The Light company

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July 1, 1985
ST-HL-AE-1296
File No.: G4.2

Mr. George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, DC 20555

South Texas Project
Units 1 & 2
Docket Nos. STN 50-498, STN 50-499
Meeting Notes from NRC MEB Audit
Question Review Portion

Dear Mr. Knighton:

On June 26, 1985 representatives of the NRC Mechanical Engineering Branch (MEB) staff met with representatives of HL&P at the Houston office to discuss the mechanical design of the South Texas Project. This meeting was conducted in two portions. Attached are the meeting notes from the portion of the meeting concerning review of FSAR questions 210.18 through 210.65.

Attachment 1 contains our responses to these questions. All of the responses were reviewed and were agreed upon with the NRC staff during the meeting. Two questions, 210.52 and 210.65, resulted in open items. Question 210.52 is open, pending NRC review of Westinghouse (W) report "STP Reactor Internals Flow Induced Vibration Assessment" which will be submitted to the staff under separate cover by July 19, 1985. Question 210.65 is open, pending NRC review of HL&P's pump and valve inservice testing program which was previously committed to be provided twelve months prior to issuance of the operating license.

Additionally, four confirmatory items were identified as follows:

- Question 210.20 - Confirmatory; pending STP submittal of pipe rupture stress summaries and design information as noted in the response.

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- o Question 210.53 - Confirmatory; pending W submittal of portions of the Roto-lok stud report (CENC-1332) to NRC (consultants) under separate cover by 7/12/85.

- o Question 210.60 - Confirmatory; pending completion of W evaluation of component support stresses.

- o Question 210.64 - Confirmatory; pending NRC review of HL&P Draft Technical Specifications (submitted by our letter ST-HL-AE-1271 dated June 17, 1985).

It was agreed at the meeting that all of the remaining questions are considered to be closed.

These question responses will be incorporated into FSAR Amendment 50. For several questions FSAR text mark-ups are included just behind the subject question. These text changes will be incorporated into FSAR Amendment 50. Any additional FSAR text changes referred to in the question responses will be incorporated into future FSAR amendments.

Attachment 2 contains slides from a presentation regarding the flow-induced vibration assessment for the STP reactor internals. This was the subject of question 210.52.

Attachment 3 contains a list of attendees.

If you should have any questions, please contact Mr. M. E. Powell at (713) 993-1328.

Very truly yours,

M. R. Wisenburg
Manager, Nuclear Licensing

CAA/as

Attachments: (1) Responses to NRC Questions 210.18 through 210.65
(2) Presentation Slides Regarding Question 210.52
(3) List of Attendees

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Attachment 1

QUESTION 210.18

Paragraph 3.6.2.1.1.4 of the FSAR stated that "In the absence of an ASME Code Class 1 stress analysis, breaks are postulated at all fittings, valves or welded attachments." Identify all the Class 1 piping systems for which an ASME Code Class 1 stress analysis is not performed.

RESPONSE

All ASME Class 1 piping systems are analyzed and all postulated breaks are based on stress analysis results. Section 3.6.2.1.1.1(b.4) will be deleted in a future amendment.

(10) and either Equations (12) or (13) of subarticle NB-3653 of ASME Code Section III, under loadings associated with the OBE and normal and upset plant conditions, exceeds 2.4 S_m, or

b) The cumulative usage factor exceeds 0.1

3) As a result of piping reanalysis, the highest stress locations may be shifted. However, once a high-energy piping system has been analyzed and break locations have been identified and evaluated, the original break locations are <u>not</u> changed unless one of the following conditions exist: 45 40 45

a) Maximum stress ranges or cumulative usage factors exceed the threshold levels specified in 2) a) and 2) b) above;

b) A change is required in pipe parameters such as major differences in pipe size, wall thickness, and routing.

4) In the absence of an ASME Code Class 1 stress analysis, breaks are postulated at all fittings, valves, or welded attachments. 40

2. ASME Code Section III Class 2 and 3 piping, breaks are postulated to occur at the following locations in each run or branch run:

a. The terminal ends.

b. At all intermediate locations between terminal ends where the primary plus secondary stresses under normal and upset conditions and an OBE event, as calculated on an elastic basis by the sum of Equations (9) and (10) (subarticle NC-3652 of the ASME Code, Section III), exceed 0.8 ($1.2S_H + S_A$).

c. For the Main Feedwater System only, there shall be a minimum of two intermediate separated break locations selected for each ASME Section III Code Class 2 Main Feedwater piping run. Where at least two intermediate break locations cannot be determined based upon the above stress criteria (see 2.b. above) additional break locations based on highest relative stress as calculated from Equations (9) and (10) shall be selected as necessary to satisfy the requirement for a minimum of two intermediate break locations for each piping run. When stresses are less than the 2.b limits above limit and the stresses differ by less than 10 percent, the two selected intermediate break locations shall be separated by a change of direction of the pipe run. 45 40 45 40

However, if the piping run has only one change or no change in direction, only one intermediate break location need be postulated.

Where a network involving several branches is modeled as a common piping system for stress analysis purposes, only two intermediate break points are required based on the highest stress within the common piping system.

QUESTION 210.19

Provide assurance that the guidance stated in BTP MEB 3-1, Section B.1.C. (1) (d) (iii) concerning changes of new highest stress locations as a result of piping reanalysis has been used in STP high energy line break location postulation.

RESPONSE

BTP MEB 3-1, Section B.1.C(1)(d)(iii) is complied with to the extent that new high stress locations exceeding the break location criteria described below are considered as break locations regardless of the degree of remoteness from previous high stress points.

FSAR Section 3.6.2.1.1 specifies the criteria for postulating pipe break locations. It states that breaks are postulated at terminal ends and at intermediate locations based on stresses and cumulative usage factors. Arbitrary intermediate breaks are not postulated in high energy piping in accordance with the letters to the NRC ST-HL-AE-1115 dated August 20, 1984, and ST-HL-AE-1202 dated March 8, 1985. FSAR Section 3.6.2.1.1.2, Tables 3.6.1-2, 3.6.1-3, 3.6.2-1 have been revised to reflect elimination of arbitrary intermediate breaks in HE piping.

The exception to the above criteria is the feedwater system, as stated in FSAR Section 3.6.2.1.1.2(c), where a minimum of two intermediate break locations are postulated based on the stress analysis results.

In response to NRC letter ST-AE-HL-90534, January 29, 1985 we have provided additional design information via ST-HL-AE-1202, March 8, 1985, concerning the feedwater system provisions to minimize waterhammer. Hence, pending NRC favorable review, the feedwater arbitrary intermediate break locations may be eliminated prior to final stress analysis. Alternatively, as the stress analysis is finalized, it is anticipated that changes in intermediate break locations, if any, would be due to the criterion contained in BTP MEB 3-1, Section B.1.c(1)(d) (i) and (ii) and thus enveloping criterion B.1.c(1)(d)(iii).

QUESTION 210.20

In order to assure that the pipe break criteria have been properly implemented, the Standard Review Plan requires the review of sketches showing the postulated rupture locations and of summaries of the data developed to select postulated break locations including, for each point, the calculated stress intensity, the calculated cumulative usage factor, and the calculated primary plus secondary stress range. The required sketches and tables for some high energy piping systems have not been provided at this time in the FSAR. Provide a schedule for submission of these data.

RESPONSE

Initial stress summaries regarding pipe break locations, stress levels, cumulative usage factors will be provided during the fourth quarter of 1985. Final design information, including as-built reconciliation, will be provided prior to fuel load.

The South Texas Project (STP) has submitted a request to the NRC for exemption to General Design Criterion 4 in order to delete postulation of Reactor Coolant Loop (RCL) pipe breaks based upon the "Leak Before Break" analyses. This has been justified in WCAP-10560. (Refer to HL&P to NRC letters ST-HL-AE-1010 dated 9/28/83, ST-HL-AE-1096 dated 7/17/84, and ST-HL-AE-1200 dated 3/1/85.) Although the NRC has not yet responded to the request, the project is sufficiently confident such that the current design is proceeding on the assumption that the exemption will be granted. Thus, RCL pipe breaks are not postulated and the information requested is not pertinent to STP for that scope. However, it should be noted that primary component supports have been designed to withstand the structural loads associated with non-mechanistic RC pipe breaks at the locations described in WCAP-8082. Upon NRC approval of the elimination of RCL pipe breaks, the STP FSAR will be revised to reflect this revised design basis.

QUESTION 210.21

FSAR Section 3.6.2.1.3.2.1 (a) and (b) described the criteria used for determining types of breaks for high energy piping other than RCL piping. These criteria do not comply with the criteria specified in BTP MEB 3-1, Section B.3.a.(1) and B.3.b.(1). Revise your FSAR to conform to the SRP criteria.

RESPONSE

Sections 3.6.2.1.3.2.1 (a) and (b) have been revised to correct the typographical error by changing 'circumferential break' to 'longitudinal break' in (a) and 'longitudinal break' to 'circumferential break' in (b). This revision will be incorporated in Amendment 49.

OUESTION 210.22

BTP MEB 3-1, Section B.3.b.(3) specifies that for longitudinal breaks, axial splits should be oriented (but not concurrently) at two diametrically opposed points on the piping circumference such that the jet reaction causes out-of-plane bending of the piping configuration. Provide assurance that this guidance has been used.

RESPONSE

Longitudinal breaks are postulated in accordance with MEB 3-1, Section B.3.b(3).

FSAR Section 3.6.2.1.4.1 will be clarified to indicate specifically that for longitudinal breaks, axial splits are oriented (but not concurrently) at two diametrically opposed points on the piping circumference such that the jet reactions cause out-of-plane bending of the piping configuration.

QUESTION 210.23

Discuss how jet impingement effects on target piping systems and components were evaluated, specifically the criteria used in determining the acceptability of the target piping systems and components.

RESPONSE

FSAR Section 3.6.2.3.1 addresses methods of analysis for jet impingement. A detailed explanation of analyses performed for the RCL is included as Section 3.6.2.3.2. Regarding RCL breaks, please refer to the response to Question 210.20.

The effects of jet impingement on safety related piping other than the RCL are analyzed using criteria established in FSAR references 3.6-5, 3.6-6, and 3.6-9. The following sentence will be added to item 14 of FSAR Section 3.6.1.1:

"For essential piping, jet impingement loads are evaluated regardless of the ratio of impinged and postulated broken pipe sizes."

Once target piping systems and components have been identified and jet impingement loads have been calculated, the piping is statically analyzed under two conditions:

1) Transient loading using a dynamic load factor of 2.0 and incorporating the effects of any dynamic supports attached to the piping.

2) Steady state loading using a dynamic load factor of 1.0 and neglecting the effects of any dynamic supports.

Piping response (stresses, deflections and support reaction loads) generated using these two conditions are enveloped and compared to the appropriate faulted allowables as defined in FSAR Section 3.9.1.1.4 and Tables 3.9-7, 3.9-7A, 3.9-7B, and 3.9-7C. Load combinations for this event are presented in Tables 3.9-2.3, 3.9-2.3A, and 3.9-2.4.

Direct jet impingement on valves and other components connected to a piping system is identified and essential targets are either requalified for jet impingement or protected from the jet impingement. Reaction end loads on valves and components due to jet impingement on piping are calculated and combined with other appropriate loads in qualifying the valves or components to the vendor's allowables.

QUESTION 210.24

The staff finds that there is insufficient information describing the jet expansion model used for evaluation of jet impingement effects of steam, saturated water or steam-water mixtures. Provide additional information to assure that the criteria described in SRP 3.6.2 Section III.3 have been met for analysis of jet impingement forces.

RESPONSE

The jet expansion model used for the evaluation of impingement effects of steam, saturated water or steam-water mixtures is described in "Design for Pipe Break Effects", Topical Report BN-TOP-2, Rev. 2, May 1974, Bechtel Power Corporation, San Francisco, California as indicated in FSAR Section 3.6.2.3.

Following is a brief description of the analytical methods used in generation of the BN-TOP-2 Rev. 2 jet expansion model.

A. Discharging fluids with superheated, two-phase, saturated or subcooled conditions at the exit plane of the pipe are expanded with the Moody model. The distance to the asymptotic plane is calculated according to the Moody methodology. However, this distance is limited to no less than five pipe diameters for longitudinal and full-separation circumferential breaks, and five times the axial separation distance for limited separation circumferential breaks.

B. Subcooled fluids with a small void fraction ($\alpha < 0.001$) or cold water (enthalpy less than the enthalpy of a saturated liquid at ambient pressure) conditions at the exit plane of the pipe are expanded at a uniform 10° half-angle.

QUESTION 210.25

SRP 3.6.2 states that rise times for jet thrust not exceeding one millisecond should be used unless justified. Provide assurance that this justification will be included in the FSAR.

RESPONSE

Rise times for jet thrust on the South Texas Project do not exceed one (1) millisecond.

FSAR section 3.6.2.2.1 will be revised to include a statement that rise times for jet thrust do not exceed one millisecond.

QUESTION 210.26

Provide assurance that the criteria described in FSAR Section 3.6.2.1.1.7 relative to a structure separating a high energy line from an essential component has been used for both inside and outside containment.

RESPONSE

For outside containment, structures that separate a high energy line from an essential component are designed to withstand the consequences of the pipe break in the high energy line which produces the greatest effect at the structure irrespective of the fact that the mechanistic criteria might not require such a break to be postulated. This conforms to the requirements of NUREG-75/087 Section 3.6. (Refer to FSAR Section 3.6.2.1.1.7).

The additional requirements for inside containment added in NUREG-0800 are not applicable to STP; however, structures inside containment are designed for the dynamic effects of postulated mechanistic breaks. FSAR Table 3.6.1-3, "Design Comparison to NRC Branch Technical Positions MEB 3-1", will be revised to clarify this partial conformance to BTP MEB 3-1, Section B.1.C(4).

QUESTION 210.27

Provide a listing of those postulated pipe breaks where limited displacements have been used to reduce break areas.

RESPONSE

Refer to the last paragraph of the response to Question 210.20. Pipe breaks were postulated in accordance with WCAP-8082 for the design of primary components. Limited non-mechanistic pipe breaks were used in the design of primary component supports.

For the calculations of loop hydraulics for primary component support designs, reduced break areas (150 in^2) were used for the reactor vessel inlet and reactor vessel outlet breaks.

Limited displacements have not been used to reduce break areas for balance of plant piping on STP.

QUESTION 210.28

Is there any unrestrained whipping pipe inside containment? If so, discuss how pipe whip and jet impingement effects were determined for those postulated breaks in high energy piping that are not restrained (unrestrained whipping pipe). Provide the acceptance criteria for the impacted safety-related structures, systems, and components.

RESPONSE

Dynamic effects such as pipe whip and jet impingement are considered for the unrestrained whipping pipe inside containment as follows:

For primary equipment supports, jets from auxiliary lines which impact reactor coolant system equipment supports are evaluated to service level D criteria (ASME Subsection NF and Appendix F-1370). Jet loads are added directly to existing faulted condition support loads. For primary loop piping and components, stresses generated from jet loads from auxiliary line jets on Westinghouse scope piping/equipment, the combination of pressure, deadweight, jet loads are compared against ASME level D condition allowables.

The effects of the unrestrained pipe whip and jet impingement on other impacted safety-related structures, systems and components have been evaluated using the guidance of SRP 3.6.1 and 3.6.2 and the methodology of Bechtel Topical Report BN-TOP-2, Rev. 2 as referenced in FSAR Section 3.6.2.3.

Therefore, in the unlikely event of a pipe break and the corresponding unrestrained pipe movement, damage will not occur to impair the safety function of essential safety-related structure, system or components.

QUESTION 210.29

Provide the loads, load combinations, and stress limits that were used in the design of pipe rupture restraints. Include a discussion of the design methods applicable to the auxiliary steel used to support the pipe rupture restraint. Provide assurance that the pipe rupture restraint and supporting structure cannot fail during a seismic event.

RESPONSE

Refer to the last paragraph of the response to Question 210.20. RCL pipe breaks have been eliminated thereby eliminating the need for RCS loop restraints.

Pipe whip restraints for other than the RCL are designed as a combination of an energy-absorbing element (EAE) and a supporting (auxiliary) structure capable of transmitting the resistance load from the EAE to the main building structures (concrete walls, slabs, and steel structures). The EAE usually is either thin gauge cellular crushable material (energy-absorbing material, EAM) or stainless steel U-bars. The design limits for EAE's are specified in FSAR Section 3.6.2.3.4.1.2.

The supporting structures typically are structural steel frames designed to the loads, load combinations, and stress limits as specified in FSAR Section 3.8.3.3 and FSAR Tables 3.8.3-2 and 3.8.4-2. Except for the main steam restraints inside the containment, the elastic working stress design method of Part I of the AISC specification 1969 (including supplements 1, 2 and 3) is used. The main steam line restraints inside the containment are designed using a nonlinear method, with allowable ductilities per FSAR Section 3.5.3 and Table 3.5-13, where the ultimate strain is taken as 50% of ASTM specified minimum.

Both the OBE and the SSE seismic events are specifically included in the loading combinations prescribed for the structural integrity of the pipe whip restraints. The restraints and their structures are treated as structural subsystems whose seismic response is determined from their frequency characteristics and the appropriate floor response spectra.

QUESTION 210.30

Provide the design criteria used for pipe rupture restraints that also support piping.

RESPONSE

As discussed in FSAR Section 3.6.2.3.4.1.1(b), generally whip restraints are designed and located on the pipe with sufficient clearances between the pipe and the restraints to prevent interaction. Exception to this criterion is when a pipe support and restraint are incorporated into the same structural steel frame, or when zero design gap is required. For example, the 5 way restraints discussed in FSAR Section 3.6.2.3.3.1.3 are classified as bending and torsional restraints according to their purpose. The loads on these restraints are combined per FSAR table 3.9-2.4. The ASME code, Section III, subsection NF allowable limits provided in FSAR table 3.9-7C are used for the parts of the restraints that are within the NF boundary. AISC portions of these restraints are designed per the criteria presented to the NRC by letter on February 25, 1985 from M. Wisenburg to H. Thompson (ST-HL-AE-1185).

In addition, a few whip restraints are used as supporting structures for pipe supports. For these restraints the loads from pipe supports are added to the pipe whip loads as discussed in FSAR Sections 3.8.3.3 and 3.8.4.3. The allowable limits of FSAR tables 3.8.3-2 and 3.8.4-2 are used.

QUESTION 210.31

SRP 3.6.2 Section III.2.a. states that the rated energy dissipating capacity shall be taken as not greater than the area under the essentially flat portion of the load deflection curve for crushable materials. Provide assurance that this guidance has been used.

RESPONSE

FSAR Section 3.6.2.3.4.1.2.(d.2) indicates that crushable energy-absorbing material is designed to crush only to 80% of the "maximum strain at uniform crushable strength", i.e. 80% of the area under the essentially flat portion of the load-deflection curve. This conforms with SRP 3.6.2, Section III.2.a.

QUESTION 210.32

No discussion could be found in the FSAR regarding design stress limits for Class 1 piping in the break exclusion zone. If there are any Class 1 lines in the break exclusion zone, provide the required design limits.

RESPONSE

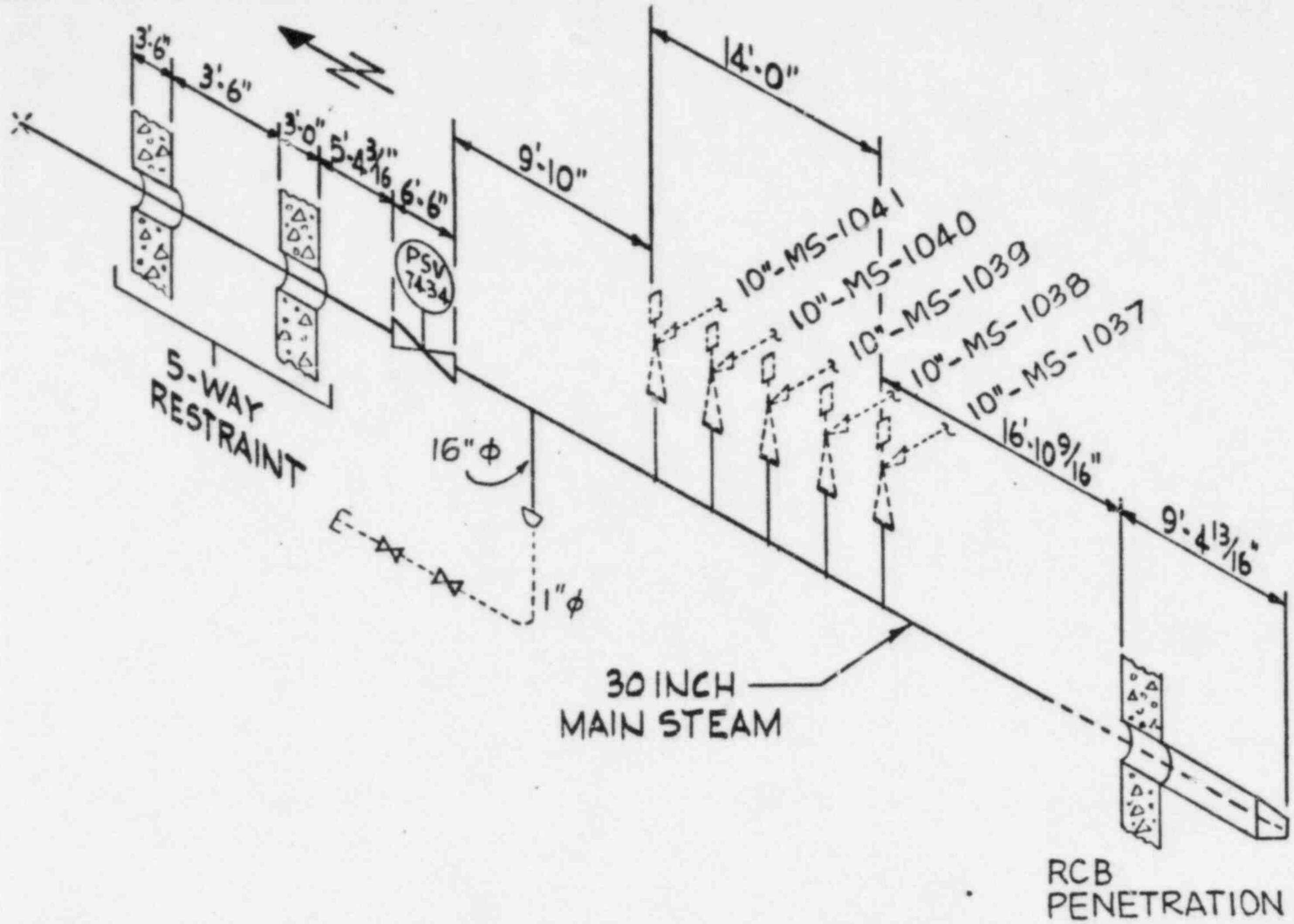
There is no Class 1 piping in the break exclusion zone for STP.

QUESTION 210.33

The criteria in the FSAR for designating the break exclusion zones on piping in the containment penetration areas require further justification. Identify all branch lines that are considered as part of the break exclusion zone. Provide drawings and/or other information quantifying the lengths of pipe for all systems including branch lines defined by criteria of Section 3.6.2.1.1.5 of the FSAR.

RESPONSE

The only lines that are designated as "break exclusion zones" are the main steam and feedwater main runs in the IVC as shown on Figures Q210.33-1 and 2. The break exclusion area extends from the containment penetrations through the five-way restraints at the IVC north walls. The only branches included are the SRV lines and the short 16" diameter stubs for main steam drain and instrumentation.



DIMENSIONS ARE APPROXIMATE

SOUTH TEXAS PROJECT UNITS 1 & 2

Pipe Break Exclusion Zone for Main Steam (Typical)

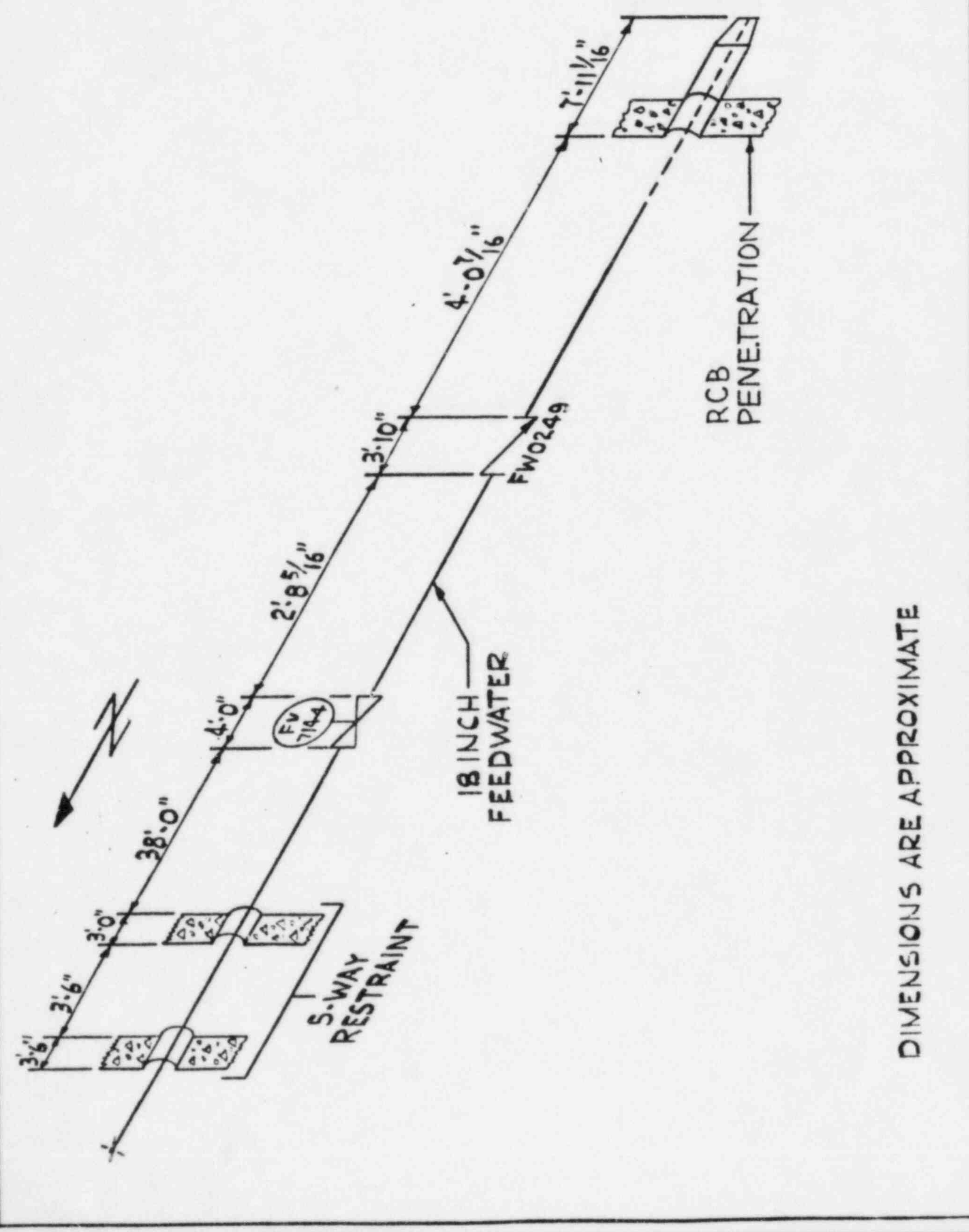
Q&R 210.33-1

Attachment 1

ST-HL-AE-1296 Page 18

SOUTH TEXAS PROJECT
UNITS 1 & 2

Pipe Break Exclusion
Zone for Feedwater
(Typical)
Q&R 210.33-2



QUESTION 210.34

Provide assurance that 100% volumetric inservice examination of all pipe welds in the break exclusion zone will be conducted during each inspection interval as defined in IWA-2400, ASME Code, Section XI.

RESPONSE

As discussed in FSAR Section 6.6.8, circumferential and longitudinal pipe welds within the break exclusion zone of high energy fluid system piping at containment penetrations will be 100% volumetrically examined during the preservice examination and during each inspection interval of the inservice inspection program in accordance with ASME Code Section XI and SRP 6.6.

QUESTION 210.35

Discuss how high energy leakage cracks were considered.

RESPONSE

STP uses the break/crack criteria contained in FSAR Section 3.6.2.1 in performing its high energy line break reviews. This conforms to the requirements of NUREG 75/087 Section 3.6. The requirement for postulating high energy leakage cracks added in NUREG 0800 is not applicable to STP.

In lieu of postulating high energy leakage cracks for environmental effects, certain non-mechanistic full circumferential breaks are postulated to establish the environmental conditions inside containment. The bulk containment effects due to leakage cracks is enveloped by these breaks.

FSAR Table 3.6.1-3, "Design Comparison to NRC Branch Technical Positions MEB 3-1", will be revised to indicate this partial conformance to BTP MEB 3-1, Section B.1.e.

QUESTION 210.36

BTP MEB 3-1, Section B.2.C specifies criteria for postulating through-wall leakage cracks for moderate energy fluid systems in areas other than containment penetration including ASME Code Class 1, 2, 3 and non-safety class piping both inside and outside containment. FSAR Section 3.6.2.1.2 states cracks in moderate energy ASME Code Class 1 piping are not postulated. Provide justification for not postulating cracks in moderate energy Class 1 piping.

RESPONSE

There are no ASME Class 1 moderate energy piping systems. FSAR Section 3.6.2.1.2 will be revised to reflect this response.

QUESTION 210.37

Provide a schedule for submission of response to NRC Question 110.6

RESPONSE

The response to Question 110.6 and the revised Section 3.6.2.1.1 will be submitted in a future amendment. Attached is a markup of the affected FSAR pages that reflect the changes.

Question 110.6

In Section 3.6.2.1.2 of the FSAR, it is stated that intersections of branch lines with the main piping run need not be considered as terminal ends when so justified in the analysis. It is our position that a branch connection to a main run need not be considered as a terminal end when all of the following conditions are met:

1. The branch and main runs are of comparable size and fixity (i.e., the nominal size of the branch is at least one-half of that of the main).

2. The branch and main runs are modeled as a common piping system during the piping stress analysis.

Expand Section 3.6.2.1.2.1(1) to correspond with this definition of terminal ends.

Response

The response is provided in revised subsection 3.6.2.1.1 ~~to this question will be provided in a later amendment.~~ 40

3.6.2.1.1 High-Energy Break Locations: With the exception of those portions of the piping identified in Section 3.6.2.1.1.5, breaks are postulated in high-energy piping at the following locations:

1. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1 - Class 1 Piping.

   a. The discrete break locations and orientations in the RCL are derived on the basis of stress and fatigue analysis. These postulated break locations and the methods used to determine them are described in Ref. 3.6-1. An analysis of each individual RCL confirms the break locations defined in Ref. 3.6-1. The stresses and cumulative usage factors resulting from seismic events are included in the stresses and cumulative usage factors which are discussed in Section 3.6.2.5 to verify the design basis break locations in the RCL noted therein.

      At all postulated RCL circumferential break locations, the piping is restrained so that the separation results in a limited flow area. Longitudinal breaks are assumed to have an opening area equal to one flow area of the pipe.

      A complete discussion of postulated RCL breaks is provided in Ref. 3.6-1.

   b. Pipe breaks are postulated to occur at the following locations in ASME Code Section III Class 1 piping runs or branch runs outside the RCL as follows: 40

      1) At terminal ends of the piping, including:

         a) Piping connected to structures, components, or anchors that act as essentially rigid restraints to piping translation and rotational motion due to static or dynamic loading.

         b) High/moderate-energy boundary such as piping runs which are maintained pressurized during normal plant conditions for only a portion of the run, i.e., up to the first normally closed valve. The terminal end of such piping is the piping connection to the closed valve.

         c) Branch intersection points are considered a terminal end for the branch line except where ~~unless~~ the branch and the main piping systems are modeled in the same ~~static, dynamic, and thermal analyses.~~

Insert

SAR CR 572) At intermediate locations where the following conditions are satisfied.

         a) The maximum stress range between any two load sets, derived on an elastically calculated basis by Equation (10) and either Equations (12) or (13) of subarticle NB-3653 of ASME Code Section III, under loadings associated with the OBE and normal and upset plant conditions, exceeds 2.4 S_m, or

Insert

"SAR CR 572"

...piping stress analysis and the branch line is shown to have a significant effect on the main run behavior (i.e. the nominal size of the branch line is at least one-half of that of the main or the ratio of the moment of inertia of main run pipe to the branch line is ~~equal to or greater~~ less than 10)

QUESTION 210.38

Justify not considering the following primary system transients for normal conditions listed in FSAR, Section 3.9.1.1.6.

1. Reactor coolant pumps startup and shutdown

2. Reduced temperature return to power

RESPONSE

The design transients for STP are based on Westinghouse internal design criteria documents, Systems Standard Design Criteria 1.3, Rev. 2, and 1.3, Appendix A. These documents do not include reactor coolant pumps startup and shutdown or reduced temperature return to power transients. These two transients were not specifically considered for Westinghouse plants designed during the time frame for which these documents are in effect.

However, in the case of the first transient, i.e., reactor coolant pumps startup and shutdown, W assumes that variations in RCS primary side temperature and in pressurizer pressure and temperature are negligible and that the steam generator secondary side is completely unaffected. It is considered by Westinghouse that due to the overall number of transient events considered in the design of STP, not including this transient in the design has a minimal effect.

Reduced temperature return to power is not considered in the RCS design basis for STP, therefore that transient is prevented from occurring by operational limitations contained in the proposed technical specifications.

The NRC MEB has reviewed and approved similar plants (Comanche Peak and Byron) which do not consider these two transients.

QUESTION 210.39

FSAR Section 3.9.1.3 states that experimental stress analysis method has been used for essential cooling water underground aluminum bronze piping. Provide the acceptance criteria used for the test program described in FSAR.

RESPONSE

The design of the Essential Cooling Water (ECW) pipe is based on analyses that satisfy the applicable load combinations and stress limits for ASME Class 3 piping as stated in the FSAR Section 3.9.1. Therefore, the experimental stress analysis method of FSAR Section 3.9.1.3 for the ECW pipe will be deleted in a future amendment.

QUESTION 210.40

Identify components for which inelastic analysis has been used. If any, provide details of methods used.

RESPONSE

Inelastic analysis has not been used to qualify any components including piping. Inelastic analysis is sometimes used to evaluate plant response due to pipe break as discussed in Section 3.6 of the FSAR.

QUESTION 210.41

SRP 3.9.2 requires a list of systems for which visual inspections and measurements (as needed) will be performed during the pre-operational piping testing program. Provide a list of systems to be included in the pre-operational testing program.

RESPONSE

The following systems will have visual inspection and measurements (as needed) during the pre-operational test program:

Auxiliary Feedwater
Component Cooling
Containment Spray (except spray header)
Chemical and Volume Control
Main Feedwater (safety-related portion only)
Main Steam (safety-related portion only)
Residual Heat Removal System
Safety Injection System
Essential Cooling Water System
Diesel Generator
Reactor Coolant
Essential Chilled Water (safety-related portion only)
Fuel Pool Cooling and Cleanup
Reactor Coolant Pressurizer System (PORV Discharge Lines)
Steam Generator Blowdown

QUESTION 210.42

Provide the acceptance criteria that will be used to determine if the vibration levels observed or measured during the pre-operational testing are acceptable. Specifically address how the vibration amplitudes will be related to a stress level and what stress levels will be used for both steady-state and transient vibration.

RESPONSE

During pre-operational testing, normal operating modes will be observed for vibration. Piping will be visually inspected to determine the acceptability of the steady state vibrations. Vibration amplitudes will be related to stress levels following the guidance of ANSI/ASME standard OM3, 1982, except as amended below. The piping will be monitored by instrumentation at locations where vibrations appear to be excessive to demonstrate that the measured pipe deflections when converted to stress will not exceed the following limits.

For steady-state vibration, the maximum calculated alternating stress intensity S_{alt} shall be limited as defined below:

(a) For ASME Class 1 piping systems:

$$S_{alt} = \frac{C_2 K_2 M}{Z} \leq \frac{S_{el}}{\alpha}$$

where

C_2 = secondary stress index as defined in the ASME Code

α = allowable stress reduction factor: 1.3 for materials covered by Fig. I-9.1; or 1.0 for materials covered by Figures I-9.2.1 or I-9.2.2 of the ASME Code

K_2 = local stress index as defined in the ASME Code

M = maximum zero to peak dynamic moment loading due to vibration only, or in combination with other loads as required by the system design specification

S_{el} = $0.8S_A$ where S_A is the alternating stress at 10^6 cycles from Figure I-9.1; or S_A at 10^{11} cycles from Figure I-9.2.2 of the ASME Code. Curves A, B and C from Figure I-9.2.2 will be used per the criteria stated in that figure. The user shall consider the influence of temperature on the Modulus of Elasticity

Z = section modulus of the pipe

RESPONSE (Continued)

(b) For ASME Class 2 and 3 piping, ANSI B31:

$$S_{alt} = \frac{C_2K_2\ M}{Z} \leq \frac{S_{el}}{\alpha}$$

where

C_2K_2 = 2i

i = stress intensification factor, as defined in Subsection NC and ND of the ASME Code or B31

QUESTION 210.43

It is the staff's position that all essential safety-related instrumentation lines should be included in the vibration monitoring program during pre-operational or startup testing. We require that either a visual or instrumented inspection (as appropriate) be conducted to identify any excessive vibration that will result in fatigue-failure. Provide a list of all safety-related small bore piping and instrumentation lines that will be included in the initial test vibration monitoring program.

RESPONSE

Small bore piping will be included in the pre-operational test program as described in the response to Question 210.41. Inspection of piping systems will include both large bore and small bore piping. Essential safety-related instrumentation lines up to the first rigid guide support will also be included in the vibration monitoring program during pre-operational testing. If observations suggest that other spans are being excited, further inspection will be conducted on a case-by-case basis.

QUESTION 210.44

Discuss how floor response spectra curves are broadened for NSSS and BOP scope.

RESPONSE

As discussed in Section 3.7.2.9 of the FSAR, the Floor Response Spectra (FRS) were broadened by $\pm$ 10 percent of each frequency point when three soil cases (upper bound, average, and lower bound) were considered in the finite element soil-structure interaction analysis. In cases where only the average soil case was considered, the FRS were broadened by at least $\pm$ 15 percent.

QUESTION 210.45

Discuss how closely spaced modes are combined for BOP scope.

RESPONSE

For BOP scope, closely spaced modes are combined in accordance with the grouping method described in Reg. Guide 1.92, EQ. 4 as discussed in FSAR Sections 3.7.3A.1.1 and 3.7.3A.7. The ten percent (10%) criteria is used.

QUESTION 210.46

On page 3.7-21 of the FSAR, it is stated that in certain cases, such as with auxiliary piping connected to the reactor coolant loop, multiple spectra have been used to reduce the excessive conservatism in supplying enveloped spectra over the entire length of piping. Discuss how multiple spectra are used.

RESPONSE

1. NSSS Scope

   For piping and components supported at multiple elevations Westinghouse uses the most limiting spectra in performing seismic analysis.

   Multiple spectra are not used in Westinghouse scope analyses.

2. BOP Scope

   Multiple response spectra are used when use of an enveloped spectra results in an excessively conservative design. In such cases supports, anchors, and nozzles are excited by their corresponding response spectra. For example, a piping system connected to the reactor coolant loop (RCL) and supported by the internal structure will have two response spectra as the forcing functions. The RCL spectrum is applied at the RCL-auxiliary piping interface and the Reactor Containment Building internal structure spectrum is used at the support locations. The responses due to multiple spectra are combined by absolute summation followed by modal summation for each direction, then combination for directions. Modal and directional summation is in accordance with Regulatory Guide 1.92. Bechtel computer program ME101 "Linear Elastic Analysis of Piping Systems" is used for multiple response spectra analysis. This computer program is discussed further in FSAR Section 3.9.1.2.2.1.

QUESTION 210.47

SRP Section 2.9.2.III.2.a.(2)(c) states that to obtain an equivalent static load on equipment or component which can be represented by a simple mode, a factor of 1.5 is applied to the peak acceleration of the applicable floor response. FSAR Section 3.7.3B.1.7 does not comply with this guidance. Provide justification for not using a factor of 1.5.

RESPONSE:

SRP Section 3.7.2 agrees with the above statement concerning a factor of 1.5 applied to the peak acceleration but also notes that a value less than 1.5 may be used if justified.

For rigid equipment, since there is no resonance or magnification of the floor response, no additional factors are applied to the high frequency acceleration levels of the applicable floor response when calculating the seismic acceleration coefficient.

Limited Flexible Equipment is defined as having only one (1) predominant mode in the frequency range subject to possible amplification (< 33 Hz). In performing the static analysis as defined in FSAR Section 3.7.3B.1.7, the total weight of the equipment or component is multiplied by the amplified response at its calculated fundamental natural frequency. This provides a conservative equivalent static load for this equipment or component.

For flexible equipment and piping Westinghouse uses dynamic analyses.

QUESTION 210.48

Provide additional information to justify the use of a multiplication factor of 1.0 in the equivalent static load method for design of cable tray hangers and heating, ventilating and air conditioning (HVAC) duct supports.

RESPONSE

As stated in Section 3.7.3A.1.2 of the FSAR, dynamic analyses using the modal response spectrum method were performed for typical cable tray and HVAC support systems. The seismic force and moment response obtained from the dynamic analyses is established to be less than the corresponding response from the equivalent static method using a factor of 1.0 times the peak acceleration of the applicable floor response spectra. Therefore, use of the multiplication factor of 1.0 in analyses by equivalent static method is justified.

This approach was reviewed by the Structural Engineering Branch during the STP audit during the week of January 7, 1985.

QUESTION 210.49

SRP 3.9.2.II.2.h specifies criteria for using constant vertical static factors. The use of constant vertical static factors is acceptable only if it can be justified that the structure is rigid in the vertical direction. Provide assurance that this guidance has been used.

RESPONSE

1. NSSS Scope

   Constant vertical static factors are not used by Westinghouse.

2. BOP Scope

   Constant vertical load factors are not used to obtain vertical response loads for the seismic design of Category I Structures, systems and components. Multimass dynamic analyses for both horizontal and vertical directions of excitation are performed to obtain the seismic responses and floor response spectra.

   For subsystems within structures, when the floor response spectra are used to define vertical input motion and/or loads for the Seismic Qualification and/or design of equipment and components, the rigidity of the structural subsystems is taken into consideration. Parametric analyses have been performed to determine the minimum subsystem frequencies required to assure effectively-rigid subsystem behavior that justifies use of the floor vertical response spectra directly without any additional amplification to account for subsystem flexibility. The established frequency limits are implemented in the Project as a specific requirement for the design of structural subsystems that support safety-related equipment. Subsystems identified to have low frequencies, if any, are stiffened to comply with the established frequency limits. FSAR Section 3.7.2.10 will be revised to reflect this response.

   This approach was reviewed by the Structural Engineering Branch during the STP audit during the week of January 7, 1985.

3.7.2.8 Interaction of Non-Category I Structures with Seismic Category I Structures. Non-Category I structures in proximity of Category I structures are checked to verify that during the extreme loading conditions of an SSE they do not collapse onto Category I structures. Interaction of seismic Category I piping with nonseismic Category I piping is described in Subsection 3.7.3.13. 34

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra. Compliance with NRC criteria regarding response spectral values at the foundation levels in the free-field required a wide variation in soil dynamic shear moduli in SSI analyses (refer to Subsection 3.7.2.4.1).

In constructing response spectra, shifting of the peaks with respect to natural frequencies by a minimum percentage is introduced to account for the uncertainties associated with computed natural frequencies of the structure. In cases where analyses have been performed for upper-bound, average and lower-bound shear moduli of soil (see Subsection 3.7.2.4.3), the frequency variation, $\pm f_j$, is determined by taking the SRSS of a minimum variation of $0.05f_j$ and the individual frequency variation $(\Delta f_j)_n$, that is:

$$\Delta f_j = \sqrt{(0.05f_j)^2 + \sum(\Delta f_j)^2_n} \qquad \text{(Equation 3.7.2-17)}$$

A value of $0.10f_j$ is used if the actual computed value of f_j is less than $0.10f_j$.

In cases where only one soil case is considered, the spectrum is shifted by at least ±15 percent of each frequency. 34

are not used to obtain

3.7.2.10 Use of Constant Vertical Load Factors. Constant vertical load factors ~~as~~ vertical response loads for the seismic design of Category I structures, systems and components. ~~are not used~~. ~~Instead,~~ Multimass dynamic analyses for both horizontal and vertical directions of excitation are performed as described in Subsection 3.7.2.1, and a combination of three component earthquake responses is made.

Insert

3.7.2.11 Method Used to Account for Torsional Effects. The actual three-dimensional soil/structure system is idealized and approximated by two-dimensional plane-strain models. Thus, the insignificant torsional motion effect on the development of the superstructural foundation motion is neglected in the SSI analyses. However, in the calculation of structural responses, the torsional effect has been incorporated in the three-dimensional mathematical model by providing a torsional spring at the foundation. Computational procedure for torsional stiffness has been given in Subsection 3.7.2.3.

Torsional effects within the structure due to eccentricities between center of mass and center of rigidity are taken into account by connecting these two centers by a rigid link. This has been discussed in Subsection 3.7.2.3.

3.7.2.12 Comparison of Responses. Since only one method of analysis (see Table 3.7-2) has been used for seismic analysis of each structure, no comparison of responses by the other method has been made. Most of the major Seismic Category I structures have been analysed using the modal analysis time-history method. Since the time-history method involved direct

Insert

P For subsystems within structures, when the floor response spectra are used to define vertical input motion and/or loads for the seismic qualification and/or design of equipment and components, the rigidity of the structural subsystems is taken under consideration. Parametric analyses have been performed to determine the minimum subsystem frequencies required to assure effectively-rigid subsystem behavior that justifies use of the floor vertical response spectra directly without any additional amplification to account for subsystem flexibility. The results from the parametric analyses indicate that for structural subsystems whose vertical natural frequencies are above 8 cps in the MEAB, 10 cps in the RCB, 12 cps in the FHB and 16 cps in the DGB, the effect of subsystem flexibility on the floor vertical response spectra is insignificant. These frequency criteria are implemented in the project design criteria as a basic design requirement satisfied either by the initial design or by subsequent stiffening of the structural subsystems.

QUESTION 210.50

FSAR Section 3.7.3A.2 states that a minimum value of five cycles per seismic event (one SSE and five OBE's) is selected for BOP seismic Category I systems and components. With respect to NSSS scope, FSAR Section 3.7.3B.2 states that a time history study has been conducted to arrive at a realistic number of maximum stress cycles per OBE occurrence. As a result of this study, 10 maximum stress cycles for flexible equipment and five maximum stress cycles for rigid equipment for each OBE occurrence are used for fatigue evaluation of Westinghouse systems and components. However, FSAR Table 3.9-8, Summary of Reactor Coolant System Design Transients, lists 400 cycles for OBE and 1 occurrence for SSE. Provide justification for the difference of earthquake cycles listed in the above referenced FSAR sections and table. Include in your discussion, the nonconformance to SRP Section 3.9.1.II.2.b criteria of a minimum of 10 maximum stress cycles per seismic event (one SSE and five OBE).

RESPONSE

Concerning the discrepancy in cycles of OBE between FSAR Sections 3.9.1.1.7.9 and 3.7.3A.2 and Table 3.9-8, the twenty (20) occurrences of twenty (20) cycles each as mentioned in Table 3.9-8 will be changed to five (5) occurrences of ten (10) cycles each. The twenty (20)-twenty (20) criteria is a conservative measure used in the design analysis for all RCS equipment and components per the applicable equipment specifications.

The reactor coolant system and the piping design in Westinghouse scope uses only the ten stress cycles per event criteria. Therefore, reference to the use of five (5) stress cycles per event for rigid equipment will be deleted from the FSAR.

For BOP systems fatique evaluation, 50 OBE cycles are considered.

3.9.1.1.7.6 Inadvertent Startup of an Inactive Loop: This transient can occur when a loop is out of service. With the plant operating at maximum allowable power level, the RCP in the inactive loop is started as a result of operator error. Reactor trip occurs on high nuclear flux. This transient is assumed to occur 10 times during the life of the plant.

3.9.1.1.7.7 Control Rod Drop: This transient occurs if a bank of control rods drops into the fully inserted position due to a single component failure. The reactor is tripped on either low pressurizer pressure or negative flux rate, depending on time in core life and magnitude of the reactivity insertion. It is assumed that this transient occurs 80 times over the life of the plant.

3.9.1.1.7.8 Inadvertent Emergency Core Cooling System Actuation: A spurious SI signal results in an immediate reactor trip followed by actuation 41
of the high-head and low-head safety injection (HHSI and LHSI, respectively) pumps. These pumps, however, do not deliver flow to the RCS, as both have shutoff heads below the minimum RCS pressure reached during the transient. This transient behaves similarly to the reactor trip from full power, with controlled steam dump and FW flow removing core residual heat after the trip. Reactor coolant temperature and pressure decrease as the control rods move into the core.

At the end of this transient, it is assumed that the plant is returned to no-load conditions, with pressure and temperature changes controlled within normal limits.

For design purposes, this transient has been assumed to occur 60 times during the 40-year design life of the plant.

3.9.1.1.7.9 Operating Basis Earthquake: The mechanical stresses resulting from the OBE have been considered on a component basis. Fatigue analysis, where required by the codes, has been performed by the supplier as part of the stress analysis report. The earthquake loads are part of the mechanical loading conditions specified in the equipment specifications. The origin of their determination is separate and distinct from those transients resulting from fluid pressure and temperature. They have been considered, however, in the design analysis. For the Nuclear Steam Supply System (NSSS) vendor scope of study, the number of occurrences for fatigue evaluation has been assumed to be 20 earthquakes at 20 cycles each (400 cycles total). For the balance-of-plant (BOP) scope of study, OBE is as defined in Section 3.7.3.2.2.
5 10 50

3.9.1.1.7.10 Excessive Feedwater Flow: This transient is defined only for the purpose of determining the adequacy of the SG, the RCS, and the pres- 30
surizer to withstand the effects of excessive FW flow. The pressure and temperature variations are considered in connection with analyzing the primary 30
and secondary sides of the SG, the RCS, and the pressurizer.

This transient is conservatively defined as an umbrella case to cover occurrence of several events of the same general nature. These include:

1. Inadvertent opening of an FW control valve.

TABLE 3.9-8 (Continued)

SUMMARY OF REACTOR COOLANT SYSTEM DESIGN TRANSIENTS

Upset Conditions (Continued)	Occurences
2. Loss of power (blackout with natural circulation in the Reactor Coolant System)	40
3. Partial loss of flow (loss of one pump)	80
4. Reactor trip from full power	
a. Without cooldown	230
b. With cooldown, without safety injection	160
c. With cooldown and safety injection	10
5. Inadvertent reactor coolant depressurization	20
6. Inadvertent startup of an inactive loop	10
7. Control rod drop	80
8. Inadvertent ECCS actuation	60
9. Operating Basis Earthquake (20 earthquakes of 20 cycles each) 5 10	~~400~~ 50 cycles
10. Excessive feedwater flow	30

Emergency Conditions*	
1. Small LOCA	5
2. Small steam break	5
3. Complete loss of flow	5

*In accordance with ASME B&PV Code Section III, emergency conditions are not included in the fatigue evaluation.

3.7.3A.1.3 Simplified Method: The simplified method involves the use of appropriate and comprehensive charts and tabulations to determine the piping spans, support loads, and types of supports. The seismic loads used in the design have been obtained by using the concept of equivalent static load method. Piping spans have been chosen to ensure that the piping stresses are within the code allowable limits.

3.7.3A.2 Determination of Number of Earthquake Cycles. The total number of significant earthquake cycles for which seismic Category I structures, systems and components are designed has been determined as a product of the number of postulated seismic events and the number of significant earthquake cycles per event.

As stated in Subsection 2.5.2.10, the duration of strong motion associated with the postulated SSE would be less than 5 seconds for which the number of significant cycles would be approximately two or three. The duration and number of cycles for the postulated OBE would be less than for the SSE. To provide a conservative design basis, a minimum ~~value~~ of ~~five~~ cycles per seismic event (one SSE and five OBEs) is selected. ten maximum stress

3.7.3A.3 Procedure Used for Modeling.

3.7.3A.3.1 Mathematical Model for Piping Systems: Modeling procedures for subsystems have been discussed in Subsection 3.7.3A.1.1.

The preparation of a mathematical model for piping dynamic analysis is based on the following guidelines:

1. The piping system is modeled as a series of finite elements with masses lumped at certain nodal points. 34
2. The mass points are selected judiciously so that their locations coincide with the locations of large valves and supporting hangers.
3. The straight piping between the mass points is divided into a large enough number of elements to obtain a good approximation of all piping frequencies and mode shapes below 33 Hz.

For piping analysis, the resonance condition in the design subsystem is eliminated whenever possible. The resonance peaks are readily identified from the appropriate response spectra. Where possible, the dominant natural frequencies are modified to avoid resonance, by providing stiffer or additional supports and changing the mass.

3.7.3A.3.2 Modeling Procedure for Cable Tray Hangers: A three-dimensional static finite element analysis is used to design the cable tray hanger. The finite element models, which represent the cable tray hangers, and transverse and longitudinal bracings, are simulated by beam elements interconnected with rotational springs at points. The dead loads, live load and seismic loads are applied simultaneously at the centers of horizontal members.

3.7.3A.3.3 Modeling Procedure for HVAC Ducts and Hangers: Design of HVAC ducts and hangers is based on the equivalent static analysis. The duct size is determined by considering the beam deformation mode (resulting from the

exceeded by the actual plant acceleration levels, the design analysis is performed again at the actual level to confirm the equipment adequacy.

3.7.3B.2 Determination of Number of Earthquake Cycles. The OBE is conservatively assumed to occur five times over the life of the plant. A time history study has been conducted to arrive at a realistic number of maximum stress cycles per OBE occurrence for all Westinghouse systems and components.

This evaluation considered both the equipment and its supporting building structure as single-degree-of-freedom systems, which tend to produce a more uniform and unattenuated response than a complex, interacting system. The natural frequencies for the building and equipment are conservatively chosen to coincide.

As a result of this study, 10 maximum stress cycles for ~~flexible~~ equipment ~~(natural frequencies less than 33 Hz) and five maximum stress cycles for rigid equipment (natural frequencies greater than 33 Hz)~~ for each OBE occurrence are used for fatigue evaluation of Westinghouse systems and components.

3.7.3B.3 Procedure Used for Modeling. Modeling technique is discussed in Subsection 3.7.3B.1.

3.7.3B.4 Basis for Selection of Frequencies. In the analysis of the Class 1 branch lines attached to the reactor coolant loop (including the surge line), the frequencies of these lines may be controlled if necessary to avoid the peak building frequencies and the lowest fundamental frequencies of the primary equipment, to maintain the equipment and support loads within allowable limits. 34

There is no specific design criteria which attempts to control the fundamental frequencies of NSSS equipment to be different from the forcing frequencies of the supporting structures. The effect of the equipment fundamental frequencies relative to the support structure forcing frequencies is, however, considered in the analysis of the NSSS equipment.

Three ranges of equipment/support behavior which affect the magnitude of the seismic acceleration are possible:

1. If the equipment is rigid relative to the structure, the maximum acceleration of the equipment mass approaches that of the structure at the point of equipment support. The equipment acceleration value in this case corresponds to the low-period region of the floor response spectra.

2. If the equipment is very flexible relative to the structure, the equipment will show very little response.

3. If the periods of the equipment and supporting structure are nearly equal, resonance occurs and must be taken into account.

Also, as noted in paragraph 3.7.3B.1, rigid equipment/support systems have natural frequencies greater than 33 Hz.

3.7.3B.5 Use of Equivalent Static Load Method of Analysis. This subject is discussed in paragraph 3.7.3B.1.7.

QUESTION 210.51

Provide the basis used for the design of piping anchors which separate seismically designed piping and non-seismic Category I piping. Include in your discussion, the loads and load combinations used and how the local pipe wall stresses are considered.

RESPONSE

In the case where an anchor is used to separate Seismic Category I piping systems from piping systems where seismic qualification is not required, the anchor is designed to meet Seismic Category I requirements. This is in agreement with Regulatory Guide 1.29, paragraph C.3 which states, "Seismic Category I design requirements should extend to the first seismic restraint beyond the defined boundaries. Those portions of structures, systems, or components that form interfaces between Seismic Category I and nonseismic Category I features should be designed to Seismic Category I requirements."

Loading conditions and load combinations for qualification of piping, components and supports are specified in Table 3.9-2.4. In the case of an anchor, the piping analysis for the piping on each side of the anchor is performed independently using the appropriate loading conditions. Anchor loads are generated for both the upstream and downstream piping runs. Anchor loads from the nonseismic Category I side include either seismic loads due to SSE or piping collapse loads. The loads from the two piping runs are then combined and used for the anchor design. Dynamic loads from the two sides are combined by SRSS. Resultant static and dynamic loads are combined absolutely as required for the appropriate plant condition as defined in FSAR Table 3.9-2.4.

Local pipe wall stresses are considered in accordance with ASME Section III, subsection NC, ND or ANSI B31.1 as appropriate. The applicable subsection is determined by the pipe class. No seismic boundary anchors are placed on ASME Class 1 piping.

QUESTION 210.52

FSAR Table 1.3-1, Comparison with Similar Facility Design, states that the new design of the reactor vessel head closure system and lower internals are different from the Comanche Peak plant. Provide additional information which describes the differences in lower internals design between STP and Comanche Peak. Specifically, describe any changes in the reactor internals design which may have resulted from utilization of the rapid Refueling concept at STP. If such changes exist, discuss the effects of these changes on the response of the reactor internals to flow-induced excitation and provide the basis for meeting the guidelines of Regulatory Guide 1.20 and maintaining Indian Point, Unit 2 as the prototype plant for STP.

RESPONSE

No changes were made to the South Texas reactor internals resulting from the utilization of the rapid Refueling concept that would impact the vibratory response of the internals. The utilization of lifting rods in the upper internals to facilitate the removal of the upper internals with the upper head has no impact on the internals vibratory response. In fact the vibration assessment, based on flow turbulence, is only concerned with the region below the upper support plate in the lower guide tube region, inlet nozzle, downcomer and outlet nozzle locations.

The South Texas plant is based on the four loop NSSS design of Indian Point insofar as Regulatory Guide 1.20 is concerned. In addition, the South Texas plant incorporates such design enhancements as have already been reviewed and approved by the NRC staff such as neutron pads versus thermal shield and the inverted top hat design. One additional modification concerns the change to the reactor internals to permit the use of a 14-foot core. To account for this the fuel no longer rests on a lower core plate but simply rests on the lower support plate. An analytical flow-induced vibration assessment has been performed and documented for the South Texas plant. It has been concluded that the vibrational response of this plant obtained from scale model tests and instrumented plant tests, shows that the internals vibration levels are low and that the South Texas reactor internals design is adequate to assure structural integrity against flow induced vibrations.

QUESTION 210.53

FSAR Section 5.3.1.7 describes the Roto-Lok reactor vessel head closure system which is used for the STP Units 1 and 2 reactor vessel head. it also states that a prototype Roto-Lok closure system has been tested to verify this closure design. Results of these tests are presented in the WCAP-8447, December, 1974. However, Section 7 of WCAP-8447 states that, "Also, it should again be noted that the program described in this report was for development hardware and testing only. The final design and analysis for a particular vessel is performed by the vessel supplier when the Roto-Lok is actually applied by production vessels." The staff's review of the WCAP-8447 as provided in the letter from J. F. Stoltz to C. Eicheldinger dated September 2, 1977, determined that WCAP-8447 provides an acceptable basis for the preliminary design of the Roto-Lok closure system. Furthermore, in that evaluation, the staff required that for the first reactor vessel to use this closure system (South Texas Plant) the results of final design and analysis of the closure system be provided in the FSAR. The applicant is requested to provide this information. Include in your discussion how the assumptions presented in the WCAP-8447 are applicable to the STP Units 1 and 2 plant specific reactor vessels.

RESPONSE

The STP reactor vessel Roto-Lok closure system configuration is shown in Figure 210.53-1. This closure assembly used the sawtooth lug design discussed in Chapters 6 and 7 of WCAP-8447 (proprietary) with the following modifications:

1. Stud fillet radii at the top of the lug to shank junctures were increased from 0.187" to 0.250",

2. Insert fillet radii at the bottom of the lug to cylindrical I.D. junctures were reduced from 0.187" to 0.125",

3. The length of each lug was increased from 1.975" to 2.095" at the shank on each stud, and

4. The lug length was increased the same amount at the cylindrical I.D. of each insert.

The final Roto-Lok closure region design was analyzed by the STP reactor vessel vendor (Combustion Engineering) using the ANSYS three-dimensional finite element computer program. The assumptions used in the proprietary WCAP-8447 (see pages 3-5 through 3-7 and 6-1) are still applicable to this analysis with the following revisions:

1. The specified stud preload is 110% of the design pressure blow-off load during normal operation or 110% of the hydrostatic blow-off load during the hydrotest versus the 120% factor used in the WCAP.

RESPONSE (Continued)

2. The full length of the closure stud was used in the analysis instead of just the portion in the vessel and closure flanges.

3. The crown portion of the closure head was modelled with two elements through the thickness in lieu of one layer.

4. The closure region was modelled as a 5 degree wedge with the width consisting of three elements. The effect of the stud holes in the circumferential direction was also handled in a more refined fashion than in the WCAP.

5. The reactor vessel's internal surfaces in contact with the primary coolant were assumed to have an infinite heat transfer coefficient instead of a finite film coefficient associated with turbulent flow.

6. Heat transfer by conduction, radiation, and convection in lieu of just convection was assumed to occur across the air gaps between the vessel components and the studs and nuts.

7. The strength reduction factor in the fatigue analysis was increased from 3.75 to 4.0.

The results of the vessel vendor's analysis of the Roto-Lok stud assembly are presented along with the corresponding ASME code allowables in the following table. This table shows the code allowable limits are met.

Category	Governing Value	ASME Code Allowable
Design Stud Membrane Stress Intensity	34.73 ksi	34.8 ksi
Maximum Average Stud Service Stress Intensity	54.3 ksi	84.0 ksi
Maximum Stud Service Stress Intensity	76.8 ksi	126.0 ksi
Usage Factor	0.502	1.0

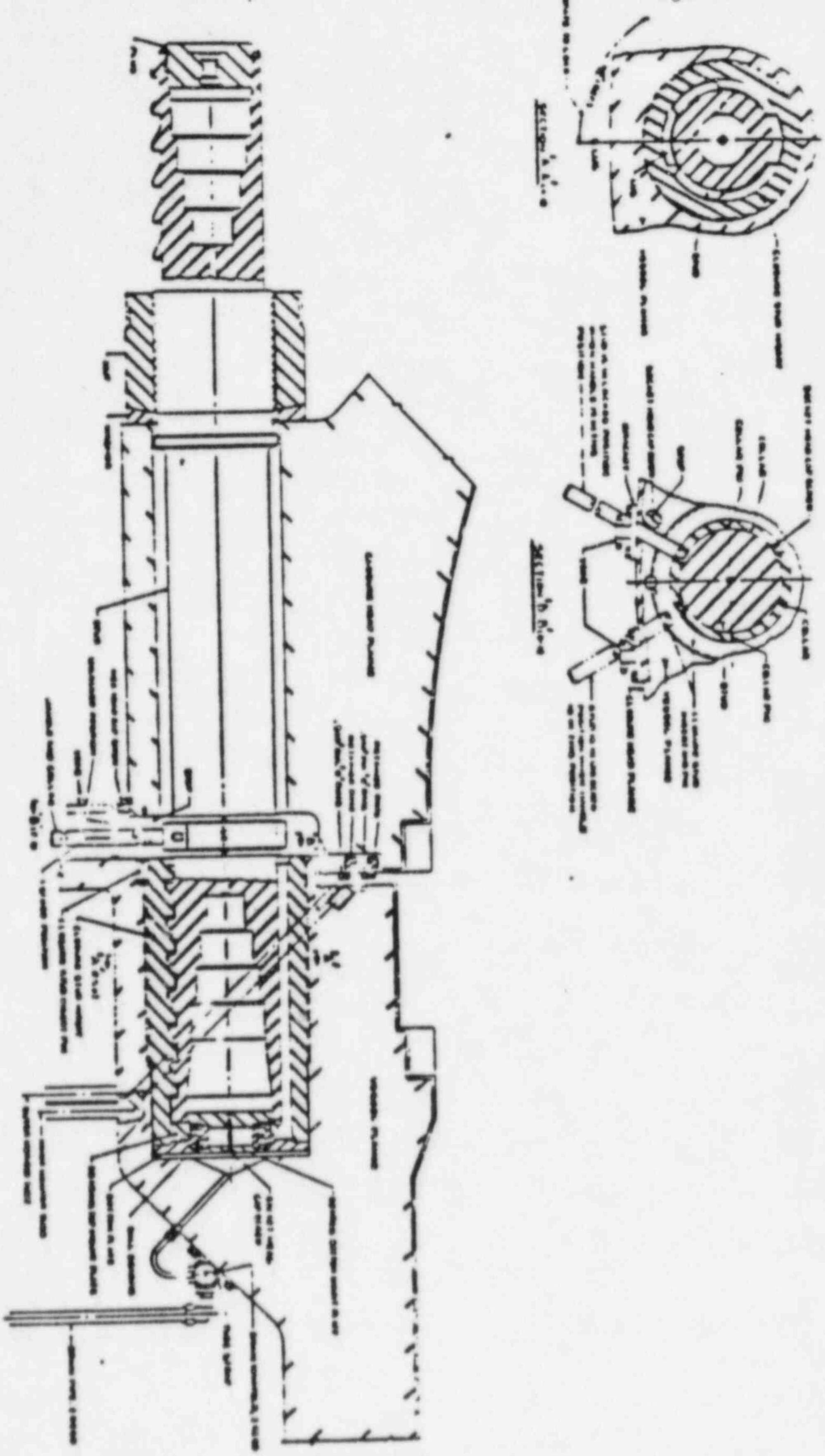


FIGURE 210.53-1. Roto-Lok Closure Assembly

QUESTION 210.54

The staff finds that there is insufficient information describing the design of safety-related HVAC ductwork and supports. Provide the design basis used for qualifying the HVAC ductwork and support structural integrity.

RESPONSE

HVAC ducts are fabricated from sheet metal and/or steel plate. The duct supports are fabricated from rolled structural shapes. All ducts and supports are galvanized.

Safety-related ducts and duct supports are designed for combinations of gravity, pressure and seismic loads utilizing allowable stresses that maintain the response within the elastic range. The seismic analysis of ducts and duct supports is based on the equivalent static method as stated in FSAR Section 3.7.3A.1.2 and 3.7.3A.3.3. Codirectional seismic responses due to longitudinal, transverse and vertical earthquakes are combined by the SRSS method or the component factor method. The component factor method is equivalent to the SRSS method, and in certain types of analyses is more practical than the SRSS method for combining codirectional responses from the three components of earthquakes. The component factor method is widely used in the industry for the design of structures, systems and components of nuclear power plants. The maximum error possible by the use of the component factor method is less than 1% with respect to the SRSS method.

The rationale for the use of the component factor method is attached. FSAR Section 3.7.3A.6 will be revised to identify the use of the component factor method as an acceptable option in addition to the SRSS method with the exception that the component factor method is not used for piping analysis.

Tables Q210.54-1 and Q210.54-2 give the load combinations and allowable stresses used in the design of duct and duct supports.

Expansion anchors (Hilti Kwik-bolts) are used occasionally in supports for safety-related HVAC ducts. As explained in response to Q210.62, the design allowable loads are based on tested ultimate load capacities with an applied factor of safety of 4 or higher. Design allowable loads are not increased for faulted or abnormal/extreme environmental loading combinations. The integrity of the expansion anchors is not compromised by normal operational vibratory motion due to the low amplitude nature of the vibration.

In specific isolated instances, as determined from pipe break analyses and/or tornado depressurization analyses, the loads due to compartment pressurization/depressurization and/or jet impingement are included in the design of safety-related ducts. The loads due to pipe break are considered as additive to the loads of combinations (2) and (3) of the above tables.

QUESTION 210.54 (Continued)

Safety-related HVAC ducts are designed using analytical guidelines established from testing results. Following are the principal codes and standards used in the design:

1. AISC - "Specification for the Design Fabrication and Erection of Structural Steel for Buildings", 1969, including Supplements 1 and 2.

2. AISC - "Code of Standard Practice for Steel Buildings and Bridges", 1976

3. AISI - "Specification for the Design of Cold-Formed Steel Structural Members" 1968 and AISI - "Supplementary Information on the 1968 Edition of the Design of Cold-Formed Steel Structural Members" 1971.

   Exception: Section 2.3.4 of AISI 1968 states that the ratio h/t of the webs of flexural members shall not exceed 500. Actual tests performed on HVAC ducts substantiate the use of w/t and h/t ratios of up to 1500 for ducts. The STP approach allows these ratios to exceed 500, but restricts these values to less than 1500.

4. AWS - Structural welding code AWS D1.1, 1977 and code for welding zinc-coated steel AWS 19.0

5. OSHA - Department of Labor, Volume 37, Number 202, Part II - Applicable sections on platforms, handrails and ladders

6. SMACNA - Sheet metal and air conditioning contractor's national association high pressure duct construction standards and low-pressure duct construction standards

Supports for safety-related HVAC ducts are designed by the working stress method using the AISC specification, 1969 edition.

TABLE Q210.54-1

LOAD COMBINATIONS FOR HVAC DUCTS

Load Case	Loading Combination	Allowable Stress
(1)	D + P	0.6 Fy
(2)	D + P + E	0.6 Fy
(3)	D + P + E_s	0.9 Fy
(4)	D + P + W_{tp}	0.9 Fy

Symbols used in load combinations:

D - Dead weight of duct
P - Maximum operating pressure inside duct
E - Operating basis earthquake load
E_s - Safe shutdown earthquake load
Fy - Minimum specified yield strength of duct material
W_{tp} - Tornado differential pressure

TABLE Q210.54-2

LOAD COMBINATIONS FOR HVAC DUCT SUPPORTS

Load Case	Loading Combination	Allowable Stress
(1)	D	F_s
(2)	D + E	F_s
(3)	D + E_s	1.5 F_s or 0.9 Fy, whichever is smaller

Symbols used in load combinations:

D - Dead load
E - Operating basis earthquake load
E_s - Safe shutdown earthquake load
F_s - Allowable stress for support material governed by AISC or AISI as applicable
Fy - Minimum specified yield strength of support material

ATTACHMENT

RESPONSE TO QUESTION No. 210.54

VALIDITY OF THE COMPONENT FACTOR METHOD

In the component factor method, the following equation is used to determine the total seismic load:

$$R_{total} = R_i + 0.4R_j + 0.4R_k \qquad (1)$$

In the following, adequacy of the above equation is demonstrated. First, consider a combined response, R' defined as follows:

$$R' = R_i + 0.414R_j + 0.318R_k \qquad (2)$$

In which

$$R_i \geq R_j \geq R_k \geq 0 \qquad (3)$$

Let

$$R_j = \bar{R}_j + R_k \quad (\bar{R}_j = 0 \text{ if } R_j = R_k)$$

$$R_i = \bar{R}_i + R_j = \bar{R}_i + \bar{R}_j + R_k \quad (\bar{R}_i = 0 \text{ if } R_i = R_j) \qquad (4)$$

The SRSS method gives:

$$R = \{(\bar{R}_i + \bar{R}_j + R_k)^2 + (\bar{R}_j + R_k)^2 + R_k^2\}^{1/2}$$

$$= \{3R_k^2 + 2\bar{R}_j^2 + \bar{R}_i^2 + 2\bar{R}_i(\bar{R}_j + R_k) + 4\bar{R}_j R_k\}^{1/2} \qquad (5)$$

According to Eq. (2)

$$R' = (\bar{R}_i + \bar{R}_j + R_k) + 0.414(\bar{R}_j + R_k) + 0.318R_k$$

$$R' = 1.732R_k + 1.414\bar{R}_j + \bar{R}_i = \{[1.732R_k + 1.414\bar{R}_j + \bar{R}_i]^2\}^{1/2}$$

$$R' = \{3R_k^2 + 2\bar{R}_j^2 + \bar{R}_i^2 + 2\bar{R}_i(1.414\bar{R}_j + 1.732R_k) + 4.9\bar{R}_j R_k\}^{1/2} \qquad (6)$$

Comparing Eqs. (5) and (6), it is obvious that the combined response calculated according to Eq. (2) is always more conservative than the combined response by the SRSS method. In the special case that $R_i = R_j = R_k$, they become identical to each other, i.e., $R = R' = \sqrt{3}R_k$.

For convenience of engineering applications, Eq. (2) can be simplified by replacing the factors 0.414 and 0.318 by a common factor of 0.4. This reduces Eq. (2) to Eq. (1). By inspection, the maximum probable error of Eq. (1) with respect to the SRSS method is less than 1%. This maximum error occurs when $R_k = 0$ and $R_i = R_j$. In this special case, the SRSS method gives $R = 1.41R_i$ and Eq. (1) gives $R = 1.4R_i$.

Question 210.54

VALIDITY OF THE COMPONENT FACTOR METHOD (continued)

In implementing Eq. (1), permutations of the component factors (1.0, 0.4, 0.4) and, positive and negative values of the seismic stresses are taken into account. The resulting 24 sub-combinations will contain the most critical case (i.e., the maximum absolute value of the total seismic response) and will be combined with stresses due to other loads using proper sign. The most critical case, thus identified, forms the basis of the final design.

vertical, transverse and longitudinal restraints) and the sheet deformation mode (resulting from the stiffener effect).

3.7.3A.4 Basis for Selection of Frequencies. In the dynamic analysis, fundamental frequencies of subsystems and equipment are calculated based on the mass and stiffness characteristics of the systems. The seismic accelerations which the system must withstand are then determined from the applicable floor design spectra.

Three ranges of dynamic behavior of systems that have been considered for the magnitude of the seismic acceleration are:

1. In cases where the system is rigid relative to the structure, the maximum acceleration of the system approaches the low-period region of the floor response spectra.

2. In cases where the equipment is very flexible relative to the structure, the internal distortion of the structure is unimportant and the system behaves as though supported on the ground. 34

3. In cases where the periods of the system and supporting structure are nearly equal, resonance occurs and is taken into account.

Rigid systems have natural frequencies greater than 33 Hz.

3.7.3A.5 Use of Equivalent Static Load Method of Analysis. The use of equivalent static load method is discussed in Subsection 3.7.3A.1.2.

3.7.3A.6 Three Components of Earthquake Motion. The subsystem and equipment responses have been determined using the modal unidirectional response spectrum analyses. The combination of modal responses from ~~undirectional~~ analyses are performed by methodology that is in accordance with RG 1.92. The total response due to three directional excitation is then obtained by using the square root of the sum of the squares (SRSS) method or the component factor method (1, 0.4 and 0.4) for the combination of codirectional responses from each excitation. ~~The resultant undirectional responses are then combined to obtain total response by using the square-root-of-the-sum-of-the-squares (SRSS) method.~~

QUESTION 210.55

Provide the basis for assuring that ASME Code Class 1, 2 and 3 piping systems are capable of performing their safety function under all plant conditions. Describe the methodology used to assure the functional capability of essential piping systems when service limites C or D are specified.

RESPONSE

Loading combinations for the various plant conditions (i.e. normal, upset, emergency, faulted) and the corresponding stress limits for ASME Code Class 1, 2, and 3 piping and pipe supports are given in the FSAR Section 3.9. These stress limits are in compliance with the code requirements and form the basis for assuring that the piping systems are capable of performing their safety functions under specified plant conditions.

NSSS Scope

For essential piping systems, the functional capability will be satisfied by analysis using the following method:

Component	Limit	Calc Method
Straight pipe, welds, reducers	1.8 Sy	NB-3650, Eq. (9)
Branches, tees	2.0 Sy	NB-3650, Eq. (9)
Elbows, 5D bends	1.8 Sy	NB-3650, Eq. (9)*

* B_1 and B_2 indices are replaced as follows:

$$0 \leq B_1 = -0.1 + 0.4h \leq 0.5$$

and $B_1 = 0.5$ for $B_2 = 1.0$

$$B_2 = \begin{bmatrix} \dfrac{1.3}{h^{2/3}} & , \text{ for } \alpha_o > 90° \\ \dfrac{0.895}{h^{0.9122}} & , \text{ for } \alpha_o = 90° \\ 1.0 & , \text{ for } \alpha_o = 0° \end{bmatrix} \text{ and } B_2 \geq 1.0$$

linear interpolation for $0 < \alpha_o < 90°$

Where

$$h = \frac{tR}{r_m^2}$$

R = bend radius

r_m = mean pipe radius

t = nominal wall thickness

RESPONSE (Continued)

The applicable loading cases for the Class 1 piping components to meet the functional capability limits for reactor coolant loop and the pressurizer safety and relief system are:

P_o + DWT + SSE

Where

P_o = Design pressure

DWT = Deadweight

SSE = Safe shutdown earthquake

To assure the functional capability of a Class 1 system not larger than 1 inch diameter which is analyzed to ASME Code Class 2 rules, the following stress limits will be used to supplement the level D requirements for ASME Class 2 stainless steel elbows.

$$B_1 \frac{PD_o}{2t} + B_2 \frac{Mj}{Z} \leq 1.8\ Sy$$

Where B_1 = $(-0.1 + 0.4h)$ and $0 \leq B_1 \leq 0.5$

and B_1 = 0.5 for B_2 = 1.0

$$B_2 = \begin{bmatrix} 1.3/(h^{2/3}) \text{ for } \alpha_o > 90° \\ 0.895/(h^{0.9122}) \text{ for } \alpha_o = 90° \\ 1.0 \text{ linear for } \alpha_o = 0° \end{bmatrix} \text{ and } B_2 \geq 1.0$$

Linear interpolation for $0 < \alpha_o < 90°$

Where $h = \frac{tR}{r_m^2}$ and α_o is the angle of the bend in degrees.

Other terms are as defined in NC-3600 of Section III of the ASME Code.

There are no Class 2 stainless steel elbows or bends with Do/t $>$ 50.

The loading combination for the reactor vessel head vent system is:

Po + DWT + SSE

BOP Scope

For essential piping systems, the functional capability will be satisfied by analysis using methods given in GE Topical Report, Functional Capability Criteria for Essential Mark II Piping, NEDO-21985, September 1978, or an equivalent analysis.

QUESTION 210.56

The staff review of SAR Section 3.9.3.3 finds that the design and installation details for mounting of pressure-relief devices require further clarification. Provide the following information for our review:

1. Clarify whether it is the intention of Section 3.9.3.3.2 to address BOP supplied components.

2. Clarify whether all the NSSS scope safety and relief valves transients are evaluated using detailed dynamic analysis techniques. Provide assurance that the most severe potential sequence of discharges, i.e., the maximum values of forces and moments are considered for multiple-valve discharges.

3. Provide a discussion of the basis for assuring that the valve end loads are acceptable. Specifically, address how the applicable design loads will be correctly reflected in the valve design specification.

RESPONSE

a. NSSS SCOPE

A description of the pressurizer safety and relief valve system is given in FSAR Section 3.9.3.3.1.1. This section also describes the analytical model of the system, the determination of forces, and method of analysis. The programs used in the dynamic analysis of the system are also provided in Section 3.9.3.3.1.1. All relevant valve discharge cases are evaluated using detailed dynamic analysis techniques. Discharge of the safety valves is the limiting design case for the downstream piping of all cases considered. The three safety valves are identical and have the same set pressure (± 1 percent). It was assumed that all three safety valves open simultaneously. The simultaneous opening of the safety valves results in peak loads in the common circular header. No appreciable impact in the tailpipe region, due to safety valve discharge, will occur if the valve sequencing is adjusted. The detailed design analyses performed for this discharge case illustrates a safety factor of 2 between the calculated stresses in the tailpipe and the allowable stresses.

The valve end loads are verified to be acceptable by ensuring that all calculated values are below conservative values specified in the piping design specification which has been approved by the valve engineer. Specific values are reconciled to the valve specification or vendor reports if any values exceed those specified in the piping design specification.

RESPONSE (Continued)

b. BOP Scope

1. Section 3.9.3.3.2 of the FSAR is intended to address the BOP supplied components. FSAR Section 3.9.3.3.2 will be revised to clarify applicability to BOP components.

2. BOP multiple valve discharge is applicable only to the main steam safety relief valves. The main steam safety relief discharge piping is designed so that the thrust force is transferred to the support structure, thus eliminating concerns regarding force transfer to the piping system.

3. Consideration of active valve end loads is discussed in FSAR Sections 3.9.3.2.1.2 and 3.9.3.2.3. Calculated valve end loads are compared for compliance with the allowable loads identified in the valve specifications or in the vendor documentation.

For each pressurizer safety and relief piping system, an analytical hydraulic model is developed. The piping from the pressurizer nozzle to the relief tank nozzle is modeled as a series of single pipes. The pressurizer is modeled as a reservoir which contains steam at constant pressure (approximately 2500 psia for safety system and approximately 2350 psia for relief system) and at approximately 680°F. The pressurizer relief tank is modeled as a sink which contains steam and water mixture.

Fluid acceleration inside the pipe generates reaction forces on all segments of the line which are bounded at either end by an elbow or bend. Reaction forces resulting from fluid pressure and momentum variations are calculated. These forces are defined in terms of the fluid properties for the transient hydraulic analysis.

Unbalanced forces are calculated for each straight segment of pipe from the pressurizer to the relief tank. The time histories of these forces are used for the subsequent structural analysis of the pressurizer safety and relief lines.

The structural model used in the seismic analysis of the safety and relief lines is modified for the valve thrust analysis to represent the safety and relief valve discharge. The time-history hydraulic forces are applied to the piping system lump mass points. The dynamic solution for the valve thrust is obtained by using a modified predictor-corrector-integration technique and normal mode theory. 41

The time-history solution is performed in subprogram FIXFM3. The input to this subprogram consists of the natural frequencies and normal modes, applied forces, and nonlinear elements. The natural frequencies and normal modes for the modified pressurizer safety and relief line dynamic model are determined with the WESTDYN program. The support loads are computed by multiplying the support stiffness matrix and the displacement vector at each support point. The time-history displacements of the FIXFM3 subprogram are used as input to the WESDYN2 subprogram to determine the internal forces, deflections, and stresses at each end of the piping elements.

The loading combinations considered in the analysis of the PSARV piping are given in Table 3.9-2.4A. These load combinations are consistant with the final recommendations of the piping subcommittee of the EPRI pressurized water reactor (PWR) PSARV performance test program.

Pressure-relieving devices have been constructed, located, and installed so that they are readily accessible for inspection and repair and so that they cannot be readily rendered inoperative. Safety or relief valves have been set to relieve at a pressure not exceeding the design pressure of the vessel at the design temperature.

(B.O.P. scope):

3.9.3.3.2 ~~Design and Installation Details for Mounting of Pressure Relief Devices~~ Pressure vessels have been protected by pressure-relieving devices to meet applicable code requirements such as ASME Code, Section III, Section VIII, ANSI B31.1, and RG 1.67 of October 1973. 4

The load due to reaction force from the opening and subsequent venting of a safety valve or relief valve(s) includes consideration of both momentum and

QUESTION 210.57

The staff review of FSAR Section 3.9.3.4 finds that there is insufficient information regarding the design of ASME Class 1, 2 and 3 equipment and component supports. Per SRP Section 3.9.3, our review includes an assessment of design and structural integrity of the supports. The review addresses three types of supports: (1) plate and shell, (2) linear, and (3) component standard types. For each of the above three types of supports, excluding pipe supports, provide the following information (as applicable) for our review:

(a) Describe (for typical support details) which part of the support is designed and constructed as component supports and which part is designed and constructed as building steel (NF vs. AISC jurisdictional boundaries).

(b) Provide the complete basis used for the design and construction of both the component support and the building steel up to the building structure. Include the applicable codes and standards used in the design, procurement, installation, examination, and inspection.

(c) Provide the loads, load combinations and stress limits used for the component support up to the building structure.

(d) Provide the deformation limits used for the component support.

(e) Describe the buckling criteria used for the design of component supports. Specifically, describe how the "A" term used in the response to NRC Question 110.19 was defined.

RESPONSE

A. NSSS SCOPE:

1. Class 2 and 3 component supports

(a) The supports are linear type or plate and shell type, and are part of the equipment. A typical support is welded to the equipment directly to the pressure boundary or wear plate, and is required to be rigidly attached to a foundation. The equipment designed to Code editions prior to the inclusion of Subsection NF into the ASME Code have the supports designed in accordance with the requirements of the AISC manual; equipment designed to the Code editions after the inclusion of Subsection NF are designed in accordance with ASME Code Subsection NF.

RESPONSE (Continued)

(b) The design, construction, examination, and inspection of the auxiliary equipment supports are in accordance with the requirements of ASME Subsection NF or AISC, depending on the procurement date of the equipment as discussed in the Part (a) response. In accordance with Westinghouse auxiliary equipment specifications, the equipment is required to be rigidly mounted.

(c) The loads and the loading combinations for the supports of the auxiliary equipment supplied by Westinghouse are the same as those of the supported component. These loads and combinations are given in FSAR Section 3.9.3.

The stress limits are in accordance with the ASME Code Subsection NF or AISC, depending on the procurement date of equipment as discussed in the response of Part (a).

(d) For passive auxiliary components, only the structural integrity of the pressure boundary and supports is required to be assured. Since passive components perform no safety function other than retaining structural integrity, there are no deformation limits specified for the supports or for the passive auxiliary components.

Deformation of supports for active pumps is limited so that certain critical clearances are maintained and the pump remains operable. These critical clearances are specified in the pump specifications.

(e) Buckling is prevented by limiting compressive stresses for linear-type auxiliary equipment supports under loadings from all service conditions to the limits of AISC Section 1.5 or ASME Appendix XVII-2210. These limits are based on the Column Research Council (CRC) buckling curve for centrally-loaded columns. Critical buckling loads are limited to two-thirds of the CRC curve.

Plate and shell type supports for Class 2 and 3 auxiliary equipment are evaluated for buckling and instability through selective use of the criteria of Appendix XVII, Subarticle XVII-200 and Subsection NC, Subparagraph NC-3133.6 of Section III of ASME Code. Subparagraph NC-3133.6 gives methods for calculating the maximum allowable compressive stress in cylindrical shells subjected to axial loadings that provide longitudinal compression stresses in the shell. Subarticle XVII-200 gives requirements for structural steel members including allowable compressive loads based on slenderness ratios and interaction equations for combined stresses.

RESPONSE (Continued)

Uses of the above requirements in the design of linear or plate and shell type supports for Westinghouse-supplied auxiliary equipment ensures the dimensional stability of the support throughout the range of applied loadings.

2. PRIMARY EQUIPMENT SUPPORTS

The following is a listing of support vs. category for the RCS equipment supports.

Reactor Vessel Support Box	Plate and Shell
Steam Generator Columns	Linear
Steam Generator Lower Lateral Support	Linear
Steam Generator Upper Lateral Support	Linear
Reactor Coolant Pump Columns	Linear
Reactor Coolant Pump Tie Rods	Linear
Pressurizer Lateral Supports	Linear
Reactor Vessel Support Shoe/Pins	Linear

(a) Figures 210.57-1 and 2 show for a typical configuration the NF boundary between component support and building structure.

(b) All parts and components of the Class 1 primary equipment supports are designed and fabricated in accordance with Subsection NF of the ASME Code. The design and construction of the primary equipment support is based on a general design specification which is amended by a plant specific specification. The specifications address the design, procurement, installation, examination and inspection of the components which make up the primary equipment supports.

(c) Design loads, load combinations and stress limits are contained in FSAR Tables 3.9-2.1 and 3.9-2.1A.

Final qualification of the RCS equipment supports is based upon loads and stresses resulting from a plant specific reactor coolant loop analysis. The results are summarized in a final as-built (P.E. stamped) design report.

(d) No deformation limits are used in the design and analysis of the primary equipment supports. The structural members are designed to the stress limits of the ASME Code, Section III, Subsection NF so that all members remain elastic. The elastic behavior of the members is then considered in the reactor coolant system loop analysis.

RESPONSE (Continued)

(e) The buckling criteria used for the design of the primary equipment supports is based on the slenderness ratio. The allowable compressive stresses are limited to the requirements of the ASME Code, Section III, Articles XVII-2110b and XVII-2213. Critical buckling is based upon CRC curves where $kl/r < C_c$ (that is, for most RCS equipment support members) and the Euler curve where $kl/r > C_c$.

C_c is the slenderness ratio corresponding to the upper limit of elastic buckling failure.

The "A" term used in the response to NRC Question 110.19 is the cross sectional area of the member.

B. BOP Scope (excluding pipe supports):

(a) The jurisdictional boundary between the pressure-retaining component and the component support is established in accordance with subsection NF of the ASME III Code.

The jurisdictional boundary for ASME Section III, Division 1, Subsection NF component supports, is the baseplate or building structure to which the component support is attached.

The typical support configurations shown in Figures 210.57-3 and 210.57-4 are samples and are only intended to show NF jurisdictional boundaries.

(b) Component supports and any supporting structure between the component and the building structure, are designed, constructed and inspected in accordance with the applicable ASME requirements. Baseplates which are supplied by the equipment vendor or owner in order to facilitate attachment to the building structure are designed and procured in accordance with AISC requirements. Welds between an NF item and non-NF item are designed, performed and inspected in accordance with the appropriate sections of ASME Section III, V, and IX. The baseplate is attached to the building structure by either welding or bolting. The welds to the building structure are considered to be AISC and as such are performed and inspected identically with the requirements delineated in the letter from M. Wisenburg to H. Thompson dated February 25, 1985 (ST-HL-AE-1185).

Refer to the response to item 210.62 for the design of anchor bolts for component supports.

RESPONSE (Continued)

(c) The loads and load combinations for component supports are presented in FSAR Table 3.9-2.4.

The allowable stress limits are presented in FSAR Tables 3.9-7b and 3.9-7c.

(d) Deformation limits for component supports are specified by the suppliers for strain sensitive equipment. The limits insure that clearance and alignment requirements are met.

There are no deformation limits specified for tanks, vessels or exchanger component supports.

(e) For component supports, the designs are in accordance with the buckling criteria given in ASME III Subsection NF.

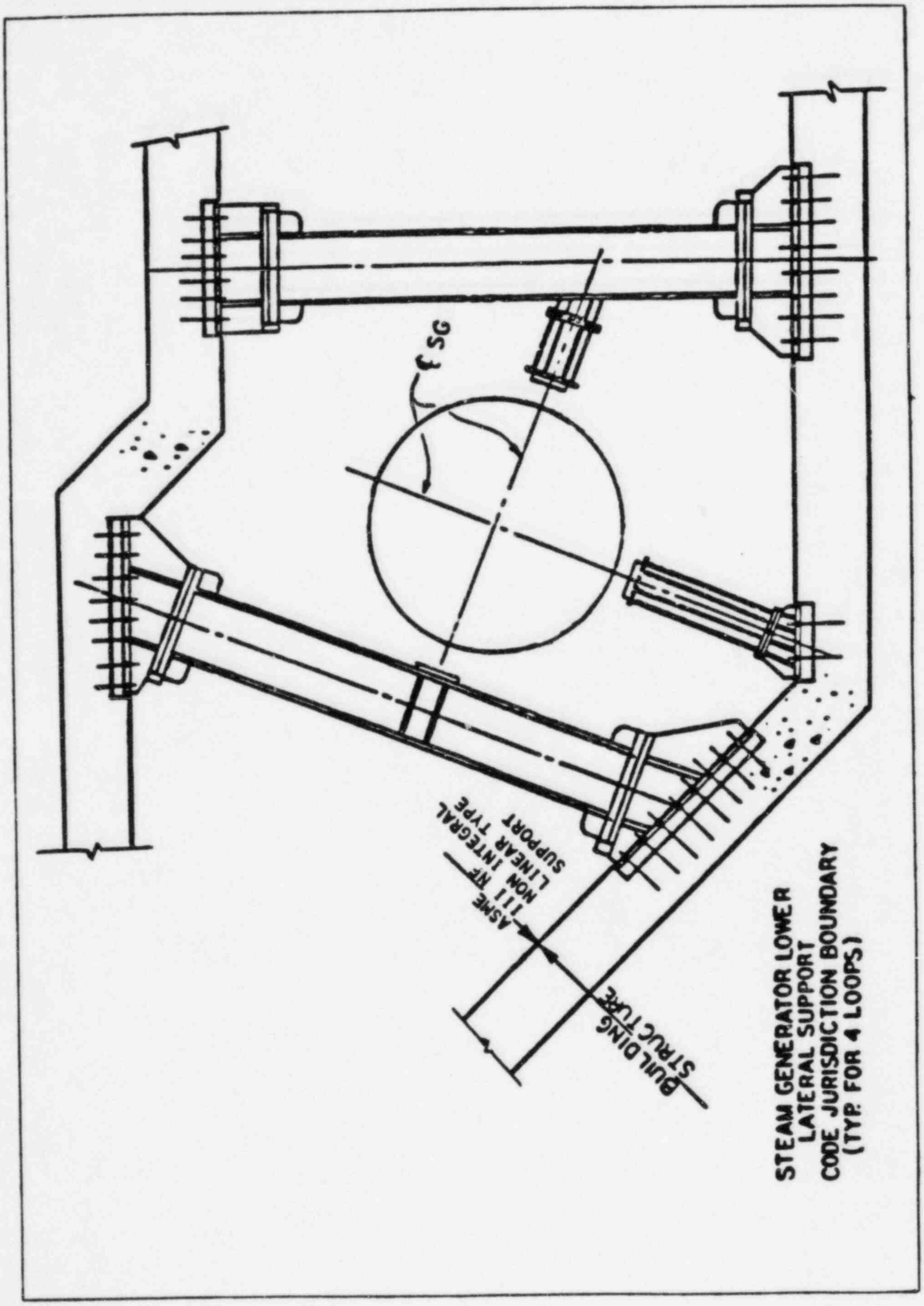


Figure 210.57-1

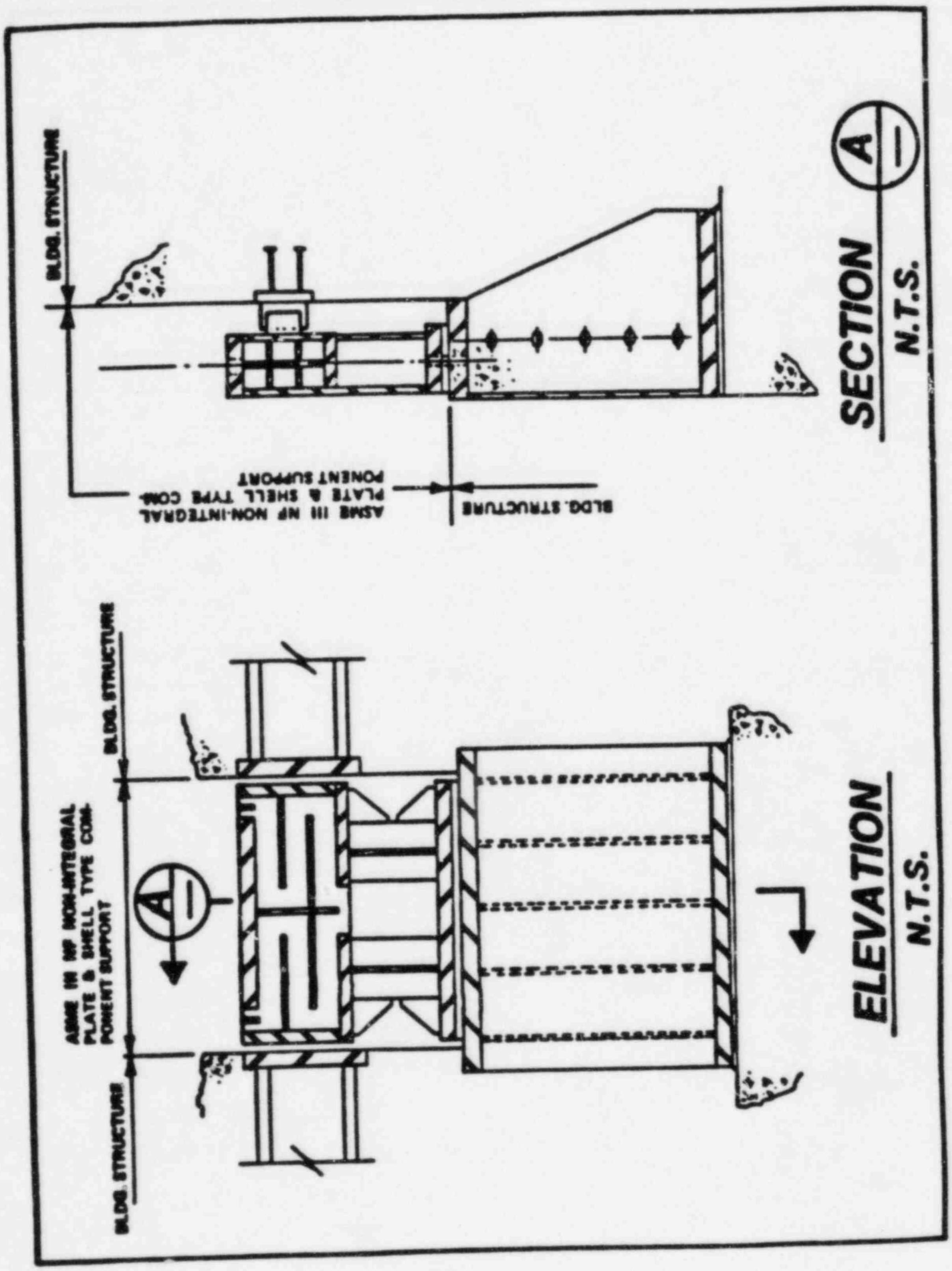


Figure 210.57-2

ASME CODE BOUNDARY DETAILS

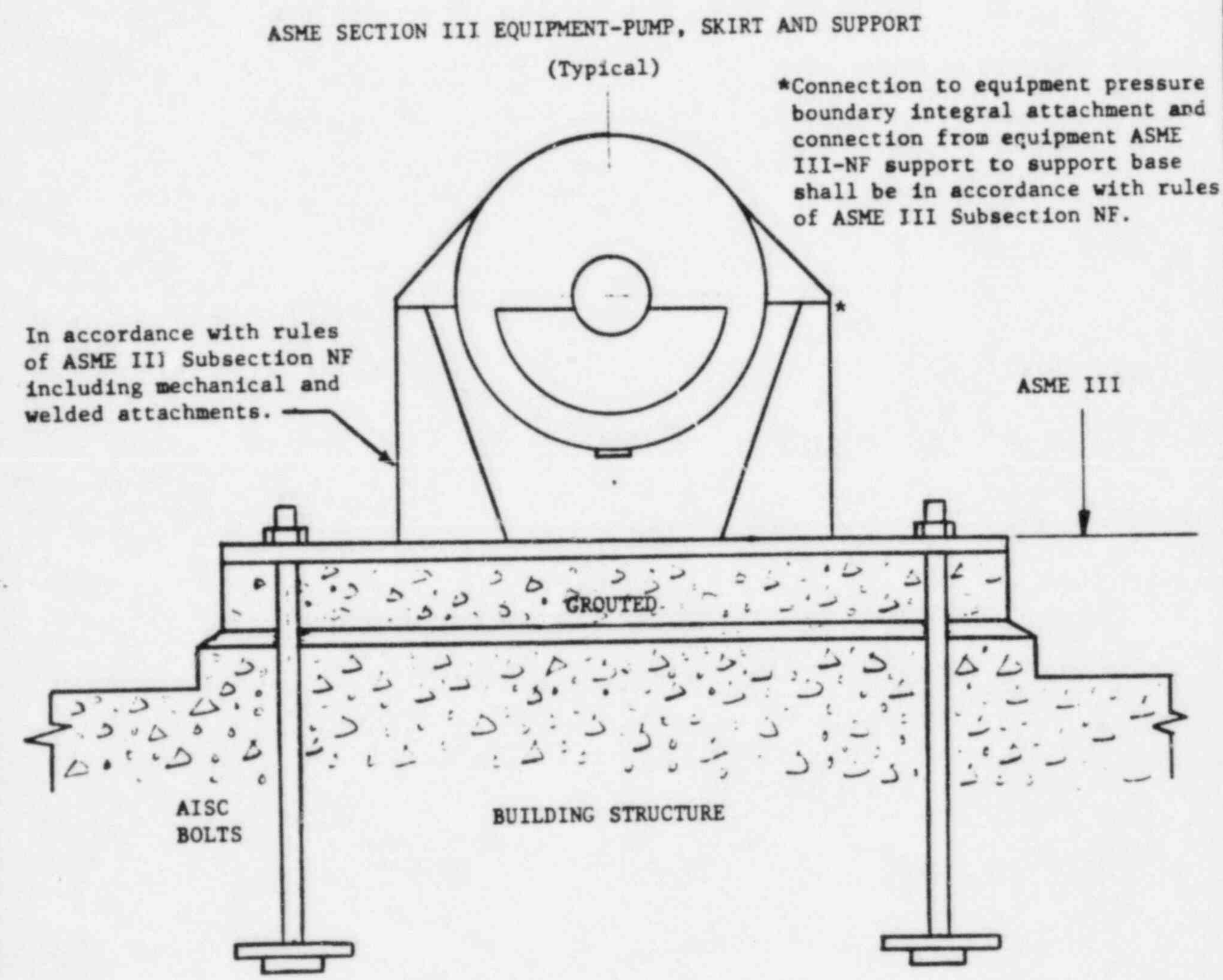
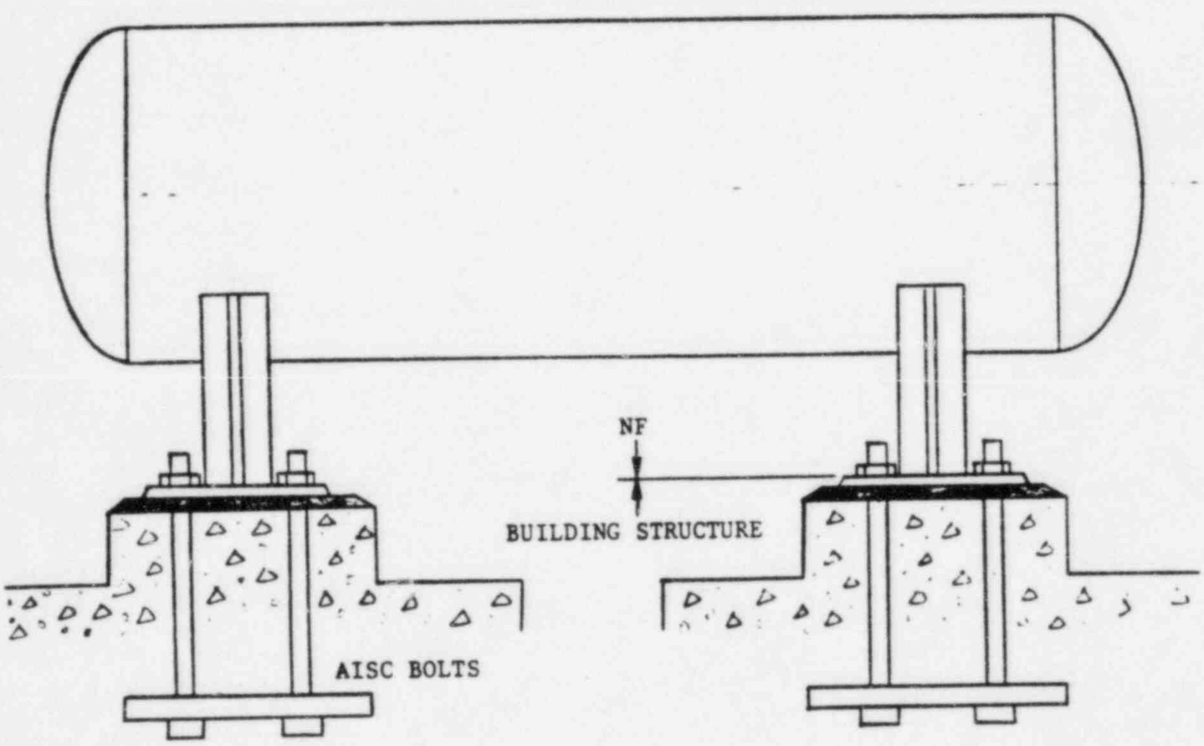


Figure 210.57-3

ASME CODE BOUNDARY DETAILS

ASME SECTION III EQUIPMENT-TANK AND SUPPORT ASSEMBLY



Integral support assembly shall be per ASME Section III, Subsection NF. Weld of integral support assembly to baseplate shall be per ASME Section III, Subsection NF. Baseplate shall be designed per AISC requirements as a minimum.

Figure 210.57-4

QUESTION 210.58

Valve discs are considered part of the pressure boundary and as such should have allowable stress limits. Provide these limits for our review.

RESPONSE

1. NSSS Scope

   The valve discs for Class 1 valves greater than 4-inch size are analyzed using the following allowables for acceptance criteria.

   Primary Membrane, $P_m \leq 1.0\ S_m$

   Primary Membrane + Bending, $P_m + P_b \leq 1.5\ S_m$

   For class 2 and 3 valves where analysis is required per the specification, the following acceptance criteria are used.

   Primary Membrane, $P_m \leq 1.0\ S$

   Primary Membrane + Bending, $P_m + P_b \leq 1.5\ S$

2. BOP Scope

   A. ASME III, Class 1 Valves

   No large bore ($>$ 4" diameter) valves are in the BOP scope. For 4 inch and smaller valves structural integrity is demonstrated by a differential pressure test across the disc and not by analysis. NB-3530 contains requirements for hydrostatic tests for the shell and disc. The hydrostatic test report for the valve includes details to show that the requirements of NB-3530 are met.

   B. ASME III, Class 2 & 3 Valves

   For Code Class 2 and 3 valve discs no design requirements are described in NC or ND-3500. However, tests for structural and pressure integrity for valve discs or plugs are described in NC/ND-3514(b). A disc hydrostatic test is required with the disc or plug in the fully closed position with a test pressure across the disc or plug equal to the pressure rating of the valve at 100°F. The ASME III design specifications may be used to stipulate a higher or lower test pressure and do require specific limiting seat leakage.

   Design requirements for valve discs and plugs are determined by the manufacturer, since items such as disc geometry, method of seating, materials, etc., are controlled by the manufacturer and subject only to review and acceptance by the owner, or his designee, as permitted in the ASME code.

QUESTION 210.59

Due to a long history of problems dealing with inoperable and incorrectly installed snubbers, and due to the potential safety significance of failed snubbers on safety-related systems and components, it is required that maintenance records for snubbers be documented as follows:

Pre-Service Examination

A pre-service examination should be made on all snubbers listed in Tables 3.7-4a and 3.7-4b of Standard Technical Specifications 3/4.7.9. This examination should be made after snubber installation but not more than six months prior to initial system pre-operational testing, and should as a minimum verify the following:

1. There are no visible signs of damage or impaired operability as a result of storage, handling, or installation.

2. The snubber location, orientation, position setting, and configuration (attachments, extensions, etc.) are according to design drawings and specifications.

3. Snubbers are not seized, frozen or jammed.

4. Adequate swing clearance is provided to allow snubber movement.

5. If applicable, fluid is to the recommended level and is not leaking from the snubber system.

6. Structural connections such as pins, fasteners and other connecting hardware such as look nuts, tabs, wire, and cotter pins are installed correctly.

If the period between the initial pre-service examination and initial system pre-operational test exceeds six months due to unexpected situations, re-examination of Items 1, 4, and 5 shall be performed. Snubbers which are installed incorrectly or otherwise fail to meet the above requirements must be repaired or replaced and re-examined in accordance with the above criteria.

Pre-operational Testing

During pre-operational testing, snubber thermal movements for systems whose operating temperature exceeds 250°F should be verified as follows:

a. During initial system heat-up and cooldown, at specified temperature intervals for any system which attains operating temperature, verify the snubber expected thermal movement.

QUESTION 210.59 (Continued)

b. For those systems which do not attain operating temperature, verify via observation and/or calculation that the snubber will accommodate the projected thermal movement.

c. Verify the snubber swing clearance at specified heat-up and cooldown intervals. Any discrepancies or inconsistencies shall be evaluated for cause and corrected prior to proceeding to the next specified interval.

The above described operability program for snubbers should be included and documented by the pre-service inspection and pre-operational test programs.

The pre-service inspection must be a prerequisite for the pre-operational testing of snubber thermal motion. This test program should be specified in Chapter 14 of the FSAR.

RESPONSE

The requirements stated in this question are identical to those contained in an NRC letter to HL&P dated October 17, 1980 (ST-AE-HL-608). HL&P has committed to meet these requirements in the reply to NRC Question 250.01N (ST-HL-AE-1219). The required examinations and tests will be performed in conjunction with either the PSI of component supports or during the pre-operational test program. The snubber pre-service inspection will be completed prior to pre-operational testing of snubber thermal motion. The test program for pre-operational testing of snubber thermal motion will consist of a visual verification of proper snubber movement, as indicated on the snubber, from room temperature to maximum operating temperature. This will be done for those systems whose normal operating temperature exceeds 250°F. If the maximum operating temperature is not attained during testing, the expected amount of movement indicated on the snubber by the ratio of the temperature rise to the maximum operating temperature to temperature rise to the test temperature (Tm/Tt) will be determined. Extrapolated results will be evaluated to assure snubbers remain within their stroke capabilities.

Chapter 14 of the STP FSAR will be revised accordingly.

QUESTION 210.60

Does the design criteria for component supports in systems categorize the stresses produced by seismic anchor point motion of piping and the thermal expansion of piping as primary or secondary? It is the staff's position that for the design of component supports, and stresses produced by seismic anchor point motion of piping and the thermal expansion of piping should be categorized as primary stresses. The application of this position is most critical for those supports which would be subjected to large deformations.

RESPONSE

The stresses produced by seismic anchor point motion of piping and thermal expansion of piping are considered as primary stresses in the design of component supports. NSSS component supports which have been designed in accordance with ASME III Subsection NF will be evaluated for compliance with this position.

QUESTION 210.61

Describe what actions have been taken to address the staff concerns regarding stiff pipe clamps as described in IE Information Notice 83-80.

RESPONSE

The applications of stiff pipe clamps on STP will be reviewed based on IE Information Notice 83-80. Section III of the ASME B&PV Code does not provide rules for evaluating stresses due to loadings from nonintegral attachments such as clamps; however, clamp-induced stresses will be evaluated by methods consistent with the intent of the Section III of the ASME B&PV Code. The procedure will include the following:

1. Identify the locations of "stiff" clamps installed on ASME Section III Nuclear Class 1 piping systems.

2. Identify the types of clamps, the loads acting on the clamps and the bolt pre-load values used in their installation. In piping, stresses due to all loading conditions at the locations of stiff clamps will also be identified and reviewed.

3. Add the primary membrane and primary bending stresses caused by the load being transmitted to the pipe through the clamp to the stresses caused by internal pressure and bending computed by equation 9 of NB-3652. Clamp-induced stresses caused by the constraint of the expansion of the pipe due to the internal pressure will be added to other secondary and peak stresses by calculating the effective increases in the C_1 and K_1 stress indices in accordance with NB-3681. Clamp induced stresses due to differential-temperature and differential-thermal-expansion coefficients will be accounted for by computing the effective C_3 and K_3 stress indices. Clamp-induced stresses on elbows caused by the constraint of pipe wall ovalization will be accounted for by computing the effective increases in C_2 and K_2 bending indices. The fatigue usage from clamp-induced plus other stresses will be calculated at governing locations.

Although bolt preloads are not addressed under the ASME B&PV Code rules for piping, bolt preloads could result in damage to a pipe if a clamp were poorly designed. Calculations will be made to ensure that bolt preloads could not result in plastic deformation of the pipe walls.

A brief summary of the criteria used and the results of the analysis will be submitted in October, 1985.

Stiff clamps were not used on STP to meet stiffness criteria. They were designed to meet the requirements for strength and load distribution using a minimum of space. The STP position is to minimize the use of stiff clamps.

The clamp design utilizes a double nut arrangement to prevent the nuts from backing off. The low temperature ($<$ 600°F) and stresses in the bolt from preloads will not cause a relaxation of the material. Consequently, no lift off from the piping will occur.

QUESTION 210.62

The staff's review of your component support design finds the additional information is required regarding the design basis used for bolts.

(a) Describe the allowable stress limits used for bolts in equipment anchorage, component supports, and flanged connections.

(b) Provide a discussion of the design methods used for expansion anchor bolts used in component supports.

RESPONSE

NSSS Scope:

For primary equipment supports the bolt design, including anchor bolts, is in accordance with subsection NF (NF-3280). Allowable stresses are per Appendix XVII-2460 and/or those of Code Case 1644. The stress allowable may be increased according to the provisions of XVII-2110(a) and F-1370(a) for emergency and faulted conditions, respectively.

For tanks and heat exchangers supplied by Westinghouse, the only bolting for supports provided by Westinghouse is on the regenerative heat exchanger. These bolts meet the requirements of Subsection NF and Code Case 1644.

Bolting on supports for Westinghouse supplied Class 2 and 3 pumps meet the requirements of ASME B&PV Code Subsection NF and Code Case 1644 or ASME pressure boundary criteria.

For flanged connections on tanks and heat exchangers the allowable stress limits are per the applicable section of the ASME Code (i.e. Section III or Section VIII).

For all bolts in the Westinghouse scope of design, an allowable stress equal to or less than the yield strength of the material at temperature is used for all loading conditions for component supports.

BOP Scope:

a. The bolts used in equipment anchorage and component supports including NSSS components, are classified as part of the building structures (i.e. non-ASME) and their embedment lengths are calculated using ACI-318. Allowable stresses for anchor bolts are in accordance with the AISC specification, except for safety-related NSSS component anchor bolts which are in accordance with the ASME Code.

   For bolts used in flanged connections on tanks and heat exchangers the allowable stress limits are per the applicable section of the ASME Code (i.e. Section III or Section VIII).

   For all bolts in the BOP scope, an allowable stress equal to or less than the yield strength of the material at temperature is used for all loading conditions for component supports.

RESPONSE (Continued)

b. In the South Texas Project two types of expansion anchor bolts are used for permanent plant installations: the wedge-type (Hilti Kwik-bolt) and the ductile-type (Maxibolts) manufactured, respectively, by Hilti Fastening Systems, Inc. and by Drillco Devices Ltd.

For Hilti Kwik-bolts the allowable design loads in shear and tension are based on tested ultimate load capacities with an applied factor of safety of 4.0 or higher. These allowable loads are for all loading combinations, and specifically, are not increased for faulted or abnormal/extreme environmental loading combinations.

For Maxibolts the allowable design loads prescribed for tension are based on 0.33 times the specified ultimate tensile strength of the bolt steel material (F_u = 125 KSI), and for shear are based on 0.17 times F_u. Comprehensive tests performed on Maxibolts demonstrate that this type of expansion anchors, which is positively anchored into an undercut hole, develops the full ductility and tensile strength of the bolt material without any concrete failure. Therefore, the ultimate load capacity is governed by the steel bolt, and accordingly, the above provisions for allowable loads in accordance with the AISC Specification are appropriate and applicable. In the case of "Abnormal/Extreme Environmental" and "Faulted" loading conditions, the allowable loads are increased by a factor of 1.5.

For evaluation of simultaneous tension and shear loads, the design loads are combined by the following interaction formulas:

$$\left(\frac{t}{T}\right) + \left(\frac{s}{S}\right) \leq 1.0$$ (For Hilti Kwik-bolts)

$$\left(\frac{t}{T}\right)^{5/3} + \left(\frac{s}{S}\right)^{5/3} \leq 1.0$$ (For Maxibolts, whose load capacity is governed by steel material. Accordingly, this interaction formula with 5/3 exponents, which is enveloped by the AISC formula with 2.0 exponents, is used).

Where:

(t, s) = design tension and shear loads, respectively

(T, S) = specified allowable tension and shear loads, respectively

RESPONSE (Continued)

The design tension in expansion anchor bolts is calculated in the component support design process utilizing either a manual calculation or a computer analysis. The baseplate flexibility and prying action effects on the bolt tension are taken into account as described in the STP response to NRC Bulletin 79-02 (Ref. letter ST-HL-AE-1073 dated 07/30/84).

QUESTION 210.63

Clarify the statement on page 3.9-80 of the FSAR that, "The allowble stress limits during the Design Basis Accident used for the core support structures are based on the 1974 edition of the ASME Code for core Support Structures, Subsection NC, and the criteria for faulted condition."

RESPONSE

The statement should read, " . . . Subsection NG, and the critiera for faulted condition."

This change will be made in a future FSAR amendment.

QUESTION 210.64

There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to protect these systems for RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure systems. The leak tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems.

Pressure isolation valves are required to be category A or AC per IWV-2000 and to meet the appropriate requirements of IWV-3420 of Section XI of the ASME Code except as discussed below.

Limiting Conditions for Operation (LCO) are required to be added to the technical specifications which will require corrective action; i.e., shutdown or system isolation when the final approved leakage limits are not met. Also, surveillance requirements which will state the acceptable leak rate testing frequency shall be provided in the technical specifications.

Periodic leak testing of each pressure isolation valve is required to be performed at least once per each refueling outage, after valve maintenance prior to return to service, and for systems rated at less than 50% of RCS design pressure each time the valve has moved from its fully closed position unless justification is given. The testing interval should average to be approximately one year. Leak testing should also be performed after all disturbances to the valves are complete, prior to reaching power operation following a refueling outage, maintenance, etc.

The staff's present position on leak rate limiting conditions for operation must be equal to or less than 1 gallon per minute (GPM) for each valve to ensure the integrity of the valve, demonstrate the adequacy of the redundant pressure isolation function and give an indication of valve degradation over a finite period of time. Significant increases over this limiting value would be an indication of valve degradation from one test to another.

The Class 1 to Class 2 boundary will be considered the isolation point which must be protected by redundant isolation valves.

In cases where pressure isolation is provided by two valves, both will be independently leak tested. When three or more valves provide isolation, only two of the valves need to be leak tested.

Provide a list of all pressure isolation valves included in your testing program along with four sets of Piping and Instrument Diagrams which describe your reactor coolant system pressure isolation valves. Also discuss in detail how your leak testing program will conform to the above staff position.

RESPONSE

Table 3.4-1 of the Proposed Technical Specifications will provide a list of all pressure isolation valves (a copy of the proposed table is attached).

The requested Piping and Instrumentation Diagrams have been included in the FSAR as Figures 5.4-6 and 6.3-1 through 6.3-4.

All the pressure isolation valves listed in Table 3.4-1 will be classified category A or AC per IWV-2000, and will meet the appropriate requirements of IWV-3420.

The Limiting Condition for Operation from proposed Technical Specification 3.4.6.2 will comply with the staff position concerning corrective action and leak rate testing frequency.

The Surveillance requirements associated with Technical Specification 3.4.6.2 will comply with the staff position concerning periodic leak testing at least once per each refueling outage, after valve maintenance, and each time a valve as been moved from its fully closed position.

The allowable leakage for any pressure isolation valve specified in Table 3.4-1 will be 1 gpm.

Pressure isolation is provided by two valves, which will be independently leak tested.

PROPOSED TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	FUNCTION
XSI0007 A, B, C	HHSI Cold Leg Injection Check Valves (RCS Loops 1, 2, 3)
XSI0009 A, B, C	HHSI Cold Leg Recirculation Check Valves (RCS Loops 1, 2, 3)
XSI0010 A, B, C	LHSI/HHSI Hot Leg Recirculation Check Valves (RCS Loops 1, 2, 3)
XRH0020 A, B, C	LHSI Hot Leg Recirculation Check Valves (RCS Loops 1, 2, 3)
XRH0032 A, B, C	LHSI Cold Leg Injection Check Valves (RCS Loops 1, 2, 3)
XSI0038 A, B, C	LHSI/HHSI/Accumulator Cold Leg Injection Check Valves (RCS Loops 1, 2, 3)
XSI0046 A, B, C	Accumulator Cold Leg Injection Check Valves (RCS Loops 1, 2, 3)

QUESTION 210.65

Provide a schedule for completion of your program for inservice testing of pumps and valves including any request relief from ASME Section XI requirements.

RESPONSE

See the response to Question 210.1N.

Attachment 2

AGENDA

- OBJECTIVE
- APPROACH
- W TEST INFORMATION ON FLOW-INDUCED VIBRATIONS OF REACTOR INTERNALS.
- W LICENSEE (FRAMATOME) TEST INFORMATION ON INTERNALS.
- DESIGN DIFFERENCES AND THEIR RELATIONSHIPS TO FLOW-INDUCED VIBRATIONS.
- TEST AND ANALYTICAL RESULTS
- PRE AND POST HOT FUNCTIONAL EXAMINATION
- SUMMARY AND CONCLUSIONS

OBJECTIVE

TO DEMONSTRATE STRUCTURAL ADEQUACY OF 4XL (SOUTH TEXAS NUCLEAR PLANT) REACTOR INTERNALS WITH RESPECT TO FLOW-INDUCED VIBRATIONS.

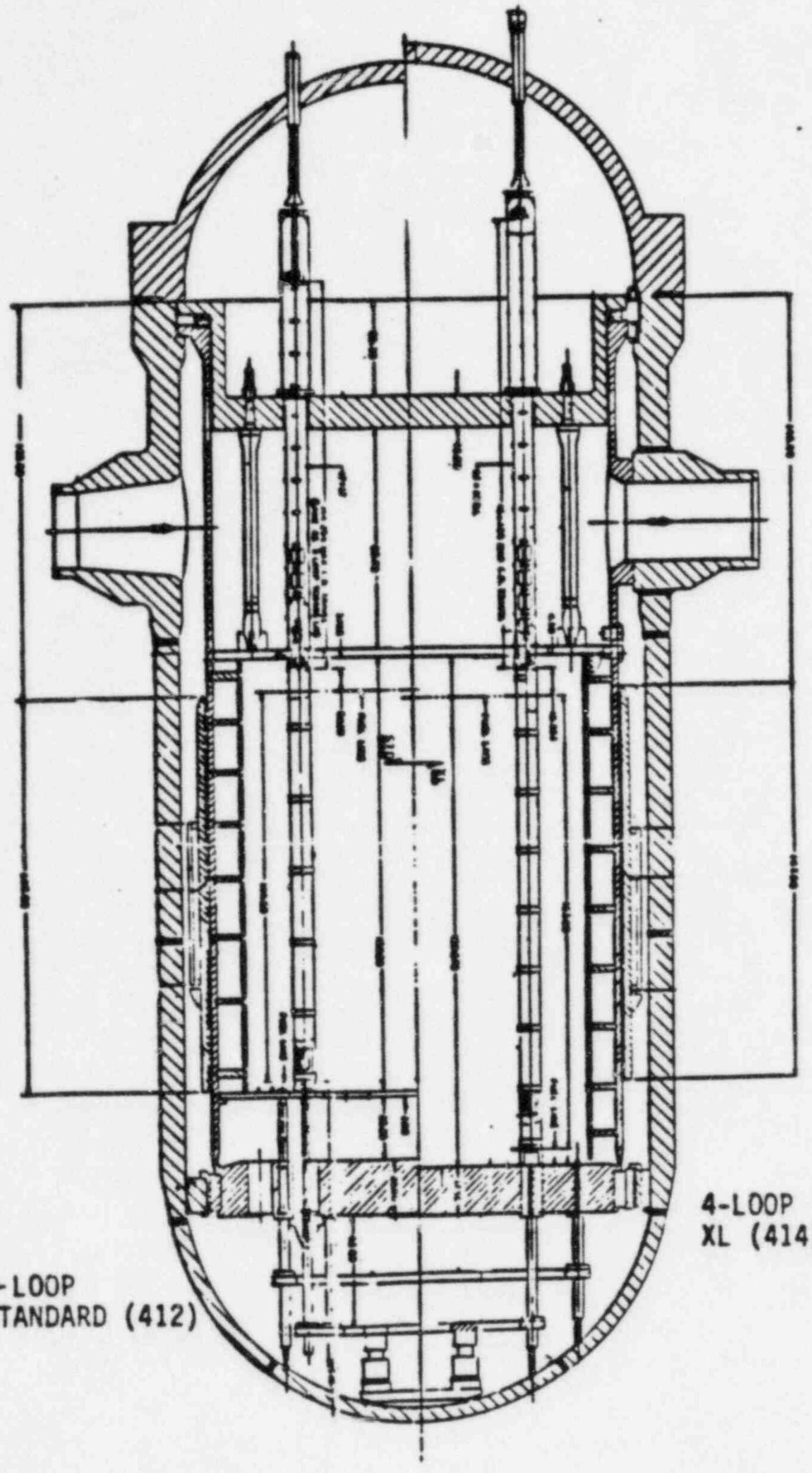


FIGURE 3-1 DESIGN DIFFERENCES BETWEEN 4-XL AND 4-LOOP STANDARD (412) LOWER INTERNALS

APPROACH

DESIGN	PLANT TEST
• PROTOTYPE 412 PLANT	INDIAN POINT 2
• NEUTRON PAD CORE BARREL WITH FLAT LOWER SUPPORT PLATE	TROJAN
• INVERTED TOP HAT UPPER INTERNALS DESIGN	SEQUOYAH 1

- NEUTRON PAD/INVERTED TOP HAT DESIGN USED IN NUMEROUS OTHER OPERATING AND NTOL PLANTS INCLUDING MCGUIRE, CATAWBA, CALLAWAY, COMMANCHE PEAK, WOLF CREEK PLUS FOREIGN PLANTS.

- DESIGN MODIFICATIONS MADE TO ACCOMMODATE INCREASED FUEL LENGTH IN XL PLANTS HAVE BEEN SHOWN TO HAVE AN INSIGNIFICANT EFFECT ON INTERNALS VIBRATIONS.

  - ANALYSIS
  - SCALE MODEL TESTS
  - CONFIRMATORY PLANT TESTS

TABLE 2-1

CHANGES TO ORIGINAL PROTOTYPE

IPP-II	INSTRUMENTED	IPP-II LICENSED
LOWER INTERNALS (NEUTRON PADS)	SCALE TESTS, ANALYSIS PLANT TESTS, OPERATING EXPERIENCE	TROJAN-1 LICENSED MCGUIRE 1 & 2 LICENSED CATAWBA 1 & 2 LICENSED OHI* LICENSED FARLEY LICENSED DOEL*3 & 4 LICENSED TWP* LICENSED
UPPER INTERNALS (INVERTED TOP HAT)	SCALE TESTS, ANALYSIS PLANT TESTS, OPERATING EXPERIENCE	SEQUOYAH LICENSED MCGUIRE 1 & 2 LICENSED CATAWBA 1 & 2 LICENSED DOEL*3 & 4 LICENSED PALUEL* LICENSED OHI* LICENSED
14 FOOT CORE	SCALE TESTS, ANALYSIS, PLANT TESTS	PALUEL* LICENSED DOEL* 4 LICENSED

*FOREIGN PLANTS

APPROACH

- ANALYTICAL STUDIES
- SCALE MODEL TESTS
- INTERNALS COMPONENT TESTS
- TESTS ON INSTRUMENTED REACTORS
- OPERATING EXPERIENCE
- PRE-AND-POST HOT FUNCTIONAL EXAMINATION ON EACH PLANT

SUMMARY OF WESTINGHOUSE

TEST INFORMATION ON FLOW INDUCED VIBRATION OF REACTOR INTERNALS

TEST	KEY RESULTS/INFORMATION
• 4-Loop Thermal Shield 1/24 Scale Model Test	• Lower internals flow induced vibration.
• 3-Loop Thermal Shield 1/22 Scale Model Test	• Lower internals flow induced vibration. • 3 and 4-loop internals vibration similar.
• Indian Point Plant Test	• Verified adequacy of internals. • Verified accuracy of scale models and analyses. • Verified use of hot functional test (no core) for vibration assessment. • Provided data for improved predictions.
• H. B. Robinson (and other prototype) Plant Tests.	• Showed similar behavior as 4-loop. • Provided data for improved predictions. • Verified adequacy of internals and analysis.
• 4-Loop Neutron Pad 1/24 Scale Model Test	• Showed neutron pad internals have behavior similar to thermal shield internals and vibration levels significantly lower.
• 3-Loop Neutron Pad 1/22 Scale Model Test	• (Same as Above) • Showed 3 and 4-Loop internals vibration similar.

SUMMARY OF WESTINGHOUSE

TEST INFORMATION ON FLOW INDUCED VIBRATION OF REACTOR INTERNALS

TEST	KEY RESULTS/INFORMATION
• TROJAN-1 PLANT TEST	• ADDITIONAL VERIFICATION OF SCALE MODEL AND ANALYSIS PREDICTIONS. • VERIFIED ADEQUACY OF 17 X 17 AND NEUTRON PAD DESIGNS. • VERIFIED REDUCED VIBRATION LEVELS OF NEUTRON PAD AND 17 X 17 DESIGN CHANGES.
• 1/7 SCALE UHI UPPER INTERNALS TEST	• FLOW INDUCED VIBRATION OF SOLID SUPPORT COLUMNS AND 17 X 17 GUIDE TUBES IN A UHI ARRAY. • SHOWED, WITH ANALYSIS, ADEQUACY OF GUIDE TUBES, AND UPPER SUPPORT COLUMNS.
• SEQUOYAH PLANT TEST	• VERIFIED ADEQUACY OF SUPPORT COLUMNS AND GUIDE TUBES IN INVERTED HAT DESIGN. ADDITIONAL VERIFICATION OF SCALE MODEL TESTING TECHNIQUE.
• 4XL 1/7 SCALE MODEL TEST	• SHOW SIMILARITY OF UHI AND 4XL UPPER INTERNAL RESPONSES.
• 3XL 1/7 SCALE MODEL TEST	• VIBRATION RESPONSE OF 3XL INTERNALS. • SHOW SIMILARITY OF 3 LOOP, 4 LOOP, AND UHI UPPER INTERNALS VIBRATORY BEHAVIOR.
• DOEL 4 (3XL) PLANT TEST	• VERIFIED ADEQUACY OF XL LOWER INTERNALS. ADDITIONAL VERIFICATION OF SCALE MODEL TESTING TECHNIQUE.

WESTINGHOUSE LICENSEE (FRAMATOME) TEST INFORMATION

TEST	KEY RESULTS/INFORMATION
• TRICASTIN PLANT	• LOWER INTERNALS FLOW INDUCED VIBRATIONS FOR 3 LOOP NEUTRON PAD CORE BARREL DESIGN.
• DOEL 3 PLANT	• FLOW INDUCED VIBRATIONS OF 17 x 17A GUIDE TUBES AND 3.5 INCH SUPPORT COLUMNS IN INVERTED TOP-HAT DESIGN.
• PALUEL PLANT	• FLOW INDUCED VIBRATIONS FOR 4 LOOP - 14 FOOT CORE PLANTS.

MAIN RESULTS FROM TESTING

- LOWER INTERNALS
  - MAIN SOURCE OF EXCITATION IS FLOW TURBULENCE AT INLET NOZZLE AND IN DOWNCOMER ANNULUS.
  - NEUTRON PAD CORE BARRELS HAVE GENERALLY LOWER AMPLITUDES THAN THOSE OF THERMAL SHIELD BARRELS.
  - VIBRATIONAL AMPLITUDES WITH CORE IN ARE LESS THAN THOSE OF WITHOUT CORE.

- UPPER INTERNALS
  - MAIN SOURCE OF EXCITATION IS FLOW TURBULENCE DUE TO AXIAL AND CROSS FLOW THAT CONVERGE ON THE OUTLET NOZZLES.
  - BASED ON SEQUOYAH, SCALE MODEL GENERALLY PREDICT HIGHER THAN IN-PLANT TESTS.

- GENERAL
  - SCALE MODEL TEST DATA ARE RELIABLE FOR PREDICTIONS OF IN-PLANT TEST.
  - PROVIDE ASSURANCE FOR STRUCTURAL INTEGRITY.

DESIGN CHANGES FROM ORIGINAL 4 LOOP CONFIGURATION AND PREVIOUS JUSTIFICATION

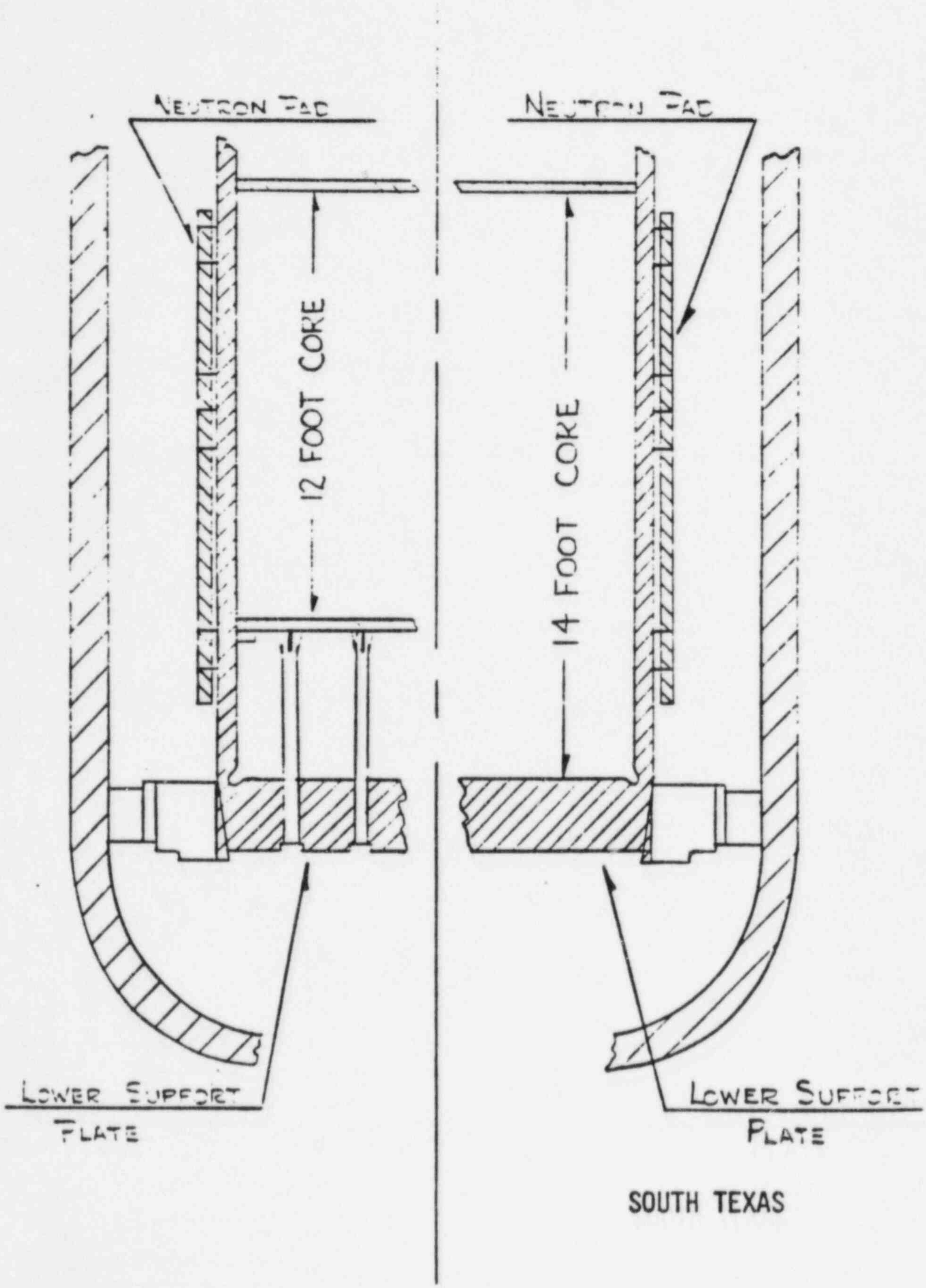
- REPLACEMENT OF THERMAL SHIELD WITH NEUTRON PADS
  - PLANT AND SCALE MODEL VIBRATION MEASUREMENT PROGRAMS COMPLETED.
- CHANGE TO (UHI STYLE) INVERTED TOP HAT UPPER INTERNALS
  - PLANT AND SCALE MODEL VIBRATION MEASUREMENT PROGRAMS COMPLETED.
- CHANGE TO 14 FOOT CORE
  - IN-PLANT TESTING OF 3 LOOP AND 4 LOOP 14 FOOT CORE PLANTS DEMONSTRATED ADEQUACY OF INTERNALS.
  - PLANT AND SCALE MODEL VIBRATION MEASUREMENT PROGRAMS COMPLETED.

DESIGN DIFFERENCE BETWEEN STP AND 412 PLANTS AND BASIS FOR JUSTIFICATION

- THE ONLY SIGNIFICANT DESIGN DIFFERENCE IS THE CHANGE OF LOWER SUPPORT STRUCTURE TO ACCOMMODATE ADDITIONAL FUEL LENGTH. THIS CHANGE IS JUSTIFIED BY

  - SCALE MODEL VIBRATION MEASUREMENT PROGRAM
  - ANALYSIS
  - PLANT MEASUREMENT PROGRAMS ON OTHER XL PLANTS

- THE CHANGE HAS INSIGNIFICANT EFFECT ON INTERNALS VIBRATION.


NEUTRON PAD
NEUTRON PAD
12 FOOT CORE
14 FOOT CORE
LOWER SUPPORT PLATE
LOWER SUPPORT PLATE

SOUTH TEXAS

EFFECTS OF REPLACEMENT OF 12 FT. CORE WITH 14 FT. CORE

- ELIMINATION OF LOWER CORE PLATE TO ACCOMMODATE ADDITIONAL FUEL LENGTH AND SLIGHTLY REDUCED THICKNESS OF LOWER SUPPORT PLATE.

• CORE BARREL DIAMETER AND THICKNESS ARE UNCHANGED; LENGTH DIFFERENCE IS INSIGNIFICANT (LESS THAN ½%).

• REACTOR VESSEL INSIDE DIAMETER, NEUTRON PAD SIZE, AND THE CORE CAVITY CROSS-SECTION ALSO REMAIN UNCHANGED.

- IN VIEW OF THE ABOVE SIMILARITIES, THE DOWNCOMER ANNULUS REMAINS THE SAME AND, THEREFORE THE CORE BARREL FORCING FUNCTION AND CORE BARREL EXCITATIONS ARE THE SAME.

TABLE 3-1

4XL (TGX) vs. 4LSTD. (412)

WECAN ANALYSIS

(With Core)

	4XL (TGX)	4LSTD. (412)
Fuel Assembly Natural Frequency 1st Mode	2.68 (Hz)	2.38 (Hz)
Core Barrel		
Cantilever Beam Mode		
- Natural Frequency	7.87/9.44 (Hz)	7.86/9.03 (Hz)
- Normalized[1] Amplitude	1.0	0.94

[1] Amplitude Normalized by (TGX) Amplitude

TABLE 3-2

4XL (TGX) vs. 4L STD. (412)

WECAN ANALYSIS

(Without Core)

Core Barrel Cantilever

Beam Model

	4XL (TGX)	4LSTD. (412)
f_n	8.16 (Hz)	8.19 (Hz)
Normalized(1) Amplitude	1.0	0.97

(1)Amplitudes Normalized by (TGX) Amplitudes

TABLE 3-3

RATIO OF 4XL vs. 4LSTD. (412) BARREL

RESPONSE FOR EQUAL EXCITATION

(WECAN ANALYSIS)

	WITH CORE	WITHOUT CORE
$\dfrac{(4XL)_{Response}}{(412)_{Response}}$ =	1.059	1.026

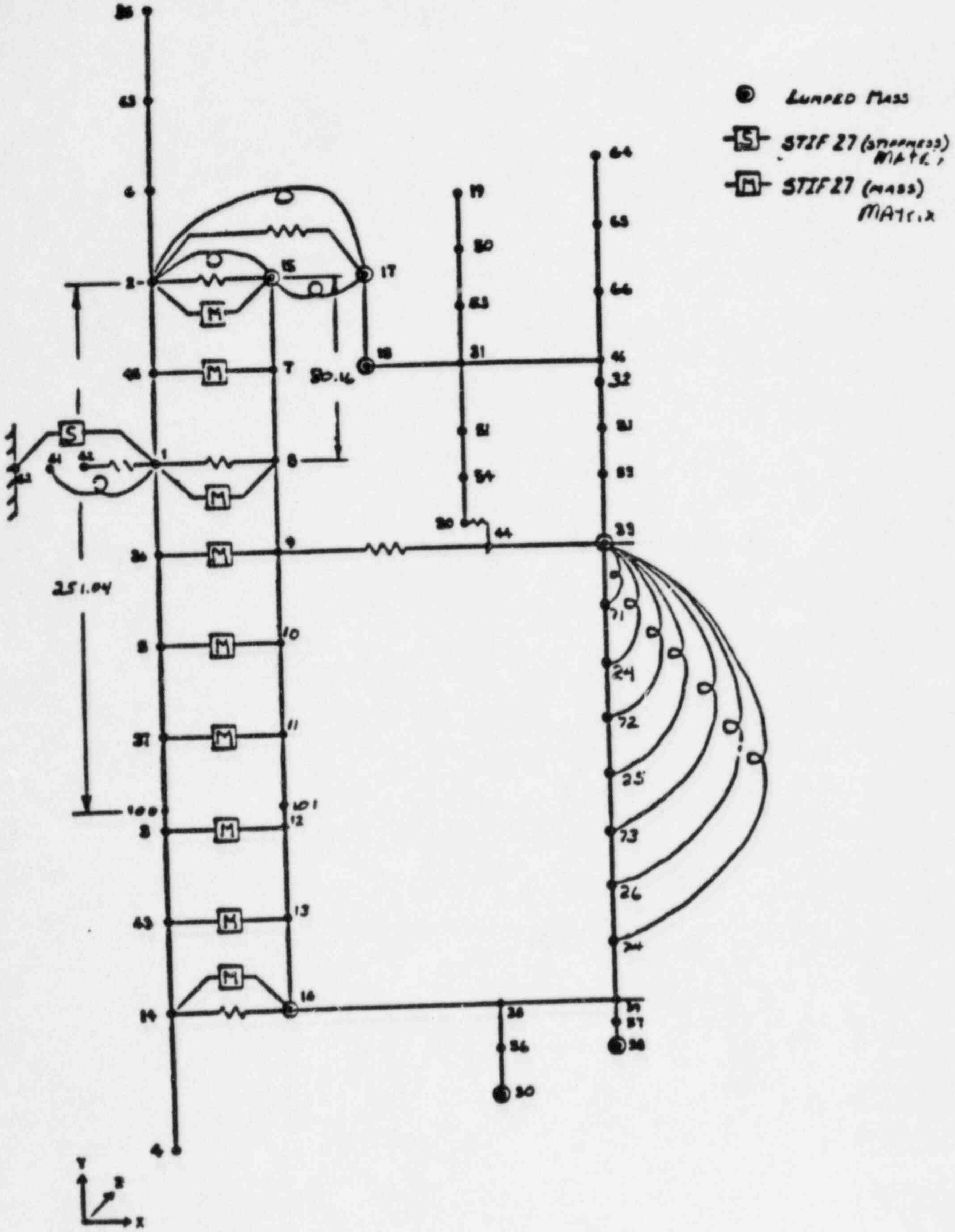


FIGURE 3-2 FINITE ELEMENT MODEL OF 4-LOOP XL PLANT

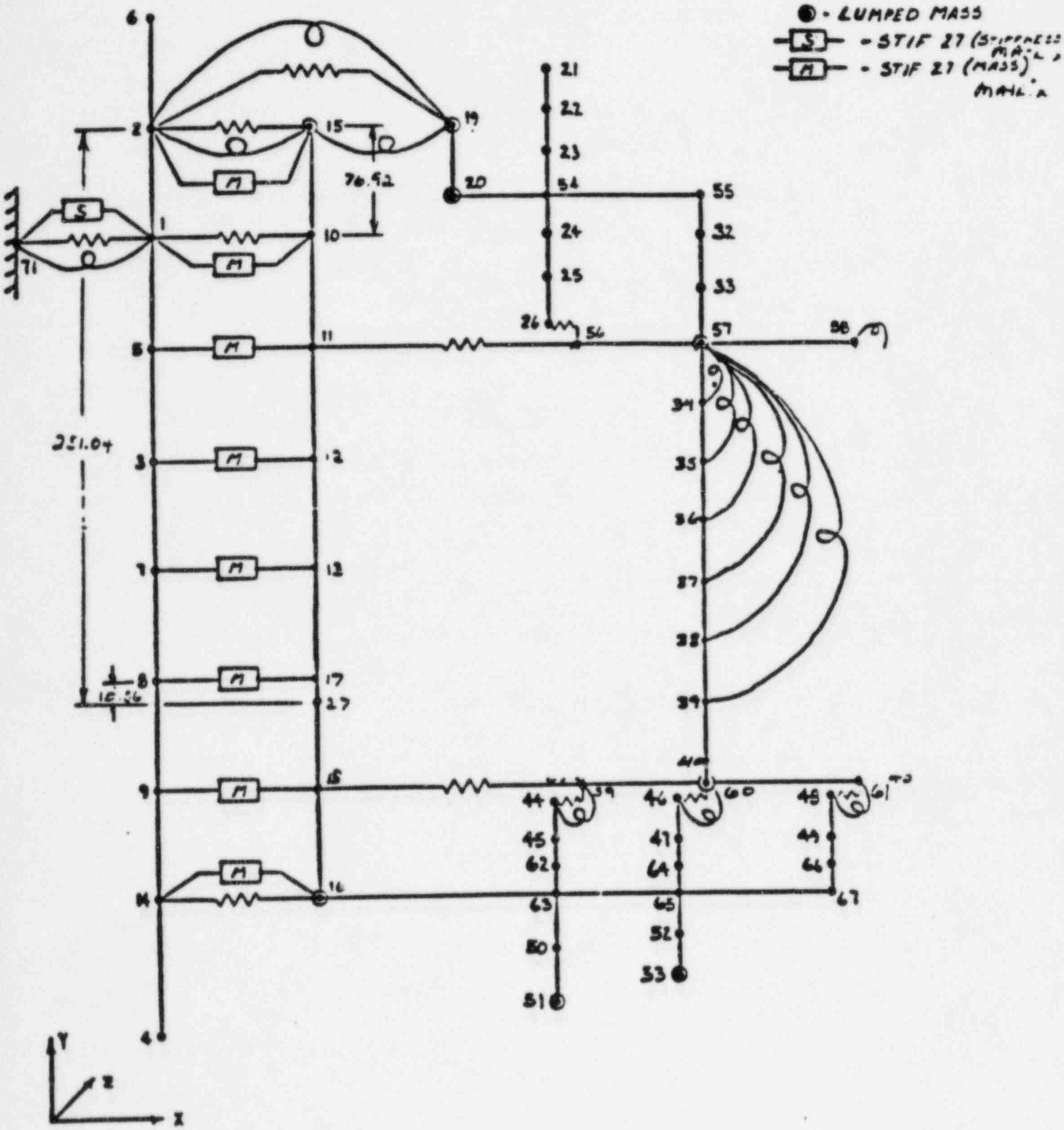
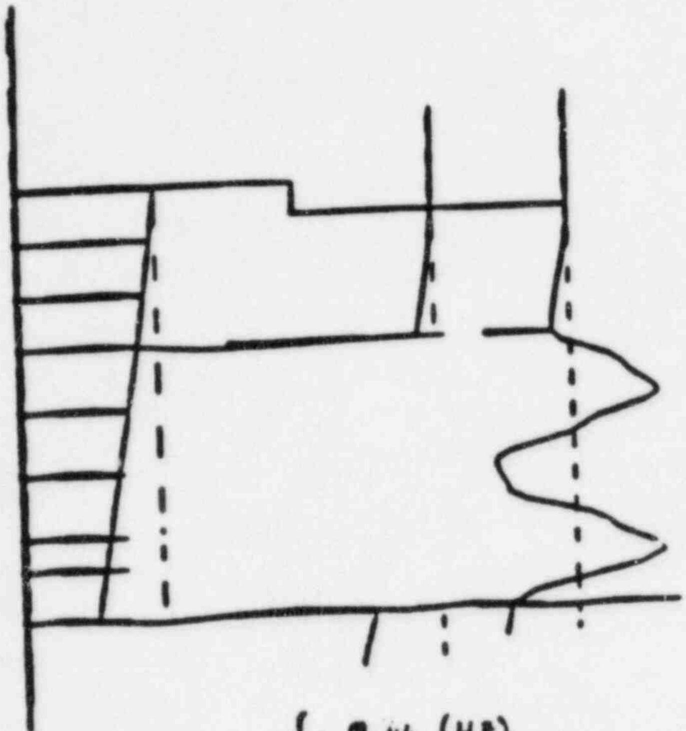
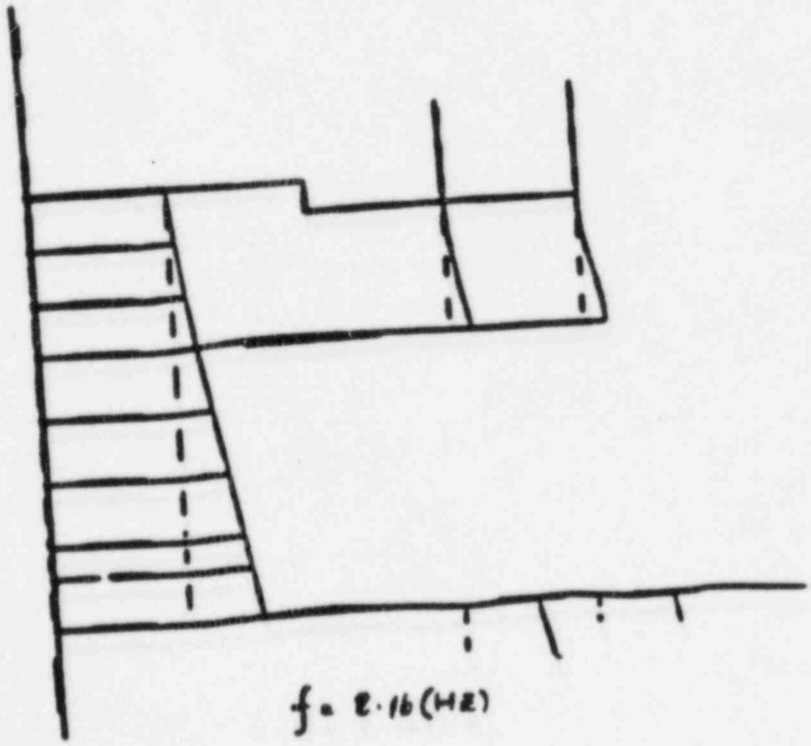


FIGURE 3-3 FINITE ELEMENT MODEL OF 4-LOOP STANDARD (412) PLANT

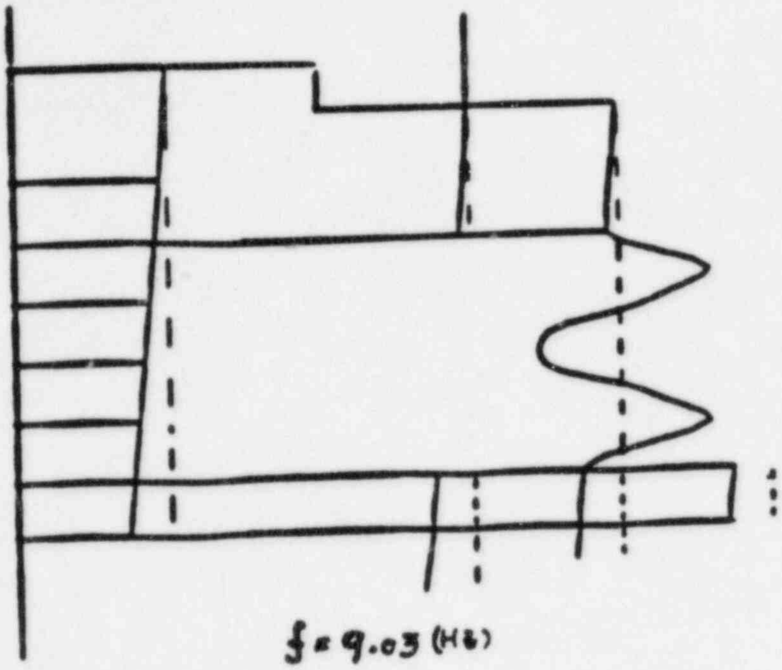


$f = 9.44$ (HZ)



$f = 8.16$ (HZ)

FIGURE 3-4 MODE SHAPES FOR 4XL WITH AND WITHOUT CORE



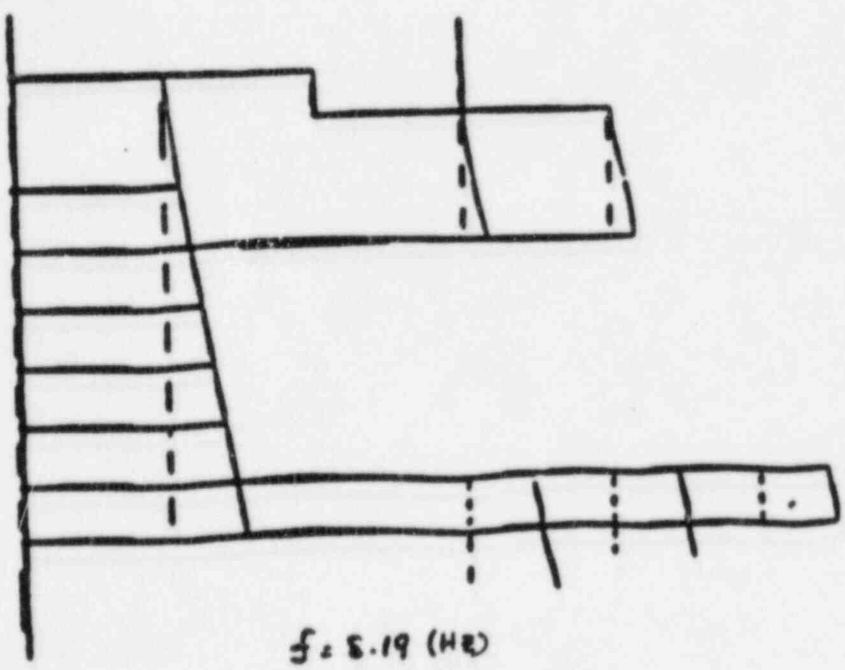


FIGURE 3-5 MODE SHAPES FOR (412) WITH AND WITHOUT FUEL

TABLE 3-4

FREQUENCY COMPARISON OF 4XL (TGX) vs. 4LSTD. (412)

USING TEST DATA

	4XL - 1/7 SCALE	1/24 SCALE	TROJAN-1 (POK)
Core Barrel Beam Mode Frequency	7.86* (Hz)	7.92 (Hz)	7.1 < f < 9 (Hz)

*With Simulated Core

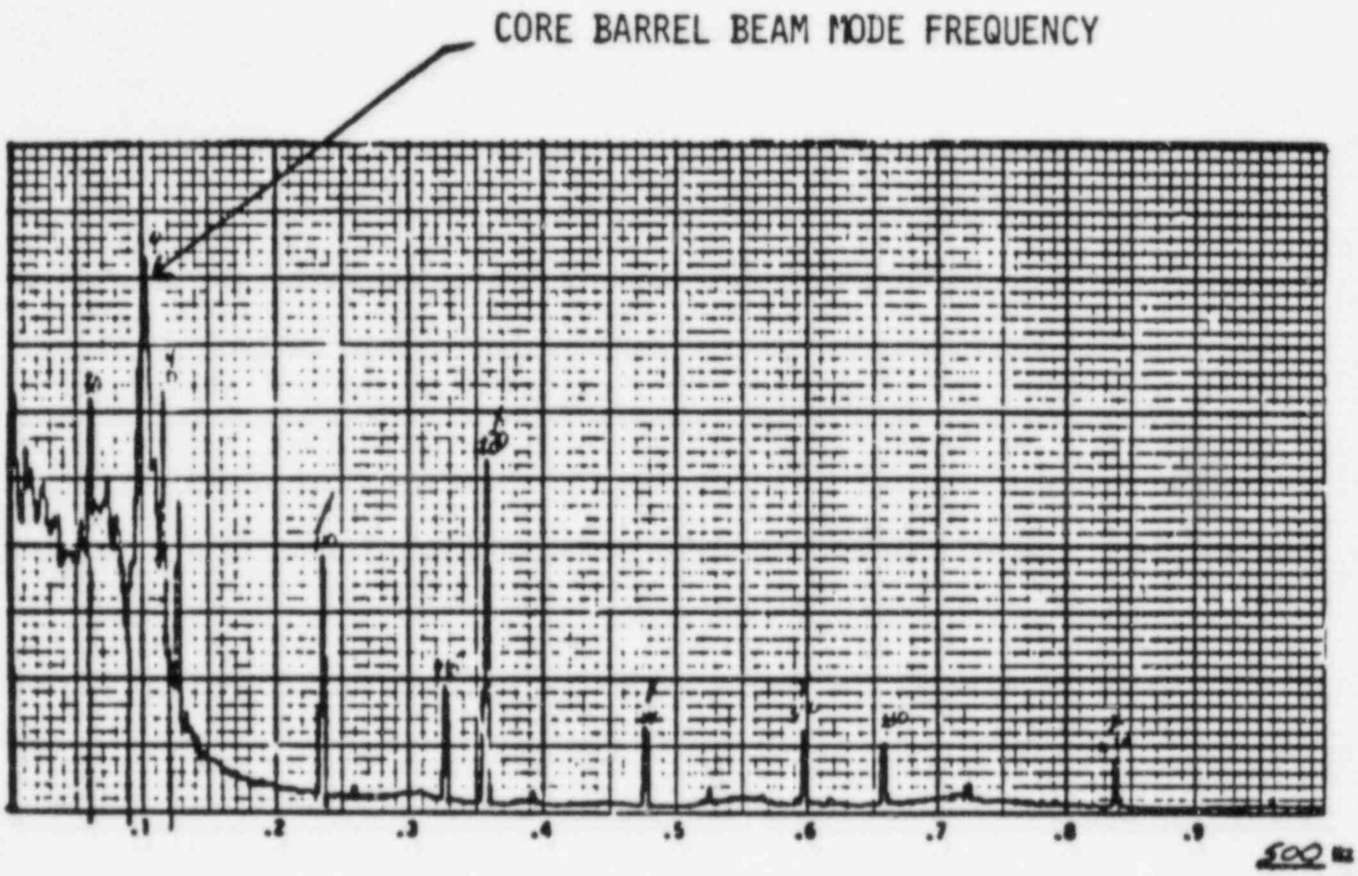


FIGURE 3-6 CORE BARREL SPECTRA FROM 4XL 1/7 SCALE MODEL TEST

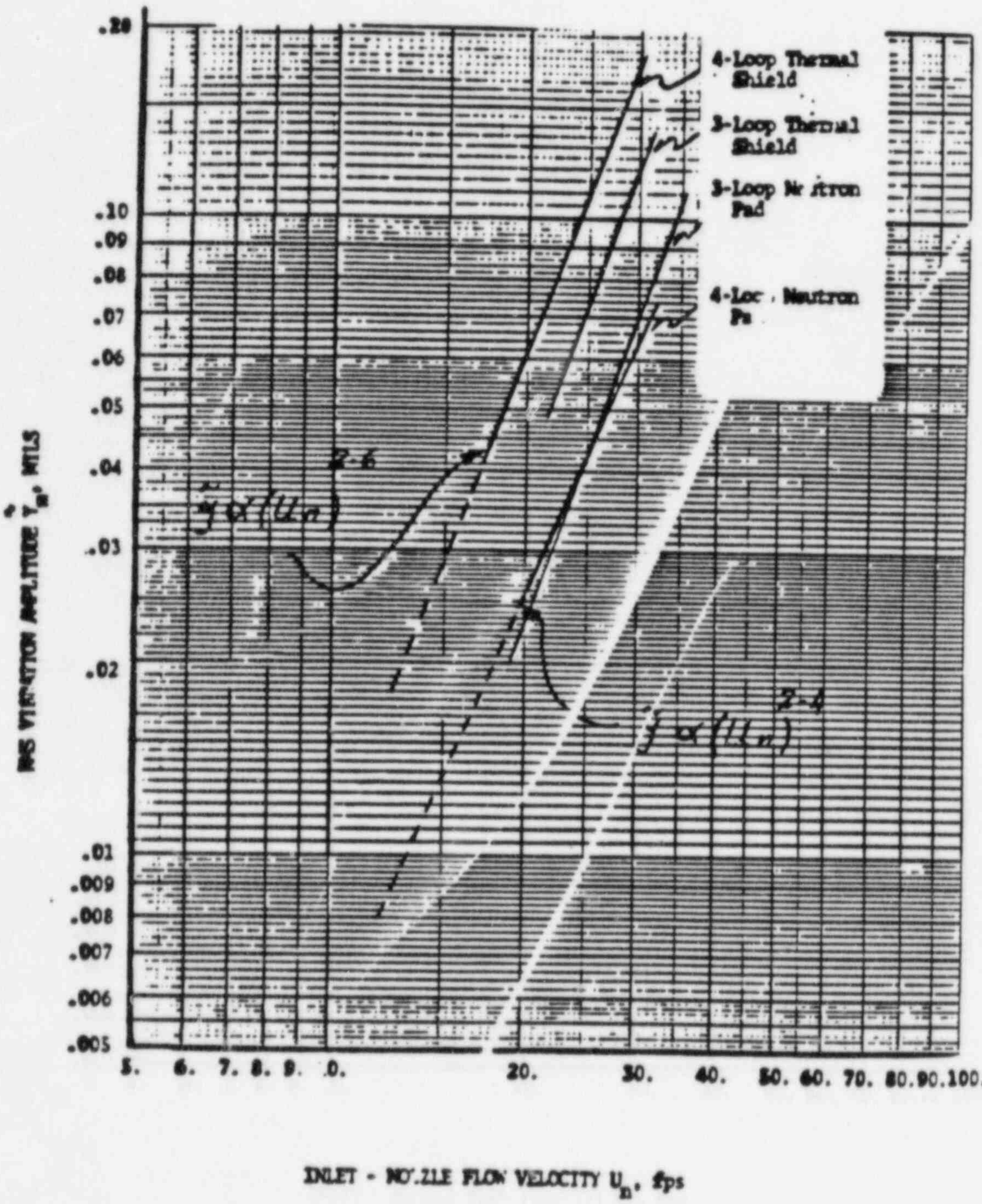


FIGURE 3-7 INLE" NOZZLE VELOCITY, Un (ft/sec) vs. CANTILEVER BEAM MODE AMPLITUDE

TABLE 3-5

CORE BARREL CANTILEVER BEAM MODE RESPONSE

AMPLITUDES (INCH)

PLANT	CORE BARREL CONFIGURATION	MAXIMUM AMPLITUDE (IN)	INLET NOZZLE FLOW-RATE (GPM)	COMMENTS
Indian Point-2	Thermal Shield	0.00406	97,700	1/24 Scale Model Measurement
Paluel-1 (4XL)	Without Neutron Pads+	0.00173	110,500	Plant Test Measurement*
Trojan-1	Neutron Pads	0.00150	102,000	1/24 Scale Model Measurement
South Texas (TGX)	Neutron Pads	0.00192	106,600	1/24 Scale Model Measurement
South Texas (TGX)	Neutron Pads	0.00160	106,600	1/7 Scale Model Measurement**

* Without Neutron Pads

**With Simulated Core

+ Tested Without Neutron Pads

TABLE 3-6

TGX CORE BARREL AMPLITUDE

PREDICTED FROM	AMPLITUDE
1/24 SCALE MODEL	1.92 Mils (rms)
1/7 SCALE MODEL*	1.60 Mils (rms)
PALUEL**	2.1 Mils (rms)

*With Simulated Core

**Neutron Pads Effects Considered

TABLE 3-7

COMPARISON OF CORE BARREL SHELL MODE FREQUENCY (HZ)

Shell Modes (n)	1/24 Scale Model 4 Loop Std. (412)	1/7 Scale Model 4 Loop XL (414)	Plant Test Trojan-1 (412)	Plant Test Paluel-1 (4XL)
n = 2	22 - 24 (Hz)	21 - 23 (Hz)	22 - 23 (Hz)	22 - 23 (Hz)
n = 3	29 - 30 (Hz)	31 - 32 (Hz)	30 - 32 (Hz)	30 - 32 (Hz)

TABLE 3-8

TGX CORE BARREL SHELL MODE AMPLITUDES

DEDUCED FROM TEST DATA	AMPLITUDE (MILS - RMS)	
	n = 2	n = 3
1/24 Scale Model Test	0.21	0.13
1/7 Scale Model Test of 4XL	0.24	0.11
Paluel Plant Test	0.23	0.11

TABLE 3-9

TGX CORE BARREL DEFORMATIONS AND STRESSES

MODE	DEFORMATIONS MILS (RMS)	PEAK STRESSES* (PSI)
Beam Mode		
n = 1	< 2.5	< 1920
Shell Modes		
n = 2	< 0.30	Negligible
n = 3	< 0.15	Negligible

* σ_{peak} = (4σ) rms

Note that the code allowable stress for high cycle fatigue evaluations is conservatively taken to be 13,200 psi for 10" cycles (40 years life); whereas the calculated maximum peak stress is only 1920 psi. Thus the fatigue usage factor, U is essentially zero ($U = \frac{n}{N} = 0$).

SUMMARY

LOWER INTERNALS VIBRATION ASSESSMENT

- 4XL LOWER INTERNALS VIBRATIONAL CHARACTERISTICS ARE SIMILAR TO THOSE OF PLANTS WITH 12 FOOT CORES.

- VIBRATIONAL BEHAVIOR OF 4XL LOWER INTERNALS CAN BE ESTABLISHED BY PREVIOUS PLANT AND SCALE MODEL TESTS.

- BECAUSE OF THE HIGHER FLOWRATE, CORE BARREL VIBRATION AMPLITUDES FOR THE 4XL PLANTS ARE EXPECTED TO BE SLIGHTLY LARGER THAN FOR SIMILAR PLANTS WITH 12 FOOT CORES.

  - THIS DIFFERENCE IS NOT SIGNIFICANT BECAUSE AMPLITUDES AND RESULTING STRESSES ARE SMALL.

COMPARISON OF 4XL AND 412 UPPER INTERNALS FOR FLOW-INDUCED VIBRATIONS

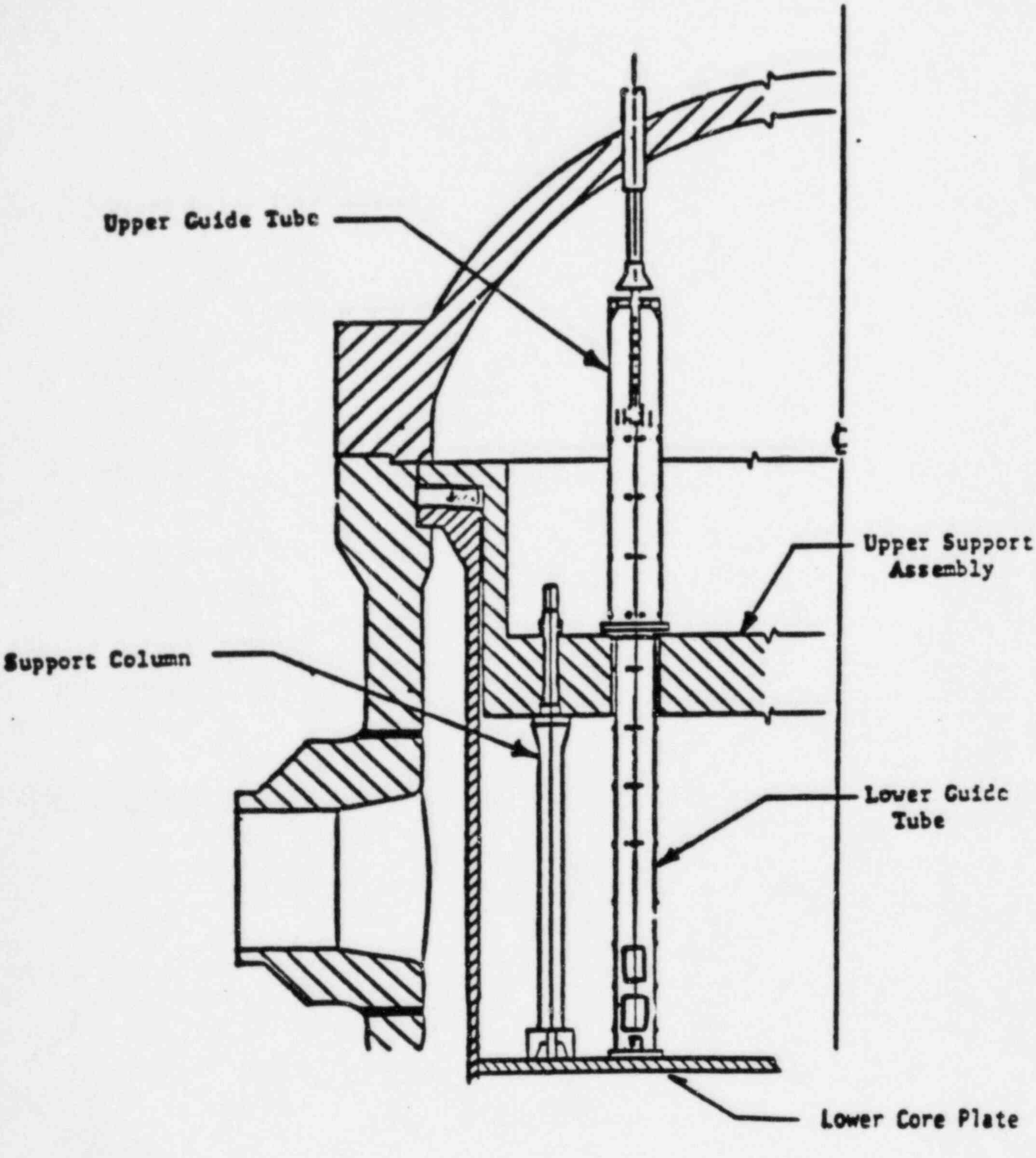
- COLUMN & GUIDE TUBES DESIGN ARE THE SAME.

- TYPE OF EXCITATION IS THE SAME

  - UPPER INTERNALS ARE EXCITED BY FLOW TURBULENCE GENERATED BY AXIAL AND CROSS FLOWS THAT CONVERGE ON THE OUTLET NOZZLES.

- ONLY DIFFERENCE IS THE EXCITATION FORCE ($PV^2/2GC$)

  - ANALYSIS SHOWS THAT THE DIFFERENCE IS SMALL I.E., STP IS APPROXIMATELY 7% LOWER.



UPPER INTERNALS

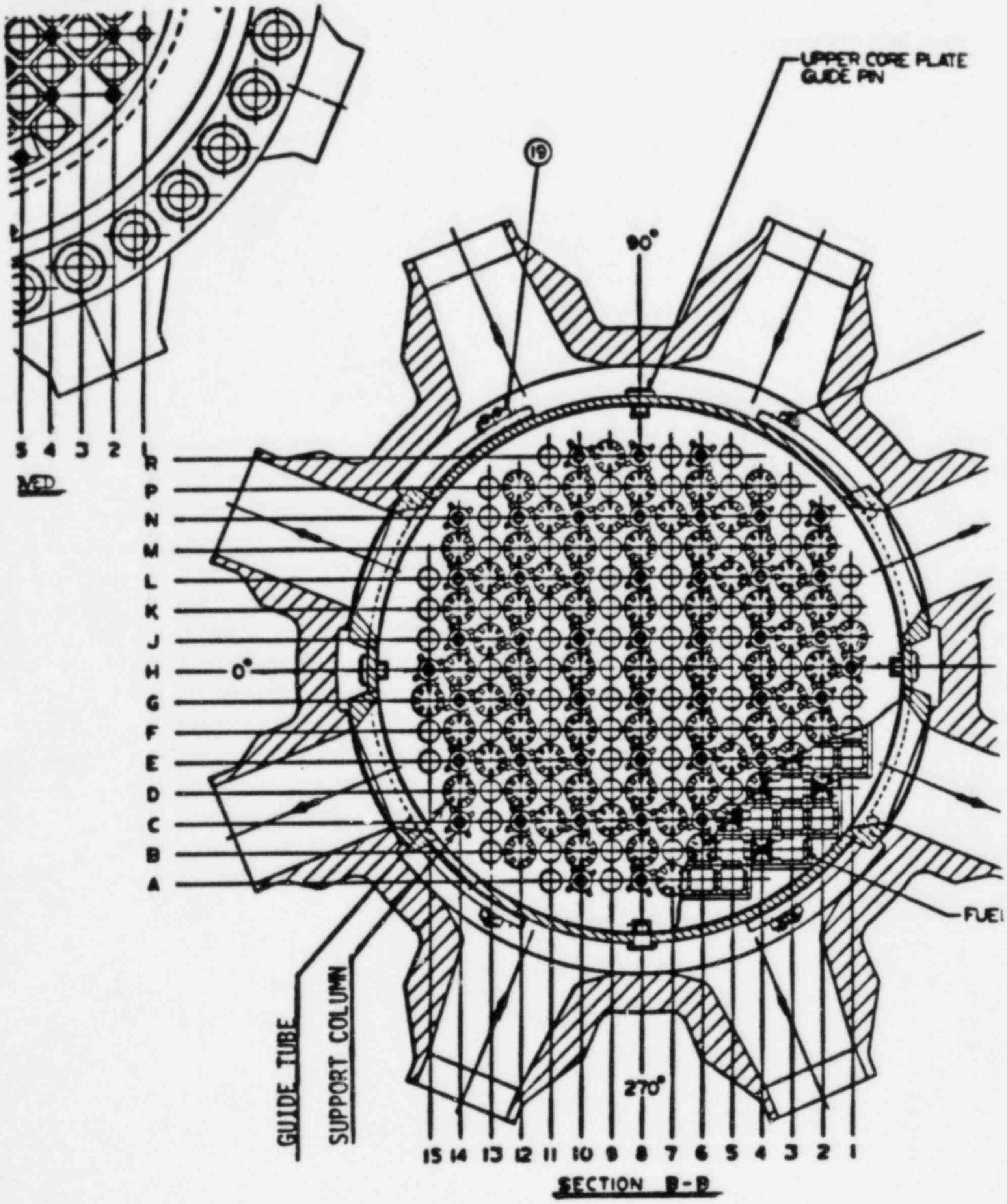


FIGURE 4-2 UPPER INTERNALS COMPONENTS (G.T. AND S.C.) IN THE DIRECTION OF CROSS FLOW AT OUTLET NOZZLES

UHI-SCALE MODEL vs. SEQUOYAH-1
UPPER INTERNALS FREQUENCIES

COMPONENT	SEQUOYAH-1 PROTOTYPE (HZ)	UHI 1/7 SCALE MODEL (HZ)
LOWER GUIDE TUBE		
0° - 180°	71.0	69.5
90° - 270°	65.5	64.0
5" SUPPORT COLUMN	116.5 - 155	111.7 - 157.6
4" SUPPORT COLUMN	109 - 155	103.7 - 155.5

UHI-SCALE MODEL vs. SEQUOYAH-1

UPPER INTERNALS STEADY FLOW LOADS

COMPONENTS	UHI 1/7 SCALE MODEL (LB_f)	SEQUOYAH-1 PROTOTYPE (LB_f)
S.C. L-2	387	385
S.C. N-2	518	297
S.C. K-1	349	402
G.T. K-2	639	672
G.T. M-2	1030	863

S.C. = SUPPORT COLUMN

G.T. = GUIDE TUBE

UHI-SCALE MODEL vs. SEQUOYAH-1 UPPER INTERNALS

RANDOM FLOW-INDUCED VIBRATORY RESPONSE

COMPONENT	UHI 1/7 SCALE MODEL (MILS)	SEQUOYAH-1 PROTOTYPE (MILS)
S.C. L-2	0.160	0.115
S.C. N-2	0.134	0.137
S.C. K-1	0.096	0.111
G.T. K-2	0.455	0.312
G.T. M-2	0.768	0.443

TABLE 4-5

UPPER INTERNALS FREQUENCY COMPARISON

COMPONENT	4XL PALUEL-PLANT (HZ)	4XL 1/7 - SCALE MODEL (HZ)
Lower Guide Tubes		
m = 1		
0° - 180°	67	70 - 74
90° - 270°	61	61 - 67
m = 2		
0° - 180°	160 - 172	-
90° - 270°	160 - 172	-
Support Column		
m = 1		
0° - 180°	110	119
90° - 270°	106 - 113	119 - 122

GUIDE TUBES

- VERY SMALL VIBRATION AMPLITUDE
- MODEL VALIDITY DEMONSTRATED BY COMPARING WITH MEASUREMENTS IN SEQUOYAH
- COMMANCHE PEAK, CALLAWAY, MAAHSHAN AND 3XL PLANTS GUIDE TUBES HAVE THE SAME DIMENSION (DESIGN) AND HAVE SUCCESSFULLY PASSED HOT FUNCTIONAL TEST
- SEQUOYAH, OHI, OKB, MCGUIRE GUIDE TUBES ARE SIMILAR AND HAVE LONG OPERATING EXPERIENCE
- WITH CORE, AT POWER, THE VIBRATION AMPLITUDE IS THE SAME AS HOT FUNCTIONAL. THE PRESENCE OF THE CORE DOES NOT AFFECT THE VIBRATION OF THE GUIDE TUBE

COLUMNS

- VERY SMALL VIBRATORY AMPLITUDE
- MODEL VALIDITY DEMONSTRATED BY COMPARING WITH MEASUREMENTS IN SEQUOYAH
- COMMANCHE PEAK, CALLAWAY AND MAANSHAN AND 3XL PLANTS COLUMNS HAVE THE SAME DIMENSIONS (DESIGN) AND HAVE SUCCESSFULLY PASSED HOT FUNCTIONAL TEST.
- SEQUOYAH, OHI, OKB & MCGUIRE COLUMNS ARE SIMILAR AND HAVE LONG OPERATING EXPERIENCE
- WITH CORE, AT POWER, VIBRATION AMPLITUDE IS THE SAME AS HOT FUNCTIONAL. THE PRESENCE OF THE CORE DOES NOT AFFECT THE VIBRATION OF THE COLUMNS.

UPPER INTERNALS QUALIFICATION

STRUCTURAL ADEQUACY OF 4XL UPPER INTERNALS IS DEMONSTRATED BY:

- PLANT TESTS ON SEQUOYAH
- TESTS ON 1/7 SCALE 4XL MODEL
- TESTS ON 1/7 SCALE UHI UPPER INTERNALS
- OPERATING EXPERIENCE
- TESTS SUPPORTED BY ANALYSIS

PRE AND POST HOT FUNCTIONAL INSPECTION

- PERFORM AN EXTENSIVE VISUAL EXAMINATION OF ALL MAJOR INTERNALS COMPONENTS AND LOAD BEARING ELEMENTS BEFORE AND AFTER HOT FUNCTIONAL TESTING (240 HOURS MINIMUM TIME AT FULL FLOW).

PRE-AND POST-HOT FUNCTIONAL INSPECTION

- REMOVE INTERNALS FROM REACTOR VESSEL.
- VISUALLY INSPECT THE FOLLOWING AREAS:
  - ALL MAJOR LOAD-BEARING ELEMENTS OF THE REACTOR INTERNALS RELIED UPON TO RETAIN THE CORE SUPPORT STRUCTURE IN POSITION.
  - THE LATERAL, VERTICAL AND TORSIONAL RESTRAINTS PROVIDED WITHIN THE VESSEL.
  - THOSE LOCKING AND BOLTING COMPONENTS WHOSE FAILURE COULD ADVERSELY AFFECT THE STRUCTURAL INTEGRITY OF THE REACTOR INTERNALS.
  - THOSE SURFACES THAT ARE KNOWN TO BE OR MAY BECOME CONTACT SURFACES DURING OPERATION.
  - THOSE CRITICAL LOCATIONS ON THE REACTOR INTERNAL COMPONENTS AS IDENTIFIED BY THE VIBRATION ANALYSIS.
  - THE INTERIOR OF THE REACTOR VESSEL FOR EVIDENCE OF LOOSE PARTS OR FOREIGN MATERIAL.

THE RESULTS OF THESE INSPECTIONS WILL BE TABULATED ON THE VIBRATIONAL CHECK-OUT FUNCTIONAL TEST INSPECTION DATA DRAWINGS.

ACCEPTANCE STANDARDS ARE THE SAME AS REQUIRED IN THE SHOP BY THE ORIGINAL DESIGN DRAWINGS AND SPECIFICATIONS.

SUMMARY AND CONCLUSIONS

- STP INTERNALS VIBRATIONAL CHARACTERISTICS ARE SIMILAR TO THOSE PLANTS WITH 12 FOOT CORES.

- VIBRATIONAL BEHAVIOR OF STP INTERNALS HAS BEEN ESTABLISHED BY PREVIOUS PLANT AND SCALE MODEL TESTS.

- VIBRATIONAL AMPLITUDES AND STRESSES ARE SMALL.

- EFFECTS OF ADDED FUEL LENGTH HAVE BEEN SHOWN TO BE INSIGNIFICANT BASED ON ANALYTICAL INVESTIGATIONS, SCALE MODEL TESTS AND PREVIOUS PLANT TESTS.

- PREDICTIONS OF VIBRATIONS FROM SCALE MODEL TESTS ARE GENERALLY CONSERVATIVE. SCALE MODEL PREDICTIONS CAPABILITY VERIFIED BY COMPARISONS TO PLANT TESTS.

- OVERALL TEST/ANALYSIS DATA BASE SHOWS INTERNALS RESPONSE TO FLOW INDUCED VIBRATIONS IS SMALL.

SUMMARY AND CONCLUSIONS

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- OVERALL TEST/ANALYSIS DATA BASE SHOWS INTERNALS RESPONSE TO FLOW INDUCED VIBRATIONS IS SMALL.

List of Attendees

Attendee	Representing
N. P. Kadambi*	NRC
Y. C. (Renee) Li	NRC MEB
C. A. Ayala*	HL&P Licensing
J. G. White*	HL&P Engineering
A. B. Poole	HL&P Engineering
C. R. Allen	HL&P Engineering
M. D. Scott	Bechtel Engineering
N. A. Joonejo*	Bechtel Engineering
W. W. Watson	Bechtel Engineering
E. W. Hunt, Jr.	Bechtel Engineering
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W. F. Guerin*	Westinghouse
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C. Boyd*	Westinghouse

*Part - time