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December 29, 1983
RBG- 16,672
File Code G9.5, G9.8.6.1

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Denton:

River Bend Station Units 1 & 2
Docket Nos. 50-458/50-459

Enclosed for your review are Gulf States Utilities Company's (GSU) responses to Draft Safety Evaluation Report (DSER) open items identified by the Nuclear Regulatory Commission's (NRC) Mechanical Engineering Branch (MEB). A response to the December 7, 1983 NRC letter requesting further clarification on the River Bend Station design regarding pipe/component supports is provided in Enclosure 21. In addition, the Licensing Review Group-II (LRG) positions 1, 2, 3 and 4-MEB are endorsed herein. Attachment 1 is a summary listing of the items discussed in Attachment 2. Attachment 2 provides the response and reference material for each item. Where indicated, these responses will be provided in a future amendment to the FSAR.

Sincerely,

William J. Cahill, Jr.
for J. E. Booker
Manager-Engineering
Nuclear Fuels & Licensing
River Bend Nuclear Group

JEB/RJK/tf

Attachments-40 copies

B001
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DSER MEB ITEMS

	DSER SECTION	RELATED QUESTION	SUBJECT	FSAR CHANGES
1.	3.2.1 pg.3-1	Q#210.30	DG system classifications-seismic	Amendment 7
2.	3.2.2 pg.3-2	Q#210.31	Exceptions to RG 1.26-QA class	Amendment 7
3.	3.6.2 pg.3-32	Q#210.32 Q#210.49	Criteria of Limits used in no break zone	Amendment 7 & Enclosure 1
4.	3.6.2 pg. 3-32	Q#410.3 Q#210.54 Q#210.77 Q#210.45 Q#210.46	Pipe break criteria HEL cracks	3rd qtr. 1984, January, 1985 Enclosure 19
5.	3.6.2 pg.3-32	Q#210.37	Pipe thrust coefficients	Amendment 7
6.	3.6.2 pg.3-32	Q#210.75	Jet impingement analysis	Amendment 7
7.	3.6.2 pg.3-32	Q#210.78 Q#210.79	Pipe whip restraint design	Amendment 7
8.	3.6.2 pg.3-32	Q#210.6 Q#210.8 Q#210.11 Q#210.65 Q#210.66	New Loads information	--------
9.	3.9.1 pg.3-49	Q#210.56 Q#210.58	Computer program verification	Amendment 7 & 9 and Enclosure 2
10.	3.9.1 pg.3-49	Q#210.60 Q#210.63	Design transients & no. of cycles	Amendment 7
11.	3.9.1 pg.3-49	Q#210.61	Component test	Amendment 7
12.	3.9.2 pg.3-50	Q#210.62	Preop. vibration testing	Amendment 7 & Enclosure 3
13.	3.9.2 pg.3-50	Q#210.6[illegible]	NUREG-0619 compliance	Amendment 7
14.	3.9.3 pg.3-53	Q#210.70	Functional capability of ASME component	Amendment 7

	DSER SECTION	RELATED QUESTION	SUBJECT	FSAR CHANGES
15.	3.9.3 pg.3-53	Q#210.66	Design limits of ASME components (New Loads)	June, 1984
16.	3.9.3 pg.3-53	Q#210.68	Support design	Amendment 7
17.	3.9.4 pg.3-55	Q#210.98	Prototype info. on CRD	Amendment 7 Enclosure 4
18.	3.9.5 pg.3-56	Q#210.67	Stress, deformation & buckling limits	Amendment 7
19.	3.9.6 pg.3-57	Q#210.19 Q#210.99	IST program specifics (pump & valve)	Amendments 7 & 1
20.	5.3.1 pg.5-27	Q#210.22 Q#210.25	Fracture toughness data	Enclosure 20
21.		Q#210.5	Pump and valve qualification data	Enclosure 5 & July, 1984
22.		Q#210.15	Pressure relief devices	Enclosure 6
23.		Q#210.43	New loads	Enclosure 7
24.		Q#210.71	Pressure isolation valves	Amendment 9
25.		Q#210.82	Feedwater check valve design	Enclosure 8
26.		Q#210.92	Asymmetric loads	Enclosure 9 & 3rd qtr. 1984
27.		Q#210.100	Out-of-roundness criteria	Enclosure 10
28.		Q#210.101	Piping tolerances	Enclosure 11
29.		Q#210.102	Codes for reactor coolant pressure boundary components	Enclosure 12
30.		Q#210.103	RCIC turbine design	Enclosure 13
31.		Q#210.104	Scram discharge volume piping classification	Enclosure 14
32.		Q#210.105	Quencher support welds	Enclosure 15
33.		Q#210.106	Quencher ASME stress design	Enclosure 15
34.		Q#210.107	Pipe support lateral instabilities design	Enclosure 16

DSER SECTION	RELATED QUESTION	SUBJECT	FSAR CHANGES
35.	Q#210.108	HCU floor design for pool swell loads	Enclosure 17
36.	Q#210.69	LRG-II 1-MEB Input criteria for use of SRSS for mechanical equipment	Amendment 7
	Q#210.98	LRG-II 2-MEB RPV internals vibration test program for BWR/6	Amendment 7 & Enclosure 4
	Q#210.63	LRG-II 3-MEB OBE stress cycles used for NSSS mechanical equipment design	Amendment 7
		LRG-II 4-MEB Kuosheng incore instrument tube break	Enclosure 18

ATTACHMENT 2
MEB

1. DSER, page 3-1 - Diesel Generator System Classification-Seismic

Response

This item is addressed in the response to FSAR Question 210.30, Chapter 3, which was submitted to the NRC in FSAR Amendment 7.

2. DSER, page 3-2 - Regulatory Guide 1.26 Exceptions

Response

This item is included in the response to FSAR Question 210.31, Chapter 3, which was submitted to the NRC in FSAR Amendment 7.

3. DSER, page 3-32 - Clarification of Limits Used for No-Break Zone

Response

This item is addressed in the response to FSAR Question 210.49 (Amendment 7) and a revised response to 210.32, Chapter 3 provided in Enclosure 1. For ASME Code, Section III Class 1 piping the RBS Architect Engineer, Stone & Webster, applies Section 3.6.2 of Standard Review Plan (SRP) NUREG-0800 for calculating the design and fatigue limits on class 1 piping under upset plant conditions and an independent OBE event transient.

4. DSER, page 3-32 - Pipe Break Criteria for High Energy Line Cracks

Response

This item is discussed in part in the response to FSAR Questions 210.54 and 210.77, Chapter 3, submitted to the NRC in FSAR Amendment 7. Revised Questions 210.45 and 210.46 regarding the class 1 pipe stress analyses is provided in Enclosure 19. A discussion on HEL cracks and breaks outside containment is provided in a supplemental response to Question 210.77 (Enclosure 19).

5. DSER, page 3-32 - Pipe Thrust Coefficient

Response

This item is addressed in the response to FSAR Question 210.37, Chapter 3, which was submitted to the NRC in FSAR Amendment 7.

Attachment 2 (cont'd)

6. DSER, page 3-32 - Jet Impingement Analysis

Response

This item is addressed in the response to FSAR Question 210.75, Chapter 3, which was submitted to the NRC in FSAR Amendment 7.

7. DSER, page 3-32 - Pipe Whip Restraint Design

Response

This item is addressed in the response to Questions 210.78 and 210.79, Chapter 3, which were submitted to the NRC in FSAR Amendment 7.

8. DSER, page 3-32 - New Loads Information

Response

The response to Questions 210.6, 210.8, 210.11, 210.65 and 210.66 will be provided during June, 1984 after completion of the hydrodynamic analysis for River Bend Station.

9. DSER, page 3-49 - Computer Program Verification

Response

This item is addressed in the response to Questions 210.56 and 210.58, Chapter 3, which were submitted to the NRC in FSAR Amendment 7 and 9, respectively. Further specific verification information for the STRUDL II, STARDYNE, LION, NUPIPE, and ANSYS computer programs is provided in Enclosure 2.

10. DSER, page 3-49 - Design Transients and Number of Cycles

Response

This item is addressed in the response to Questions 210.60 and 210.63, Chapter 3, which were submitted to the NRC in FSAR Amendment 7.

11. DSER, page 3-49 - Component Tests

Response

This item is addressed in the response to FSAR Question 210.61, Chapter 3, which was provided to the NRC in FSAR Amendment 7.

Attachment 2 (cont'd)

12. DSER, page 3-50 - Preoperational Vibration Testing

Response

This item is addressed in part in the response to FSAR Question 210.62, Chapter 3, which was provided to the NRC in FSAR Amendment 7. The response addressed the NSSS scope. The BOP preoperational vibration testing program is provided in Enclosure 3.

13. DSER, page 3-50 - NUREG-0619 Compliance

Response

This item is addressed in the response to FSAR Question 210.64, Chapter 3, which was provided to the NRC in FSAR Amendment 7.

14. DSER, page 3-53 - Functional Capability of ASME Component

Response

This item is addressed in the response to FSAR Question 210.70, Chapter 3, which was provided to the NRC in FSAR Amendment 7.

15. DSER, page 3-53 - Design Limits of ASME Components

Response

This item is addressed in part in the response to FSAR Question 210.66, Chapter 3, which was submitted to the NRC in FSAR Amendment 7. Upon the new loads update of the FSAR, currently scheduled for June 1984, the guidance provided in NUREG-0800 for piping, vessel, internals, component support, and pipe and floor-mounted equipment will be addressed.

16. DSER, page 3-53 - Support Design

Response

This item is addressed in the response to FSAR Question 210.68, Chapter 3, which was submitted to the NRC in FSAR Amendment 7.

17. DSER, page 3-55 - Prototype Information on the Control Rod Drive System

Response

This item is addressed in the revised response to FSAR Question 210.98, Chapter 3, which was initially submitted to the NRC in FSAR Amendment 7. LRG-II issue 2-MEB discusses the reactor pressure vessel internals vibration test program for BWR/6's. River Bend Station is scheduled to perform the required non-prototype tests in September 1984 and submit its results with a

reference to the Kuo Sheng test report during November 1984 (Enclosure 4).

18. DSER, page 3-56 - Stress, Deformation & Buckling Limits

Response

This item is address in the response to FSAR Question 210.67, Chapter 3, which was submitted to the NRC in FSAR Amendment 7.

19. DSER, page 3-57 - Inservice Testing Program for Pumps and Valves

Response

This item is addressed in part in the response to FSAR Questions 210.99 and 210.19, Chapter 3, which was provided to the NRC in FSAR Amendments 7 and 1, respectively. The RBS Inservice Testing Program will be provided within 6 months of the anticipated date of commercial operation.

20. The responses to Questions 210.22 and 210.25, "Fracture Toughness Data", are provided in Enclosure 20. The responses will be submitted in a future FSAR amendment.

21. A revised response to FSAR Question 210.5, "Additional Qualification Information on Pumps and Valves", is provided in Enclosure 5. FSAR Tables 3.9A-5, 10 and 11 will be updated during July, 1984.

22. The revised response to FSAR Question 210.15, "Pressure Relief Devices", is provided in Enclosure 6. The new response will be submitted in a future FSAR amendment.

23. Updated data for Tables 3.6A-1 through 3.6A-11 in response to FSAR Question 210.43 is provided in Enclosure 7. This information will be submitted in a future FSAR amendment.

24. A response to FSAR Question 210.71, Chapter 3, was submitted in Amendment 9. A discussion of the leak testing program will be provided within 6 months of the anticipated date of commercial operation.

25. A response to FSAR Question 210.82, "Feedwater Isolation Check Valves", is provided in Enclosure 8. The response will be provided in a future amendment to the FSAR.

26. A revised submittal date of the third quarter 1984, for the reactor asymmetric loads analysis (Enclosure 9) is provided in response to FSAR Question 210.92. This new date will be incorporated into a future FSAR amendment.

Attachment 2 (cont'd)

27. Technical justification for the assumptions used in calculating the stress analysis for class 1 piping is provided in a response to new FSAR Question 210.100, "Class 1 Piping Out-of-Roundness Tolerances" (Enclosure 10). This information will be submitted in a future FSAR amendment.

28. A discussion of the RBS program which ensures that counterbore of piping is in accordance with the Stone & Webster Specification No. 228.000, "Specification for Piping Engineering and Design", and that minimum wall thickness is not violated is provided in a response to new FSAR Question 210.101 (Enclosure 11). The response to FSAR Question 210.101 will be submitted in a future FSAR amendment.

29. The response to new FSAR Question 210.102, "Codes for Reactor Coolant Pressure Boundary Components", is provided in Enclosure 12. New Table 3.2-9 will be submitted in a future FSAR amendment.

30. A response to new Question 210.103, "RCIC Turbine Design", is provided in Enclosure 13. This response will be provided in a future FSAR amendment.

31. The response to new Question 210.104, "Scram Discharge Volume Piping Classification" is provided in Enclosure 14. This information will be submitted in a future FSAR amendment.

32. The response to new FSAR Question 210.105, "Quencher Base Support", is provided in Enclosure 15. Section ND-3645 of the ASME Boiler and Pressure Vessel Code was taken into consideration during the design of the RBS X-quencher. This new response will be submitted in a future FSAR amendment.

33. A response to new FSAR Question 210.106, "Quencher ASME Stress Design", is provided in Enclosure 15. Stresses at critical locations in the X-quencher configuration have been evaluated and determined to be within the ASME code allowables for quencher design. This information will be submitted in a future FSAR amendment.

34. A response to new FSAR Question 210.107, "Pipe Supports Lateral Instability Design", is provided in Enclosure 16. The only ECCS piping inside containment located within twenty-five feet above the maximum suppression pool water level is short horizontal segments of piping at the penetration elevation. These segments are analyzed for pool swell impact loads.

35. A response to new FSAR Question 210.108, "Hydraulic Control Unit (HCU) Floor Design for Pool Swell Loads", is provided in Enclosure 17. The Control Rod Drive (CRD) piping has been designed to withstand the vibratory responses due to hydrodynamic loadings including the loads on the HCU floor due to a pool swell event. This information will be submitted in a future FSAR amendment.

Attachment 2 (cont'd)

36. The endorsement of LRG-II positions 1-MEB through 4-MEB is provided below. The discussion of these items is provided in the enclosures.

ITEM	TITLE	ENDORSED	FSAR DISCUSSED
1-MEB	Input Criteria for Use of SRSS for Mechanical Equipment	Yes	3.9.3.4.1.2A
2-MEB	RPV Internals Vibration Test Program for BWR/6	Yes	3.9.2.4B
3-MEB	OBE Stress Cycles Used for NSSS Mechanical Equipment Design	Yes	3.7.3.2B 3.9.1.1.5B 3.9.1.1.6B
4-MEB	Kuosheng Incore Instrument Tube Break	Yes*	5.3.3.1.4.5

* The operational controls specified in the LRG-II position are not being implemented at RBS. GSU believes the stresses due to flow induced vibration are acceptable based on the new flow deflector design.

QUESTION 210.32 (3.6.2.1.5.2.1A.2.a)

The design stress and fatigue limits for Class 1 piping in the containment penetration areas are not in compliance with Standard Review Plan 3.6.2 and BTP MEB 3-1. If the maximum stress range of Equation (10) exceeds 2.4 Sm, both Equations (12) and (13) must be less than 2.4 Sm. In all cases the cumulative usage factor must be less thar 0.1.

RESPONSE

The response to this request is provided in revised Section 3.6.2.1.5.2.1A, Item 2.

2. Fluid System Piping in Containment Penetration Areas

Breaks are not postulated in the portions of high energy piping between the containment isolation valves, outside and inside containment. Breaks are not postulated in the portions of high energy piping between the isolation valve and the first restraint or groups of restraints designed to protect these portions of piping from breaks outboard of this area both inside and outside containment.

These pipe whip restraints are capable of resisting bending and torsional moments produced by a postulated piping failure outboard of the first restraint or group of restraints beyond the containment isolation valves.

The restraints are designed to withstand the loadings resulting from a postulated piping failure beyond these portions of piping, so that neither the isolation valve operability nor the leaktight integrity of the associated containment penetration will be impaired. These portions of piping are designed to meet the requirements of ASME Code, Section III, Subarticle NE-1120, and the following additional design requirements, which are in conformance with Revision 01 (July, 1981) ~~(November 24, 1975)~~ of SRP 3.6.2 and BTP MEB 3-1, the documents applicable at the time the analysis was performed: 7

a. The following design stress and fatigue limits are not exceeded for Class 1 piping:

INSERT →

~~(1) The maximum stress ranges between any two load sets (including the zero load set) calculated by Equation (10) in Paragraph NB-3653, ASME Code, Section III, for normal and upset plant conditions and an OBE do not exceed 2.4 S_m. If the maximum stress range of Equation (10) exceeds 2.4 S_m, but is not greater than 3 S_m, the cumulative usage factor should be less than 0.1. If the calculated maximum stress range of Equation (10) exceeds 3 S_m, the stress range calculated by both Equations (12) and (13) in Paragraph NB-3653 should not exceed 2.4 S_m and the cumulative usage factor should be less than 0.1.~~

~~(2) The cumulative usage factor should be less than 0.1 if consideration of fatigue limits is required according to Item a.(1).~~

~~(3) The maximum stress, as calculated by Equation (9) in Paragraph NB-3652 under the loading resulting from a postulated piping failure beyond these portions of piping, does not exceed 2.25 S_m.~~

b. The following design stress limits are not exceeded for Class 2 piping:

(1) The maximum stress ranges do not exceed 0.8 $(1.2S_h+S_A)$, as calculated by Equations (9) and (10) in Paragraph NC-3652, ASME Code, Section III, considering normal and upset plant conditions (i.e., sustained loads, occasional loads, and thermal expansion) and an OBE event.

(2) The maximum stresses do not exceed $1.8S_h$, as calculated by Equation (9) in Paragraph NC-3652 under the loadings resulting from a postulated piping failure of fluid system piping beyond these portions of piping.

c. Welded attachments for pipe supports or other purposes, to these portions of piping are avoided, except where detailed stress analysis demonstrates compliance with the limits previously discussed in Section 3.6.2.1.5.2A, Items 2a and 2b.

d. The number of circumferential and longitudinal piping welds and branch connections is minimized.

e. The length of these portions of piping is reduced to the minimum length practicable.

f. The design of pipe anchors or restraints (e.g., connections to containment penetrations and pipe whip restraints) does not require welding directly to the outer surface of the piping (e.g., flued integrally forged pipe fittings are used), except where such welds are capable of 100-percent volumetric

(1) The maximum stress range between any two load sets (including the zero load set), calculated by Equation (10) in Paragraph NB-3653, ASME Code, Section III for those loads and conditions thereof, for which level A and level B stress limits have been specified in the design specification including an operating basis earthquake (OBE) event transient, do not exceed 2.4 Sm. If the calculated maximum stress range of Eq. (10) exceeds 2.4 Sm, the stress ranges calculated by both Eq (12) and Eq (13) in Paragraph NB-3653 should meet the limit of 2.4 Sm.

(2) The cumulative usage factor is less than 0.1

(3) The maximum stress, as calculated by Eq. (9) in Paragraph NB-3652 under the loading resulting from a postulated piping failure beyond these portions of piping do not exceed 2.25 Sm except that following a failure outside containment, the pipe between the outboard isolation valve may be permitted higher stresses provided a plastic hinge is not formed.

QUESTION 210.56 (APPENDIX 3A, 3.9.1.2.2B)

NUREG-0800 requires that computer programs used in analyses of seismic Category I Code and non-Code items have the following information provided to demonstrate their applicability and validity:

a. The author, source, dated version and facility.

b. A description and the extent and limitations of its applications.

c. Solutions to a series of test problems which shall be demonstrated to be substantially similar to solutions obtained from any one of sources 1 through 4, and source 5:

   1. Hand calculations

   2. Analytical results published in the literature

   3. Acceptable experimental tests

   4. By a MEB acceptable similar program

   5. The benchmark problems prescribed in Report NUREG/CR-1677, "Piping Benchmark Problems."

Please demonstrate compliance with these requirements and provide summary comparisons for the computer programs used in seismic Category I analyses.

RESPONSE

The response to this request is provided in revised Appendix 3A, ~~Introduction~~ (for BOP systems) and in revised Section 3.9.1.2B (for NSSS systems).

~~Further specific verification information for the STRUDL II, STARDYNE, LION, NUPIPE, and ANSYS computer programs will be provided by June 1983.~~

The following computer programs are used for the analysis of Seismic Category I piping systems, including pipe supports:

1. NUPIPE
2. TRHEAT
3. HTLOAD
4. WATAIR
5. PSPECTRA
6. STEHAM
7. WATHAM
8. PITRUST
9. PILUG
10. ANSYS

For each computer program, there is a brief description of the program's theoretical basis, the assumptions and references used in the program, the extent of its application, and a summary of manual or comparison qualification.

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The computer programs used in the analyses of Seismic Category I code and non-code items are documented in accordance with the requirements of S&W Corporate Engineering Department Procedures. In meeting these requirements, compliance to the following has been demonstrated:

a. Documentation of computer programs includes the author, source dated version, and facility.

b. Documentation of computer programs includes the extent and limitations of its application.

c. Test problems demonstrate the qualification of each program.

RBS FSAR

~~3A.5 GTRUDL II (Structural Design Language)~~

~~GTRUDL II has been designed as a modified subsystem of the Integrated Civil Engineering System (ICES) which was designed and formulated by the Dept. of Civil Engineering at Massachusetts Institute of Technology(1).~~

~~The finite element method provides for the solution of a wide range of solid mechanics problems(2). Its implementation within the context of the GTRUDL analysis facilities expands these for the treatment of plane stress, plane strain, plate bending, shallow shell, and three dimensional stress analysis problems.~~

~~GTRUDL II also provides a dynamic analysis capability for linear elastic structures undergoing small displacements. Either free or forced vibrational response may be obtained, and in the latter case, the forcing functions may be in the form of time histories or response spectra.~~

~~The three-dimensional finite element capability of GTRUDL is used to analyze the drywell at the region of the equipment hatch and personnel door assembly and other regions of interest.~~

~~Seismic Category I structures are analyzed for seismic effect using the dynamic analysis capability of GTRUDL. The analysis yields frequencies of vibration, mode shapes, displacements, velocities, accelerations, and forces. GTRUDL II is a recognized program in the public domain. Version 2, Modification 2 (June 1972), is used. The Software System is IBM-MVS - Release 3.8. The hardware configuration is IBM-3033.~~

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RBS FSAR

3A.5 STRUDL (Structural Design Language)

3A.5.1 General Description

The STRUDL Computer code used within SWEC was developed from Version 2, Modification 2 (June 1972) of the Integrated Civil Engineering System (ICES) STRUDL II program which was designed and formulated by the Department of Civil Engineering at the Massachusetts Institute of Technology. STRUDL II is a recognized program in the public domain. The software system is IBM-MVS Release 3.8. The hardware configuration is IBM-3033.

The finite element method provides for the solution of a wide range of solid mechanics problems(2). Its implementation within the context of the STRUDL analysis facilities expands these for the treatment of plane stress, plane strain, plate bending, shallow shell, and three-dimensional stress analysis problems.

STRUDL also provides a dynamic analysis capability for linear elastic structures undergoing small displacements. Either free or forced vibrational response may be obtained, and in the latter case, the forcing functions may be in the form of time histories or response spectra.

The three-dimensional finite element capability of STRUDL is used to analyze the drywell at the region of the equipment hatch and personnel door assembly and other regions of interest.

Seismic Category I structures are analyzed for seismic effect using the dynamic analysis capability of STRUDL. The analysis yields frequencies of vibration, mode shapes, displacements, velocities, accelerations, and forces.

3A.5.2 Program Verification

Comparisons of results for five test problems performed by both 'STRUDL' and 'GT STRUDL' are provided herein. 'GT-STRUDL' is a recognized program in the public domain, developed by the GT-ICES systems laboratory, school of Civil Engineering, Georgia Institute of Technology, Atlanta, Georgia. In all cases, there is excellent agreement of results between 'STRUDL' & 'GT-STRUDL'.

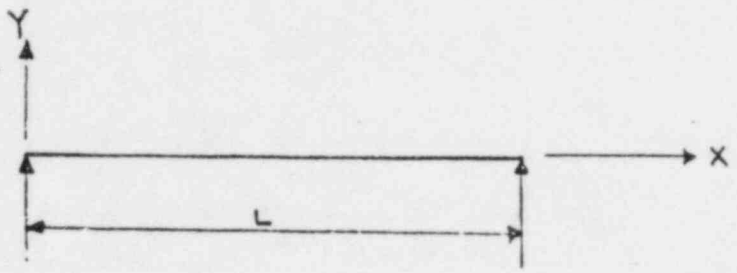
3A.5.2.1 Comparison of 'STRUDL' vs 'GT-STRUDL' results for Dynamic Analysis capability of 'STRUDL'

PROBLEM NO. 1

Find the natural frequencies F (I) of vibration for an I-Beam with simply supported end, vibrating in the Plane of its web,

The pertinent parameters of the beam:

Length	= 30 ft.
Modulus of Elasticity	= 30×10^6 psi
Moment of Inertia	= 3021 in^4
Weight perfoot	= 100 lb.

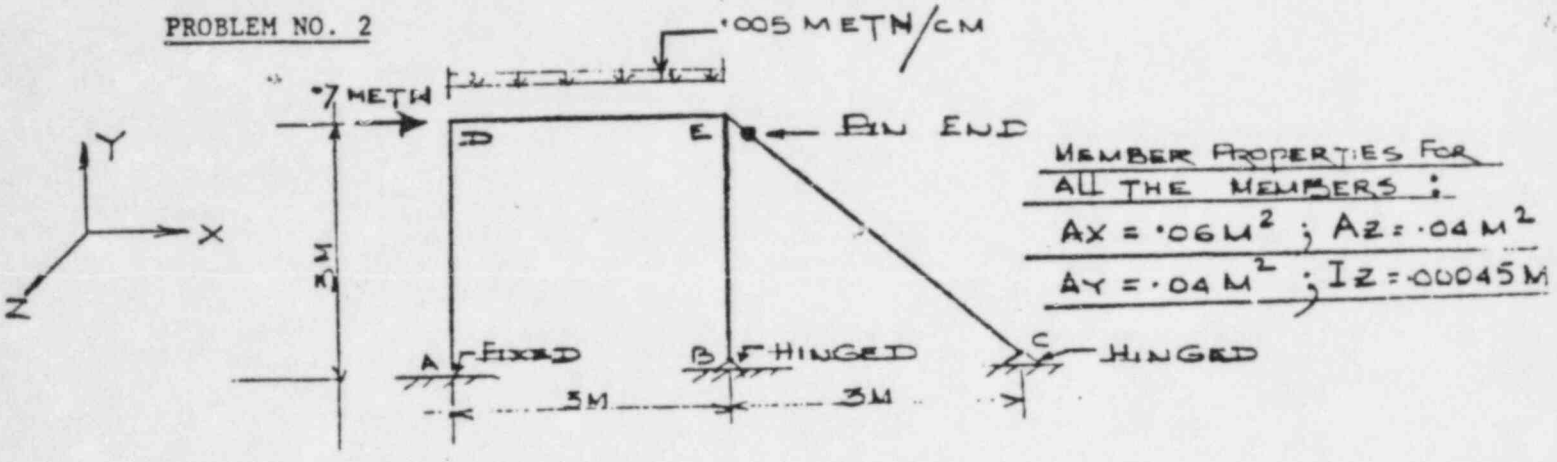


The theoretical results can be verified from "Vibration Problems in Engineering"; Fourth Edition, S. Timoshenko, D.W. Young, W. Weaver, Page 423, Problem 1.

Results: Natural Frequency F(I), where "I" = the mode number
= 24.8 $(I)^2$ Cycles/sec.
=155.82 $(I)^2$ Rad/sec.

The comparison of results (i.e. eigen values and eigenvectors) of the theoretical values, STRUDL and GT-STRUDL are tabulated in the tables 3A.5-1 and 3A.5-2. The eigenvalues for STRUDL and GT-STRUDL agree with each other (Table 3a.5-1). The eigenvectors for STRUDL and GT-STRUDL agree with each other (Table 3A.5-2).

3A.5.2.2 Comparison of 'STRUDL vs 'GT-STRUDL' Results for Static Analysis capability of 'STRUDL'



The Frame as shown in the sketch is tested for the loads as shown in the sketch. Also the frame was tested for joint displacement of joint A and B in the Y direction and also the joint displacement of joint A in the X direction. The member forces and the joint forces of the STRUDL run agreed with the GT-STRUDL run. The comparison of the results are tabulated in the Tables 3A.5-3 and 3A.5-4.

LOADING CONDITION 1

Member DE force Y uniform W-.005 Metn/cm

Joint D load force X 0.7 Metn

LOADING CONDITION 2

Joint A displaced Y -0.8 cm

Joint B displaced Y -0.3 cm

LOADING CONDITION 3

Joint A displaced x -0.2 cm

3A.5.2.3

Comparison of STRUDL vs. GT-STRUDL results for Finite Element Capability of STRUDL.

PROBLEM NO. 3

A foundation mat was analyzed using finite element capability of STRUDL, for a variety of loading combinations. A comparison check is performed for a loading condition which combines the self weight of the substructure and super structure, dead load of 2.5 ft. of soil above the mat and East-West tornado loading, by using finite element capability of GT-STRUDL, a computer program in the public domain. A finite element model is provided in Fig. 3A.5-1. Sign convention details are provided in Fig. 3A.5-2. Refer to the Tables 3A.5-5, & 3A.5-6 for comparison between the results obtained from STRUDL & GT-STRUDL.

3A.5.2.4

Comparison of STRUDL vs. GT-STRUDL results for Static Analysis Capability of STRUDL.

PROBLEM NO. 4

A comparison check is performed for suspended ceiling design using static analysis capability of STRUDL vs. GT-STRUDL. A model is provided in Fig. 3A.5-3. The loading condition accounts for the dead loads of the ceiling. Refer to the Tables 3A.5-7 & 3A.5-8 for comparison between the results obtained from STRUDL & GT-STRUDL.

COMPARISON OF STRUDL VS. GT-STRUDL RESULTS FOR DYNAMIC ANALYSIS (RESPONSE SPECTRA) CAPABILITY OF STRUDL

PROBLEM NO. 5

A comparison check is performed for suspended ceiling design using dynamic analysis capability of STRUDL vs. GT STRUDL. A model is provided in Fig. 3A.5-3. The loading condition accounts for the dynamic seismic loads resulting from ceiling dead load. Refer to the Tables 3A.5-9, 3A.5-10, and 3A.5-11 for comparison between the results obtained from STRUDL & GT-STRUDL.

References - Section 3A.5

1. ICES STRUDL - II Structural Design Language Engineering Users' Manual, Vol. I Frame Analysis, Nov. 1968, Vol. II Addition Design and Analysis Facilities, (Chapters III and IV), June 1971.

2. Zienkiewicz, O.C. and Cheung, Y.K. The Finite Element Method. McGraw-Hill Book Company, Inc., New York, NY, 1967.

3. GT-STRUDL-GT ICES Systems Laboratory, School of Civil Engineering, Georgia Institute of Technology, Atlanta, Georgia.

TABLE: 3A.5-1

INSERT HERE

EIGEN VALUES*:

"I" - The Mode Numbers	Theoretical Results: Frequency Cycles/Sec.	STRUDL Results Frequency Cycles/Sec.	GT-STRUDL Results Frequency Cycles/Sec.
1	24.8	24.84	24.84
2	99.2	99.33	99.33
3	223.2	223.37	223.38
4	396.8	396.39	396.40

*Note: For comparison purpose, the results of four modes are tabulated.

Comparison of Eigen Values from Theoretical results
STRUDL Results and GT-STRUDL Results (Problem #1)

TABLES: 3A.5-2

COMPARISON OF EIGEN VECTORS FROM 'STRUDL' & 'GT-STRUDL'

(Problem #1)

EIGEN VECTORS:

Joint		Y-Disp.	
		STRUDL	GT-STRUDL
MODE 1	1	0.0	0.0
	2	.309	.309
	3	.588	.588
	4	.809	.809
	5	.951	.951
	6	1.000	1.000
	7	.951	.951
	8	.809	.809
	9	.588	.588
	10	.309	.309
	11	0.0	0.0
MODE 2	1	0.000	0.000
	2	.618	.618
	3	1.000	1.000
	4	1.000	1.000
	5	.618	.618
	6	0.000	0.000
	7	-.618	-.618
	8	-1.000	-1.000
	9	-1.000	-1.000
	10	-.618	-.618
	11	0.000	0.000
MODE 3	1	0.0	0.0
	2	-0.809	-.809
	3	-.951	-.951
	4	-.309	-.309
	5	0.588	.588
	6	1.000	1.0
	7	.588	.588
	8	-.309	-.309
	9	-.951	-.951
	10	-.809	-.809
	11	0.0	0.0
MODE 4	1	0.0	0.0
	2	1.0	-1.0
	3	.618	-.618
	4	-.618	.618
	5	-1.0	1.0
	6	0.000	.00
	7	1.0	-1.0
	8	.618	-.618
	9	-.618	.618
	10	-1.0	1.0
	11	0.0	0.0

NOTE: FOR COMPARISON PURPOSES, THE RESULTS OF FOUR MODES ARE TABULAR

TABLE 3A.5-3

THE MEMBER FORCES FROM STRUDL AND GT-STRUDL COMPUTER RUNS FOR DIFFERENT LOADING CONDITIONS

(Problem #2)

Member	Loading Condition	Joint	STRUDL			GT-STRUDL		
			AXIAL	SHEAR Y	MOM Z	AXIAL	SHEAR Y	MO
AD	1	A	1652.25	-237.33	-7958.75	1652.25	-237.33	-7958
		D	-1652.25	237.33	-20071.84	-1652.25	237.33	-20071
	2	A	-583.75	-675.01	-104030.94	-583.75	-675.01	-10403
		D	583.75	675.01	24305.86	583.75	675.01	2430
	3	A	-694.20	1480.59	107811.12	-694.20	1480.59	10781
		D	694.20	-1480.59	67060.81	694.20	-1480.59	6706
DE	1	D	1780.56	1652.25	20071.84	1780.56	1652.25	2007
		E	-1780.56	1654.68	-20215.77	-1780.56	1654.69	-2021
	2	D	675.01	-583.75	-24305.86	675.01	-583.75	-2430
		E	-675.01	583.75	-44640.70	-675.01	583.75	-4464
	3	D	-1480.59	-694.20	-67060.81	-1480.59	-694.20	-6706
		E	1480.59	694.20	-14930.68	1480.59	694.20	-1493

TABLE - 3A.5-3 (continued)

Member	Loading Condition	Joint	STRUDL			GT-STRUDL		
			AXIAL	SHEAR Y	MOM Z	AXIAL	SHEAR Y	MOM
BE	1	B	45.28	171.16	0.0	45.29	171.16	0.
		E	-45.28	-171.16	20215.80	-45.29	-171.16	20215.
	2	B	286.70	377.96	-.01	286.70	377.96	0.0
		E	-286.70	-377.96	44640.70	-286.70	-377.96	44640.8
	3	B	2301.20	126.41	0.00	2301.20	126.41	0.0
		E	-2301.20	-126.41	14930.65	-2301.20	-126.41	14930.7
EC	1	E	2276.03	0.0	0.0	2276.04	0.0	0.0
		C	-2276.03	0.0	0.0	-2276.04	0.0	0.0
	2	E	420.09	0.0	0.0	420.09	0.0	0.0
		C	-420.09	0.0	0.0	-420.09	0.0	0.0
	3	E	-2272.64	0.0	0.0	-2272.65	0.0	0.0
		C	2272.64	0.0	0.0	2272.65	0.0	0.0

TABLE 3A.5-4

THE JOINT LOADS (AT SUPPORTS) FROM STRUDL AND GT-STRUDL COMPUTER RUNS FOR DIFFERENT LOADING CONDITIONS

(Problem #2)

Joint	Loading	STRUDL			GT-STRUDL		
		X FORCE	Y FORCE	Z MOM	X FORCE	Y FORCE	Z MOM
A	1	237.33	1652.25	-7958.75	237.33	1652.25	-7958.75
	2	675.01	-583.75	-104030.94	675.01	-583.75	-104031.48
	3	-1480.59	-694.20	107811.12	-1480.59	-694.20	107811.62
B	1	-171.16	45.28	0.00	-171.16	45.29	0.00
	2	-377.96	286.70	-0.01	-377.96	286.70	0.00
	3	-126.41	2301.20	0.00	-126.41	2301.20	0.00
C	1	-1609.40	1609.40	0.0	-1609.40	1609.40	0.00
	2	-297.05	297.05	0.0	-297.05	297.05	0.00
	3	1607.00	-1607.00	0.0	1607.00	-1607.00	0.00

TABLE 3A.5-5

COMPARISON OF ELEMENT (Randomly Selected) STRESSES (Problem #3)

ELEMENT	NODE	Mxx		Myy		Mxy		Vxx		Vyy	
		STRUDL	GT STRUDL	STRUDL	GT STRUDL	STRUDL	GT STRUDL	STRUDL	GT STRUDL	STRUDL	GT STRUDL
1	1	-12.546	-12.543	-78.334	-78.354	19.545	19.547	13.417	13.412	3.617	3.618
	2	-4.098	-4.109	-18.611	-18.642	19.107	19.109	13.417	13.412	-8.769	-8.756
	10	-13.852	-13.831	-69.887	-69.860	46.293	46.298	-3.549	-3.538	-8.769	-8.756
	9	-8.936	-8.952	-56.769	-56.764	46.730	46.735	-3.549	-3.538	-3.617	3.618
6	6	-2.145	-2.146	-16.671	-16.678	13.396	13.400	-0.595	-0.597	-.158	-.158
	7	-2.876	-2.879	-19.390	-19.409	10.630	10.635	-0.595	-0.597	.916	.924
	15	-3.956	-3.940	-11.933	-11.916	9.555	9.558	0.694	.701	.916	.924
	14	-6.974	-6.986	-12.939	-12.935	12.320	12.323	0.694	.701	-.158	-.158
10	11	-10.414	-10.417	-4.335	-4.336	65.119	65.122	1.027	1.026	-13.922	-13.924
	12	-15.984	-15.983	7.190	7.180	13.283	13.285	1.027	1.026	-9.953	-9.948
	20	-74.387	-74.369	-3.683	-3.672	22.101	22.102	5.789	5.798	-9.953	-9.948
	19	-99.338	-99.357	-12.307	-12.310	73.937	73.939	5.789	5.798	-13.922	-13.924
16	18	3,765	3.747	26.738	26.721	94,743	94.475	-15.849	-15.844	.332	.335
	19	-56.181	-56.179	-5.239	-5.250	71.624	71.624	-15.849	-15.844	-9.083	-9.078
	27	-117.077	-117.062	-7.560	-7.547	49.705	49.704	-27.147	-27.139	-9.083	-9.078
	26	1.424	1.396	31.393	31.403	72.554	72.555	-27.147	-27.139	.332	.335
22	25	-2.301	-2.323	56.058	56.053	74.477	74.481	-3.966	-3.962	-1.428	-1.429
	26	0.446	0.451	33.162	33.148	71.634	71.637	-3.966	-3.962	-2.312	-2.305
	34	-11.675	-11.652	29.188	29.207	49.249	49.251	-5.178	-5.163	-2.312	-2.305
	33	-1.561	-1.594	45.381	45.381	52.093	52.094	-5.178	-5.163	-1.428	-1.429
45	51	-149.275	-149.281	-2.222	-2.223	29.027	29.028	6.244	6.243	0.660	0.658
	52	-113.469	-113.472	-1.815	-1.825	26.628	26.629	6.244	6.243	-1.319	-1.314
	60	-124.368	-124.354	-.097	-.085	23.243	23.244	3.869	3.877	-1.319	-1.314
	59	-144.766	-144.787	-2.139	-2.142	25.642	25.643	3.869	3.877	0.660	0.658

CONTINUED TABLE 3A.5.5

COMPARISON OF ELEMENT (Randomly Selected) STRESSES

		Mxx		Myy		Mxy		Vxx		Vyy	
ELEMENT	NODE	STRUDL	GT STRUDL	STRUDL	GT STRUDL	STRUDL	GT STRUDL	STRUDL	GT STRUDL	STRUDL	GT STRUDL
	59	-144.756	-144.773	-2.047	-2.055	24.031	24.030	3.874	3.879	2.303	2.305
	60	-124.346	-124.339	.0146	.0123	22.659	22.658	3.874	3.879	-2.621	-2.620
52	68	-143.165	-143.158	0.591	.595	24.111	24.111	-2.034	-2.030	-2.621	-2.620
	67	-129.535	-129.548	-1.240	-1.237	25.483	25.482	-2.034	-2.030	2.303	2.305
	68	-130.197	-130.198	2.811	2.810	22.296	22.298	22.805	22.805	-4.389	-4.388
60	69	-20.373	-20.368	25.257	25.246	11.012	11.013	22.805	22.805	5.429	5.433
	77	-19.106	-19.097	61.775	61.791	34.229	34.230	34.586	34.591	5.429	5.433
	76	-157.545	-157.548	-.386	-.384	45.512	45.514	34.586	34.591	-4.389	-4.388
	77	-24.513	-24.517	54.034	54.028	35.968	35.968	14.908	14.911	.413	.414
68	78	11.657	11.666	104.330	104.326	39.457	39.457	14.908	14.911	1.380	1.384
	86	21.811	21.826	103.782	103,797	82.856	82.858	16.069	16.074	1.380	1.384
	85	-24.572	-24.578	56.964	56.973	79.367	79.369	16.069	16.074	.413	.414
	91	-36.410	-36.423	2.194	2.186	30.708	30.708	1.185	1.190	7.587	7.593
80	92	-35.756	-35.747	8.412	8.413	65.545	65.546	1.185	1.190	6.573	6.572
	100	-7.093	-7.089	8.341	8.342	69.355	69.356	.424	.424	6.573	6.572
	99	-4.321	-4.324	3.110	3.117	34.518	34.518	.424	.424	7.587	7.593

TABLE 3A.5-6

COMPARISON OF RESULTANT JOINT DISP. SUPPORTS (Randomly Selected)
(GLOBAL)
(Problem # 3)

JOINT	Z DISPLACEMENT		X-ROTATION		Y ROTATION	
	STRUDL	GT STRUDL	STRUDL	GT STRUDL	STRUDL	GT STRUDL
1	-.0447149	-.0447119	.0004818	.0004817	.0001927	.0001925
5	-.0484141	-.0484130	.0003599	.0003598	.0001907	.0001908
15	-.0483233	-.0483236	.0003430	.0003429	.0002064	.000206
20	-.0427901	-.0427890	.0003791	.0003790	.0002582	.000258
30	-.0435545	-.0435547	.0002832	.0002832	.0002928	.000292
40	-.0442976	-.0442990	.0002317	.0002317	.0002164	.0002163
70	-.0354676	-.0354688	.0002586	.0002586	.0004367	.0004369
100	-.0259728	-.0259738	.0001780	.0001779	.0008314	.0008313
112	-.0426962	-.0426994	-.0001656	-.0001656	.0008937	.000893

TABLE 3A.5-7

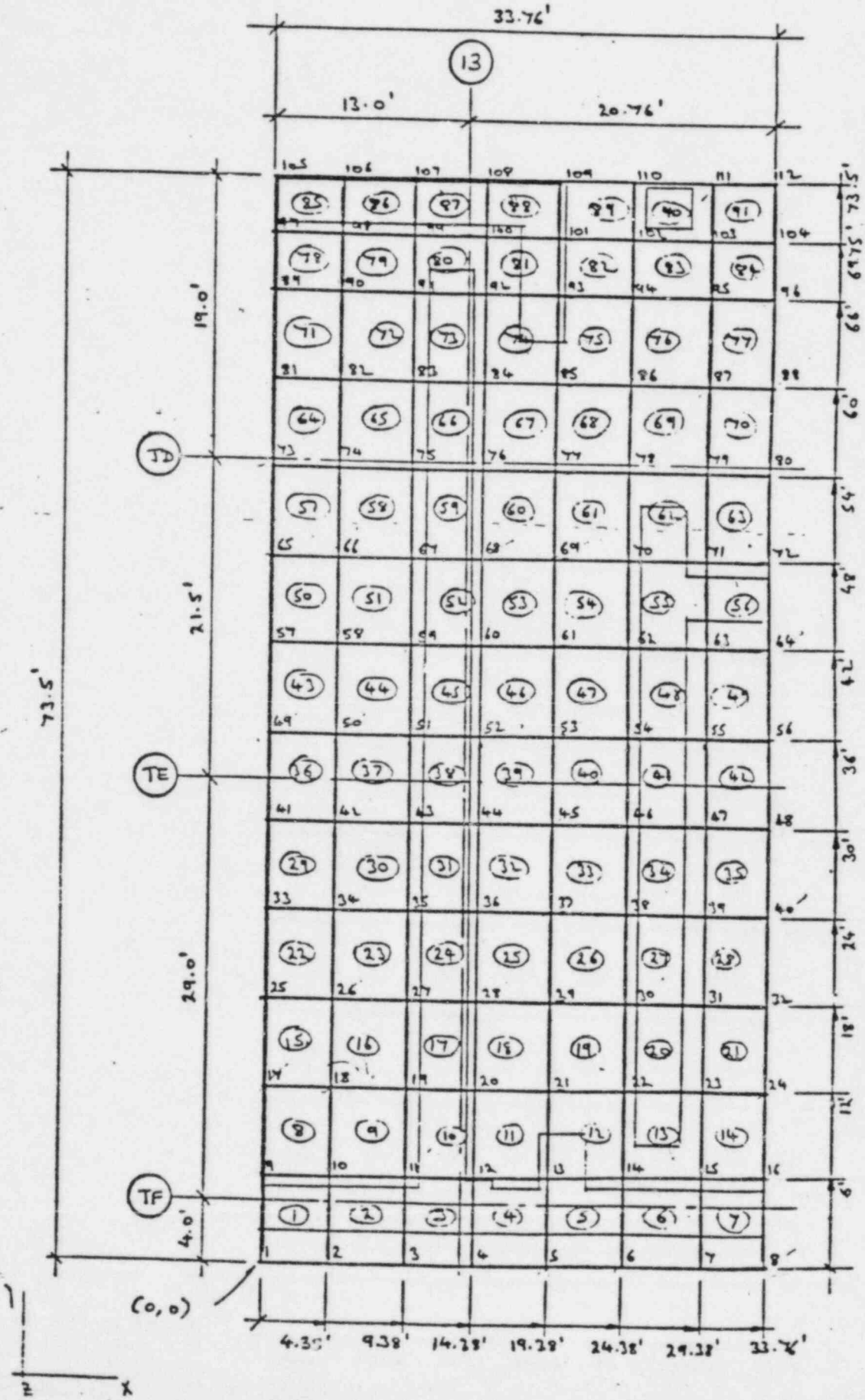
COMPARISON OF MEMBERS (Randomly Selected) FORCES AND MOMENTS (Problem #4)

MEMBER	JOINT	AXIAL		SHEAR Y		SHEAR Z		TORSIONAL MOMENT		BENDING Y MOMENT		BENDING Z MOMENT	
		STRUDL	GTSTRUDL	STRUDL	GTSTRUDL	STRUDL	GTSTRUDL	STRUDL	GTSTRUDL	STRUDL	GTSTRUDL	STRUDL	GTSTRUDI
1	1	0.0	0.0	0.0	0.0	0.527	0.527	0.0	0.0	0.0	0.0	0.0	0.0
	2	0.0	0.0	0.0	0.0	-0.371	-.371	0.0	0.0	-21.567	-21.568	0.0	0.0
4	4	0.0	0.0	0.0	0.0	-.713	-.713	0.0	0.0	5.158	5.158	0.0	0.0
	5	0.0	0.0	0.0	0.0	.869	.869	0.0	0.0	32.818	32.818	0.0	0.0
10	10	0.0	0.0	0.0	0.0	0.226	.226	0.0	0.0	5.627	5.627	0.0	0.0
	11	0.0	0.0	0.0	0.0	-0.070	-.070	0.0	0.0	-12.738	-12.738	0.0	0.0
20	3	0.0	0.0	0.0	0.0	0.158	.158	0.0	0.0	0.0	0.0	0.0	0.0
	23	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	-5.670	-5.670	0.0	0.0
25	14	0.0	0.0	0.0	0.0	.117	.117	0.0	0.0	1.818	1.818	0.0	0.0
	17	0.0	0.0	0.0	0.0	-.093	-.093	0.0	0.0	-4.338	-4.338	0.0	0.0
30	36	0.0	0.0	0.0	0.0	-.093	-.093	0.0	0.0	4.338	4.338	0.0	0.0
	39	0.0	0.0	0.0	0.0	0.117	.117	0.0	0.0	-1.818	-1.818	0.0	0.0
50	12	0.0	0.0	0.0	0.0	.426	.426	0.0	0.0	0.0	0.0	0.0	0.0
	54	0.0	0.0	0.0	0.0	.000	.000	0.0	0.0	-15.336	-15.336	0.0	0.0
60	47	0.0	0.0	0.0	0.0	-.069	-.069	0.0	0.0	12.621	12.621	0.0	0.0
	48	0.0	0.0	0.0	0.0	.225	.225	0.0	0.0	-5.569	-5.569	0.0	0.0

Enclosure 2 (Cont'd.)

FIG. 3A.5-1

FINITE ELEMENT MODEL OF THE FOUNDATION MAT



10 - Joints
10 - Members

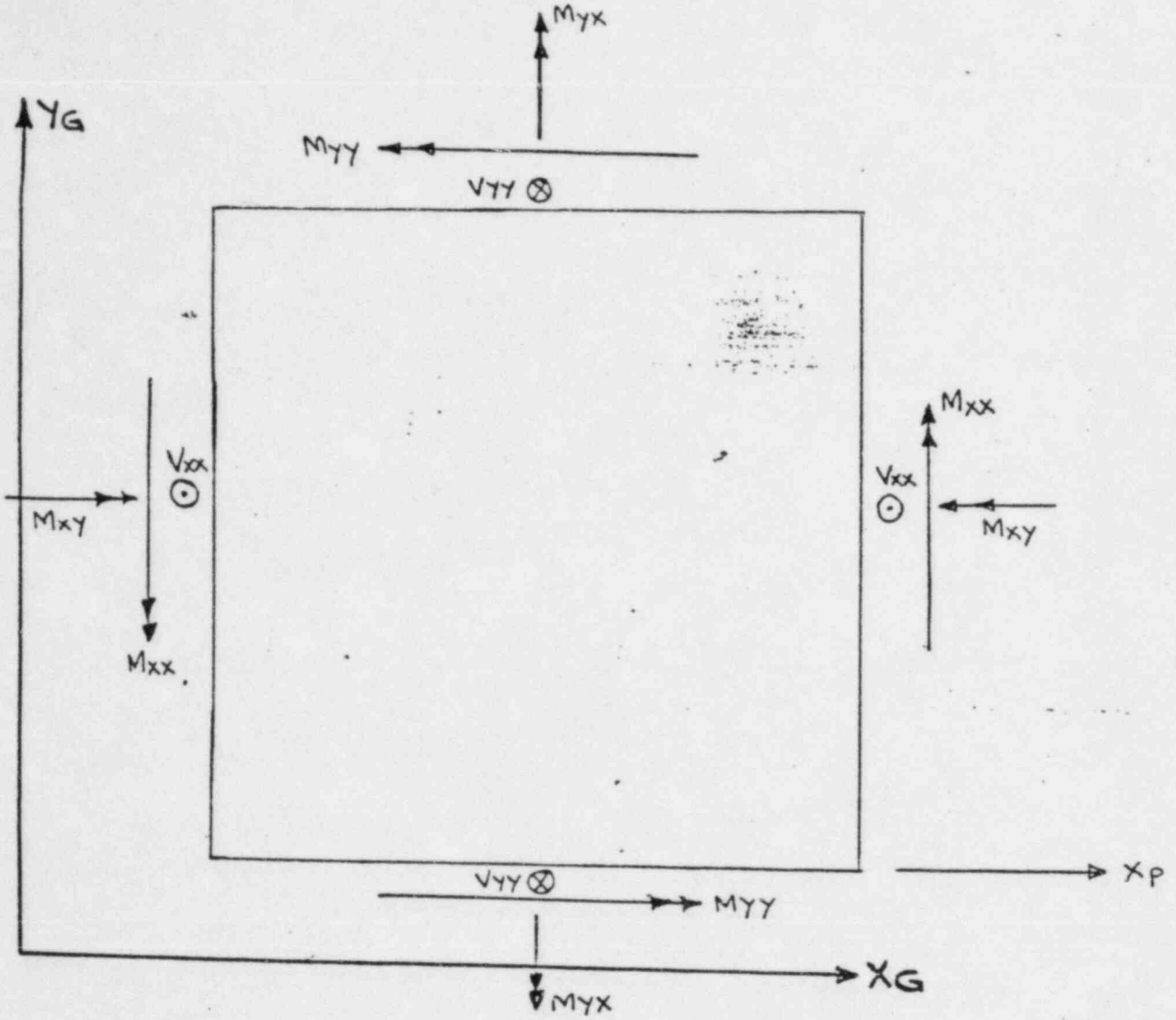


PLATE BENDING

POSITIVE SIGN CONVENTION FOR RESULTS OF PLATE BENDING ELEMENT

- **Mxx, Myy, Mxy, the moment resultants are output at the available locations on the elements.**
- **Vxx, Vyy are also output for the transverse shear resultants.**

. Positive direction coming out of paper

x Positive direction going into the paper

xp,yp - Planer coordinate system

xg,yg - Global coordinate system

xp is parallel to xg

yp is parallel to yg

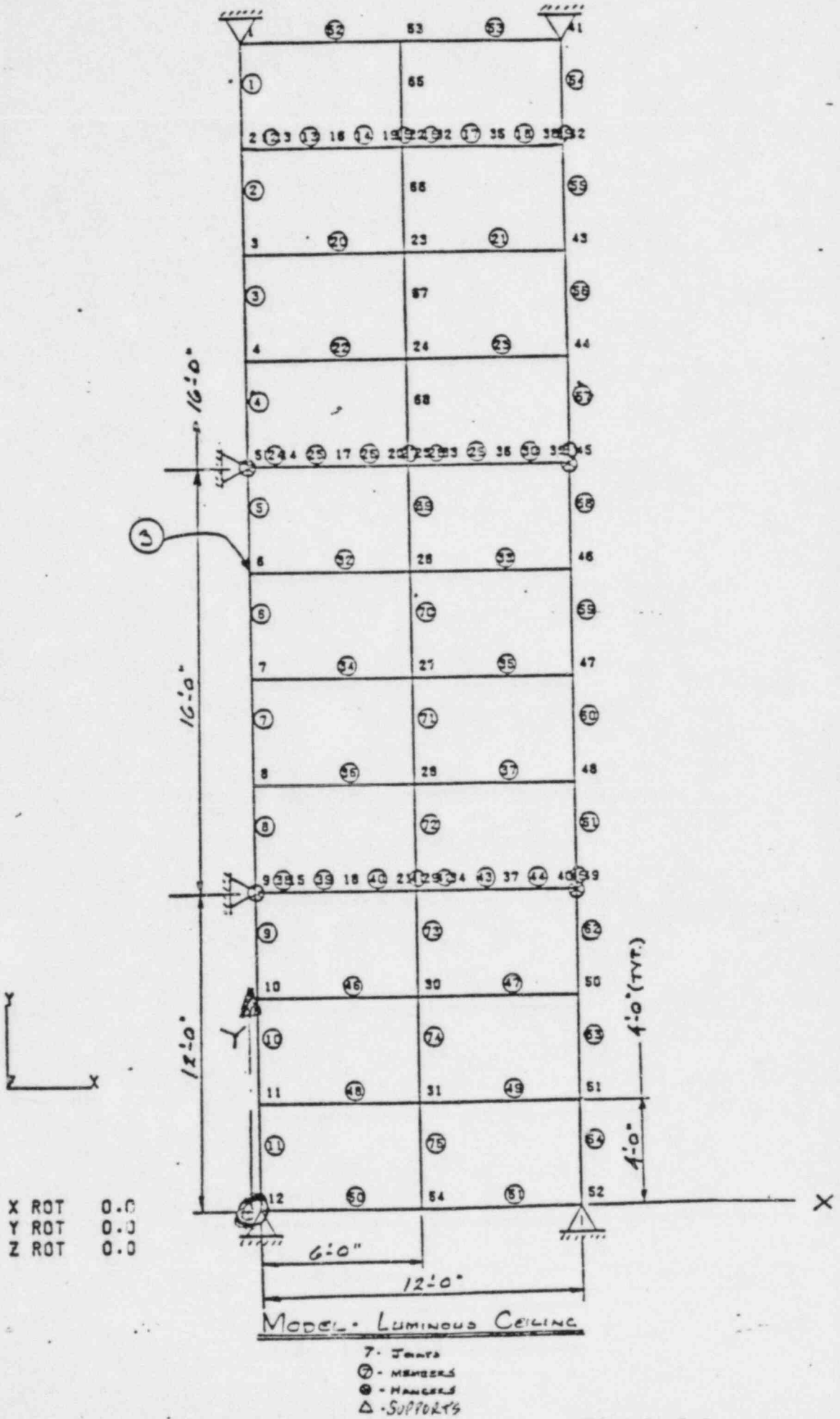

16'-0"
16'-0"
12'-0"
4'-0" (TYP.)
4'-0"
6'-0"
12'-0"
X ROT 0.0
Y ROT 0.0
Z ROT 0.0
MODEL - LUMINOUS CEILING
J - JOINTS
M - MEMBERS
H - HANGERS
S - SUPPORTS

Fig. 3A.5-3

TABLE 3A.5-8

COMPARISON OF RESULTANT JOINT LOADS - SUPPORTS (Problem #4)
(GLOBAL)

JOINT	X FORCE		Y FORCE		Z FORCE		X MOM		Y MOM		Z MOM	
	STRUDL	GTSTRUDL	STRUDL	GTSTRUDL	STRUDL	GTSTRUDL	STRUDL	GTSTRUDL	STRUDL	GTSTRUDL	STRUDL	GTSTRUDL
1	0.0	0.0	0.0	0.0	.953	.953	0.0	0.0	0.0	0.0	0.0	0.0
5	0.0	0.0	0.0	0.0	1.885	1.885	0.0	0.0	0.0	0.0	0.0	0.0
9	0.0	0.0	0.0	0.0	1.536	1.536	0.0	0.0	0.0	0.0	0.0	0.0
12	0.0	0.0	0.0	0.0	.769	.769	0.0	0.0	0.0	0.0	0.0	0.0
41	0.0	0.0	0.0	0.0	.953	.953	0.0	0.0	0.0	0.0	0.0	0.0
45	0.0	0.0	0.0	0.0	1.885	1.885	0.0	0.0	0.0	0.0	0.0	0.0
49	0.0	0.0	0.0	0.0	1.536	1.536	0.0	0.0	0.0	0.0	0.0	0.0
52	0.0	0.0	0.0	0.0	.769	.769	0.0	0.0	0.0	0.0	0.0	0.0

TABLE 3A.5-9

COMPARISON OF EIGEN VALUES, FREQUENCIES AND PERIODS

(Problem #5)

MODE	EIGENVALUE		FREQUENCY (CYCLES/TIME UNIT)		PERIOD (TIME UNIT/CYCLE)	
	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL
1	1.7911230 03	1.7911710+03	6.7357030 00	6.7357910+00	1.4846260-01	1.4846070-01
2	1.8925150 03	1.8925960+03	6.9238160 00	6.9238730+00	1.4442900-01	1.4442780-01
3	1.9256830 03	1.9257150+03	6.9841340 00	6.9841910+00	1.4318170-01	1.4318050-01
4	1.9381170 03	1.9381480+03	7.0056450 00	7.0067020+00	1.4272170-01	1.4272050-01
5	1.9474530 03	1.9474850+03	7.0235010 00	7.0235580+00	1.4237910-01	1.4237800-01
6	1.9490310 03	1.9490630+03	7.0263460 00	7.0264030+00	1.4232150-01	1.4232030-01
7	1.9494540 03	1.9494860+03	7.0271080 00	7.0271660+00	1.4230600-01	1.4230490-01
8	1.9499060 03	1.9499400+03	7.0279270 00	7.0279840+00	1.4228950-01	1.4228830-01
9	2.8949550 03	2.8950240+03	8.5632970 00	8.5634000+00	1.1677750-01	1.1677600-01
10	2.9351170 03	2.9351880+03	8.6224930 00	8.6225970+00	1.1597570-01	1.1597430-01
11	2.9351170 03	2.9351880+03	8.6224930 00	8.6225970+00	1.1597570-01	1.1597430-01
12	3.5667430 03	3.5668360+03	9.5050860 00	9.5052100+00	1.0520680-01	1.0520550-01
13	3.9414380 03	3.9415420+03	9.9918860 00	9.9920180+00	1.0008120-01	1.0007990-01
14	8.7323670 03	8.7326000+03	1.4872570 01	1.4872770+01	6.7237860-03	6.7236960-02
15	1.3457670 04	1.3457930+04	1.8463120 01	1.8463300+01	5.4162040-02	5.4161510-02
16	1.3464190 04	1.3464450+04	1.8467590 01	1.8467770+01	5.4148920-02	5.4148390-02
17	1.3465970 04	1.3466230+04	1.8468810 01	1.8468990+01	5.4145340-02	5.4144810-02
18	2.5732370 04	2.5733140+04	2.5530540 01	2.5530920+01	3.9168770-02	3.9168190-02
19	2.6503930 04	2.6504630+04	2.5910470 01	2.5910810+01	3.8594440-02	3.8593930-02
20	2.7457710 04	2.7458540+04	2.6372560 01	2.6372960+01	3.7918200-02	3.7917630-02
21	3.1771050 04	3.1771670+04	2.8368470 01	2.8368750+01	3.5250400-02	3.5250060-02
22	3.2486990 04	3.2487620+04	2.8686320 01	2.8686600+01	3.4859320-02	3.4859480-02
23	3.5258450 04	3.5259160+04	2.9884890 01	2.9885200+01	3.3461720-02	3.3461380-02
24	3.9539620 04	3.9540660+04	3.1647280 01	3.1647690+01	3.1598290-02	3.1597880-02
25	5.1708760 04	5.1710010+04	3.6191130 01	3.6191570+01	2.7631080-02	2.7630740-02
26	5.5929490 04	5.5930580+04	3.7639220 01	3.7639580+01	2.6568040-02	2.6567780-02
27	5.5929490 04	5.5930580+04	3.7639220 01	3.7639580+01	2.6568040-02	2.6567780-02
28	5.9599630 04	5.9601650+04	3.8854620 01	3.8855210+01	2.5736970-02	2.5736570-02
29	6.5382550 04	6.5384280+04	4.0695640 01	4.0696480+01	2.4572480-02	2.4572150-02
30	8.2065290 04	8.2067580+04	4.5593180 01	4.5593620+01	2.1933110-02	2.1932800-02

TABLE 3A.5-10

COMPARISON OF EIGENVECTORS
FOR FEW RANDOMLY SELECTED MODES AND JOINTS

(Problem #5)

EIGENVECTORS (GLOBAL)

MODE	JOINT	X-DISPL		Z-DISPL	
		STRUDL	GT STRUDL	STRUDL	GT STRUDL
1	1	0.0	0.0	0.0	0.0
	2	0.721	-.721	0.000	0.000
	13	0.721	-.721	0.000	0.000
	23	1.00	-1.00	0.000	0.000
	43	0.999	-0.999	0.000	0.000
6	1	0.0	0.0	0.0	0.0
	2	0.000	0.000	-.0009	-.0009
	13	0.000	0.000	-.0014	-.0014
	23	0.000	0.000	-.567	-.567
	43	0.000	0.000	-.0005	-.0005
3	1	0.0	0.0	0.0	0.0
	2	0.000	0.000	.0095	.0095
	13	0.000	0.000	.0153	.0153
	23	0.000	0.000	1.000	1.000
	43	0.000	0.000	.013	.013

* Y DISPL. ARE APPROXIMATELY 0.000 FOR ALL THREE MODES

TABLE 3A.5-11

COMPARISON OF JOINT DISPLACEMENTS AT THE FREE JOINTS FOR RANDOMLY SELECTED JOINTS (GLOBAL)

(Problem #5)

JOINTS	RESPONSE TYPE	X-DISPL.		Y-DISPL.		Z-DISPL.	
		STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL
7	RMS	0.000	0.000	0.000	.000	.00099	.00099
	ABS SUM	0.000	0.000	.000	.000	.00213	.00212
	CSM	0.000	0.000	.000	.000	.00117	.00116
15	RMS	0.000	0.000	0.000	0.000	.0105	.0104
	ABS SUM	0.000	0.000	0.000	0.000	.0105	.0104
	CSM	0.000	0.000	0.000	0.000	.0105	.0104
22	RMS	0.000	0.000	0.000	0.000	.0424	.0424
	ABS SUM	0.000	0.000	0.000	0.000	.0449	.0449
	CSM	0.000	0.000	0.000	0.000	.0424	.0424
36	RMS	0.000	0.000	0.000	0.000	.0287	.0288
	ABS SUM	0.000	0.000	0.000	0.000	.0287	.292
	CSM	0.000	0.000	0.000	0.000	.0287	.292

RMS = Root Mean Square
ABS SUM = Absolute Sum
CSM = Closely Spaced Mode

3A.11 STARDYNE

The STARDYNE Structural Analysis System, written by Mechanics Research, Inc., of Los Angeles, California, is a fully warranted and documented computer program available at Control Data Corporation.

The MRI STARDYNE Analysis System consists of a series of compatible, digital computer programs designed to analyze linear and nonlinear elastic structural models. The system encompasses the full range of static and dynamic analyses.

The static capability includes the computation of structural deformations and member loads and stresses caused by an arbitrary set of thermal, nodal-applied loads, and prescribed displacements.

Utilizing the normal mode technique, linear dynamic response analyses can be performed for a wide range of loading conditions, including transient, steady-state harmonic, random, and shock spectra excitation types. Dynamic response results can be presented as structural deformations and internal member loads.

The nonlinear dynamic analysis program is integrated in the rest of the STARDYNE system. The equations of motion for the linear portion of the structural model are generated and modified to account for the nonlinear springs. The resulting nonlinear equations of motion are directly integrated, using either the Newmark or Wilson implicit integration operators. The user may enter sets of structural loadings, which vary with time, and specify time points at which the program is to output the structural response.

INSERT "A"

INSERT A

This computer program is considered verified by constant use and by the vendor's original documentation and qualification.

3A.13 LION

LION is a digital computer program which is used to solve three-dimensional transient and steady-state temperature distribution problems. The program may also consider subcooled nucleate boiling and coolant heat transfer effects. The surface conditions may be forced convection, free convection, or radiation and heat may be externally or internally generated. Input to the program consists of structural geometry, physical properties, boundary conditions, internal heat generation rates, coolant flow properties, and flow rates.

The program solves the transient heat conduction equations for a three-dimensional field using a first forward difference method. To ensure the temperature calculation stability, LION can determine the suitable time increment, if the specified input time increment is too large.

Since the original program was developed, subsequent versions have evolved to solve larger and more complex problems[1,2,3,4,5,6].

LION is a recognized program in the public domain and has been used extensively.

INSERT "A"

RBS FSAR

3A.15 NUPIPE

NUPIPE was developed by the Nuclear Services Corporation and is fully documented. The S&W version of NUPIPE used differs slightly from the public domain program NUPIPE in the postprocessing of the analytical results.

NUPIPE performs a linear elastic analysis of three-dimensional piping systems subjected to thermal, static,and dynamic loads. It utilizes the finite-element method of analysis with special features incorporated to accommodate specific requirements in piping analysis. In addition, it checks analytical conformance to ASME Section III and ANSI B31.1.0. This program accepts the complete geometric and physical description of the piping system, provides a complete error and coordinate check for the inputs, and computes internal forces and moments, support and equipment reactions, and displacements and stress values for a variety of loading cases .

NUPIPE has been verified with ADLPIPE[3] for thermal, weight, and response spectrum seismic analyses. The results from both programs are presented in Tables 3A.15-1 through 3A.15-7. The model used for this comparison is presented in Fig. 3A.15-1.

The comparison is also made with ASME Benchmark solution for force time-history dynamic response[2]. The model used for this comparison is shown on Fig. 3A.15-2. The results for comparisons are presented in the form of plots on Fig. 3A.15-2. The natural frequencies are given in Table 3A.15-8.

The Class 1 piping stress conforms with the hand calculations. The model used is shown on Fig. 3A.15-3. The results are tabulated in Tables 3A.15-9 and 3A.15-10.

In addition, NUPIPE has been verified in accordance with NRC IE Bulletin 79-07, which was accepted by the NRC in Reference 4. Finally, NUPIPE has been verified using problem numbers 1,2,4, and 7 of NUREG/CR-1677.

RBS FSAR

References - Section 3A.15

1. ASME Boiler and Pressure Vessel Code - Section III - Nuclear Power Plant Components. American Society of Mechanical Engineers, 1971; Including Summer 1972 Addenda.

2. American Society of Mechanical Engineers. Pressure Vessel and Piping; 1972 Computer Programs Verification. Problem No. 5.

3. Arthur D. Little, Inc. ADLPIPE: Static, Dynamic, Thermal Pipe Stress Analysis.

4. Letter dated Dec. 18, 1979, L.C.Shao (NRC) to Stone & Webster Engineering Corp. (Attn. W. G. White, Jr.).

3A.25 ANSYS

The ANSYS engineering analysis system, developed by the Swanson Analysis System, Inc., is a fully warranted and documented computer program available at Control Data Corporation's 6600 data centers.

The ANSYS computer program, which has been used for production analysis since early 1970, is a large-scale, general-purpose computer program for the solution of several classes of engineering analysis problems. Analysis capabilities include: static and dynamic; plastic, creep, and swelling; small and large deflections; steady-state and transient heat transfer; and steady-state fluid flow.

The matrix displacement method of analysis, based upon finite element idealization, is employed throughout the program. The library of finite elements available contains more than 30 elements for static and dynamic analyses and more than 10 for heat transfer and fluid flow analyses. This variety of elements gives the ANSYS program the capability of analyzing frame structures (two-dimensional frames, grids, and three-dimensional frames), piping systems, two-dimensional plane and axisymmetric solids, flat plates, three-dimensional solids, axisymmetric and three-dimensional shells and nonlinear problems, including interfaces and cables.

Loading on the structure may be forces, displacements, pressures, temperatures, or response spectra. Loadings may be arbitrary time functions for linear and nonlinear dynamic analyses. Loadings for heat transfer analyses include: internal heat generation, convection, and radiation boundaries and specified temperatures or heat flows.

This computer program is considered verified by constant use and by the vendor's original documentation and qualification.

QUESTION 210.62 (3.9.2)

The discussion of the preoperational testing program does not discuss the acceptance limits for steady state and transient vibrations. What criteria will be used in developing these limits. If a stress limit will be used, what basis will be used to determine the actual stress from the measured values? Please provide a list of flow transients and a list of selected locations for visual inspections and measuring devices.

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 start-up 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.

The essential instrumentation lines to be inspected should include (but are not limited to) the following:

a. Reactor pressure vessel level indicator instrumentation lines (used for monitoring both steam and water levels).

b. Main steam instrumentation lines for monitoring main steam flow (used to actuate main steam isolation valves during high steam flow).

c. Reactor core isolation cooling (RCIC) instrumentation lines on the RCIC steam line outside containment (used to monitor high steam flow and actuate isolation).

d. Control rod drive lines inside containment (not normally pressurized but required for scram).

RESPONSE

The response to this request for the NSSS scope is provided in revised Sections 3.9.2.1.4.3B and 3.9.2.1.6B, and in Section 3.9.2.1.1.3B.

The response to this request for the BOP scope ~~will be~~ is provided ~~by December 1983.~~ in Section 3.9.2.1.2A.

3.9.2.1.1A Flow Modes

Tabulated flow modes for various systems are provided as part of the above test program.

3.9.2.1.2A Preoperational Vibration Testing

Safety-related piping systems designated as Class 1, 2, or 3 are designed in accordance with ASME Section III. Each system is designed to withstand dynamic loadings from operational transient conditions that are encountered during expected service as required by NB-3622, NC-3622, and ND-3622 of the code.

During the preoperational test program, vibration testing is performed using a portable vibration meter on the following high energy systems located in Seismic Category I structures:

1. Reactor recirculation system
2. Residual heat removal system
3. High pressure core spray system
4. Low pressure core spray system
5. Reactor core isolation cooling system
6. Feedwater system
7. Condensate system
8. Other piping systems which have exhibited significant vibration response based upon past operating experiences with similar systems or similar system operating conditions.

Vibration measurements are conducted for steady-state and transient conditions such as pump starts and valve operation. Also, visual inspections are performed on these systems with emphasis placed on vents, drains, and branch piping. Determining vibration response is the objective of these visual inspections.

INSERT →

3.9.2.1.3A Preoperational Thermal Expansion Testing

Preoperational tests for BWRs are conducted near ambient conditions; therefore, thermal expansion testing during the preoperational test phase is very limited.

BOP steady-state vibration testing consists of two separate testing phases. Phase I testing consists of visual screening of ASME Class 1, 2, and 3, and selected high and moderate energy piping systems at preselected locations. A specific list of monitored systems and locations at which visual observations will be made will be contained in start-up test and preoperational test procedure. These vibration visual observations are performed by engineers trained for excessive vibration screening. Any piping system viewed from the recommended observation distance which does not exhibit excessive vibration, is considered acceptable.

Calculations for observation distances (eg, displacement) are based upon deflection equations given in the ANSI/ASME OM-3 Requirements for Preoperational and Initial Start-Up Vibration Testing of Nuclear Power Plant Piping Systems with an allowable stress of 10,000 psi and using $(0.8)S_{el}$ for carbon steel piping and $(0.6)S_{el}$ for stainless steel piping.

Phase II testing is performed on piping systems which exhibit excessive vibration during the visual screening. Phase II testing consists of taking a velocity and/or displacement reading using hand-held vibration monitors. Criteria for displacement measurements are based on the ANSI/ASME OM-3 with assumptions as stated above.

BOP transient vibration testing is performed at preselected data points. Two levels of acceptance criteria, Level I and Level II are imposed. Level I and II criteria are defined per RBS FSAR Sections 3.9.2.1.4.1B and 3.9.2.1.4.2B. Acceptance limits for Level I are based upon the ASME Boiler and Pressure Vessel Code Section III Equation 9 for Category I, II and III systems or the ANSI B31.1 Equation 12 for Category IV systems. Acceptance limits restrict the bending stress due to deflection plus stresses due to deadweight and pressure to a value less than the normal/upset allowable stress for occassional loads. Level II criteria are based on pipe stress and support loads not to exceed design basis predictions. Flow transients monitored are pump starts, pump stops, changes to system flows due to rapid valve position changes, and FSAR designated system trips. A specific list of flow transients is contained in start-up test and preoperational test procedures.

Small Bore Pipe Testing Small bore piping branch connections on systems monitored for steady-state vibration is included as part of the visual observations. Control rod drive lines are instrumented for transient vibration with level 1 and level 2 acceptance limits as defined above.

Insert (cont'd.)

Essential instrument lines chosen for additional monitoring points are instrumented or visually examined for vibration. Acceptance criteria limits for these test points are based upon Equation 9 of the ASME Boiler and Pressure Vessel Code. The reactor pressure vessel level indicator instrumentation lines for moritoring both steam and water levels, main steam instrumentation lines for monitoring main steam flow, reactor core isolation cooling (RCIC) instrumentation lines on the RCIC steam line outside containment and instrumentation lines and systems identified in the INPO SER 64-83, "Fatigue Cracks and Leaks in Small Diameter Piping", dated September 12, 1983, are possible additional monitoring points for excessive vibration.

RBS FSAR

QUESTION 210.98 (3.9)

The River Bend plant references Kuosheng I as the prototype for the BWR/6 - 218" reactor. The staff has not yet received the reactor internals vibration test program for Kuosheng I. The Kuosheng test program must be reviewed and approved by the staff prior to its use as a prototypical program applicable to River Bend.

RESPONSE

BWR/4 and BWR/5 reactors now in service provide the basis for the prediction of vibrational behavior of reactor internals during the component design stage. GE Licensing Topical Report NEDE-24057 presents results of such tests and measurements. In addition, confirmation test data has been obtained from the Kuosheng test program and is contained in NEDE-22146. Kuosheng is the first 218-inch BWR/6 to become operational and is the prototype for River Bend Station. LRG-II position 2-MEB discusses the reactor pressure vessel internals vibration test program for BWR/6's.

of-a-kind reactor internals configuration*, in accordance with guidelines of Regulatory Guide 1.20 for prototype reactor internals. These programs are conducted in the prototype plants in three phases, described as follows:

1. Preoperational Tests Prior to Fuel Loading. Steady-state test conditions include balanced (two-pump) recirculation system operation and unbalanced (single-pump) operation, over the full range of flow rates up to rated flow. Transient flow conditions include single and two-pump trips from rated flow. The specified test duration is to be 35 hr of balanced operation, plus 14 hr of single-pump operation of each recirculation loop, for a total of 63 hr. The major components are subjected to a minimum of 10^6 cycles of vibration at the anticipated dominant response frequency and at the maximum response amplitudes. Vibration measurements are obtained during this test, and a close visual inspection of internals is conducted before and after the test.

2. Precritical Testing with Fuel. This vibration measurement series is conducted with the reactor assembly complete but prior to reactor criticality. Flow conditions include balanced, unbalanced, and transient conditions as for the first test series. This test series verifies the anticipated effect of the fuel on the vibration response of internals. Previous vibration measurements in BWRs* have shown that the fuel adds damping and reduces vibration amplitudes of major internal structures. Thus, the first test series (without fuel) is a conservative evaluation of the vibration levels of these structures.

3. Initial Startup Testing. Vibration measurements are made during reactor startup at conditions up to 100 percent rated flow and power. Balanced, unbalanced, and transient conditions of recirculation system operation are evaluated. The primary purpose of this test series is to verify

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*NEDE-24057-P (Class III) and NEDO-24057 (Class I), Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants, November 1977. Also, NEDO-24057-1-P (Ammendment No. 1 dated December 1978) and NEDE 24057-2-P (Amendment No. 2 dated June 1979.) NEDE-22146 (Class III), Kuosheng-1 Reactor Intervals Vibration Measurements, July 1982.

the anticipated effect of two-phase flow on the vibration response of internals. Previous vibration measurements in BWRs* have shown that the effect of the two-phase flow is to broaden the frequency-response spectrum and diminish the maximum response amplitude of the shroud and core support structures.

Vibration sensor types include strain gages, displacement sensors (linear variable transformers), and accelerometers. Accelerometers are provided with double integration signal conditioning to give a displacement output. Sensor locations include the following:

Top of shroud head, lateral acceleration (displacement)
Top of shroud, lateral displacement
Jet pump riser braces, bending and extension strains
Jet pump diffuser, bending strain
Control rod drive housings, bending strain
Incore housings, bending strain
Core spray internal piping, bending strain

In addition to the above components, vibration of the core spray sparger is measured during preoperational testing of that system at the designated prototype 218 size BWR/6 plant. In all prototype plant vibration measurements, only the dynamic component of strain or displacement is recorded. Data are recorded on magnetic tape, and provision is made for selective on-line analysis to verify the overall quality and level of the data. Interpretation of the data requires identification of the dominant vibration modes of each component by the test engineer, using frequency, phase, and amplitude information from the component dynamic analyses. Comparison of measured vibration amplitudes to predicted and allowable amplitudes is then made on the basis of the analytically obtained normal mode which best approximates the observed mode.

The visual inspections conducted prior to and following preoperational testing are for the purpose of detecting evidence of vibration, wear, or loose parts. At the completion of preoperational testing, the reactor vessel

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*NEDE-24057-P (Class III) and NEDO-24057 (Class I), Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants, November 1977. Also, NEDO-24057-1-P (Ammendment No. 1 dated December 1978) and NEDE 24057-2-P (Amendment No. 2 dated June 1979.) NEDE-22146 (Class III), Kuosheng-1 Reactor Intervals Vibration Measurements, July 1982.

QUESTION 210.5 (3.9A)

Provide the missing information mentioned in Tables 3.9A-5, 3.9A-10, and 3.9A-11 or provide a schedule for submittal of the information.

RESPONSE

The information required to complete Tables 3.9A-5, 3.9A-10, and 3.9A-11 ~~is not available. Those qualification programs are ongoing and the information will be provided as it becomes available.~~ will be provided during July 1984.

RBS FSAR

QUESTION 210.15 (3.9.3.3A)

Provide the missing analyses as mentioned in the FSAR or provide a schedule for submittal of this information.

RESPONSE

~~The results of these analyses will be provided in 1983.~~

In reviewing Section 3.9.3.3A, no mention of missing analyses is found.

QUESTION 210.43 (3.6A)

Tables 3.6A-1 through 3.6A-11 - Please provide a schedule for completion of the stress analyses and updating of these tables. Will cumulative usage factors be limited to less than 1.0?

RESPONSE

The stress analyses is updated in Tables 3.6A-1 through 3.6A-11, ~~will be completed by December 1983.~~ As noted in revised Section 3.6.2.5A, cumulative usage factors are limited to less than 1.0. The analysis is scheduled for completion in the third quarter 1984.

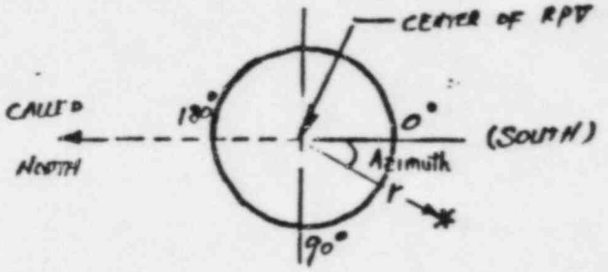
RBS FSAR

TABLE 3.6A-1

SUMMARY OF STRESSES IN HIGH-ENERGY ASME CLASS 1 PIPING

MAIN STEAM SYSTEM - INSIDE CONTAINMENT - LOOP A

Break Point	Location: Elevation (ft-in)	Location: Azimuth (deg)	Location: r (ft-in)	Maximum Stress Range(1): Eq. 12 (psi)	Maximum Stress Range(1): Eq. 13 (psi)	Cumulative Usage Factor(1)	Pipe Break Stress Limit 2.4 S_m (psi)	Description of Break Points	Break Type
1A	155-0 1/2	72	11-10 1/2	14,800	18,740	0.0068	42,210	TP	CB
5A	155-0 1/2	72	14-6	57,150	20,950	0.5832	42,210	IP (Elbow)	CB & LB
10A	152-0 1/2	72	17-6	46,420	19,110	0.2699	42,210	IP (Elbow)	CB & LB
25A	132-4 3/4	72	17-6	10,580	35,910	0.0972	42,210	IP (Elbow)	CB & LB
30A	129-4 3/4	61	17-9	14,560	37,760	0.1098	42,210	IP (Elbow)	CB & LB
38A	130-10 3/4	51	19-0	19,900	32,070	0.4632	42,210	TP	CB
42A	130-10	7	20-0	18,430	31,710	0.3409	42,210	TP	CB
60A	129-0 7/16	[illegible]	25-0	7,190	19,110	0.0023	42,210	TP	CB
38A-1	130-5 1/2	51	19-0	30,350	48,750	0.862	42,210	TP	CB
42A-1	130-4 3/4	34.5	20-0	28,220	47,680	0.625	42,210	TP	CB



Key: IP = Intermediate point
TP = Terminal point
CB = Circumferential break
LB = Longitudinal break

(1) Stresses were calculated in accordance with equations 12 and 13 of ASME Section III, paragraphs NB-3653.6(a) and NB-3653.6(b), respectively. Cumulative usage factors were calculated in accordance with ASME Section III, subarticle NB-3650.

NOTES: See Fig. 3.6A-12 for break locations.

The data presented in this table were used in conjunction with Section 3.6.2A to determine the break locations shown in Fig. 3.6A-12.

The data in this table are preliminary for main steam Loop A, but have been provided as a typical example of a completed stress summary, including notes, for Class 1 high-energy piping. This table will be updated as required upon completion of the Class 1 analysis.

Reference: 12210-AX-2G-1

TABLE 3.6A-2

SUMMARY OF STRESSES IN HIGH-ENERGY ASME CLASS 1 PIPING

MAIN STEAM SYSTEM - INSIDE CONTAINMENT - LOOP B

Break Point	Location: Elevation (ft-in)	Location: Azimuth (deg)	Location: r (ft-in)	Maximum Stress Range(1): Eq. 12 (psi)	Maximum Stress Range(1): Eq. 13 (psi)	Cumulative Usage Factor(1)	Pipe Break Stress Limit 2.4 S_m (psi)	Description of Break Points	Break Type
1B	155-0 1/2	252	11-10 1/2	12,010	20,770	0.0053	42,210	TP	CB
5B	155-0 1/2	252	14-6	45,730	22,850	0.0455	42,210	IP (Elbow)	CB & LB
30B	132-1 19/64	256	22-0	12,940	36,950	0.1059	42,210	IP (Elbow)	CB & LB
35B	129-6 1/8	264.5	23-8	15,330	40,840	0.1157	42,210	IP (Elbow)	CB & LB
37B	131-1 1/4	272	23-6	23,470	27,750	0.8866	42,210	TP	CB
38B	131-0 1/4	278	23-9	20,390	27,610	0.5175	42,210	TP	CB
39B	130-11 3/4	284	24-4	18,360	27,830	0.3403	42,210	TP	CB
47B	130-11 1/8	297	25-6	24,130	27,340	0.9763	42,210	TP	CB
48B	130-11	302	25-5	-	-	-	42,210	TP	CB
70B	129-0 5/8	337	28-0	7,690	19,690	0.0019	42,210	TP	CB
37B-1	130-8	272	23-6	36,190	42,380	2.106	42,210	TP	CB
38B-1	130-7	278	23-9	31,410	38,540	1.449	42,210	TP	CB
39B-1	130-6 1/2	284	24-4	21,180	36,270	1.193	42,210	TP	CB
47B-1	130-5 7/8	297	25-6	36,940	42,930	2.234	42,210	TP	CB
48B-1	130-5 3/4	302	25-5	41,410	46,360	3.189	42,210	TP	CB

Key: IP = Intermediate point
TP = Terminal point
CB = Circumferential break
LB = Longitudinal break

(1) Stresses were calculated in accordance with equations 12 and 13 of ASME Section III, paragraphs NB-3653.6(a) and NB-3653.6(b), respectively. Cumulative usage factors were calculated in accordance with ASME Section III, subarticle NB-3650.

NOTES: See Fig. 3.6A-14 for break locations.

The data presented in this table were used in conjunction with Section 3.6.2A to determine the break locations shown in Fig. 3.6A-14. The reference coordinate system is shown in Table 3.6A-1.

The data in this table are preliminary for Main Steam-Loop B, but have been provided as a typical example of a completed stress summary, including notes, for Class 1 high-energy piping. This table will be updated as required upon completion of the Class 1 analysis.

Reference: 12210-AX-2F-1

TABLE 3.6A-3

SUMMARY OF STRESSES IN HIGH-ENERGY ASME CLASS 1 PIPING

MAIN STEAM SYSTEM - INSIDE CONTAINMENT - LOOP C

Break Point	Location Elevation (ft-in)	Azimuth (deg)	r (ft-in)	Maximum Stress Range[1] EQ.12 (psi)	EQ.13 (psi)	Cumulative Usage Factor[1]	Pipe Break Stress Limit 2.4 Sm (psi)	Description of Break Points	Break Type	
1C	155-0 1/2	108	11-10 1/2	12,010	20,770	0.0053	42,210	TP	CB	1.47
5C	155-0 1/2	108	14-6	45,730	22,850	0.0455	42,210	IP (Elbow)	CB & LB	1.48
30C	132-1 19/64	104	22-0	12,940	36,950	0.1059	42,210	IP (Elbow)	CB & LB	1.49
35C	129-6 1/8	95.5	23-8	15,330	40,840	0.1157	42,210	IP (Elbow)	CB & LB	1.50
37C	131-1 1/4	88.1	23-6	23,470	27,750	0.8866	42,210	TP	CB	1.51
38C	131-0 1/4	82.1	23-9	20,390	27,610	0.5175	42,210	TP	CB	1.52
39C	130-11 3/4	76.2	24-4	18,360	27,830	0.3403	42,210	TP	CB	1.53
40C	130-11 1/2	70.6	25-0	-	-	-	42,210	TP	CB	1.54
47C	130-11 1/8	63	25-6	24,130	27,340	0.9763	42,210	TP	CB	1.55
48C	130-11	58	25-5	-	-	-	42,210	TP	CB	1.56
70C	129-0 5/8	23	28-0	7,690	19,690	0.0019	42,210	TP	CB	1.57
37C-1	130-8	88.1	23-6	36,190	42,380	2.106	42,210	TP	CB	1.58
38C-1	130-7	82.2	23-9	31,410	38,590	1.449	42,210	TP	CB	2.1
39C-1	130-6 1/2	76	24-4	21,180	36,270	1.193	42,210	TP	CB	2.2
40C-1	130-6 1/4	71	25-0	-	-	-	42,210	TP	CB	2.3

Key: IP - Intermediate point
TP - Terminal point
CB - Circumferential break
LB - Longitudinal break

(1) Stresses were calculated in accordance with equations 12 and 13 of ASME Section III, paragraphs NB-3653.6(a) and NB-3653.6(b), respectively. Cumulative usage factors were calculated in accordance with ASME Section III, subarticle NB-3650.

NOTES: See Fig. 3.6A-13 for break locations.

The data presented in this table were used in conjunction with Section 3.6.2A to determine the break locations shown in Fig. 3.6A-13. The reference coordinate system is shown in Table 3.6A-1.

The data in this table are preliminary for Main Steam-Loop C, but have been provided as a typical example of a completed stress summary, including notes, for Class 1 high-energy piping. This table will be updated as required upon completion of the Class 1 analysis.

Reference: 12210-AX-2F-1

TABLE 3.6A-3

SUMMARY OF STRESSES IN HIGH-ENERGY ASME CLASS 1 PIPING

MAIN STEAM SYSTEM - INSIDE CONTAINMENT - LOOP C

Break Point	Location: Elevation (ft-in)	Location: Azimuth (deg)	Location: r (ft-in)	Maximum Stress Range[1]: EQ.12 (psi)	Maximum Stress Range[1]: EQ.13 (psi)	Cumulative Usage Factor[1]	Pipe Break Stress Limit 2.4 Sm (psi)	Description of Break Points	Break Type	
47C-1	130-5 7/8	63	25-6	36,940	42,930	2.234	42,210	TP	CB	2.4
48C-1	130-5 3/4	58	25-5	41,410	46,360	3.189	42,210	TP	CB	2.5

Key: IP - Intermediate point
TP - Terminal point
CB - Circumferential break
LB - Longitudinal break

[1]Stresses were calculated in accordance with equations 12 and 13 of ASME Section III, paragraphs NB-3653.6(a) and NB-3653.6(b), respectively. Cumulative usage factors were calculated in accordance with ASME Section III, subarticle NB-3650.

NOTES: See Fig. 3.6A-13 for break locations.

The data presented in this table were used in conjunction with Section 3.6.2A to determine the break locations shown in Fig. 3.6A-13. The reference coordinate system is shown in Table 3.6A-1.

The data in this table are preliminary for Main Steam-Loop C, but have been provided as a typical example of a completed stress summary, including notes, for Class 1 high-energy piping. This table will be updated as required upon completion of the Class 1 analysis.

Reference: 12210-AX-2F-1

TABLE 3.6A-4

SUMMARY OF STRESSES IN HIGH-ENERGY ASME CLASS 1 PIPING

MAIN STEAM SYSTEM-INSIDE CONTAINMENT-LOOP D

Break Point	Location: Elevation (ft-in)	Location: Azimuth (deg)	Location: r (ft-in)	Maximum Stress Range(1): Eq. 12 (psi)	Maximum Stress Range(1): Eq. 13 (psi)	Cumulative Usage Factor(1)	Pipe Break Stress Limit 2.4 Sm (psi)	Description of Break Points	Break Type
1D	155-0 1/2	288	11-10 1/2	14,800	18,740	0.0068	42,210	TP	CB
5D	155-0 1/2	288	14-6	57,150	20,950	0.5832	42,210	IP (Elbow)	CB & LB
10D	152-0 1/2	288	17-6	46,420	19,110	0.2699	42,210	IP (Elbow)	CB & LB
25D	132-0	288	17-6	10,580	35,910	0.0972	42,210	IP (Elbow)	CB & LB
30D	129-4 3/4	297.5	17-9	14,560	37,760	0.1098	42,210	IP (Elbow)	CB & LB
38D	130-10 3/4	309	19-0	19,900	32,070	0.4632	42,210	TP	CB
39D	130-10 1/4	316	20-0	-	-	-	42,210	TP	CB
42D	130-10	325.5	19-6	18,430	31,710	0.3409	42,210	TP	CB
60D	126-0 7/16	352	25-0	7,190	19,110	0.0023	42,210	TP	CB
38D-1	130-5 1/2	309	19-0	30,350	48,750	0.862	42,210	TP	CB
42D-1	130-5	325.5	19-6	28,220	47.680	0.625	42,210	TP	CB
39D-1	130-5	316	20-0	-	-	-	42,210	TP	CB

Key: IP = Intermediate point
TP = Terminal point
CB = Circumferential break
LB = Longitudinal break

(1) Stresses were calculated in accordance with equations 12 and 13 of ASME Section III, paragraphs NB-3653.6(a) and NB-3653.6(b), respectively. Cumulative usage factors were calculated in accordance with ASME Section III, subarticle NB-3650.

NOTES: See Fig. 3.6A-14 for break locations.

The data presented in this table were used in conjunction with Section 3.6.2A to determine the break locations shown in Fig. 3.6A-14. The reference coordinate system is shown in Table 3.6A-1.

The data in this table are preliminary for Main Steam-Loop D, but have been provided as a typical example of a completed stress summary, including notes, for Class 1 high-energy piping. This table will be updated as required upon completion of the Class 1 analysis.

Reference: 12210-AX-2G-1

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TABLE 3.6A-4a

SUMMARY OF STRESSES IN HIGH-ENERGY ASME CLASS 1 PIPING

MAIN STEAM VENT SYSTEM

Break Point	Location: Elevation (ft-in)	Location: Azimuth (deg)	Location: r (ft-in)	Maximum Stress Range(1): EQ. 12 (psi)	Maximum Stress Range(1): EQ. 13 (psi)	Cumulative Usage Factor	Pipe Break Stress Limit 2.4 Sm (psi)	Description of Break Points	Break Type	
1	173-0 15/16	90	1-3 13/16					TP	CB	1.38 1.39
2	173-0 15/16	90	2-2 5/16					IP	CB & LB	1.40 1.41
3	173-1 3/8	90	2-6 5/16					IP	CB	1.42 1.43
4	163-1	90	13-3					IP	CB	1.44
5	162-5	90	13-3					IP	CB	1.45
6	156-11	90	13-3					IP	CB	1.46
7	156-8	90	13-6					IP	CB	1.47
8	156-8	90	15-5 1/2	(LATER)				IP	CB	1.48 1.49
9	156-8	90	15-10 1/2					IP	CB	1.50 1.51
10	156-8	90	16-3 3/4					IP	CB	1.52
11	156-8	90	16-11					IP	CB	1.53
12	156-8	99.5	20-6					IP	CB	1.54
13	156-8	101	20-6					IP	CB	1.55
14	156-8	105	19-6					IP	CB	1.56
15	156-5	106	19-3					IP	CB	1.57
19	150-9 1/2	106	19-3					IP	CB	1.58
20	150-6 1/2	105	19-6					IP	CB	2.1
21	150-6 1/2	101	19-3					IP	CB	2.2
22	150-6 1/2	99.5	20-6					IP	CB	2.3

Key: IP = Intermediate point
TP = Terminal point
CB = Circumferential break
LB = Longitudinal break

(1)Stresses were calculated in accordance with equations 12 and 13 of ASME Section III, paragraphs NB-3653.6(a) and NB-3653.6(b), respectively. Cumulative usage factors were calculated in accordance with ASME Section III, subarticle NB-3650.

NOTES: See Fig. 3.6A-33e for break locations.

The data presented in this table were used in conjunction with Section 3.6.2A to determine the break locations shown in Fig. 3.6A-33e. The reference coordinate system is shown in Table 3.6A-1.

TABLE 3.6A-4a

SUMMARY OF STRESSES IN HIGH-ENERGY ASME CLASS 1 PIPING

MAIN STEAM VENT SYSTEM

Break Point	Location: Elevation (ft-in)	Location: Azimuth (deg)	Location: r (ft-in)	Maximum Stress Range(1): EQ. 12 (psi)	Maximum Stress Range(1): EQ. 13 (psi)	Cumulative Usage Factor	Pipe Break Stress Limit 2.4Sm (psi)	Description of Break Points	Break Type	
23	150-6 1/2	90	16-9					IP	CB	2.4
24	150-3 1/2	90	16-6					IP	CB	2.5
25	148-9 1/2	90	16-6					IP	CB	2.6
27	148-6 1/2	90	16-7					IP	CB	2.7
29	148-6 1/2	84.5	16-8					IP	CB	2.8
31	148-6 1/2	82	16-10	(LATER)				IP	CB	2.9
33	148-6 1/2	76	17-0					TP	CB	2.10
34	156-8	90	15-6					IP	CB	2.11
39	156-8	126	16-4					IP	CB	2.12
41	156-5	126	16-5					IP	CB	2.13
43	148-9	126	16-5					IP	CB	2.14
45	148-6	126	16-4					IP	CB	2.15
47	148-6	130	16-3					TP	CB	2.16

Key: IP = Intermediate point
TP = Terminal point
CB = Circumferential break
LB = Longitudinal break

(1)Stresses were calculated in accordance with equations 12 and 13 of ASME Section III, paragraphs NB-3653.6(a) and NB-3653.6(b), respectively. Cumulative usage factors were calculated in accordance with ASME Section III, subarticle NB-3650.

NOTES: See Fig. 3.6A-33e for break locations.

The data presented in this table were used in conjunction with Section 3.6.2A to determine the break locations shown in Fig. 3.6A-33e. The reference coordinate system is shown in Table 3.6A-1.

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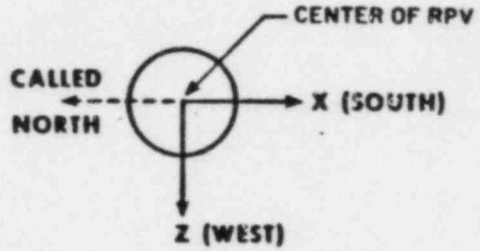
TABLE 3.6A-5

SUMMARY OF STRESSES IN HIGH-ENERGY ASME CLASSES 2 AND 3 PIPING

MAIN STEAM SYSTEM - OUTSIDE CONTAINMENT - LOOP A

Break Point	Location Elevation (ft-in)	Location X (ft-in)	Location Z (ft-in)	Maximum Stress[1] Total Additive Stress Eq. 9 and Eq. 10 (psi)	Pipe Break Stress Limit $0.8(1.2S_h + S_A)$ (psi)	Description of Break Points	Break Type	
121A	128-3 1/4	107-9 1/2	3-6	--	32,400	TP	CB	1.42
122A	128-1 13/16	116-5	3-6	23,875	32,400	IP (Elbow)	CB	1.43
140A	127-6 1/16	138-5	43-6	23,515	32,400	IP (Elbow)	CB	1.44
168A	114-0	147-11	3-3	13,600	32,400	TP	CB	1.45
401A	127-11 1/4	119-5	22-0 1/2	24,202	32,400	IP (Valve)	CB	1.46
402A	127-11 11/16	119-5	25-11 1/2	27,968	32,400	IP (Valve)	CB	1.47



Key: IP = Intermediate point
TP = Terminal point
CB = Circumferential break
LB = Longitudinal break

[1]Stresses were calculated in accordance with Equations 9 and 10 of ASME Section III, paragraph NC-3652.

NOTES: See Fig. 3.6A-15 for break locations.

The data presented in this table were used in conjunction with Section 3.6.2A to determine the break locations shown in Fig 3.6A-15.

Reference: 12210-AX-2G-2

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TABLE 3.6A-6

SUMMARY OF STRESSES IN HIGH-ENERGY ASME CLASSES 2 AND 3 PIPING

MAIN STEAM SYSTEM - OUTSIDE CONTAINMENT - LOOP B

Break Point	Location Elevation (ft-in)	Location X (ft-in)	Location Z (ft-in)	Maximum Stress(1) Total Additive Stress Eq. 9 and Eq. 10 (psi)	Pipe Break Stress Limit $0.8(1.25 S_h + S_A)$ (psi)	Description of Break Points	Break Type	
348B	128-3 1/4	107-9 1/2	-(10-6)	--	32,400	TP	CB	1.43
365B	121-7 11/32	118-7 13/32	-(10-6)	22,694	32,400	IP (Elbow)	CB	1.44
370B	119-5 3/4	119-5	-(10-6)	22,759	32,400	IP (Elbow)	CB	1.45
375B	117-11 1/8	119-5	-(10-6)	21,967	32,400	IP (Elbow)	CB	1.46
425B	114-7	122-5	-(43-6)	22,663	32,400	IP (Elbow)	CB	1.47
466B	114-0	147-11	-(9-9)	10,070	32,400	TP	CB	1.48
502B	114-11 1/8	119-5	-(31-0 1/2)	27,109	32,400	IP (Valve)	CB	1.49
503B	114-10 4/16	119-5	-(34-11 1/2)	25,983	32,400	IP (Valve)	CB	1.50
350B	128-2 3/4	110-8	-(10-6)	25,058	32,400	IP (Elbow)	CB	1.51
355B	127-4 5/32	112-9 19/32	-(10-6)	22,641	32,400	IP (Elbow)	CB	1.52
506B	114-9 3/4	119-5	-(21-8 7/8)	27,798	32,400	IP	CB	1.53

Key: IP = Intermediate point
TP = Terminal point
CB = Circumferential break
LB = Longitudinal break

(1)Stresses were calculated in accordance with Equations 9 and 10 of ASME Section III, paragraph NC-3652.

NOTES: See Fig. 3.6A-16 for break locations.

The data presented in this table were used in conjunction with Section 3.6.2A to determine the break locations shown in Fig 3.6A-16. The reference coordinate system is shown in Table 3.6A-5.

Reference: 12210-AX-2P-2

TABLE 3.6A-7

SUMMARY OF STRESSES IN HIGH-ENERGY ASME CLASSES 2 AND 3 PIPING

MAIN STEAM SYSTEM - OUTSIDE CONTAINMENT - LOOP C

Break Point	Location Elevation (ft-in)	Location X (ft-in)	Location Z (ft-in)	Maximum Stress(1) Total Additive Stress Eq. 9 and Eq. 10 (psi)	Pipe Break Stress Limit $0.8(1.2S_h + S_A)$ (psi)	Description of Break Points	Break Type	
348C	128-3 1/4	107-9 1/2	10-6	--	32,400	TP	CB	1.42
365C	121-7 11/32	118-7 13/32	10-6	21,861	32,400	IP (Elbow)	CB	1.43
370C	119-5 3/4	119-5	10-6	22,160	32,400	IP (Elbow)	CB	1.44
375C	117-11 1/8	119-5	10-6	21,831	32,400	IP (Elbow)	CB	1.45
420C	114-7	119-5	40-6	22,433	32,400	IP (Elbow)	CB	1.46
425C	114-7	122-5	43-6	21,201	32,400	IP (Elbow)	CB	1.47
466C	114-0	147-11	9-9	9,537	32,400	TP	CB	1.48
502C	114-11 1/8	119-5	31-0 1/2	25,286	32,400	IP (Valve)	CB	1.49
503C	114-10 1/4	119-5	34-11 1/2	24,894	32,400	IP (Valve)	CB	1.50
350C	128-2 3/4	110-8	10-6	23,758	32,400	IP (Elbow)	CB	1.51
355C	127-4 5/32	112-9 19/32	10-6	21,785	32,400	IP (Elbow)	CB	1.52
506C	114-11 1/4	119-5	19-5 3/4	25,178	32,400	IP	CB	1.53

Key: IP = Intermediate point
TP = Terminal point
CB = Circumferential break
LB = Longitudinal break

(1)Stresses were calculated in accordance with Equations 9 and 10 of ASME Section III, paragraph NC-3652.

NOTES: See Fig. 3.6A-15 for break locations.

The data presented in this table were used in conjunction with Section 3.6.2A to determine the break locations shown in Fig 3.6A-15. The reference coordinate system is shown in Table 3.6A-5.

Reference: 12210-AX-2F-2

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TABLE 3.6A-8

SUMMARY OF STRESSES IN HIGH-ENERGY ASME CLASSES 2 AND 3 PIPING
MAIN STEAM SYSTEM - OUTSIDE CONTAINMENT - LOOP D

Break Point	Location Elevation (ft-in)	X (ft-in)	Z (ft-in)	Maximum Stress (1) Total Additive Stress Eq. 9 and Eq. 10 (psi)	Pipe Break Stress Limit $0.8(1.25S_h + S_A)$ (psi)	Description of Break Points	Break Type	
121D	128-3 1/4	107-9 1/2	-(3-6)	--	32,400	TP	CB	1.40
122D	128-1 13/16	116-5	-(3-6)	22,632	32,400	IP (Elbow)	CB	1.41
123D	128-1 7/16	119-5	-(6-6)	20,133	32,400	IP (Elbow)	CB	1.42
140D	127-6 1/16	138-5	-(43-6)	18,038	32,400	IP (Elbow)	CB	1.43
168D	114-0	147-11	-(3-3)	10,131	32,400	TP	CB	1.44
401D	127-11 1/4	119-5	-(22-0 1/2)	23,732	32,400	IP (Valve)	CB	1.45
402D	127-11 11/16	119-5	-(25-11 1/2)	28,009	32,400	IP (Valve)	CB	1.46

Key: IP = Intermediate point
TP = Terminal point
CB = Circumferential break
LB = Longitudinal break

(1) Stresses were calculated in accordance with Equations 9 and 10 of ASME Section III, paragraph NC-3652.

NOTES: See Fig. 3.6A-16 for break locations.

The data presented in this table were used in conjunction with Section 3.6.2A to determine the break locations shown in Fig. 3.6A-16. The reference coordinate system is shown in Table 3.6A-5.

Reference: 12210-AX-2G-2

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TABLE 3.6A-9a

SUMMARY OF STRESSES IN HIGH-ENERGY ASME CLASS 1 PIPING

FEEDWATER SYSTEM - INSIDE CONTAINMENT

Break Point	Location: Elevation (ft-in)	Location: Azimuth (deg)	Location: r (ft-in)	Maximum Stress Range[1]: Eq. 12 (psi)	Maximum Stress Range[1]: Eq. 13 (psi)	Usage Factor[1]	Cumulative Stress Limit 2.4Sm (psi)	Pipe Break Description of Break Points	Break Type	
1W	142-3 1/2	135	12-0 3/16	11,530	25,380	0.0067	47,340	TP	CB	1.24
35W	122-0	93.6	17-3 7/8	2,970	26,960	0.0948	47,340	IP	CB & LB	1.25
60AW	122-0	32.08	20-4 1/16	50,950	49,610	0.4027	47,340	IP	CB & LB	1.26
60BW	122-0	27.75	19-6 7/16	50,950	49,610	0.4027	47,340	IP	CB & LB	1.27
60CW	122-0	25.44	21-0 1/2	50,950	49,610	0.4027	47,340	IP	CB & LB	1.28
70W	123-9	24.59	17-9 11/16	52,260	26,250	0.1211	54,180	IP	CB & LB	1.29
80W	140-9 1/2	24.59	17-9 11/16	56,880	25,660	0.1223	54,180	IP	CB & LB	1.30
85W	142-3 1/2	29.24	17-4 1/16	47,800	26,270	0.1088	54,180	IP	CB & LB	1.31
90W	142-3 1/2	39.86	16-9 1/16	34,710	27,400	0.1067	54,180	IP	CB & LB	1.32
95W	142-3 1/2	45	15-2 1/4	58,070	28,460	0.1202	54,180	IP	CB & LB	1.33
100W	142-3 1/2	45	12-0 3/16	27,860	27,900	0.0418	54,180	TP	CB	1.34
105W	122-0	20	22-3 3/4	8,730	56,100	0.7273	47,340	IP	CB & LB	1.35
120W	122-0	11.7	24-2 5/8	7,920	53,730	0.7472	47,340	IP	CB & LB	1.36
130W	122-0	7.5	26-4 13/16	21,330	24,320	0.0339	47,340	TP	CB	1.37

1.39

Key: IP = Intermediate point
TP = Terminal point
CB = Circumferential break
LB = Longitudinal break

1.41
1.42
1.43
1.44

[1] Stresses were calculated in accordance with Equations 12 and 13 of ASME Section III, paragraphs NB-3653.6(a) and NB-3653.6(b), respectively. Cumulative usage factors were calculated in accordance with ASME Section III, subarticle NB-3650.

1.46
1.47
1.48

NOTES: See Fig. 3.6A-17 for break locations.

1.50

The data presented in this table were used in conjunction with Section 3.6.2A to determine the break locations shown in Fig. 3.6A-17. The reference coordinate system is shown in Table 3.6A-1.

1.52
1.53

1.55
1.56
1.57

Reference: 12210-AX-17D-1

2.1

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TABLE 3.6A-9b

SUMMARY OF STRESSES IN HIGH-ENERGY ASME CLASS 1 PIPING

FEED WATER SYSTEM - INSIDE CONTAINMENT

Break Point	Location: Elevation (ft-in)	Location: Azimuth (deg)	Location: r (ft-in)	Maximum Stress Range: Eq.12 (psi)	Maximum Stress Range: Eq.13 (psi)	Cumulative Usage Factor(1)	Pipe Break Stress Limit 2.4Sm (psi)	Description of Break Points	Break Type
1E	142-3 1/2	225	12-0 3/16	11,530	25,380	0.0067	47,340	TP	CB
35E	122-0	266.4	17-3 7/8	2,970	26,960	0.0948	47,340	IP	CB & LB
60AE	122-0	327.91	20-4 1/16	50,950	49,610	0.4027	47,340	IP	CB & LB
60BE	122-0	332.25	19-6 7/16	50,950	49,610	0.4027	47,340	IP	CB & LB
60CE	122-0	334.56	21-0 1/2	50,950	49,610	0.4027	47,340	IP	CB & LB
70E	123-9	335.41	17-9 11/16	52,260	26,250	0.1211	54,180	IP	CB & LB
80E	140-9 1/2	335.41	17-9 11/16	56,880	25,660	0.1223	54,180	IP	CB & LB
85E	142-3 1/2	330.76	17-4 1/16	47,800	26,270	0.1088	54,180	IP	CB & LB
90E	142-3 1/2	320.14	16-9 1/16	34,710	27,400	0.1067	54,180	IP	CB & LB
95E	142-3 1/2	315	15-2 1/4	58,070	28,460	0.1202	54,180	IP	CB & LB
100E	142-3 1/2	315	12-0 3/16	27,860	27,900	0.0418	54,180	TP	CB
105E	122-0	340	22-3 3/4	8,730	56,100	0.7273	47,340	IP	CB & LB
120E	122-0	348.3	24-2 5/8	7,920	53,730	0.7472	47,340	IP	CB & LB
130E	122-0	352.5	26-4 13/16	21,330	24,320	0.0339	47,340	TP	LB

Key: IP = Intermediate point
TP = Terminal point
CB = Circumferential break
LB = Longitudinal break

(1) Stresses were calculated in accordance with equations 12 and 13 of ASME Section III, paragraphs NB-3653.6(a) and NB-3653.6(b), respectively. Cumulative usage factors were calculated in accordance with ASME Section III, subarticle NB-3650.

NOTES: See Fig. 3.6A-17 for break locations.

The data presented in this table were used in conjunction with Section 3.6.2A to determine the break locations shown in Fig. 3.6A-17. The reference coordinate system is shown in Table 3.6A-1.

Reference: 12210-AX-17D-1

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TABLE 3.6A-10a

SUMMARY OF STRESSES IN HIGH-ENERGY ASME CLASSES 2 AND 3 PIPING

FEEDWATER SYSTEM - OUTSIDE CONTAINMENT (FWS-020-62-2, FWS-020-32-4)

Break Point	Location Elevation (ft-in)	Location X (ft-in)	Location Z (ft-in)	Stress(1) EQ.9 (psi)	Stress(1) EQ. 10 (psi)	Total Additive Stress(1) (psi)	Pipe Break Stress Limit $0.8(1.2S_h + S_A)$ (psi)	Description of Break Points	Break Type
1	121-8 1/16	107-9 1/2	3-6				32,400	TP	CB
2	121-6 15/16	116-11	3-6	11,720	15,490	27,210	32,400	IP	CB
3	119-0 5/8	119-5	3-6	13,010	20,590	33,600	32,400	IP	CB & LB
6	108-7 15/16	119-5	35-8	11,120	12,030	23,150	32,400	IP	CB
7	108-3 43/64	149-5	38-2				32,400	TP	CB
20	115-1 5/8	119-5	3-6	13,970	15,210	29,180	32,400	IP	CB
22	108-6 13/32	129-2	38-2	17,790	11,530	29,320	32,400	IP	CB

Key: IP = Intermediate point
TP = Terminal point
CB = Circumferential break
LB = Longitudinal break

(1)Stresses were calculated in accordance with Equations 9 and 10 of ASME Section III, paragraph NC-3652.

NOTES: See Fig. 3.6A-18 for break locations.

The data presented in this table were used in conjunction with Section 3.6.2A to determine the break locations shown in Fig. 3.6A-18. The reference coordinate system is shown in Table 3.6A-5.

Reference: 12210-AA-17D-2

RBS FSAR

TABLE 3.6A-10b

SUMMARY OF STRESSES IN HIGH-ENERGY ASME CLASSES 2 AND 3 PIPING

FEEDWATER SYSTEM - OUTSIDE CONTAINMENT (FWS-020-63-2, FWS-020-31-4)

Break Point	Location			Stress(1)		Total Additive Stress(1)	Pipe Break Stress Limit $0.8\ (1.2S_h + S_A)$	Description of Break Points	Break Type	
	Elevation (ft-in)	X (ft-in)	Z (ft-in)	EQ.9 (psi)	EQ. 10 (psi)	(psi)	(psi)			
11	121-8 1/16	107-9 1/2	-(3-6)	-	-	-	32,400	TP	CB	1.43
12	121-6 5/16	119-5	-(3-6)	11,720	15,490	27,210	32,400	IP	CB	1.44
13	119-0 5/16	121-11	-(3-6)	13,010	20,590	33,600	32,400	IP	CB & LB	1.45
16	108-4	149-5	34-0	-	-	-	32,400	TP	CB	1.46
30	115-1 5/16	121-11	-(3-6)	13,970	15,210	29,180	32,400	IP	CB	1.47
31	108-7 3/4	121-11	31-6	11,120	12,030	23,150	32,400	IP	CB	1.48
33	108-5 15/16	134-2	34-0	17,790	11,530	29,320	32,400	IP	CB	1.49

Key: IP = Intermediate point
TP = Terminal point
CB = Circumferential break
LB = Longitudinal break

(1)Stresses were calculated in accordance with Equations 9 and 10 of ASME Section III, paragraph NC-3652.

NOTES: See Fig. 3.6A-18 for break locations.

The data presented in this table were used in conjunction with Section 3.6.2A to determine the break locations shown in Fig. 3.6A-18. The reference coordinate system is shown in Table 3.6A-5.

Reference: 12210-AX-17E-2
Amendment 11

January 1984

RBS FSAR

TABLE 3.6A-11a

SUMMARY OF STRESSES IN HIGH-ENERGY ASME CLASS I PIPING

RCIC SYSTEM - INSIDE CONTAINMENT

Break Point	Location Elevation (ft-in)	Location Azimuth (deg)	Location r (ft-in)	Maximum Stress Range Eq.12 (psi)	Maximum Stress Range Eq.13 (psi)	Cumulative Usage Factor	Pipe Break Stress Limit 2.4Sm (psi)	Description of Break Points	Break Type	
1	146-3 3/4	67.5	16-6 3/8	-	-	-	42,210	TP	CB	1.47
5	125-3 3/8	0	17-7 1/8	11,196	34,549	0.1189	42,210	IP (Elbow)	CB & LB	1.48
12	122-6 3/4	24	26-10	-	-	-	42,210	TP	CB	1.49
4	126-3 3/8	0	16-6 3/8	11,569	32,675	0.08	42,210	IP (Elbow)	CB	1.50
7	125-2 11/16	2	21-11 25/32	6,757	30,915	0.0851	42,210	IP (Elbow)	CB	1.51
										1.52

Key: IP = Intermediate point
TP = Terminal point
CB = Circumferential break
LB = Longitudinal break

NOTES: Stresses were calculated in accordance with equations 12 and 13 of ASME Section III, Paragraphs NB-3653.6(a) and NB-3653.6(b), respectively. Cumulative usage factors were calculated in accordance with ASME Section III, Subarticle NB-3650.

See Fig. 3.6A-12 for break locations.

The data presented in this table were used in conjunction with Section 3.6.2A to determine the break locations shown in Fig. 3.6A-12. The reference coordinate system is shown in Table 3.6A-1.

Reference: 12210-AX-2G-1.

RBS FSAR

TABLE 3.6A-11B

SUMMARY OF STRESSES IN HIGH-ENERGY ASME CLASS 1 PIPING

RCIC HEAD SPRAY SYSTEM - INSIDE CONTAINMENT

Break Point	Location: Elevation (ft-in)	Location: Azimuth (deg)	Location: r (ft-in)	Maximum Stress Range(1): EQ. 12 (psi)	Maximum Stress Range(1): EQ. 13 (psi)	Cumulative Usage Factor	Pipe Break Stress Limit 2.4Sm (psi)	Description of Break Points	Break Type	
1	172-11 1/4	270	1-0 7/16					TP	CB	1.37 1.38
2	172-11 1/4	270	4-0		(LATER)			TP	CB	1.40 1.41

Key: IP = Intermediate point
TP = Terminal point
CB = Circumferential break
LB = Longitudinal break

(1)Stresses were calculated in accordance with equations 12 and 13 of ASME Section III, paragraphs NB-3653.6(a) and NB-3653.6(b), respectively. Cumulative usage factors were calculated in accordance with ASME Section III, subarticle NB-3650.

NOTES: See Fig. 3.6A-20 for break locations.

The data presented in this table were used in conjunction with Section 3.6.2A to determine the break locations shown in Fig. 3.6A-20. The reference coordinate system is shown in Table 3.6A-1.

QUESTION 210.82 (3.6.2)

Provide the bases for assuring that the feedwater isolation check valves can perform its function following a postulated pipe break of the feedwater line outside containment.

Response

The response to this request is provided in Appendix 3C.2.2.

(SRV), CRD lines, the primary shield wall, and the drywell wall.

To preclude damage caused by a whipping feedwater pipe, a total of 22 restraints have been installed on the feedwater system inside the drywell.

All restraints on the feedwater system are omnidirectional except 1FWS-PRR-811 and 831 which are moment-limiting "zero gap" restraints to keep the stresses within acceptable limits in the isolation valves and break exclusion zones.

All equipment inside the drywell, whose operation during or after a LOCA is required for safe shutdown, is qualified for the post-LOCA drywell environment as discussed in Section 3.11.

Inside the Steam Tunnel

All feedwater piping from inboard of the first moment-limiting (zero gap) restraint in the drywell to outboard of the jet impingement wall is defined as a break exclusion zone.

In the Auxiliary Building

INSERT →

From the drywell the two 20-in feedwater lines enter the auxiliary building from the north side of the north-south centerline (steam tunnel) at approximate el 121 ft-6 in, and then drop vertically to approximate el 109 ft-0 in. A 90° elbow directs them horizontally west along the north auxiliary building wall. At a mean distance approximately 13 ft from the west wall the lines turn south, cross the auxiliary building, and enter the turbine building at approximate el 108 ft-4 in.

Seven circumferential breaks and one longitudinal break have been postulated for each line. The targets associated with these breaks are nonessential for safe shutdown. They are: RHR lines, RWCU lines, main steam drain (DTM) lines, service air lines (SAS), condensation lines (CNS), the auxiliary building wall, and structural steel at various elevations.

However, to preclude the generation of secondary missiles, and to maintain structural integrity, nine restraints have been installed on the feedwater system in the auxiliary building.

A check valve dynamic analysis is performed for the main feedwater isolation check valve to assure its functional performance following a postulated pipe break outside the containment. A dynamic forcing function is used in the valve stress analysis. First a flow transient analysis is performed for the feedwater system to simulate the pipe break condition. The reverse flow condition at the check valve location is determined using the SWEC computer program WATHAM (Appendix 3A). Hydrodynamic torque exerted on the valve disk by the reverse flow is calculated. The valve closing time and the disk alarm velocity is determined from the valve dynamic model.

The structural analysis of the check valve consists of a disc-to-shaft analysis utilizing finite element methods, and assuming that the valve body is rigid. This method of analysis is based on past experience of the valve vendor that indicates that the valve internals are limiting under this design condition.

QUESTION 210.92 (3.9)

Using the guidance of NUREG-0609, provide the methodology used and the results of the annulus pressurization (AP) analysis (asymmetric LOCA loads) for the reactor system and affected components including the following:

1) reactor pressure vessel and supports,

2) core supports and other reactor internals,

3) control rod drives,

4) ECCS piping attached to the reactor coolant system,

5) primary coolant piping, and

6) piping supports for affected piping system.

The results of the above analysis should specifically address the effects of the combined loadings due to annulus pressurization and an SSE.

RESPONSE

The reactor asymmetric loads analysis is due to be completed in the third quarter of 1984 and will be documented in a self-contained appendix to Section 3.9.

The following is a brief description of the methodology.

1. Pressure-Time Histories

   The pressure time histories in the annulus region between the RPV and shield wall are generated from a feedwater line break and a recirculation line break. The RELAP code using nodalized mass and energy balance is used in this analysis.

2. Concentrated Force-Time Histories

   The forcing function of jet impingement on the shield wall is obtained from the break flow transient caused by a feedwater line break and a recirculation line break. Forcing functions of jet reaction on RPV, jet impingement on RPV, and pipe whip restraint load on restraint anchors are obtained from the feedwater line break, the recirculation line break, and main steam line break.

210.100 In your Specification No. 228.000, "Specification for Piping Engineering and Desigr" Revision 1, dated 3/5/81 including Addenda 1 and 2, you have addressed piping out-of-roundness tolerances based on the fabrication limits established by the ASME and ANSI Codes. You have further stated that stress analysis is based on the assumptions that the out-of-roundness is within the 0.08t limit that is specified in Table KB-3681(a)-1, unless otherwise specified.

Out-of-roundness tolerances established by the ASME Code (e.g., for pipe bends KB-42223.2), ANSI Standards (e.g. ANSI B16.9 for elbows) and pipe specifications (e.g., SA-106, SA-155) are not sufficient to assure that the out-of-roundness is less than 0.08t.

Your response, except for the final clause "unless otherwise specified", indicates that your piping analysis ignores the Code requirement concerning increasing the K_1-index to account for out-of-roundness. If this is correct, provide your technical justification for the un-modified K_1-indices. If, by the clause "unless otherwise specified", you mean that where the out-of-roundness may be greater than 0.08t, you do include out-of-roundness effects in K_1; provide a description of where and when this "unless otherwise specified" is accomplished.

Response

The response to this request is provided in new Section 3.9.1.7A.

210.101 In your Specification No. 228.000, "Specification for Piping Engineering and Design", Revision 1, dated 3/15/81 including Addenda 1 and 2, you have addressed the minimum allowable wall thickness at girth butt welds in piping. You have stated that the minimum wall thickness at girth butt welds is established by the counterbore requirements and it will be found on analysis that this value will not decrease to less than the manufacturer's minimum wall thickness requirements. Using a 12 inch schedule 160 pipe as an example you have provided the following example:

Sketch	STD-SP-1056-15
Pipe O.D.	12.719 in. (including 1/32 in. tolerance)
Pipe I.D.	10.126 in.
"C" Bore (per sketch)	10.423 in. (including 0.010-in. tolerance)
Final wall thickness	(12.719 - 10.423)/2 = 1.148 in.
(0.875)tn	(0.875)(1.312) = 1.148 in.

The use of C-dimensions as illustrated by your example, in conjunction with a pipe specification that has a tolerance of (minus) -1/32" on outside diameter (such as SA-106), would assure $t_m \geq (7/8)t$; provided the counterbore is concentric with respect to the minimum outside diameter (t_m = minimum wall, t_n = nominal wall). If you do not control concentricity of the counterbore, explain how you are assured that $t_m \geq (7/8)t_n$. Include in the explanation consideration of 1) ANSI B16.9 fittings, 2) pipe specification SA-155, and 3) any other specification you use for ASME Code piping at River Bend which does not have the same minus tolerance on outside diameter as SA-106.

Response

Stone & Webster Engineering Corporation's (SWEC) specification 228.150, Shop Fabricated Piping, page 1-54, Item 19, provides quality control measures when counterboring pipe. A discussion of the information requested is provided in new Section 3.9.1.7A.

3.9.1.7A Piping Engineering and Design

A quality control representative witnesses, on a surveillance basis, the Seller's dimensional checks to ensure that the counterbore of pipe is in accordance with the specification and that minimum pipe wall thickness has not been violated. In order to assure records on actual field end counterbores performed by the shop fabricator, the fabricator includes on his shop traveler the required and actual counterbore dimension as well as the required and actual pipe wall thickness after counterboring.

The stress analysis of class 1 piping system is based on the assumption that the out-of-roundness is within 0.08t limit that is specified in Table NB-3681(a)-1 of ASME Section III. Technical justification for these assumptions are described as follows:

(1) While the out-of-roundness tolerances, established by ASME Code and piping specifications, are not sufficient to assure the 0.08t limit, the pipes are judged to be generally round (i.e. within the 0.08t limit to require no increase of k-index).

(2) Out-of-roundness affects only the pressure term in the calculation of peak stresses, its contribution to the overall peak stresses are not significant even if out-of-roundness conditions permitted by the design/fabrication specification is considered.

However, all class 1 pipe stress analysis are reviewed to account for out-of-roundness based on the fabrication limits established by the ASME Code and pipe specification, unless more realistic field measurements are available.

CHAPTER 3

TABLE OF CONTENTS (Cont)

Enclosure 12

QUESTION 210.102

In order to demonstrate compliance with 10CFR50.55a, identify the code, code edition, and addenda used in the construction of each Quality Group A (ASME Section III, Class I) component within the reactor coolant pressure boundary.

Response

The response to this request is provided in new Table 3.2-9.

Enclosure 12 (cont'd)

Table 3.2-9

Reactor Coolant Pressure
Boundary - Quality Group A
(ASME III, Class 1)
Components

BOF System/Components	CODE	CODE EDITION	ADDENDA
HPCS System			
Piping	ASME III	1974	No Addenda
Valves	"	"	"
LPCS System			
Piping	"	"	"
Valves	"	"	"
RHR System			
Loop A			
Piping	"	"	"
Valves	"	"	"
Loop B			
Piping	"	"	"
Valves	"	"	"
Loop C			
Piping	"	"	"
Valves	"	"	"
Common Loop A&B			
Piping	"	"	"
Valves	"	"	"
RCIC System			
Piping	"	"	"
Valves	"	"	"
RWCU System			
Piping	"	"	"
Valves	"	"	"
Reactor Plant Equipment Drains			
Piping	"	"	"
Valves		"	"
Feedwater System			
Piping	"	"	"
Valves	"	"	"

Enclosure 12 (cont'd)

Table 3.2-9

Reactor Collant Pressure
Boundary - Quality Group A
(ASME III, Class 1)
Components

BOP System/Components	CODE	CODE EDITION	ADDENDA
Turbine Plant Miscellaneous Drains			
Piping	ASME III	1974	No Addenda
Valves	"	"	"
Main Steam System			
Piping (Vent Line)	"	"	"
Valves	"	"	"
Standby Liquid Control			
Piping	"	"	"
Valves	"	"	"
NSSS System/Components			
Reactor Pressure Vessel	"	1971	Summer 1972
Main Steam System	"	"	Winter 1972
Recirculation System	"	"	"
High Pressure Core Spray System	"	"	"
SLC Explosive Valve	"	"	"

Enclosure 13

QUESTION 210.103

In note 13 of Table 3.2-1, identify the specific design requirement that GE has established for the RCIC turbine that provides a level of quality equivalent to Safety Class 2. Footnote "m" of GESSAR II is a typical example of the type of information requested.

Response

The response to this request is provided in revised note 13 of Table 3.2-1

TABLE 3.2-1 (Cont)

NNS from the outer isolation valve or the process shutoff valve (root valve) to the sensing instrumentation.

4. All other instrument lines:

   a. through the root valve are of the same classification as the system to which they are attached.

   b. beyond the root valve, if used to actuate a safety system, are of the same classification as the system to which they are attached.

5. All sample lines from the outer isolation valve or the process root valve through the remainder of the sampling system are classified NNS.

Insert →

(13) The RCIC turbine does not fall within the applicable design codes. To assure that the turbine is fabricated to the standards commensurate with their safety and performance requirements, GE has established specific design requirements for this component.

(14) The condensate storage tank is designed, fabricated, and tested to meet the intent of ANSI-B96.1.

(15) ASME Section VIII-1 and ANSI B31.1 apply downstream of outermost isolation valves.

(16) The gaseous radwaste system piping, pumps, and valves containing gaseous radwaste are constructed in accordance with the applicable codes of classification NNS.

(17) The hydraulic control unit (HCU) is a GE factory-assembled engineered module of valves, tubing, piping, and stored water which controls a single CRD by the application of precisely timed sequences of pressures and flows to accomplish slow insertion or withdrawal of the control rods for power control and rapid insertion for reactor scram.

Although the HCU, as a unit, is field installed and connected to process piping, many of its internal parts differ markedly from process piping components because of the more complex functions they must provide. Thus, although the codes and standards invoked for Safety Classes 1, 2, 3, and NNS

Enclosure 13 (cont'd)

TABLE 3.2-1 - (Continued)

EQUIPMENT CLASSIFICATION

(13) The turbine does not fall within the applicable design codes. To assure the turbine is fabricated to the standards commensurate with their safety and performance requirements. General Electric has established specific design requirements for this component which are as follows:

a. All welding shall be qualified in accordance with Section IX, ASME Boiler and Pressure Vessel Code,

b. All pressure-containing castings and fabrications shall be hydrotested at 1.5 X design pressure,

c. All high-pressure castings shall be radiographed according to:

ASTM E-94	
E-142	maximum feasible volume
E-71, 186 or 280	Severity level 3

d. As-cast surfaces shall be magnetic particle or liquid penetrant tested according to ASME, Section III, Paragraphs NB-2575, NC-2575, NC-2576, or NB-2576, NC-2576,

e. Wheel and shaft forgings shall be ultrasonically tested according to ASTM A-388,

f. Buff-welds shall be radiographed, and magnetic particle or liquid penetrant tested according to ASME Boiler and Pressure Vessel Code. Acceptance standards shall be in accordance with ASME Boiler and Pressure Vessel Code Section III Paragraph NB-5340, NC-5340, NB-5350, NC-5350, respectively,

g. Notification to be made on major repairs, and records maintained, thereof,

h. Record system and traceability according to ASME Section III, NA-4000,

i. Control and identification according to ASME Section III, NA-4000,

j. Procedures shall conform to ASME Section III, NB-5100, NC-5100,

k. Inspection personnel shall be qualified according to ASME Section III, NB-5500, NC-5500.

Enclosure 14

QUESTION 210.104

In Table 3.2-1 (Page 2 of 19), add the Scram Discharge Volume Piping as a line item of the CRD Hydraulic System. This component should be classified Safety Class 2, Seismic Category I, and Quality Assurance Category B.

RESPONSE

The Scram Discharge Volume Piping (header and lines) is included in Table 3.2-1 IV, Item 4, piping, scram discharge volume lines.

RBS FSAR

TABLE 3.2-1 (Cont)

	Safety[1] Class	Seismic[2] Category	Quality[3] Assurance Category	Tornado[4] Protection Designation	Location[5]	Scope[6] of Supply	Design[7] Detail	Notes
13. Mechanical modules, instrumentation, with safety function	2	I	B	E	C	GE	GE	
14. Electrical modules with safety function	2	I	B	E	C	GE	GE	
15. Cable, with safety function	2	NA	B	E	-	P	S	
III. Recirculation System								
1. Piping	1	I	B	E	D	GE	GE	(12)
2. Pipe suspension, recirculation line	1	I	B	E	D	GE	GE	
3. Pipe restraints, recirculation line	2	I	B	E	D	GE	GE	
4. Pumps	1	I	B	E	D	GE	V	
5. Valves	1	I	B	E	D	GE	V	
6. Motor, pump	2	I	B	E	D	GE	V	
7. Electrical modules, with safety function	2	I	B	E	C	GE	GE	
8. Cable with safety function	2	NA	B	E	-	P	S	
9. LFMG set	NNS	NA	NA	N	T	GE	GE	
IV. CRD Hydraulic System								
1. Valves, scram discharge volume lines	2	I	B	E	C	GE & P	V	
2. Valves insert and withdraw lines	2	I	B	E	C	P	V	(9)
3. Valves, other	NNS	NA	S	E	C,F/A,C	P/GE	V	
4. Piping, scram discharge volume lines and header	2	I	B	E	C	P	S	(20)
5. Piping, insert and withdraw lines	2	I	B	E	C,D	P	S	(9)
6. Piping, other	NNS	NA	S	E	C,D,F	P	S	(12)
7. Hydraulic control unit	2	I	B	E	C	GE	GE	(17)
8. Electrical modules, with safety function	2	I	B	E	C	GE	GE	
9. Cable, with safety function	2	NA	B	E	-	P	S	
10. CRD pumps, filters, and strainers	NNS	NA	S	E	F	GE	V	

Enclosure 15

QUESTION 210.105

In a comparison of the quencher and supports per NUREG-0763, you have shown that the quencher base support is welded to the basement and is provided with rigid supports from the quencher body to the drywell wall. The rigid supports appear to be welded directly to the conical reducer of the quencher body. In the staff's opinion, if the supports are welded directly to the quencher body, the welded attachment could cause excessive localized bending stresses and potentially harmful thermal gradients in the quencher body following actuations of the safety-relief valves. Furthermore, the large number of stress cycles could result in fatigue failure in the quencher-to-support weld. Provide the quencher support details and the results of your stress analyses for the support attachment to the quencher body. Provide assurance that the structural integrity of the quencher support capabilities is not compromised.

RESPONSE

The response to this request is provided in Section A.6A.7.2

Enclosure 15 (cont'd)

QUESTION 210.106

The quencher device is classified as an ASME Code Class 3 component. Provide the basis for assuring that the quencher device has been adequately designed for all applicable stresses resulting from the actuation of the safety-relief valves including localized bending stresses and thermal gradients.

RESPONSE

The response to this request is provided in Section A.6A.7.2

A.6A.7 QUENCHER ARRANGEMENT

Fig. A.6A.4-1 and A.6A.4-2 show the general arrangement of the quenchers in the suppression pool.

A.6A.7.1 Quencher Arm/Strut Loads and Loading Application

The types of loads used in the design of the quencher bottom anchoring and side support struts are identified in Tables A.6A.7-1 and A.6A.7-2 and Fig. A.6A.7-1 through A.6A.7-3.

A.6A.7.2 Quencher Design Information

The quencher used in River Bend Station is designed as specified by GE in Section A7.2 of CLR, except as given in the following criteria.

Insert A ——→

A.6A.7.2.1 SRVDL Geometry

See Fig. A.6A.10-2a through A.6A.10-2e for River Bend Station SRVDL routing.

A.6A.7.2.2 Quencher Design Criteria

Forces - See Fig. A.6A.7-1 through A.6A.7-3 and Tables A.6A.7-1 and A.6A.7-2

Fatigue - See Section A.6A.9 and Fig. A.6A.5-13

Cycles of Operation - See Section A.6A.9 and Fig. A.6A.5-13.

Enclosure 15 (cont'd)

Insert A

The X-quencher was designed and analyzed as a Class 3 piping component in accordance with the requirements of Section ND-3600 of the ASME Boiler and Pressure Vessel Code. The evaluation included all required loads associated with safety relief valve operation including both mechanical and hydrodynamic loads. The quencher is capable of withstanding loading from connecting safety relief valve discharge piping for sustained, occasional, thermal expansion, emergency and faulted conditions.

This section ND-3645 of the ASME Boiler and Pressure Vessel Code Section III-1983 Edition was considered during the design of the quenchers. Cyclic loading was accounted for by equations 10 and 11 of ND-3652.3 and found to be acceptable. Stress concentrations and the potential for local bending stresses and flattening were minimized during design.

Although ND-3645 does not require any separate analysis, the bracket-quencher interface connection was analyzed to the requirements of Subsection NB-3600 of the ASME Code. The analysis consisted of calculating the fatigue usage factor for 40 years of plant life at the support bracket connection.

The analysis was restricted to the quencher bracket interface because this is the cross-section which is most affected by the normal and upset condition loads. A fatigue usage factor was calculated using alternating stress "S_a" and peak stress "S_p" obtained from Equations 11 and 14 of Subsection NB-3600 respectively. Equation 11 was modified to include the stresses due to lug 1 effects by incorporating ASME Code Case N-122. To calculate the worst thermal gradients across the bracket connection, the most severe transient was used. The transient consisted of a step change from 70°F to 350°F. Appropriate heat transfer coefficients were calculated for the transient. Other normal and upset condition loads being transmitted by SRV piping were included in Equation 11.

The results of the analysis showed a usage factor of 0.47 at the support bracket connection. We can, therefore, safely conclude that the X-Quencher for the River Bend Station can safely withstand the expected normal and upset condition loads.

The stresses at critical locations in the quencher configuration were determined to be within the ASME code allowables for quencher design. Thermal gradients and localized stresses are considered in the fatigue factor used in the stress analysis for ASME Code Class 3 piping components. The requirements of ND-3645 have been satisfied.

Section 3.9.3.4.3A (cont'd.)

results for each of the components listed below will be provided in a later amendment. The specified component operating condition is consistent with the plant operating condition for each transient event.

1. RHR heat exchanger supports and restraints
2. Fuel pool cooling and cleanup heat exchanger supports and restraints
3. Standby diesel generator fuel oil day tank supports
4. Quencher supports and restraints

Insert B→

5. SSW pump supports.

The RWCU regenerative and nonregenerative heat exchanger supports are designed to the AISC code.

U-bolts are not used in the installation of component supports. High strength bolts (A325 steel) are used for the mounting of the following equipment:

1. Main steam isolation valve accumulators located at el 142 ft in the reactor building and el 134 ft in the auxiliary building.
2. Chilled water compression tanks located at el 98 ft in the control building.

3.9.4A Control Rod Drive Systems

See Section 3.9.4B.

3.9.5A Reactor Pressure Vessel Internals

See Section 3.9.5B.

3.9.6A Inservice Testing of Pumps and Valves

Inservice testing of ASME Code Class 1, 2, and 3 pumps and valves is performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10CFR50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10CFR50, Section 50.55a(g)(6)(i).

The examinations are in accordance with ASME Section XI, 1977 Edition, up to and including the Summer 1978 Addenda.

Enclosure 15 (cont'd)

Insert B

Quencher design information is discussed in Section A.6A.7.2.

Question 210.107

The staff has been informed that several piping systems inside the suppression pool wetwell area which are subjected to pool swell impact and drag loadings are restrained using linear (vertical) pipe supports with pinned-pinned end connections. Provide the basis for assuring that the design of these piping systems have properly considered the lateral instability of those pipe supports with pinned-pinned end connections when subjected to the non-cyclic compressive loadings as experienced during a pool swell event.

Response

The response to this request is provided in Appendix 6A.16.5.3.

6A.16.5.3 Impact Load on Piping above Pool Surface

INSERT→

~~Only the short horizontal segment of piping at the penetration elevation is subjected to pool swell impact.~~ Its length is the distance from the containment wall to the farthest point on the inner wall of the riser. In the case of the RHR discharge line, the pipe axis is 2.5 ft inward from the containment wall, and the inner pipe radius is 6.0 in = 0.5 ft. Therefore, the effective length is 2.5+0.5 = 3.0 ft.

The impact load is a spike whose normalized shape is given in CLR Fig. 10-2. The amplitude at time t is

$P_I(t)$ = 60 psi x (cross section of pipe exposed to bulk water) x (value from plot at time t)

For the segment under consideration:

$$P_I(t) = 60 \frac{\text{lb}}{\text{sq in}} \times (24 \times 12.75) \text{ sq in} \times (\text{value from plot})$$

$$= 18{,}360 \text{ lb} \times (\text{value from plot})$$

Assuming a uniform impact line load, including the inside of the pipe, the moment is adequately represented by applying a force at the upper elbow end, at 1 ft from the wall, representing a 2-ft length, and a force at the lower end of the elbow, representing the remaining 1-ft length.

Thus, with node points as defined in Fig. 6A.16-3

$$F_{10} = \frac{2}{3} \cdot 18{,}360 = 12{,}240 \text{ lb}$$

$$F_{15} = \frac{1}{3} \cdot 18{,}360 = 6{,}120 \text{ lb}$$

They are applied in the upward direction.

The impact load is applied either as a force history or as a static case with dynamic load factor 1.5.

In the case of the RHR pipe a static analysis was performed.

Linear (vertical) pipe supports with pinned-pinned connections were avoided in the design of piping above the suppression pool surface due to pool swell impact drag loads. Only the short horizontal segment of ECCS piping at the penetration elevation, within 25 feet above the suppression pool surface, is subjected to pool swell impact.

Question 210.108

Provide the basis for assuring that the CRD piping have been adequately designed to withstand the HCU floor dynamic response to the impact loadings associated with a LOCA pool swell event.

Response

The response to this request is provided in Appendix 6A.12.

RBS FSAR

The potential for circumferential variations in the pressure transient in the wetwell region beneath the HCU floor has been examined, and on the basis of bounding calculations it is concluded that the pressure variation is less than 0.5 psid.

Since the air tests were performed, additional PSTF tests have been conducted with the specific objective of providing further data on the interaction of pool swell with the HCU floors. The test results are in CLR Reference 11. Supplement 1 to CLR Reference 1 describes the GE analytical model used to simulate the HCU floor flow pressure differential and presents a comparison of model predictions with test data. The model is shown to be conservative. The LOCTVS model described in Reference 1 to Section 6.2.1 is similar to the GE model and yields comparable results.

INSERT

The HCU floor has been designed to withstand loads from a pool swell event. CRD piping is capable of withstanding the vibratory responses due to all hydrodynamic loadings.

For the pool swell froth impact load, the loading, as shown in Figure 3B-74 of GESSAR II, Appendix 3B, was applied to a finite element model of the HCU floor. Dynamic responses of the floor were generated in the form of amplified response spectra (ARS) and maximum displacements at various locations on the floor.

The ARS for this loading was found to be insignificant.

3.9.4.2B Applicable CRDS Design Specifications

The control rod drive system (CRDS) is designed to meet the functional design criteria as outlined in Section 4.6 and consists of the following:

1. Locking piston control rod drive
2. Hydraulic control unit
3. Hydraulic power supply (pumps)
4. Interconnecting piping
5. Flow and pressure and isolation valves
6. Instrumentation and electrical controles.

Those components of the CRD forming part of the primary pressure boundary are designed according to ASME Code Section III.

The safety classification of the CRD and the CRD hydraulic system is outlined in Table 3.2-1, and the components are designed according to the codes and standards governing the individual safety classes.

Pertinent aspects of the design and qualification of the CRD components are discussed in the following locations: transients in Section 3.9.1.1B, faulted conditions in Section 3.9.1.4B, seismic testing in Section 3.9.2.2B, loading combinations and stress limits in Table 3.9B-2t.

3.9.4.3B Design Loads, Stress Limits, and Allowable Deformation

The ASME Code components and the CRDs have been evaluated analytically, and the design load combinations and stress limits are listed in Table 3.9B-2t. For the noncode components, experimental testing was used to determine the CRD performance under all possible conditions as described in paragraph 3.9.4.4B.

INSERT BELOW

3.9.4.4B CRD Performance Assurance Program

The CRD test program consists of the following tests:

1. Development tests
2. Factory Quality Control tests

The CRD piping has been designed to withstand the HCU floor dynamic response to the impact loadings associated with a LOCA pool swell event.

nozzle. The vessel top head nozzle is provided with a flange with large groove facing. The drain nozzle is of the full penetration weld design. The recirculation inlet nozzles (located as shown in Fig. 5.3-1), feedwater inlet nozzles, core spray inlet nozzles, and the LPCI nozzles all have thermal sleeves. Nozzles connecting to stainless steel piping have safe ends, or extensions made of stainless steel. These safe ends or extensions were welded to the nozzles after the pressure vessel was heat treated to avoid furnace sensitization of the stainless steel. The material used is compatible with the material of the mating pipe.

The nozzle for the standby liquid control pipe is designed to minimize thermal shock effects on the reactor vessel in the event that use of the standby liquid control system is required.

Insert A →

5.3.3.1.4.6 Materials and Inspections

The reactor vessel was designed and fabricated in accordance with the appropriate ASME Boiler and Pressure Vessel Code as defined in Section 5.2.1. Table 5.2-3 defines the materials and specifications. Section 5.3.1.6 defines the compliance with reactor vessel material surveillance program requirements.

5.3.3.1.4.7 Reactor Vessel Schematic (BWR)

The reactor vessel schematic is contained in Fig. 5.3-1. Trip system water levels are indicated as shown in Fig. 5.3-2.

5.3.3.2 Materials of Construction

All materials used in the construction of the RPV conform to the requirements of ASME Section II materials. The vessel heads, shells, flanges, and nozzles are fabricated from low alloy steel plate and forgings purchased in accordance with ASME specifications SA533 Grade B, Class 1 and SA508, Class 2. Special requirements for the low alloy steel plate and forgings are discussed in Section 5.3.1.2. Cladding employed on the interior surfaces of the vessel consists of austenitic stainless steel weld overlay.

These materials of construction were selected because they provide adequate strength, fracture toughness, fabricability, and compatibility with the BWR environment. Their suitability has been demonstrated by long term successful operating experience in reactor service.

The three LPCI inlets are each designed with a flow deflector to prevent horizontal flow impingement upon the core and instrument tubes. The flow deflectors are designed with a conical flow splitter which redirects the LPCI flow upward, downward and in the two horizontal directions tangential to the core. The flow deflectors are fabricated of 316C stainless steel plate material attached to the shroud wall by full penetration welds at the four corners of the deflector plate.

The design, analysis, and testing of the deflectors and its effects on the LPCI system parameters were considered in Reference 4.

In addition to the use of flow deflectors, the Intermediate Range Monitor (IRM) instrument tube nearest each LPCI inlet is reinforced.

RBS FSAR

References - 5.3

1. An Analytical Study on Brittle Fracture of GE-BWR Vessel Subject to the Design Basis Accident. NEDO-10029.

2. Watanabe, H. Boiling Water Reactor Feedwater Nozzle/Sparger Final Report. NEDO-21821-2, Proprietary Version, and NEDO-21821-2, Non-Propietary Version, August 1979.

3. Cooke, F. Transient Pressure Rises Affecting Fracture Toughness Requirements for Boiling Water Reactors, NEDO-21778-A, December 1978.

4. Artigas, Dr. R., Flow Deflector Effects on LPCI System Parameters and Plant Safety, to R. L. Tedesco (NRC), May 18, 1982.

RBS FSAR

QUESTION 210.77 (3.6.2)

Discuss how high energy leakage cracks were considered.

RESPONSE

The evaluations of pipe breaks and cracks are in accordance with Revision 0 of SRP 3.6.2 and Branch Technical Position MEB 3-1, November 24, 1975, the documents applicable at the time the evaluations were done. Therefore, high energy leakage cracks were not considered.

High energy line leakage cracks would not be postulated in fluid system piping located in containment penetration areas since the design stress and fatigue limits specified in BTP MEB 3-1, Section B.1.b, are met for high energy piping in these areas (see Section 3.6.2.1.5.2A).

The effects of high energy line leakage cracks in other areas would generally be bounded by the analyses performed for high energy line breaks, moderate energy line cracks, and inadvertent fire suppression system actuation.

RBS FSAR

QUESTION 210.46 (3.6A)

Figures 3.6A-12 Through 3.6A-19 and 3.6A-21 through 3.6A-33 - Provide a schedule for completion of pipe stress analyses as they effect pipe break location selections.

RESPONSE

The pipe stress analyses, and any resulting changes to Figures 3.6A-12 through 3.6A-19 and 3.6A-21 through 3.6A-33 will be ~~completed by December 1983.~~ updated during the third quarter 1984.

RBS FSAR

QUESTION 210.45 (3.6A)

Tables 3.6A-12 through 3.6A-20, 3.6A-27 Through 3.6A-42 and 3.6A-45 Through 3.6A-51 - Provide a schedule for completion of these tables.

RESPONSE

Tables 3.6A-12 through 3.6A-20 will be ~~completed by December 1983.~~ Tables 3.6A-27 through 3.6A-42, and 3.6A-45 through 3.6A-51 will be completed by January 1985.

INSERT BELOW

updated during the third quarter 1984.

QUESTION 210.22 (5.2.3.3.1)

Submit the fracture toughness data in tabular form including calibration of instruments and equipment.

RESPONSE

7 The response to this request ~~will be~~ is provided ~~by June 1983.~~ in Section 5.2.3.3.1.1 and Table 5.3-1.

exhibit 50 ft-lb absorbed energy and 35 mil lateral expansion at 60°F above the RT_{NDT}. The core beltline material meets 70 ft-lb absorbed upper shelf energy.

INSERT A

5.2.3.3.2 Control of Welding

5.2.3.3.2.1 Control of Preheat Temperature Employed for Welding of Low Alloy Steel (Regulatory Guide 1.50)

Regulatory Guide 1.50 delineates preheat temperature control requirements and welding procedure qualifications supplementing those in ASME Sections III and IX.

Welding of low alloy steel is restricted to the reactor pressure vessel. Other ferritic components in the RCPB to be welded are fabricated from carbon steel materials. New welding procedure qualifications were qualified with the test assembly preheat temperature maintained within 50 degrees above the minimum preheat temperature.

Preheat temperatures employed for welding of low alloy steel meet or exceed the recommendations of ASME Section III, Subsection NB. Components were either held for an extended time at preheat temperature to assure removal of hydrogen, or preheat was maintained until post weld heat treatment. The minimum preheat and maximum weld interpass temperatures were specified and monitored.

5.2.3.3.2.2 Control of Electroslag Weld Properties (Regulatory Guide 1.34)

No electroslag welding was performed on BWR components.

5.2.3.3.2.3 Welder Qualification for Areas of Limited Accessibility (Regulatory Guide 1.71)

Qualification for areas of limited accessibility is discussed in Section 5.2.3.4.2.3.

5.2.3.3.2.4 Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components (Regulatory Guide 1.43)

Reactor pressure vessel specifications require that all low alloy steel be produced to fine grain practice. The requirements of this regulatory guide are not applicable to the River Bend Station RCPB.

4. Fracture toughness data is provided in Table 5.3-1. The calibration of instruments and equipment such as temperature instruments and Charpy V-notch impact test machines used in impact testing is performed in accordance with ASME Code Section III, 1971 Edition, Summer 1973 Addenda, subsection NB-2360, "Calibration of Instruments and Equipment".

TABLE 5.3-1

RIVER BEND STATION UNIT 1 REACTOR VESSEL CHARPY TEST RESULTS
VESSEL BELTLINE CHEMICAL COMPOSITION

1) Vessel Plate (Beltline)

Heat Number	C	Mn	Si	P	S	Percent Ni	Percent Cu	Mo	V
C3138-2	0.19	1.37	0.25	0.012	0.015	0.63	0.08	0.58	-
C3058-1	0.19	1.30	0.26	0.007	0.020	0.70	0.09	0.57	-
C3058-2	0.19	1.30	0.26	0.007	0.012	0.70	0.09	0.57	-

2) Vessel Welds (Beltline)

Heat/Lot No.	C	Mn	Si	P	S	Percent Ni	Percent Cu	Mo	V
492L4871/ A421B27AE	0.07	1.06	0.37	0.018	0.025	0.95	0.04	0.50	0.02
492L4871/ A421B27AF	0.07	1.17	0.32	0.020	0.020	0.98	0.03	0.51	0.02
5P6756/0342(1)	0.078	1.24	0.53	0.010	0.012	0.92	0.09	0.46	0.006
5P6756/0342(2)	0.063	1.27	0.57	0.010	0.011	0.93	0.09	0.45	0.006

1) Vessel Plate (Beltline)

Heat Number	Start RT_{NDT} °F	R.G. 1.99 Extrap. RT_{NDT} °F	EOL RT_{NDT} °F	Transverse Charpy Upper Shelf (ft-lb)
C3138-2	+9	42	51	86, 74, 78
C3058-1	-20	35	15	94, 93, 93
C3058-2	+2	35	37	92, 102, 92

2) Vessel Welds (Beltline)

Heat/Lot No.	Start RT_{NDT} °F	R.G. 1.99 Extrap. RT_{NDT} °F	EOL RT_{NDT} °F	Transverse Charpy Upper Shelf (ft-lb)
492L4871/ A421B27AE	-60	64	4	151, 160, 161
492L4871/ A421B27AF	-50	71	21, limiting weld	126, 129, 136
5P6756/0342(1)	-50	42	-8	95, 99, 96
5P6756/0342(2)	-60	42	-18	89, 94, 91

(1) Tandem wire process
(2) Single wire process

QUESTION 210.25 (5.3.2.1)

Describe the procedures that will be used to update the actual fracture toughness limits, e.g., Figure 5.3-4.

RESPONSE

The response to this request ~~will be~~ is provided ~~by June 1983.~~ in Section 5.3.2.1 and Figure 5.3-4. 7

final heat treatment and threading, as specified in the guide, in accordance with NB-2583 of the applicable ASME Code.

Radial scan calibration is based on a 1/2-in (12.7 mm) diameter flat bottom hole of a depth equal to 10 percent of the material thickness. End scan calibration is in accordance with NB-2585. Angle beam examination is performed on the outer cylindrical surface of nuts and washer in accordance with ASME SA388 in both axial and circumferential directions. Any indication greater than the indication from the applicable calibration feature is unacceptable. A distance-amplitude correction curve in accordance with NB-2585 is used for the longitudinal wave examination.

In relationship to Regulatory Position C.3, GE practice allows exposure to stud bolting surfaces to high purity fill water; nuts and washers are dry stored during refueling.

5.3.2 Pressure-Temperature Limits

5.3.2.1 Limit Curves

The limit curves presented in Fig. 5.3-4 are based on the requirements of 10CFR50, Appendix G with the modification to Paragraph IV A.2.c in accordance with GE BWR Licensing Topical Report NEDO-21778-A. All the vessel shell and head areas remote from discontinuities plus the feedwater nozzles were evaluated, and the operating limit curves are based on the limiting location. The boltup limits for the flange and adjacent shell region are based on a minimum metal temperature of RT_{NDT}+60°F. The maximum through-wall temperature gradient from continuous heating or cooling at 100°F per hr was considered. The safety factors applied were as specified in ASME Code Appendix G and GE Licensing Topical Report NEDO-21778-A. ← INSERT B

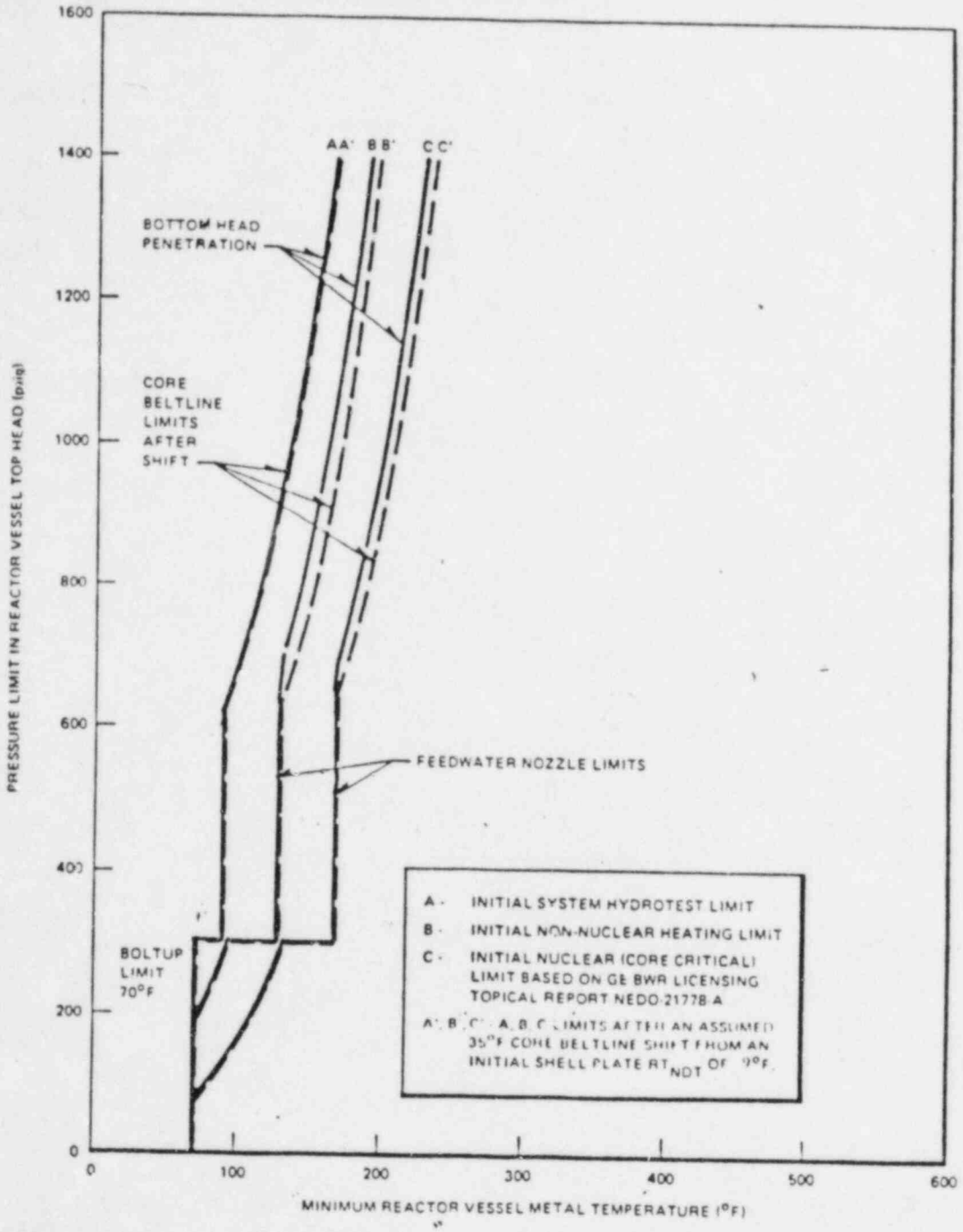
5.3.2.1.1 Temperature Limits for Boltup

A minimum temperature of 70°F is required for the closure studs for River Bend Station. A sufficient number of studs (up to 10 percent) may be fully tensioned at this minimum closure stud temperature to seal the closure flange O-rings for the purpose of raising the reactor water level above the closure flanges in order to assist in warming them. The flanges and adjacent shell are required to be warmed to minimum temperature of 70°F before they are stressed by the full intended bolt preload (all bolts fully tightened). The fully preloaded boltup limits are shown on Fig. 5.3-4.

Insert B

To account for overstress and thermal shock during anticipated operational occurrences in the control of reactivity, the surveillance of irradiated capsule specimen are tested at 10 years-first capsule and 30 years-second capsule, with a third capsule for standby. The procedure used to update the actual fracture toughness limits in Figure 5.3-4 is in accordance with 10CFR50, Appendix G and ASME Code Section III, Appendix G. Fracture toughness properties of ferritic materials are determined by performing Charpy V-notch tests, and when required dropweight tests, as specified in ASME Code Section III subsection NB-2321.2 and NB-2321.1, respectively. After testing, the RTndt level will be compared to the acceptable value as specified in 10CFR50, Appendix G (IV)(C). If the results are within the allowable limit, Figure 5.3-4 will not change.

Any shifts occurring in the core beltline transition temperature due to neutron irradiation, restricted will be plotted on Curves A, B and C. The operation of RBS would be if Curve C were to exceed the saturation limits.



Note: Curves A, B & C are predicted to apply as the limits for 20 years (16EFPY) of operation.

FIGURE 5.3-4

MINIMUM TEMPERATURES REQUIRED VERSUS REACTOR PRESSURE

RIVER BEND STATION

FINAL SAFETY ANALYSIS REPORT

ENCLOSURE 21

Response to NRC's
December 7, 1983 Letter

On March 4, 1982 the NRC Staff Mechanical Engineering Branch (MEB) and their consultants met with Gulf States Utilities Company (GSU) and the River Bend Station (RBS) architect engineer, Stone & Webster (SWEC), to perform an audit. During the audit, the design of RBS ASME equipment supports was reviewed and found to be in compliance with ASME Code Section III, Subsection NF, 1977 Edition up to and including the Winter 1978 Addenda. This design includes thermal and anchor displacement loads as primary loads.

The methodology and procedures used in the design of Seismic Category I ASME Code Class 1, 2 and 3 pipe supports are in accordance with the requirements of ASME Section III, Subsection NF, 1974 Edition including the Summer 1974 Addenda. The RBS FSAR discusses component support design including piping, in Section 3.9.3.4A and Table 3.9A-13.

Subsection NF of the 1974 Edition of the ASME Section III Code classifies the stresses induced in a pipe support from restraint of free end displacement of the piping system as secondary stresses. Explicit definitions of free end displacement, expansion stresses, and secondary stresses are given in articles NF-3213.10, 3213.11, 3222.2, Table NF-3217-1, and Figure 3221-1. The load conditions currently being used in compliance with the 1974 edition of the ASME Code and as delineated in the current FSAR Table 3.9A-13 are listed below, along with the corresponding plant service level conditions A-normal, B-upset, C-Emergency, and D-faulted.

Plant Service Level	RBS Load Condition	Decription
A	1	DL
B	2	DL+SRSS (OBEI, OCCU)
B	3	DL+THER+SRSS (OBET,OCCU)
C	4	DL+SRSS (OBEI,OCCE)
D	5	DL+SRSS (SSEI, OCCF)

Since the ASME Code limits for primary stresses in service level A are the same as for service level B (articles NF-3220, -3230, -3320, -3330) service level A loads do not govern the design of the Code Class 1, 2 and 3 pipe supports. Service level B loads are evaluated separately for primary loads (Load Condition 2) and for primary-plus-secondary loads (Load Condition 3). The Code does not require evaluation of secondary stresses for service levels C and D (article NF-3231.1(b) and (c)).

Tab A gives a summary of the allowable stresses used in designing structural members of linear type pipe supports.

Enclosure 21 (Cont'd)

TAB A

PIPE SUPPORT DESIGNS FOR RIVER BEND STATION

SUMMARY OF ALLOWABLE STRESSES FOR STRUCTURAL MEMBERS

Plant Service Levels A and B - RBS Load Conditions 1 and 2

Tension	$F_t = 0.60\ S_y$	Code Article XVII-2211
Shear	$F_v = 0.40\ S_y$	Code Article XVII-2212
Bending	$F_b = 0.60\ S_y$	Code Article XVII-2214
Axial Compression	Eq's 4&5	Code Article XVII-2213
Axial Comp + Bending	Eq's 19-21	Code Article XVII-2215

Plant Service Level B - RBS Load Condition 3

Tension (See Note 1)

Shear $F_v = 1.33 \times 0.40\ S_y$

Bending $F_b = 1.11 \times 0.60\ S_y$ (See Note 2)

Axial Compression $(2/3)\ S_y \{1-(Kl/r)^2/2(C_c)^2\}$ for $Kl/r \leq C_c$

and $(2/3)\ \pi^2\ E/(Kl/r)^2$ for $200 \geq Kl/r > C_c$

Axial Comp. + Bending Eq's 19-21 Code Article XVII-2215
with 1.11 factor applied to allowable bending stress.

Plant Service Levels C and D - RBS Load Conditions 4 and 5

The allowable stresses used for load conditions 4 and 5 are the same as those for load condition 3.

Enclosure 21 (Cont'd)

NOTE 1

The Code permits use of three times the stress limits of XVII-2000 for stresses due to primary and secondary loads combined. However, the support members are normally subjected to both tension and compression loads in which case compression allowable becomes the limiting design stress. Therefore, this stress limit is used for load condition 3 in place of the tensile stress allowable.

NOTE 2

Critical buckling stress is the stress which results in column buckling in compression members, or in local buckling in compression flanges and webs of flexural members. Since critical buckling is a function of the length and cross sectional shape of the member, detailed analysis may be cumbersome. Therefore, a factor of 1.11 applied conservatively to all shapes assures that 2/3 of critical buckling stress is never exceeded. The buckling criterion implemented in this manner generally governs the design of pipe support members for all service conditions instead of the stress factors allowed by the Code.