of Yankee, the staff's review was based on preliminary analyses and several working-level meetings between YAEC and NRC personnel and their consultants. Therefore, the conclusions presented in the evaluation may be revised should new information be presented in the final YAEC seismic report.

As discussed in the evaluation, YAEC has evaluated the hot shutdown piping systems to the NRC site specific spectra and the balance of the piping systems for cold shutdown and accident mitigation to the Yankee composite spectra. With the proposed addition of a "Dedicated Hot Shutdown System," the YAEC has proposed not to upgrade the balance of piping systems for cold shutdown and accident mitigation where modifications to restore design allowables have been identified . This proposal will be evaluated during the integrated assessment of Yankee.

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The evaluation identifies structures, major mechanical equipment and their supports, and electrical equipment and other mechanical equipment (including supports) which either do not meet current design criteria for postulated seismic loads, or have not been adequately evaluated by YAEC. In order for the staff to complete its evaluation of the seismic design of Yankee, the outstanding requested analyses and results should be provided to the staff. It is our understanding that YAEC will submit all of the final reevaluation reports by the end of October 1982.

ADD: G. Staley

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

Sincerely,

Original signed by:

Ralph Caruso, Project Manager
Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

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DATE	9/30/82	9/30/82	9/30/82	9/30/82	10/4/82	~~9/ /82~~ 10/4/82	9/ /82

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YANKEE ATOMIC ELECTRIC COMPANY

DOCKET NO. 50-29
YANKEE-ROWE ATOMIC
POWER STATION

Mr. James A. Kay
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Yankee Atomic Electric Company
1671 Worcester Street
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cc:

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Resident Inspector Yankee Rowe Nuclear Power Station c/o U.S. NRC P. O. Box 28 Monroe Bridge, Massachusetts 01350	Resident Inspector
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SYSTEMATIC EVALUATION PROGRAM

TOPICS III-6 AND III-11

YANKEE NUCLEAR POWER STATION

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

I. INTRODUCTION

The nuclear power plant facilities under review in SEP received construction permits between 1956 and 1967. Seismic design procedures evolved significantly during and after this period. The Standard Review Plan (SRP), first issued in 1975, along with the 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 necessary to make a reassessment of the seismic capability 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 plants in each group. The approaches used 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 Yankee plant was placed in Group II for review.

The Yankee plant, a four loop, pressurized water reactor (PWR) of 175 MWe capacity, is located in Rowe, Massachusetts, adjacent to the Dearfield River alongside the pond formed by Sherman Dam. The Nuclear Steam Supply System (NSSS) was supplied by Westinghouse and the plant was designed and constructed by Stone & Webster Engineering Corporation. The plant received its construction permit in November 1957, and the provisional and full-term operating licenses were issued on July 9, 1960 and June 23, 1961, respectively.

The Yankee plant originally was not designed for seismic loads. Neither structures nor equipment were classified into seismic categories such as "Seismic Category I" or equivalent, but instead, were classified as Safety-Related or Non-Safety Related. For structures, the design of lateral load resisting systems was predicated by wind loading requirements. However, provisions were not made for internal structures and equipment to resist lateral loads. Attachment 1 to this report, "Seismic Design Bases and Criteria for Yankee Nuclear Generating Station," summarizes the details of the original criteria and design.

The SEP seismic review of the Yankee 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 integrity of mechanical and electrical equipment and engineered safety features systems (including containment). By letters dated August 4, 1980 and April 24, 1981 (Ref. 1 and 2), the licensee, Yankee Atomic Electric Company, was required to reanalyze and upgrade, if necessary, the seismic resistance of the safety-related structures, systems and components to an earthquake level that is acceptable to the staff. Then, the staff reviewed the licensee's reanalysis criteria, scope, methods, and results to assess the overall capacity of the facility.

II. REVIEW CRITERIA

Since the SEP plants were not designed to current codes, standards and NRC requirements, it was necessary to perform "more realistic" or "best estimate" assessments of the seismic capacity of the facility and to consider the conservatisms associated with original analysis methods and design criteria. A set of review criteria and guidelines was developed for the SEP plants. The 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 J.A. Kay, YAEC, "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 J.A. Kay, YAEC, "SEP Topic III-6, Seismic Design Considerations, Staff Guidelines for Seismic Evaluation Criteria for the SEP Group II Plants," dated July 26, 1982 (Ref. 3).

E. Revision 1 of Criteria #D above - Letter from D.M. Crutchfield, NRC to J.A. Kay, YAEC, dated September 20, 1982.

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

III. RELATED TOPICS AND INTERFACES

The related SEP topics to the review of seismic design considerations and component integrity are Topics II-4, II-4.A, II-4.B, and II-4.C. These topics relate to specification of seismic hazard at the site, i.e., site specific ground response spectrum for the Yankee plant site. The staff recommended site specific spectra and the basis for it are found in Ref. 4 and 5.

IV. EVALUATION

A. General Approach

The seismic reevaluation of the Yankee Nuclear Power Station 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 letters (Ref. 1 and 2) were issued to require the licensee to complete a seismic reevaluation program. The program included: (1) providing a justification to demonstrate that the plant could continue to operate in the interim until the program was completed, (2) proposing a program plan that addressed the scope, criteria and schedule for completion of the program; and (3) after the staff accepted the proposed program plan, performing seismic analysis and providing final results to the staff for review. The staff's review of results would serve as the basis for seismic safety assessment of the plant facility.

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

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

(1) The analyses were 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 (Ref. 3) at appropriate service conditions. A 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 is required to demonstrate design adequacy. Review Criteria A through E provide the basic guidelines for all evaluations, in conjunction with the previously referenced SRP and Regulatory Guide guidelines.

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

B. Detailed Evaluation

1. Seismic Input

As a result of NRC Seismic Hazard Analysis (Ref. 4) program conducted by the staff and its consultant, Lawrence Livermore National Laboratory (LLNL), the site specific ground spectrum, which is acceptable to the staff as the input for the seismic reevaluation of the Yankee plant, was recommended to the licensee by NRC letters dated August 4, 1980 (Ref. 1) and June 17, 1981 (Ref. 5). As indicated in these letters, the local site effects (or local site amplification) were not considered in the development of these spectra. The spectra may be modified in the future when the review of local site amplification is complete. In these letters, the staff also encouraged the licensee to propose its own site specific ground response spectra before the final decision about seismic input of this site was made. In its letters dated April 10, 1981 and December 14, 1981 (Ref. 6 and 7), the licensee proposed a Design Basis Spectrum (the Yankee Composite Spectrum) that it contends represents the SSE. The licensee committed to perform: (1) analyses to both NRC site specific and Yankee Composite Spectra to demonstrate the integrity of the reactor coolant pressure boundary and the integrity of structures, systems and components necessary for safe hot shutdown, and (2) an analysis to the Yankee composite spectrum for the evaluation of accident mitigating and cold shutdown systems. This approach was accepted by the staff (Ref. 8).

2. Justification for Continued Operation

As requested by Reference 1, the licensee provided a basis for continued operation of the Yankee plant on December 5, 1980 (Ref. 9) and December 14, 1981 (Ref. 7). The NRC safety evaluation report (SER) to allow Yankee to continue to operate until the seismic reevaluation program is complete, was issued December 31, 1981 (Ref. 8).

3. Review of Licensee's Seismic Reevaluation Program Plan

A detailed seismic reevaluation program plan including scope, criteria, analytical procedure, and schedule for completion, was submitted by the licensee through its letters dated September 12, 1980 (Ref. 10), October 15, 1980 (Ref. 11), December 5, 1980 (Ref. 9), January 30, 1981 (Ref. 12), February 4, 1981 (Ref. 13), June 30, 1981 (Ref. 14), and December 14, 1981 (Ref. 7). The review of this program plan was completed as detailed in an NRC letter dated January 22, 1982 (Ref. 15).

4. Staff Review of Criteria and Scope

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

(a) selection of the earthquake hazard,
(b) design seismic loadings,
(c) soil-structure interaction,
(d) damping and energy absorption,
(e) methods of dynamic analyses 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 accurately represent the actual stress level in structures, systems and components during a postulated earthquake event and consider, to a certain extent, nonlinear behavior of the systems.

The SEP seismic reevaluation of the Yankee 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 covered by the scope discussed above were reviewed on an audit basis.

5. <u>Review of Reevaluation Criteria and Scope Proposed by the Licensee</u>

The licensee presented its seismic reevaluation criteria and scope through their letters (Refs. 7, 9 thru 14) and a series of working-level review meetings. From a comparison with the staff's guidelines and review of other factors, the criteria proposed by the licensee appear reasonable for reevaluation of the plant facility (safety-related structures, systems and components). The details of criteria reviewed are found in the staff's contractor reports (Attachments 2 and 3).

The scope of the licensee's seismic reevaluation program was proposed in a letter dated December 14, 1981 (Ref. 7). The identification of all structures, systems and components included in the scope are found in Enclosures 1 and 2 of Reference 7. As proposed in Reference 7, all structures, systems and components listed in Enclosure 1 were to be analyzed using both the site specific ground response spectrum recommended by the staff and Yankee composite spectrum as inputs and upgraded as required. The additional structures, systems and components listed in Enclosure 2 were to be analyzed using only Yankee composite spectrum as input.

The staff agreed that the scope of analysis of structures, systems and components committed to by the licensee in their submittal (Ref. 7) was adequate. Since the licensee initially proposed its program, it has reconsidered how it intends to provide a seismically qualified hot shutdown system. The licensee has currently presented a conceptual approach for a "Dedicated Shutdown System" that would provide water to the steam generators for removal of residual heat and to the primary system to make-up shrinkage during cooldown. The "Dedicated Shutdown System" is based on the premise that non-isolable portions of the primary system as well as portions of the main steam and feed systems would be seismically qualified to the site specific spectrum level to preclude a seismically-induced pipe break. This system would include the upgrading of the fire tank, the addition of new seismically qualified piping to a new diesel powered make-up pump, and piping to connect to paths for providing water to the steam generators and the reactor vessel.

The concept proposed by the licensee does not meet the scope of seismic reevaluation approved by the staff (Ref. 8). The proposal would provide a method for reaching hot shutdown but would not provide seismically qualified means to achieve cold shutdown or mitigate accidents. The staff recommends that this issue be resolved as part of the integrated assessment.

6. Review of Structures

The structural review of the Yankee plant was based on licensee's presentations at a series of working-level meetings documented as handouts and discussions between the NRC staff (the staff and its consultants) and the licensee (licensee and its consultants). The following factors were included in the review: criteria (both analysis criteria and performance criteria), basic assumptions, modeling techniques, analysis methods, and general appropriateness of the results. The details of the staff's review of criteria assumptions, modeling techniques, and analysis methods, including the neglecting of soil-structure interaction effect, are described in the staff's contractor report (Ref. 15 and Attachment 2). The licensee's reevaluation of structures were found generally to be acceptable. The review of reevaluation results are briefly discussed below:

(a) Vapor Container

The staff agrees that the results from the dynamic analysis for the vapor container, presented by the licensee in the review meetings, are indicative that integrity of the vapor container would be maintained under the postulated SSE loadings. In addition, the staff and its consultant also performed a seismic confirmatory analysis for the vapor container (Attachment 3). The results confirmed the fact that the analysis results presented by the licensee are reasonable. However, the licensee should provide additional information to address the staff's concern regarding their buckling stress allowable which is significantly higher than those allowed by AISC and SRP that was used for the evaluation of the vapor container steel columns.

(b) Concrete Reactor Support Structure

Two models were used by the licensee for the analysis of reactor support structures. The first was a three-dimensional linear hybrid model that consisted of lumped mass-spring system and finite elements for the generation of the structural responses (dynamic shear, axial forces and moments) for the assessment of structural integrity and the in-structure response spectra for the input to subsystems. The staff considers the analysis model and the preliminary results to be reasonable. In order to account for the expected nonlinear behavior at the junction between the support columns and internal structures, the second model, a two-dimensional finite element model was developed to assess the design adequacy of these joints. A detailed review of the licensee's theoretical basis, modeling methods and preliminary results was performed by the staff and its consultants since this item was considered significant. As a result of this review, the staff concludes that the general approach and the preliminary results appear reasonable, i.e., there is no likelihood of a catastrophic collapse of the reactor support structure due to the failure of supporting columns under the postulated SSE loads. However, the following staff concerns should be addressed:

i. Justify the appropriateness of the stress-strain curve for No. 14 rebar dowels used in the connections between the reactor support structure and its support columns. Also, the effect on the hysteretic behavior of the joints from the steel yield strength and the shape of the stress-strain diagram should be clarified. Provide the steel grade for the rebar dowels.

ii. The size and type of stud bolts must be identified and an analysis of load transfer from the steel shell through the stud bolts and into the concrete and the dowel bars must be carried out. In particular, consequences of semi-ductile or brittle behavior on the local load transfer must be evaluated.

iii. What is the maximum bond stress which can be reasonably expected in the connection? The value of 1000 psi used in the CYGNA analysis appears to be somewhat high. Justifications of the appropriate maximum bond stress value requires further documentation of the geometry of the surface deformations in the actual dowel bars. This information would then allow a better estimate of bond strength under monotonic loads.

In arriving at bond strength, credit has been taken for effects of aging on the concrete compressive strength. However, concrete compressive strength at the top of a column is likely to be below average because of the normal amount of bleeding and dilution of paste.

Below the construction joint, circular ties are spaced at 12-inches and the dowel bars in the interior columns are too closely spaced. Local cracking around the stud bolts may further reduce the ultimate bond strength.

Degradation in bond strength due to cyclic loading, particularly when maximum tension stress in the reinforcement reaches yield strength, must also be taken into account.

On the positive side, there is the confinement effect of the steel shell on the bond strength and on the overall capacity of the connection. Under these conditions, and in the absense of directly applicable test results, it is not possible to predict reliably the value of maximum bond strength. However, it would be prudent to evaluate the effect of a lower bound bond stress value on the behavior of the structure.

iv. Anchorage length of 42 inches is assumed in developing a bilinear force-displacement curve for the No. 14 bar dowel. This anchorage length is based on the arrangement of steel in the interior columns. Some bars in the exterior columns have an embedment length of only about 24-inches. The following questions must be resolved: can yield strength of No. 14 bar be developed in a 24-inch anchorage length and with reduced bond stress, and what effect, if any does this condition have on the overall behavior of the structure?

v. Values of 7% and 10% damping used in the seismic analysis appear to be excessive, in light of low stresses throughout most of the structure and confinement of concrete columns provided by the steel shells. Verification of the concept that foundation lift-off and associated behavior may justify the higher damping values requires a detailed review of the foundation and of its effect on the general behavior of the structure during a seismic event.

(c) Turbine Building

As discussed by the licensee, the turbine building was modeled as three-dimensional, fixed base, linear lumped mass-spring system to generate structural responses and in-structure response spectra. The licensee's analyses and preliminary results are considered by the staff to be reasonable. The results showed that some steel frames, bracings, joints, and the masonry walls under the control room were found to be overstressed under the postulated seismic motion defined by NRC site specific spectrum. The licensee has committed to upgrade this building.

(d) Primary Auxiliary Building

This building was modeled as a three-dimensional, fixed base, lumped mass-spring system and analyzed by response spectrum analysis method with NRC site specific spectra as input. The results showed that seismically induced stresses in general are low. However, local overstresses were identified at some of the connections of the steel beams to the concrete walls. The licensee has committed to upgrade these connections. Based on the results of the review, the staff considers the modeling techniques, analysis methods and results to be reasonable.

(e) Diesel Generator Building

A three-dimensional, fixed base, lumped mass-spring model and response spectrum analysis method were used by the licensee for the seismic analysis of this building. The preliminary results showed that the building and annex would be adequate to withstand the postulated seismic loads with the exception that some high bending stresses could occur in the supporting columns. The licensee has committed to upgrade the structure to alleviate this potential problem. From results of the staff's review we conclude that the model, analysis approach and the results are reasonable.

(f) MS and FW Piping Support Structures

The licensee stated that the MS/FW piping support structure was considered an integral part of piping systems and would, therefore, be analyzed together with the two piping systems. The analyses were not completed when the review meetings were conducted. Therefore, the review of this structure has not been performed and the design adequacy of this structure will be considered as an open item.

(g) Spent-Fuel Pool and Spent-Fuel Chute

According to the licensee, the pool structure and chute were assumed to be uncoupled. For the purpose of application for spent pool expansion the pool structure was modeled as a fixed base finite element system and analyzed by response spectrum analysis method with 0.2g R.G. 1.60 spectrum (7% damping) as seismic input. The combined loads (combination of seismic loads with water sloshing, soil pressure, hydrostatic pressure and thermal loads) were used for the evaluation of design adequacy. The chute was modeled as fixed base, lumped mass-spring system and was analyzed by response spectrum analyses method using the NRC site specific spectrum as input. The preliminary results presented by the licensee showed that the stress level in the structure is low and the effect of the relative motion between reactor support structure and spent-fuel pool on the chute is small. The staff agrees that the models, analysis method, and results are reasonable.

(h) Field-Erected Tanks and Buried Piping or Tunnels

The licensee stated that there are no safety-related buried piping or tunnels at the Yankee plant site. The fire water tank is the only tank that the licensee intends to seismically justify. This tank is a major component of the proposed "Yankee Dedicated Shutdown System." The analysis of this tank had not been completed and, therefore, no review of the item has been performed. The design adequacy of this tank is considered as an open item. Further, the staff's position is the safety injection tank should be included in the scope of review, and analyzed and upgraded, if required.

(i) Masonry Walls

Based on the licensee's analyses, a total of approximately 2,157 ft. of safety-related masonry walls which are located adjacent to the hot shutdown system or components were analyzed and require upgrading. The design of wall modifications is underway. The licensee has committed to complete all required modifications to the masonry walls that could effect the hot shutdown system. However, neither analysis of, nor committment to, upgrade when required have been made for masonry walls near safety-related equipment for cold shutdown or accident mitigation.

(7) Review of Safety-Related Piping Systems

As described in Reference 7, a total of fifteen (15) piping systems (the identifications of these piping systems are found in Attachment 3 to this report) have been or are currently being analyzed by the licensee's consultant. The analyses were performed in accordance with the "Seismic Reevaluation and Retrofit Criteria," which has been revised and accepted by the staff (Ref. 8). The staff's review of the analysis of safety-related piping systems as well as their supports was performed by selecting samples out from these 15 systems and conducting detailed audit review of the preliminary results during the working-level review meetings. The piping systems sampled were pressure relief, main steam and high pressure safety injection lines. In general, the analysis criteria, analysis methods and preliminary results for the piping and its supports are considered to be reasonable. However, justifications need to be provided by the licensee to address the possible impact loads at some pipe supports which were not evaluated.

(8) Major Mechanical Equipment (Including Supports)

The components to be evaluated under this category are: reactor pressure vessel (RPV), steam generators, reactor coolant pumps, and pressurizer. The licensee presented an overall screening criteria based on a linear interaction formula (interaction between nozzle reaction shear and moments) for the qualification of these components in the review meetings. The axial stress at the nozzle due to internal pressure was not included. The licensee agreed to reconsider the effects of axial tension force on the adequacy of nozzles. In addition, no evaluation was performed for the structural integrity (other than nozzles) of components.

(9) Electrical and Mechanical Equipment (Including Supports)

To qualify the structural integrity of equipment items (including equipment supports), the licensee proposed a so called "Correlation of Inservice Seismic Experience to Qualification of Nuclear Power Plant Equipment" approach. The method consists of a detailed walkdown to identify safety-related equipment items and subsequent comparison of these equipment items with similar items found in fifteen (15) non-nuclear facilities which have survived earthquakes exceeding the Yankee plant reevaluation basis, namely, NRC site specific ground response spectrum. The licensee has concluded, based on the above stated approach, that all safety-related equipment possess the capability to withstand the postulated seismic loads. This approach is not acceptable to the staff. The staff believes that the comparison results obtained will serve as a basis to qualify safety-related equipment by a sampling approach. It is the staff's position that the licensee should perform an evaluation of structural integrity on at least one sample for each category of safety-related equipment. The structural integrity is defined as: (1) the integrity of anchorage and support systems; and (2) the integrity of load path from an internally mounted element to the component anchorage and support systems. Further, equipment frequencies of related items should be checked when extrapolating the results. Therefore, the structural integrity of all equipment items will be considered as open items and will be resolved during the integrated assessment.

V. CONCLUSION

Since the licensee had not completed its seismic reevaluation of the facility consistent with the staff's review schedule, this evaluation was based on the preliminary analysis results presented, and discussion of the out-standing questions identified, both of which were contained in the handouts of the view graphs presented during the working-level meetings. Therefore, the conclusions drawn here are preliminary and could be revised should changes be identified from the review of the licensee's final evaluation reports. The licensee has committed to submit all final reevaluation reports by the end of October 1982.

With regard to the scope of the Yankee seismic reevaluation, the licensee has evaluated the non-isolable parts of hot shutdown piping systems to the NRC site specific spectra and the balance of piping systems for cold shutdown and accident mitigation to the Yankee Composite Spectra. The licensee has stated that it intends to add a "Dedicated Hot Shutdown System" to the plant and not upgrade the balance of system for cold shutdown and accident mitigation where modifications to restore design allowables have been identified. This issue should be resolved as a part of the integrated assessment.

Structures

From the evaluation of results presented by the licensee, the staff has concluded that generally all safety-related structures are considered capable of withstanding the postulated seismic loads. Some areas have been identified to be over-stressed:

(1) Steel bracing in turbine and primary auxiliary buildings.

(2) Some columns due to bending in diesel generator building.

The licensee has committed to upgrade these structures. The schedule should be established as part of the integrated assessment.

Additional information, as identified in the previous section, is needed for resolving issues related to the joint between the reactor support structure and support columns and justification for the buckling stress allowable used as criteria for the vapor container columns. However, the staff also concludes that the reactor support structures would not fail catastrophically at postulated seismic loads. The licensee committed to upgrade the steel bracing in the turbine and primary auxiliary buildings. The proposed modifications will ensure that the structures will retain their integrity with an earthquake motion to the NRC site specific spectrum level. According to the licensee, the identified open items will be addressed in the final reevaluation reports at the end of October 1982. As far as the issue related to the reactor support structure and column joints is concerned, the staff considers this item still open until the additional information requested is provided and reviewed to ascertain if the staff's concerns are adequately addressed.

The staff recommends that the licensee should complete and submit the analyses and upgrade, if required, the MS and FW piping structure, the fire tank and the safety injection tank. These will be resolved as part of the integrated assessment.

The licensee has identified by analysis and committed upgrade masonry walls located in close proximity to the hot shutdown systems. However, analyses of or committments to upgrade masonry walls near other safety-related equipment have not been made. This item should be resolved as a part of the Integrated Assessment.

Piping Structures

Based on a detailed audit review of three of fifteen safety-related piping systems, the staff concludes generally that all safety-related piping systems were adequately reevaluated and considered to be capable of withstanding the postulated seismic loads. In some cases, however, modifications (addition of supports) were required to meet design allowables. Information should be provided to address the staff's concern regarding the impact loads at some pipe supports.

Major Mechanical Equipment and Their Supports

No information regarding the ability of major mechanical equipment and their respective supports to retain their structural integrity under postulated seismic load was provided. The evaluations performed for component nozzle integrity was not acceptable to the staff. The staff does not find the treatment of major mechanical equipment acceptable. No committments have been made by the licensee for the resolution of this issue. This item should be resolved as part of the integrated assessment.

Electrical and Other Mechancial Equipment (Including Supports)

As discussed previously, the equipment similarity and earthquake experience approach proposed by the licensee is not acceptable to the staff by itself. It is the staff's conclusion that the structural integrity of all safety-related equipment items is considered as an open item. The licensee has committed to perform analyses on one sample (or samples) selected from each category of safety-related equipment. This sample analysis work in addition to the similarity approach would be acceptable to the staff. This item should be resolved as part of the integrated assessment.

For the qualification of electrical cable trays, the licensee intends to have the evaluation completed by testing through the SEP Owners Group program and the results applied later specifically to their plant. This program is scheduled for completion by December 1982, and plant specific application will be completed subsequently.

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

VI. REFERENCES

1. Letter from D.G. Eisenhut, NRC to J.A. Kay, YAEC dated August 4, 1980.

2. USNRC letter to YAEC, dated April 24, 1981.

3. USNRC letter to YAEC, dated July 26, 1982.

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

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

6. YAEC letter to USNRC, FYR 81-58, dated April 10, 1981.

7. YAEC letter to USNRC, FYR-161, dated December 14, 1981.

8. USNRC letter to YAEC, dated December 31, 1981.

9. YAEC letter to USNRC, WYR 80-127, dated December 5, 1980.

10. YAEC letter to USNRC, WYR , dated September 12, 1980.

11. YAEC letter to USNRC, WYR 80-114, dated October 15, 1980.

12. YAEC letter to USNRC, FYR 81-14, dated January 30, 1981.

13. YAEC letter to USNRC, FYR 81- , dated February 4, 1981.

14. YAEC letter to USNRC, FYR 81-101, dated June 30, 1981.

15. Letter from D.M. Crutchfield, NRC to J.A. Kay, YAEC dated January 22, 1982.