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PERRY NUCLEAR POWER PLANT PROJECT ORGANIZATION
CEI NUCLEAR ENGINEERING DEPARTMENT
AND KAISER ENGINEERS, INC.

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U.S. NUCLEAR REGULATORY COMMISSION

July 24, 1981

Mr. Dean Houston
U. S. Nuclear Regulatory Commission
Office of Nuclear Regulation
Washington, D. C. 20555

Re: MEB SER

Dear Mr. Houston:

Attached is our preliminary response to open items identified in the Draft MEB SER for sections 3.2, 3.6, 3.7, and 3.9. These responses will be used as a starting point for our August 4 meeting.

Upon completion of the meeting, these responses will be finalized and appropriate FSAR changes will be made.

Very truly yours,

W E Coleman

W. E. Coleman
Senior Licensing Engineer

WEC:dlp

cc: Mr. D. Terao (NRC)
Mr. G. Beeman (PNL)

B001
S 1/1

Note to Reviewers

1. A copy of Appendix A to "Request for Resolution of Open Issues - Mechanical Engineering" has been provided with this package with DSER question numbers assigned in the margin.

2. Many of the responses to questions have been answered by providing revised text pages. Where appropriate, question numbers have been provided in the margins to facilitate review.

3. Changes have been provided in Section 3.6 based on the current design. These changes appear with side-bars and no question reference.

APPENDIX

QUESTIONS ON PERRY FSAR

QUESTIONS ON PERRY FSAR

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

3.2.1 Seismic Classification

3.2.1, Page 3.2-1

3.2-1 It states in the FSAR that structures, components and systems designated as Safety Class 1, 2, or 3 are classified as Seismic Category I except for some portions of the radioactive waste treatment handling and disposal systems. There are several items in Table 3.2-1 in conflict with this statement.

3.2.1, Page 3.2-2

-2 "The seismic classification indicated in Table 3.2-1 meets the requirements of Regulatory Guide 1.29." It is also stated in Section 1.8 that the Perry plant complies with all the requirements of Regulatory Guide 1.29. Does this mean that seismic Category I cooling water is provided to the recirculation pump during normal operation and following LOCA?

Table 3.2-1, Page 3.2-9

-3 Quality assurance requirements should be addressed in this table.

Table 3.2-1, Page 3.2-9

-4 What design requirements were used in the design of the reactor pressure vessel skirt?

Table 3.2-1, Page 3.2-9

-5 Justify the non-seismic classification of the control rods. Note 7 does not apply to the control rods.

Table 3.2-1, Page 3.2-9

-6 Provide an explanation for the "I, NA" seismic classification for relief valve discharge piping.

Table 3.2-1, Page 3.2-10

-7 How much of the main steam piping, between the M.O. stop valve and the turbine stop valve, is located in the Auxillary Building?

Table 3.2-1, Page 3.2-24

-8 There appears to be a discrepancy in the seismic classification of the discharge tunnel. The discharge tunnel and the diffusor nozzle are seismic Category I. The tunnel entrance structure and downshaft are not. Provide clarification for this apparent contradiction.

Table 3.2-1, Page 3.2-25

-9 What is the seismic classification of the Containment Vessel Cooling Units?

Table 3.2-1, Page 3.2-34

-10 Note 19 is an exception to Regulatory Guide 1.29 and should be included in Section 1.8.

3.6 PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.2, Page 3.6-7

3.6-1 In Section 3.6.1 references are made to "elastic/plastic pipe whip restraints or pipe supports which eliminate pipe whip damage". Details of how pipe supports are designed for pipe whip protection and an example of such an analysis are needed.

3.6.2.1.4, Page 3.6-10

-2 How is it determined that "The internal energy associated with whipping is insufficient to impair the safety function of any structure, system or component to an unacceptable level"?

[redacted]

-3

3.6.2.1.5, Page 3.6-11

-4 Plant loading conditions for evaluating pipe break are to include normal and upset conditions plus an OBE. Assurance must be provided that SRV discharge loads are included in the upset conditions.

3.6.2.1.5, Page 3.6-11

-5 For ASME, Section III, Class 1 piping designed to seismic Category I standards, breaks due to stress are to be postulated at the following locations:

(1) If Eq. (10), as calculated by Paragraph NB-3653, ASME Code Section III, exceeds 2.4 S_m, then Eqs. (12) and (13) must be evaluated. If either Eq. (12) or (13) exceeds 2.4 S_m, a break must be postulated. In other words, a break is postulated if

$$\text{Eq. (10)} > 2.4\ S_m \text{ and Eq. (12)} > 2.4\ S_m$$

or

$$\text{Eq. (10)} > 2.4\ S_m \text{ and Eq. (13)} > 2.4\ S_m$$

(2) Breaks must also be postulated at any location where the cumulative usage factor exceeds 0.1.

The above criteria is evaluated under loadings resulting from normal and upset plant conditions including the OBE.

Any deviations from the above criteria must be justified.

3.6.2.1.5, Page 3.6-11

Are there any high energy Class 2, Class 3, or B31.1 lines? If so, what criteria is used for postulating breaks in these lines? -6

3.6.2.1.6, Page 3.6-13

Any instances where longitudinal break areas are less than one circumferential pipe area must be identified. The analytical methods representing test results and based on a mechanistic approach must be explained or justified. Provide examples of a typical analysis. -7

3.6.2.1.6, Page 3.6-14

How are energy reservoirs of sufficient capacity to develop a jet flow determined? What are justifiable line restrictions? Provide the justification. Any instances where flow limiters are used should be identified and justified. -8

3.6.2.1.7.1, Page 3.6-15

For ASME, Section III, Class 1 piping designed to seismic Category I standards, breaks need not be postulated providing the following stress criteria is met.

(1) If Eq. (10) as calculated by Paragraph NB-3653, ASME Code, Section III does not esceed 2.4 S_m, a break need not be postulated.

(2) If Eq. (10) does exceed 2.4 S_m, Then Eqs. (12) and (13) must be evaluated. If neither Eq. (12) or (13) exceeds 2.4 S_m, a break need not be postulated. In other words, a break need not be postulated if

$$\text{Eq. (10)} < 2.4\ S_m$$

or

-9

$$\text{Eq. (10)} > 2.4\ S_m \text{ and Eq. (12)} < 2.4\ S_m$$
$$\text{and Eq. (13)} < 2.4\ S_m$$

(3) Breaks need not be postulated as long as the cumulative fatigue usage factor is less than 0.1.

(4) For plants with isolation valves inside containment, the maximum stress, as calculated by Eq. (9) in ASME Code Section III, Paragraph NB-3652 under the loadings of internal pressure, dead weight and a postulated piping failure of fluid systems upstream or downstream of the containment penetration area must not exceed 2.25 S_m.

The above criteria is evaluated under loadings resulting from normal and upset plant conditions including the OBE.

In addition, augmented inservice inspection is required on all piping in the break exclusion area.

The applicant must provide assurances that their criteria for piping in the break exclusion areas complies with the requirements outlined above and those of Standard Review Plan 3.6.2.

3.6.2.1.7.1, Page 3.6-15

-10 Are there any Class 2, Class 3 or B31.1 piping in the break exclusion areas? If so, what criteria is used for their design?

3.6.2.1.7.1, Page 3.6-15

-11 A list of all systems in the break exclusion area is needed. Break exclusion area should be shown on the appropriate piping drawings.

3.6.2.1.7.2, Page 3.6-15

-12 Provide an example of the detailed stress analysis done on a welded attachment to the process pipe. In addition, provide details of the stress analysis done on the head fitting for the main steam line.

3.6.2.2.1, Page 3.6-17

-13 Provide a list of all locations where limited break opening areas have been used. Provide justification for each location and details of any inelastic analysis used.

3.6.2.2.1, Page 3.6-17

-14 Provide a list of all locations where break opening times greater than one millisecond have been used. Provide and justify any experimental data and analytical theory.

3.6.2.2.2, Page 3.6-20

-15 Provide assurance that all potential targets are evaluated when considering pipe whip.

3.6.2.2.2, Page 3.6-20

-16 Provide a definition for limits of strain which are similar to strain levels allowed in restraint plastic members.

3.6.2.2.2, Page 3.6-20

"Piping systems are designed so that plastic instability does not occur
in the pipe at the design dynamic and static loads unless damage studies
-17 are performed which show the consequences do not result in direct damage
to any essential system or component." Provide a list of where this
technique has been used and an example of the studies performed.

-18 [redacted]

3.6.2.3.1, Page 3.6-23

It is the staff's position that when evaluating jet impingement loads
<u>all</u> potential targets must be evaluated. Provide assurances that your
-19 analysis for jet impingement effects have included all possible targets.

3.6.2.3.1, Page 3.6-29

What service limits are used for piping when evaluating jet impingement
-20 loads?

3.6.2.3.1, Page 3.6-30

How is it determined that the dynamic load factor (DLF) is suitable?
-21 Provide an example of its use.

3.6.2.3.1, Page 3.6-30

For snubbers, what are the "other simultaneous loads" that are combined
-22 by the SRSS method?

3.6.2.3.3, Page 3.6-33

-23 "Piping integrity usually does not depend upon the pipe whip restraints for any loading combination." List all those locations where integrity of the piping depends upon the pipe whip restraints.

3.6.2.3.3, Page 3.6-33

-24 What service limits are used in the design of the pipe whip restraints?

3.6.2.3.3.1, Page 3.6-33

-25 What critical locations inside containment are monitored during hot functional testing?

3.6.2.3.3.1, Page 3.6-40

-26 Any locations where the increase in the yield or ultimate strengths, of the material used for pipe whip restraints, exceeds 10% must be identified. Justification for any increase greater than 10% must also be provided.

3.6.2, Tables

-27 Provide a schedule for the completion of any table that is incomplete.

3.6.2, Figure 3.6-66

-28 Are all postulated break locations in the recirculation system shown?

3.6.2, Figures 3.6-71, 3.6-73, 7.3-74, 3.6-77, 3.6-78, 3.6-79, 3.6-80

-29 Where are breaks postulated in these figures?

3.6.2, Figure 3.6-75

-30 Indicate the location of valves in this line.

3.7.3 Seismic Subsystem Analysis

3.7 2.1.2.5, Page 3.7-11

The discussion on "Different Seismic Movement of Interconnected Components" requires some clarification. "The stresses thus obtained for
3.7-1 each natural mode are then superimposed for all modal displacements of the structure by the SRSS (square root sum of the squares) method." Provide an example of what was done here.

3.7.2.1.2.5, Page 3.7-11

-2 What criteria was used to determine whether or not a mode was significant?

3.7.2.1.2.5, Page 3.7-11

"When a component is covered by the ASME Boiler and Pressure Vessel Code, the stresses due to relative displacement as obtained above are
-3 treated as secondary stresses." Does this statement pertain to piping or supports?

3.7.3.1.1, Page 3.7-20

"Seismic analyses were performed for those subsystems that could be
-4 modeled to correctly predict the seismic response." What procedure was used for the other systems? Provide an example of some of those systems.

3.7.3.1.1, Page 3.7-21

-5 What is meant by "Closely spaced in phase modes"?

3.7.3.2.1, Page 3.7-21

-6 How many stress cycles are used in the BOP design?

-7 [redacted]

3.7.3.3.2.1, Page 3.7-23

Part (a) discussing decoupling of main steam and branch lines is not
-8 a criteria.

3.7.3.3.2.2, Page 3.7-24

Mention is made of using 33 hertz as a frequancy cutoff for seismic analysis. At some point in the FSAR the applicant must address the
-9 frequencies of 50 to 60 hertz and greater than come from the suppression pool hydrodynamics.

3.7.3.5, Page 3.7-25

"For flexible equipment, the equivalent static load is taken as the product of 1.5 times the equipment mass and the peak floor response spectrum
-10 value." Regulatory Guide 1.100 allows the use of the 1.5 factor for verifying the integrity of frame type structures. For equipment having configurations other than a frame type structure, justification is required for use of the 1.5 factor.

3.7.3.7.1, Page 3.7-26

What procedure is used for combining closely spaced modes of systems
-11 in the BOP scope?

3.7.3.7.2, Page 3.7-26

The referenced equation should be as follows

$$R = \left[\sum_{K=1}^{N} \sum_{S=1}^{N} \left| R_K \, R_S \right| \varepsilon_{KS} \right]^{1/2}$$

-12

3.7.3.8.1, Page 3.7-28

Justification must be provided that the modeling of valves with off-
-13 set motor operators is detailed enough to provide acceleration values to be used for valve qualification.

3.7.3.8.1, Page 3.7-28

"In addition, the effects of the modes not included are added to the
-14 SRSS response as one term, using the acceleration at the highest frequency from the SRSS response under 33 hertz to obtain the total response." Provide an example of what was done here.

Table 3.7-11, Page 3.7-54

-15 Provide a detailed explanation of the information in this table.

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9, Page 3.9-1

Any references to the ASME Boiler and Pressure Vessel Code should
3.9-1 indicate what part is being referenced.

3.9.1.2, Page 3.9-1

Methods of verification are required for all NSSS computer codes
-2 used in the analysis.

3.9.1.2.6, Page 3.9-16

All computer programs used in the design and analysis of systems and components within the BOP scope must be listed. Methods of verifica-
-3 tion are required for all BOP programs.

3.9.1.4.12, Page 3.9-26

It is stated that elastic-plastic methods of analysis may be used
-4 for some components. We would like to review the analysis procedures that would be used if an elastic-plastic analysis was done.

3.9.2, Page 3.9-27

More detail is needed for the NSSS and BOP preoperational vibration testing
-5 program. What locations will be monitored. What types of instrumentation will be used. What are the actual values that will be used for deflection and stress limits.

The staff's position is that acceptance limits for vibration should be based on half of the endurance limit as defined by the ASME Code at 10^6 cycles. We will require a copy of your results from your preoperational vibration testing program.

3.9.2.1.2, Page 3.9-29

"The piping system does 'shakedown' after a few thermal expansion
-6 cycles." Provide an explanation of this statement.

3.9.2.4, Page 3.9-65

"In addition to the above components, vibration measurements of the core spray sparger will be measured during preoperational testing of
-7 that system at the designated prototype 251 BWR/6 plant (Grand Gulf)." Show how this is applicable to Perry.

3.9.2.4.1, Page 3.9-66

Provide a commitment that Perry will be in compliance with Regulatory
-8 Guide 1.20 for prototype reactors.

3.9.2.5, Page 3.9-67

"These periods will be determined from a comprehensive dynamic model
-9 of the RPV and internals with 12 degrees of freedom." It is not clear what is actually done here. How can a model be comprehensive and have only 12 degrees of freedom?

3.9.2.6, Page 3.9-68

It appears that some results from Grand Gulf will be used in the evaluation and qualification of the reactor internals at Perry. Show
-10 that the similarity between the two sets of internals is sufficient to allow direct comparisons.

3.9.3, Page 3.9-68

Several references are made throughout this section to allowable stresses for bolting. Specifically, what allowable stress limits are
-11 used for bolting for (a) equipment anchorage, (b) component supports, and (c) flanged connections? Where are these limits defined?

3.9.3.1.2, Page 3.9-78

-12 Are there any Class 1 systems in the BOP scope of responsibility?

3.9.3.4.1, Page 3.9-107

"For the NSSS scope of supply, all valve operators which are mounted on Class 1 piping will not be used as attachment points for component
-13 supports." What about Class 2 and 3 piping? This question also applies to the BOP scope of responsibility.

3.9.3.4.1, Page 3.9-109

-14 Provide more detail on the testing done on snubbers.

3.9.3.4.4, Page 3.9-112

What elastic-plastic analysis has been done on supports? Provide
-15 an example of this analysis.

3.9.4.3, Page 3.9-114

Reference is made to allowable deformation in the title of this
-16 section but there is no discussion of allowable deformations in the text.

3.9.5.1.1.8, Page 3.9-120

Recently, cracking has been observed in BWR jet pump holddown beams. The resolution of this problem may affect the design or testing of the
-17 Perry jet pumps (see I&E Bulletin 80-07).

3.9.5.1.1.10, Page 3.9-121

What feedwater sparger design and Control Rod Drive Return Line modifications is used at Perry? Provide a commitment
-18 to NUREG-0619.

3.9.5.3.3, Page 3.9-129

-19 Have the reactor internals placed in the "other internals" category been seismically analyzed to show that they will not compromise the integrity of seismically qualified reactor internals?

3.9.6, Page 3.9-131

-20 There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to pretect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure systems. The leak-tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems thus causing an intersystem LOCA.

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

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

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

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

Leak rates higher than 1 GPM will be considered if the leak rate changes are below 1 GPM above the previous test leak rate or system design precludes measuring 1 GPM with sufficient accuracy. These items will be reviewed on a case by case basis.

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

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

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

Table 3.9-1, Page 3.9-134

-21 Does this table apply to Perry?

Table 3.9-1, Page 3.9-135

-22 What does "1*****" refer to?

Table 3.9-1, Page 3.9-135

-23 How many ADS cycles are included in the design of Perry?

Table 3.9-1, Page 3.9-136

-24 Standard Review Plan 3.9 requires 5 OBEs of 10 cycles each. If fewer cycles are used, justification must be provided.

Table 3.9-3, Page 3.9-141

-25 The acceptance criteria should reference the ASME Code Service Limits. A similar table is needed for the BOP.

Table 3.9-3a, Page 3.9-143

-26 "The results of stress and fatigue usage analysis are given in detail in the vessel manufacturer's stress report and in new loads evaluation by GE within the code limits." Provide clarification of this statement.

Table 3.9.3m, 3.9.3o, 3.9.3q and 3.9.3h

-27 Some values in these tables are missing. Provide a schedule for their completion.

Table 3.9-3s, Page 3.9-225

-28 Provide an explanation for the results in this table.

Table 3.9-28, Page 3.9-282

-29 Where are the loads used in this table defined? How are these loads combined?

Table 3.9-32, Page 3.9-297

-30 Has Eq. b) been used? If so, provide the supporting data. If not, delete the equation from the table.

Table 3.9-33, Page 3.9-298

-31 Have Eqs. e), f), or g) been used? If so, provide the supporting data. If not, delete these equations from the table.

Table 3.9.34, Page 3.9-301

-32 Has Eq. c) been used. If so, provide the supporting data. If not, delete the equation from the table.

ADDITIONAL QUESTIONS

Table 3.2-1, Page 3.2-9

3.2-11 What design requirements were used in the design of the core support structures?

3.6.2.1.6, Page 3.6-13

3.6-31 Regardless of the ratio of longitudinal to hoop stress, both a longitudinal split and a circumferential break should be postulated at any location where the cumulative usage factor is greater than 0.1.

3.9.1.1.1, Page 3.9-1

3.9-33 How many cycles due to SRV discharge are included in the analysis?

3.9.2.5, Page 3.9-67

Previous analyses for other nuclear plants have shown that certain reactor system components and their supports may be subjected to previously under-estimated asymmetric loads under the conditions that result from the postulation of ruptures of the reactor coolant piping at various locations.

-34 The applicant has described the design of the reactor internals for blowdown loads only. The applicant should also provide information on asymmetric loads. It is, therefore, necessary to reassess the capability of these reactor system components to assure that the calculated dynamic asymmetric loads resulting from these postulated pipe ruptures will be within the bounds necessary to provide high assurance that the reactor can be brought safely to a cold shutdown condition. The reactor system components that require reassessment shall include:

a. Reactor pressure vessel
b. Core supports and other reactor internals
c. Control rod drives
d. ECCS piping that is attached to the primary coolant piping
e. Primary coolant piping
f. Reactor vessel supports

The following information should be included in the FSAR about the effects of postulated asymmetric LOCA loads on the above mentioned reactor system components and the various cavity structures.

1. Provide arrangement drawings of the reactor vessel support systems in sufficient detail to show the geometry of all principal elements and materials of construction.

2. If a plant-specific analysis will not be submitted for your plant, provide supporting information to demonstrate that the generic plant analysis under consideration adequately bounds the postulated accidents at your facility. Include a comparison of the geometric, structural mechanical, and thermal-hydraulic similarities between your facility and the case analyzed. Discuss the effects of any differences.

3. Consider all postulated breaks in the reactor coolant piping system, including the following locations:

   a. Steam line nozzles to piping terminal ends.
   b. Feedwater nozzle to piping terminal ends.
   c. Recirculation inlet and outlet nozzles to recirculation piping terminal ends.

   Provide an assessment of the effects of asymmetric pressure differentials* on the systems and components listed above in combination with all external loadings including safe shutdown earthquake loads and other

*Blowdown jet forces at the location of the rupture (reaction forces), transient differential pressures in the annular region between the component and the wall, and transient differential pressures across the core barrel within the reactor vessel.

faulted condition loads for the postulated breaks described above. This assessment may utilize the following mechanistic effects as applicable:

a. Limited displacement -- break areas
b. Fluid-structure interaction
c. Actual time-dependent forcing function
d. Reactor support stiffness
e. Break opening times.

4. If the results of the assessment on item 3 above indicate loads leading to inelastic action of these systems or displacement exceeding previous design limits, provide an evaluation of the inelastic behavior (including strain hardening) of the material used in the system design and the effect of the load transmitted to the backup structures to which these systems are attached.

5. For all analyses performed, include the method of analysis, the structural and hydraulic computer codes employed, drawings of the models employed and comparisons of the calculated to allowable stresses and strains or deflections with a basis for the allowable values.

6. Demonstrate that safety-related components will retain their structural integrity when subjected to the combined loads resulting from the loss-of-coolant accident and the safe shutdown earthquake.

7. Demonstrate the functional capability of any essential piping when subjected to the combined loads resulting from the loss-of-coolant accident and the safe shutdown earthquake.

The applicant has outlined his approach for determining the forcing functions considered in the system and component dynamic analyses of reactor structures for normal operation and anticipated transients. These methods are a combination of analytical methods and predictions based on data from previously tested reactor internals of a similar design. The forcing function information is combined with dynamic modal analysis to form a basis for interpretation of the pre-operational and initial startup test results. Modal stresses are calculated and relationships are obtained between sensor responses and peak component stresses for each of the lower modes.

3.9.3.3-2, page 3.9-106

Provide justification for using a modified static ~~anlaysis~~ analysis on the safety relief valve piping in the suppression pool and explain what is used for the "conservative dynamic load factor" in the analysis.

-35 Provide the time-history transient forces resulting from the SRV actuation used in the SRV piping and support design including the loads developed from the discharging water slug.

Discuss the types of supports used on the SRV piping in both the drywell and suppression pool and provide drawings of the supports.

Provide the type of safety relief valves used in the plant , the valve opening time, and the sequences of valve actuation used in the analysis.

3.9.3.4.6, page 3.9-113

-36 Are the stress due to differential anchor movements considered as primary or secondary stresses for BOP supports?

DSER 3.2-1 (3.2.1, Page 3.2-1)	It states in the FSAR that structures, components and systems designated as Safety Class 1, 2, or 3 are classified as Seismic Category I except for some portions of the radioactive waste treatment handling and disposal systems. There are several items in Table 3.2-1 in conflict with this statement.

Response

The response to this question is provided in revised Section 3.2.1 and Table 3.2-1.

DSER 3.2-2 (Table 3.2.1, Page 3.2-2

"The seismic classification indicated in Table 3.2-1 meets the requirements of Regulatory Guide 1.29." It is also stated in Section 1.8 that the Perry plant complies with all the requirements of Regulatory Guide 1.29. Does this mean that Seismic Category I cooling water is provided to the recirculation pump during normal operation and following LOCA?

<u>Response</u>

The response to this question is provided in revised Section 3.2.1 and Note 19 to Table 3.2-1.

DSER 3.2-3 (Table 3.2-1, Page 3.2-9) Quality assurance requirements should be addressed in this table.

Response

Quality Assurance requirements are addressed in revised Note 2 to Table 3.2-1 and Section 3.2.4.

DSER 3.2-4 (Table 3.2-1, Page 3.2-9)	What design requirements were used in the design of the reactor pressure vessel skirt?

Response

The response to this question is provided in revised Table 3.2-1.

DSER 3.2-5 (Table 3.2-1, Page 3.2-9)	Justify the non-seismic classification of the control rods. Note 7 does not apply to the control rods.

Response

The response to this question is provided in revised Table 3.2-1.

DSER 3.2-6 (Table 3.2-1, Page 3.2-9)

Provide an explanation for the "I, NA" seismic classification for relief valve discharge piping.

Response

The response to this question is provided in revised Table 3.2-1.

DSER 3.2-7 (Table 3.2-1, Page 3.2-10)	How much of the main steam piping, between the M.O. stop valve and the turbine stop valve, is located in the Auxiliary Building?

Response

A length of 4'-8-3/8" of main steam piping is located in the Auxiliary Building. Refer to revised Figure 10.3-1.

DSER 3.2-8 (Table 3.2-1, Page 3.2-24)	There appears to be a discrepancy in the seismic classification of the discharge tunnel. The discharge tunnel and the diffuser nozzle are Seismic Category I. The tunnel entrance structure and downshaft are not. Provide clarification for this apparent contradiction.

Response

The discharge tunnel and diffuser nozzle are Seismic Category I because the alternate emergency service water system intake is through the discharge tunnel and diffuser nozzle. Refer to Figures 3.8-65 and 3.8-70 for clarification.

DSER 3.2-9 (Table 3.2-1, Page 3.2-25)	What is the seismic classification of the Containment Vessel Cooling Units?

Response

The response to this question is provided in revised Table 3.2-1.

DSER 3.2-10 (Table 3.2-1, Page 3.2-34)	Note 19 is an exception to Regulatory Guide 1.29 and should be included in Section 1.8.

Response

The response to this question is provided in revised Table 1.8-1.

DSER 3.2-11 (Table 3.2-1, Page 3.2-9)	What design requirements were used in the design of the core support structures?

Response

The response to this question is provided in revised Table 3.2-1.

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.;RRRC Category)	Degree of Conformance	Reference
1.29 - (Revision 3 - 9/78;RRRC Cat. 1)		
Seismic design classification	PNPP design conforms to this guide, with exception as stated in Note 19 of Table 3.2-1	3.2.1, 3.8, 8.1.1
1.30 - (Revision 0 - 8/72;RRRC Cat. 1)		
Quality assurance requirements for the installation, inspection, and testing of instrumentation and electrical equipment	PNPP conforms to this guide to the extent required by ANSI N18.7-1976.	17.2
1.31 - (Revision 3 - 5/78;RRRC Cat. 1)		
Control of ferrite content in stainless steel weld metal	Conformance evaluation was based on an extensive test program which demonstrates that controlling weld filler metal ferrite at 5% minimum produces production welds which meets the regulatory requirements. All austenitic stainless steel weld filler material for PNPP is supplied with a minimum of 5% ferrite material.	3.8.3, 4.5.1, 4.5.2, 5.2.3
1.32 - (Revision 2 - 2/77;RRRC Cat. 1)		
Criteria for safety-related electric power systems for nuclear power plants	The design of the PNPP Class IE power system conforms to IEEE Standard 308-1974 as modified by the positions of Regulatory Guide 1.32	7.1.2, 8.1.1
1.33 - (Revision 2 - 3/78;RRRC Cat. 1)		
Quality assurance program requirements (operations)	PNPP Project conforms to this guide.	17.2

-0

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Certain structures, components and systems of the nuclear plant are considered important to safety because they perform safety actions required to avoid or mitigate the consequences of abnormal operational transients or accidents. The purpose of this section is to classify structures, components and systems according to the importance of the safety function they perform. In addition, design requirements are placed upon such equipment to assure the proper performance of safety actions, when required.

3.2.1 SEISMIC CLASSIFICATION

Plant structures, systems, and components important to safety are designed to withstand the effects of a Safe Shutdown Earthquake (SSE) and remain functional if they are necessary to assure:

a. The integrity of the reactor coolant pressure boundary,

b. The capability to shut down the reactor and maintain it in a safe condition, or

c. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 100.

Plant structures, systems, and components (including their foundations and supports) designed to remain functional in the event of a SSE are designated as Seismic Category I, as indicated in Table 3.2-1.

Structures, components, equipment, and systems designated as Safety Class 1, Safety Class 2, or Safety Class 3 (see Section 3.2.3 for a discussion of safety classes) are classified as Seismic Category I except for (1) those noted in Table 3.2-1 and (2) those portions of the radioactive waste treatment handling and disposal systems whose postulated simultaneous failure would not result in conservatively calculated offsite exposures comparable to the guideline exposures of 10 CFR 100. 1

TABLE 3.2-1

EQUIPMENT CLASSIFICATION

Principal Component[1]	Safety Class[2]	Location[3]	Quality Group Classification[4]	Principal Construction Code[5]	Seismic Category[6]	Comment
I. Reactor System						
1. Reactor vessel	1	D	A	III-1	I	
2. Reactor vessel support skirt	1	D	N/A	III-1*	I	
3. Reactor vessel appurtances, pressure retaining portions	1	D	A	III-1	I	
4. CRD housing supports	2	D	N/A	None	I	
5. Reactor internal structures, engineered safety features	2	D	N/A	None	I	
6. Reactor internal structures, other	NSC	D	N/A	None	N/A	
7. Control Rods	2	D	N/A	None	I	
8. Control rod drives	2	D	N/A	None	I	(10)
9. Core support structure	2	D	N/A	III-NG	I	
10. Fuel assemblies	2	D	N/A	None	I	
II. Nuclear Boiler System						
1. Vessels, level instrumentation condensing chambers	1	D	A	III-1	I	
2. Vessels, air accumulators	2	-	B	III-2	I	
3. Piping, relief valve discharge	3	C/D	C	III-3	I	(7)
4. Piping, main steam, within outermost isolation valve	1	D	A	III-1	I	
5. Piping, feedwater within outermost isolation valve	1	C/D	A	III-1	I	
6. Pipe supports, main steam	1	D	A	III-1	I	
7. Pipe restraints, main steam	2	D	N/A	N/A	I	
8. Piping, main steam, between isolation valve and M.O. stop valve	2	A	B	III-2	I	

*ASME Code III-1, 1971 Edition, up to and including Winter 1972 addenda.

TABLE 3.2-1 (Continued)

Principal Component(1)	Safety Class(2)	Location(3)	Quality Group Classification(4)	Principal Construction Code(5)	Seismic Category(6)	Comment
7. Valves, other	2	A	B	III-2	I	(8)
8. Turbine	2	A	N/A	None	I	(12)
9. Electrical modules, with safety function	2	M,A,C	N/A	None	I	
10. Cable, with safety function	2	M,A,C	N/A	N/A	I	
XIII. Fuel Service Equipment	(See Chapter 9, Table 9.1-3)					
~~1. Fuel preparation machine~~	~~3~~	~~M~~	~~N/A~~	~~None~~	~~I~~	
~~2. General purpose grapple~~	~~2~~	~~M~~	~~N/A~~	~~None~~	~~I~~	
XIV. Reactor Vessel Service Equipment	(See Chapter 9, Table 9.1-5)					
~~1. Steam line plugs~~	~~NSC~~	~~C~~	~~N/A~~	~~None~~	~~N/A~~	
~~2. Dryer and separator sling and head strongback~~	~~2~~	~~C~~	~~N/A~~	~~None~~	~~N/A~~	
XV. In-Vessel Service Equipment						
1. Control rod grapple	2	C	N/A	None	N/A*	
XVI. Refueling Equipment						
1. Refueling equipment assembly platform	2	C/M	N/A	None	I	
2. Refueling bellows	NSC	D/C/M	N/A	None	N/A	
3. ~~Fuel transfer tube~~	~~2~~	~~D/C/M~~	~~B~~	~~III-2~~	~~I~~	(13)
4. Fuel transfer system	~~2~~	~~C/M~~	~~N/A~~	~~None~~	~~I~~	(13)
	(See Chapter 9, Table 9.1-4)					

* The control rod grapple is classified as N/A, i.e., exempt from seismic evaluation because it is suspended from a cable which mitigates the seismic effect.

TABLE 3.2-1 (Continued)

Principal Component[1]		Safety Class[2]	Location[3]	Quality Group Classification[4]	Principal Construction Code[5]	Seismic Category[6]	Comment
XXXV.	Heating, Cool ng and Ventilation Systems						
1.	Annulus Exhaust Gas Treatment System Units	2	M	N/A	AMCA, ERDA 76-21 UL-586, ANSI N509 RDT M 16-IT, ANSI N101.1	I	
2.	Drywell Cooling Units	NSC	D	N/A	AMCA, ARI-410	N/A	
3.	Containment Vessel Cooling Units	NSC	C	N/A	AMCA, ARI-410	N/A	
4.	Purge Supply Units	NSC	M	N/A	AMCA, ARI-410	N/A	
5.	Purge Exhaust Units	NSC	M	N/A	AMCA, ERDA 76-21 UL 586, ANSI N509 RDT M16-1T, ANSI N101.1	I	
6.	Piping & Isolation Valves from Containment Vessel through outer Isolation Valves	2	C,E	B	III-2	I	
7.	(ECCS) Pump Rooms Cooling Units	3	A	C	AMCA, ARI-410, III-3	I	
8.	Emergency Closed Cooling Pump Area Cooling Units	3	M	C	AMCA, ARI-410, III-3	I	
9.	Radwaste Building Supply Units	NSC	W	N/A	AMCA, ARI-410	I	

TABLE 3.2-1 (Continued)

NOTES:

1. A module is an assembly of interconnected components which constitute an identifiable device or piece of equipment. For example, electrical modules include sensors, power supplies, and signal processors and mechanical modules include turbines, strainers, and orifices.

2. 1, 2, 3 - safety classes defined in Section 3.2.3
NSC - No Safety Class
All Safety class 1, 2 or 3 systems and components meet the quality assurance requirements of 10CFR50, Appendix B. Additional details of the quality assurance program are provided in Section 3.2.4.

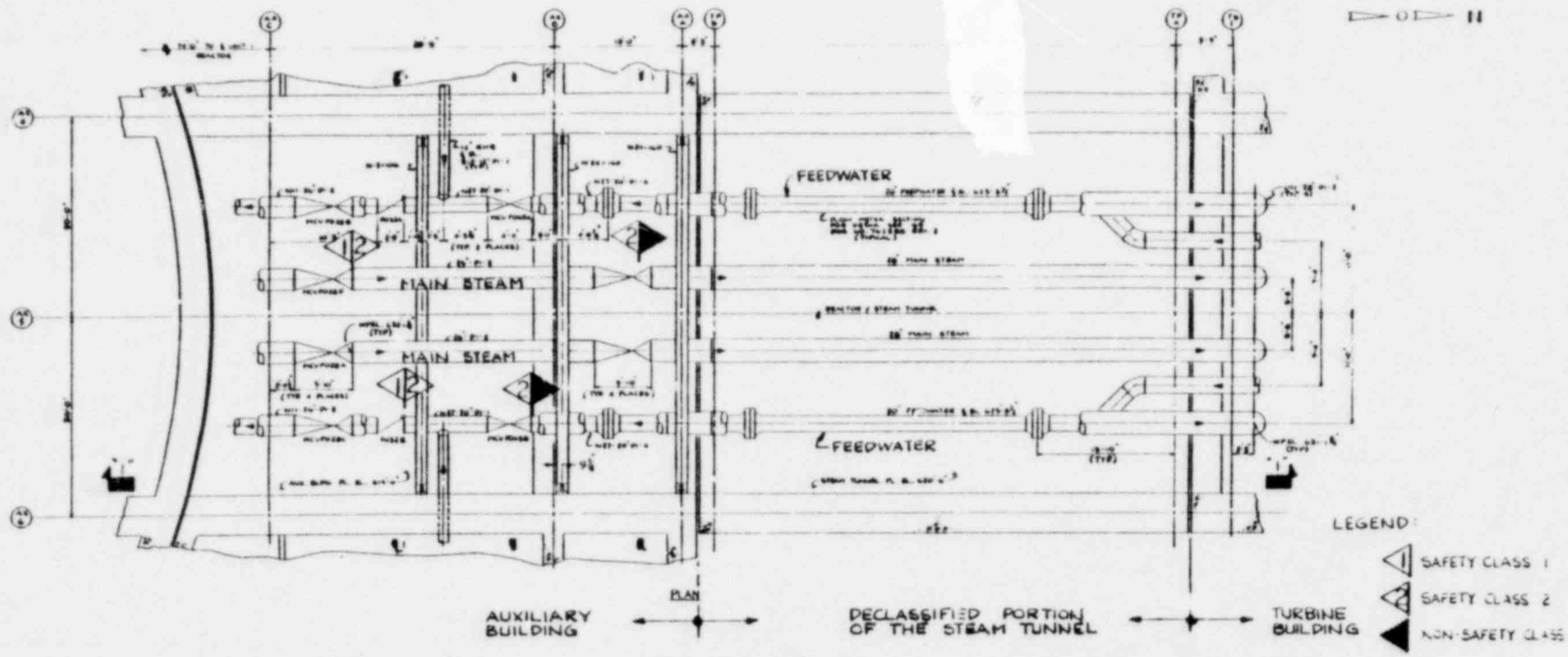
3. A - auxiliary building
C - inside containment, but outside drywell
D - drywell
E - within shield building annulus
L - offsite locale
M - any other location
O - outdoors onsite
P - pump houses
S - service building
T - turbine building
W - radwaste building

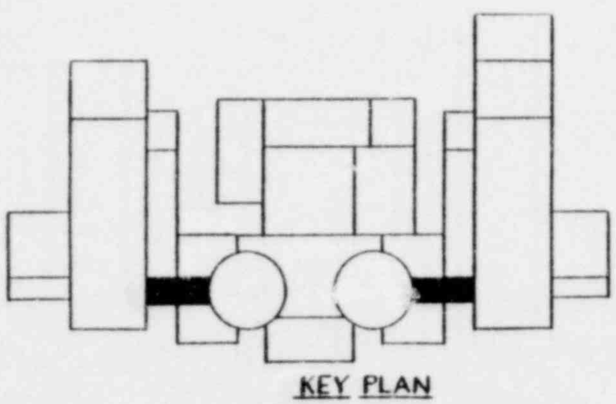
4. A, B, C, D - NRC quality groups defined in Regulatory Guide 1.26. The equipment is constructed in accordance with the codes listed in Table 3.2-2.
N/A - Quality Group Classification not applicable to this equipment.

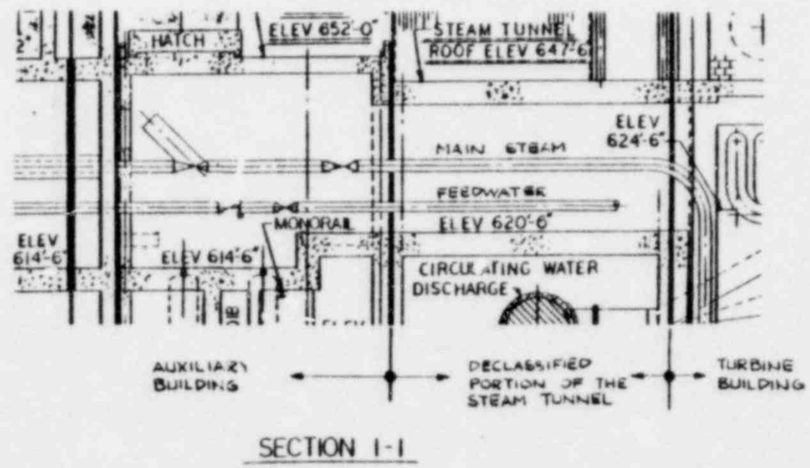
5. Notations for principal construction codes are:

III-1, 2, 3, MC - ASME Boiler and Pressure Vessel Code, Section III, class 1, 2, 3, or MC
VIII-1 - ASME Boiler and Pressure Vessel Code, Section VIII, Div. 1
API-650 - API 650, Welded Steel Tanks for Oil Storage
TEMA-C - Tubular Exhanger Manufactures Association, Class C
B31.1.0 - ANSI B 31.1.0, Code for Pressure Piping
SRcq - Nondestructive Tests Examination Requirements per ASME Section VIII, Div. 1
D100 - AWWA-D100, Standard for Steel Tanks, Standpipes, Reservoirs, and Elevated Tanks for Water Storage

6. I - Constructed in accordance with the requirements of Seismic Category I structures and equipment as described in Section 3.7, Seismic Design.
N/A - The seismic requirements are not applicable to the equipment.







PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Steam Tunnel Arrangement

Figure 10.3-1

DSER 3.6-1 (3.6.2, Page 3.6-7)	In Section 3.6.1 references are made to "elastic/plastic pipe whip restraints or pipe supports which eliminate pipe whip damage". Details of how pipe supports are designed for pipe whip protection and an example of such an analysis are needed.

Response

The response to this question is provided in revised Section 3.6.1.

No elastic/plastic pipe supports exist for pipe whip protection.

DSER 3.6-2 (3.6.2.1.4, Page 3.6-10)	How is it determined that "The internal energy associated with whipping is insufficient to impair the safety function of any structure, system or component to an unacceptable level"?

Response

The response to this question is provided in revised Section 3.6.2.1.4.c.

DSER 3.6-3 Question deleted.

DSER 3.6-4 (3.6.2.1.5, Page 3.6-11)	Plant loading conditions for evaluating pipe break are to include normal and upset conditions plus an OBE. Assurance must be provided that SRV discharge loads are included in the upset conditions.

<u>Response</u>

SRV discharge loads are included in code design specification for piping stress analysis, as used for determination of break locations. The response to this question is provided in revised Section 3.6.2.1.5 and Table 3.9-3.

DSER 3.6-5 (3.6.2.1.5, Page 3.6-11)

For ASME, Section III, Class 1 piping designed to seismic Category I standards, breaks due to stress are to be postulated at the following locations:

(1) If Eq. (10), as calculated by Paragraph 3-3653, ASME Code Section III, exceeds 2.4. S_m, then Eqs. (12) and (13) must be evaluated. If either Eq. (12) or (13) exceed 2.4 S_m, a break must be postulated. In other words, a break is postulated if

$$\text{Eq. (10)} > 2.4\ S_m \text{ and Eq. (12)} > 2.4\ S_m$$

or

$$\text{Eq. (10)} > 2.4\ S_m \text{ and Eq. (13)} > 2.4\ S_m$$

(2) Breaks must also be postulated at any location where the cumulative usage factor exceeds 0.1.

The above criteria is evaluated under loadings resulting from normal and upset plant conditions including the OBE.

Any deviations from the above criteria must be justified.

Response

The response to this question is provided in revised Section 3.6.2.1.5. Perry has committed to meet the requirements of BTP-MEB 3-1, revision 0 of November 24, 1975. The response to this question is in accord with Section B.1.b of this revision. BTP MEB 3-1 Rev. 0 does not require evaluation of equations (12) and (13) of NB-3653 unless equation (10) exceeds 3.0 S_m, nor does it require evaluation of cumulative usage factors unless equation (10) exceeds 2.4 S_m.

Final code stress analyses for major class 1 lines have been reviewed (see Tables 3.6-7 and 3.6-9). No locations were found where Equation (12) or (13) exceeds 2.4 S_m unless Equation (10) also exceeds 3.0 S_m. No locations were found where the cumulative usage factor exceeds 0.1 unless (10) also exceeds 2.4 S_m, except at terminal ends or torsional and moment restraints. Breaks are posulated at terminal ends regardless of stress range or usage factor. Breaks had also been postulated at each torsional and moment restraint where the cumulative usage factor exceeds 0.1.

DSER 3.6-6 (3.6.2.1.5, Page 3.6-11)	Are there any high energy Class 2, Class 3, or B31.1 lines? If so, what criteria is used for postulating breaks in these lines?

RESPONSE

The response to this question is provided in revised Section 3.6.2.1.5.

DSER 3.6-7 (3.6.2.1.6, Page 3.6-13)

Any instances where longitudinal break areas are less than one circumferential pipe area must be identified. The analytical methods representing test results and based on a mechanistic approach must be explained or justified. Provide examples of a typical analysis.

Response

Section 3.6.2.1.6 has been revised to eliminate the reference to longitudinal break areas less than one circumferential pipe area, since none have been postulated. No mechanistic approach is used.

DSER 3.6-8 (3.6.2.1.6, Page 3.6-14)	How are energy reservoirs of sufficient capacity to develop a jet flow determined? What are justifiable line restrictions? Provide the justification. Any instances where flow limiters are used should be identified and justified.

Response

The response to this question is provided in revised Sections 3.6.2.1.6.d and 3.6.2.2.1. The term "justifiable" has been replaced by more exact descriptions of the means of analysis.

DSER 3.6-9 (3.6.2.1.7.1, Page 3.6-15)

For ASME, Section III, Class 1 piping designed to seismic Category I standards, breaks need not be postulated providing the following stress criteria is met.

(1) If Eq. (10) as calculated by Paragraph NB-3653, ASME Code, Section III does not exceed 2.4 S_m, a break need not be postulated.

(2) If Eq. (10) does exceed 2.4 S_m, Then EQ. (12) and (13) must be evaluated. If neither Eq. (12) or (13) exceed 2.4 S_m, a break need not be postulated. In other words, a break need not be postulated if

Eq. (10) < 2.4 S_m

or

Eq. (10) > 2.4 S_m and Eq. (12) < 2.4 S_m

and EQ. (13) < 2.4 S_m

(3) Breaks need not be postulated as long as the cumulative fatigue usage factor is less than 0.1.

(4) For plants with isolation valves inside containment, the maximum stress, as calculated by Eq. (9) in ASME Code Section III, Paragraph NB-3652 under the loadings of internal pressure, dead weight and a postulated piping failure of fluid systems upstream or downstream of the containment penetration area must not exceed 2.25 S_m.

The above criteria is evaluated under loadings resulting from normal and upset plant conditions including the OBE.

In addition, augmented inservice inspection is required on all piping in the break exclusion area.

The applicant must provide assurances that their criteria for piping in the break exclusion areas complies with the requirements outlined above and those of Standard Review Plan 3.6.2.

Response

The response to this question is provided in revised Sections 3.6.2.1.5 and 3.6.2.1.7.1. Perry has committed to meet the requirements of BTP-MEB 3-1, revision 0 of November 24, 1975. The response to this question is in accord with Section B.1.b of this revision. BTP-MEB 3-1 Rev. 0 does not require evaluation of equations (12) and (13) of NB-3653 unless equation (10) exceeds 3.0 S_m, nor does it require evaluation of cumulative usage factors unless equation (10) exceeds 2.4 S_m.

An augmented in-service inspection program is provided, as referenced by new Section 3.6.2.1.7.5.

Final code stress analyses for major class 1 lines have been reviewed (see Tables 3.6-7 and 3.6-9). No locations were found where Equation (12) or (13) exceeds 2.4 S_m unless Equation (10) also exceeds 3.0 S_m. No locations were found where the cumulative usage factor exceeds 0.1 unless (10) also exceeds 2.4 S_m, except at terminal ends or torsional and moment restraints. Breaks are postulated at terminal ends regardless of stress range or usage factor. Breaks are postulated at terminal ends regardless of stress range or usage factor. Breaks had also been postulated at each torsional and moment restraint where the cumulative usage factor exceeds 0.1.

DSER 3.6-10 (3.6.2.1.7.1, Page 3.6-15) Are there any Class 2, Class 3 or B31.1 piping in the break exclusion areas? If so, what criteria is used for their design?

Response

The response to this question is provided in revised Section 3.6.2.1.7.1 and new Section 3.6.2.1.8.

DSER 3.6-11 (3.6.2.1.7.1, Page 3.6-15) A list of all systems in the break exclusion area is needed. Break exclusion area should be shown on the appropriate piping drawings.

Response

The response to this question is provided in revised Section 3.6.2.1.7.1 and Figures 3.6-5, 3.6-6 and 3.6-7.

The following high energy systems are "in the vicinity of" the containment penetration area; they are within guard pipe in the penetration area:

1. Feedwater
2. Main Steam
3. Reactor Water Cleanup
4. Main Steam Drain
5. RCIC Steam Line

DSER 3.6-12 (3.6.2.1.7.2) Page 3.6-15)	Provide an example of the detailed stress analysis done on a welded attachment to the process pipe. In addition, provide details of the stress analysis done on the head fitting for the main steam line.

Response

The response to this question is provided in revised Sect.on 3.6.2.1.7.2.

DSER 3.6-13 (3.6.2.2.1, Page 3.6-17)	Provide a list of all locations where limited break opening areas have been used. Provide justification for each location and details of any inelastic analysis used.

Response

No credit has been taken for limited break opening areas in high-energy piping. See revised Section 3.6.2.2.1. The details of the inelastic pipe whip analysis are provided in Section 3.6.2.2.2, which discribes methodology of the GE computer program PDA used for this analysis.

DSER 3.6-14 (3.6.2.2.1, Page 3.6-17)	Provide a list of all locations where break opening times greater than one millisecond have been used. Provide and justify any experimental data and analytical theory.

Response

The response to this question is provided in revised Section 3.6.2.2.1.

DSER 3.6-15 (3.6.2.2.2, Page 3.6-20)	Provide assurance that all potential targets are evaluated when considering pipe whip.

Response

The response to this question is provided in revised Sections 3.6.2.2.2 and 3.6.2.3.2.

DSER 3.6-16 (3.6.2.2.2, Page 3.6-20) Provide a definition for limits of strain which are similar to levels allowed in restraint plastic members.

Response

The response to this question is provided in revised Section 3.6.2.2.2.

DSER 3.6-17 (3.6.2.2.2, Page 3.6-20)	"Piping systems are designed so that plastic instability does not occur in the pipe at the design dynamic and static loads unless damage studies are performed which show the consequences do not result in direct damage to any essential system or component." Provide a list of where this technique has been used and an example of the studies performed.

<u>Response</u>

The response to this question is provided in revised Sections 3.6.2.2.2.e, 3.6.2.2.4c, 3.6.2.3.2, and Table 3.6-17. No cases exist where more than 50% of the minimum actual ultimate uniform strain (at the maximum stress on an engineering stress-strain curve) has been allowed.

DSER 3.6-18   Question deleted.

DSER 3.6-19 (3.6.2.3.1, Page 3.6-23)	It is the staff's position that when evaluating jet impingement loads all potential targets must be evaluated. Provide assurances that your analysis for jet impingement effects have included all possible targets.

Response

The response to this question is provided in revised Section 3.6.2.3.1.

DSER 3.6-20 (3.6.2.3.1, Page 3.6-29)	What service limits are used for piping when evaluating jet impingement loads?

Response

The response to this question is provided in revised Section 3.6.2.3.1.

DSER 3.6-21 (3.6.2.3.1, Page 3.6-30)	How is it determined that the dynamic load factor (DLF) is suitable? Provide an example of its use.

Response

The response to this question is provided in revised Section 3.6.2.3.1 for BOP systems.

DSER 3.6-22 (3.6.2.3.1, Page 3.6-30)

For snubbers, what are the "other simultaneous loads" that are combined by the SRSS method?

Response

The response to this question is provided in revised Section 3.6.2.3.1. Only concurrent vibratory dynamic load cases are combined by SRSS.

DSER 3.6-23 (3.6.2.3.3, Page 3.6-33) "Piping integrity usually does not depend upon the pipe whip restraints for any loading combination." List all those locations where integrity of the piping depends upon the pipe whip restraints.

Response

The response to this question is provided in revised Section 3.6.2.3.3. Piping integrity does not depend on pipe whip restraints.

DSER 3.6-24 (3.6.2.3.3, Page 3.6-33) What service limits are used in the design of the pipe whip restraints?

RESPONSE

The response to this question is provided in revised Section 3.6.2.3.3.

DSER 3.6-25 (3.6.2.3.3.1, Page 3.6-33)

What critical locations inside containment are monitored during hot functional testing?

Response

The response to this question is provided in revised Section 3.6.2.3.3.1.

All supports and pipe whip restraints are monitored during pre-operational and startup testing for adequate clearances to accommodate thermal expansion.

DSER 3.6-26 (3.6.2.3.3.1, Page 3.6-40) Any locations where the increase in the yield or ultimate strengths, of the material used for pipe whip restraints, exceeds 10% must be identified. Justification for any increa e greater than 10% must also be provided.

Response

The answer to this question is provided in revised Section 3.6.2.3.3.1.4.c.

No increases greater than 10% have been used.

DSER 3.6-27 (3.6.2, Tables)	Provide a schedule for the completion of any table that is incompl[illegible]

Response

Table:	Status as of this response:
3.6-1	Revised
3.6-2	Revised
3.6-3	Revised
3.6-4	Unchanged
3.6-5	Unchanged
3.6-6	Unchanged
3.6-7	Provided with this response
3.6-8	On or about September 20, 1981
3.6-9	Provided with this response
3.6-10	On or about July 1, 1982
3.6-11	On or about July 1, 1982 (Unit 1), July 1, 1984 (Unit 2).
3.6-12	Provided with this response
3.6-13	Provided with this response
3.6-14	Replaced by Figures 3.6-95 through 3.6-98
3.6-15	Provided with this response
3.6-16	Unchanged
3.6-17	New with this response.

Tables 3.6-8, 3.6-10 and 3.6-11 will contain only confirmatory data to show that break location and break exclusion region stress criteria are met.

DSER 3.6-28 (3.6.2, Figure 3.6-66) Are all postulated break locations in the recirculation system shown?

Response

All postulated break locations are shown on revised Figure 3.6-66, as determined by the final ASME Code stress analysis (General Electric document 22A7140, Rev. 0).

DSER 3.6-29 Where are breaks postulated in these figures?
(3.6-2,
Figures 3.6-71,
3.6-73,
7.3-74,
3.6-77,
3.6-78,
3.6-79,
3.6-80)

Response

Breaks are located at each fitting and valve weld, as indicated by standard piping symbols ("hash marks"). Notes on these figures have been revised to clarify break locations. Figures 3.6-79 and 3.6-80 have been completely redrawn to reflect redesigned piping.

DSER 3.6-30 Indicate the location of valves in this line.
(3.6.2,
Figure 3.6-75)

RESPONSE

Figure 3.6-75 has been revised to show valve locations.

DSER 3.6-31 (3.6.2.1.6, Page 3.6-13) Regardless of the ratio of longitudinal to hoop stress, both a longitudinal split and a circumferential break should be postulated at any location where the cumulative usage factor is greater than 0.1.

Response

The response to this question is provided in revised Sections 3.6.2.1.5 and 3.6.2.1.6. Perry has committed to meet the requirements of BTP-MEB 3-1, revision 0 of November 24, 1975. The response to this question is in accord with sections B.3.a and B.3.b of this revision.

The criteria of BTP-MEB 3.1, sections B.3.a and B.3.B, provide means of discriminating between types of breaks, but refer only to stress criteria for class 2 and 3 piping. The ASME code makes no provision for calculation of cumulative usage factors for class 2 and 3 piping.

Revised Section 3.6.2.1.6 extends these criteria to class 1 piping by employing the corresponding class 1 stress criteria.

3.6 PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

3.6.1 POSTULATED PIPING FAILURES IN FLUID SYSTEMS

3.6.1.1 Design Bases

The foremost requirement for protection against the effects of a postulated pipe rupture event is conformance to the offsite dose limits given by 10 CFR 100. This objective is achieved by plant arrangements which permit habitability of the control room, assure essential shutdown system operation and mitigate the consequences of the piping failure. The plant design provides this protection primarily by physical separation through spatial arrangement or enclosure within structures or compartments. Structural barriers and jet impingement shields are employed as required to further mitigate the consequences of pipe break events. Where separation or barriers are either insufficient or impractical, protection by means of pipe whip restraints is used to ensure the operability of equipment and structures essential for safe plant shutdown.

A comprehensive review of plant fluid systems with respect to postulated rupture of piping is presented in Section 3.6.1.2. Spatial arrangements of high and moderate energy lines are shown relative to equipment required for safe shutdown. Section 3.6.1.3 provides a discussion of the effects of postulated pipe ruptures coincident with single active failures in required systems. The ability to safely shut down the plant is discussed with respect to the various combinations of pipe rupture and single active failure. Environmental conditions for which equipment is designed to operate in the accident mode are addressed in Section 3.11.

3.6.1.2 Description

High and moderate energy lines are listed in Tables 3.6-1 and 3.6-2, respectively. These lines are illustrated by Figures 3.6-1 through 3.6-47 in relation to plant layout. These figures also identify systems and components required for safe shutdown (see Table 3.6-3). As illustrated by Figures 3.6-1 through 3.6-47, systems and components required for safe shutdown are protected from postulated pipe rupture, to a large extent, by physical arrangement. Detailed descriptions of these physical arrangements are presented in Sections 3.6.1.2.1 and 3.6.1.2.2.

3.6.1.2.1 Physical Arrangement Inside the Reactor Building

a. Inside the Drywell

To the greatest possible extent, the piping, the electrical, and structural arrangement within the drywell provides for safe shutdown capability in the event of high energy pipe rupture by means of spatial separation. Both the main coolant piping (recirculation and feedwater) and the ECCS piping are evenly distributed around the reactor. Furthermore, the electrical power divisions serving the various ECCS systems govern the location of system pipe routing to prevent any single high energy pipe break from jeopardizing any additional ECCS. A limited number of postulated ruptures could potentially jeopardize the functioning of an adequately redundant number of ECCS due to limitations of spatial and barrier separation. Each such case is discussed in Section 3.6.2.5.3 and resolved either by means of a jet shield or by analytically establishing the adequacy of separation. These high energy lines within the drywell are restrained from whipping by elastic/plastic pipe whip restraints which prevent pipe whip damage to essential systems and limit structural loads.

b. Between the Drywell and the Reactor Building Wall

Between the drywell and the reactor building wall, portions of three high energy systems constitute potential pipe rupture sources: the reactor water cleanup system; control rod drive supply line and standby liquid control system. In all cases postulated ruptures are located so that spatial separation provides protection to ECCS from the effects of postulated ruptures.

In all cases high energy lines between the drywell and the reactor building wall are restrained from whipping by pipe whip restraints or pipe supports.

3.6.1.2.2 Physical Arrangement Outside the Reactor Building

Building arrangements outside the reactor building are characterized by the following areas for purposes of the pipe rupture analysis:

a. Inside the Steam Tunnel

A significant design feature of the plant with regard to postulated rupture of high energy piping is the provision of the steam tunnel. This structure serves as a conduit for essentially all high energy piping between the reactor building and turbine building. The steam tunnel is designed to contain the environmental effects (pressure and temperature) resulting from a full circumferential pipe break (double ended rupture) of either a main steam or feedwater pipe. Following such a postulated event, the steam tunnel vents the blowdown from the break to the turbine building. Rapid closing isolation valves close to limit the release of mass and energy from the break. These valves and their operation are discussed in Sections 5.4.5 and 6.2.4. The pressure rise analysis for this design basis event is discussed in Section 3.6.2.2. A description of the structural design and analysis of the steam tunnel is presented in Section 3.6.2.3.

High energy piping routed through the steam tunnel is shown in Figure 3.6-24. Pipe whip restraints are provided to prevent consequential damage following a postulated pipe break.

b. Inside the Fuel Handling Building

The fuel handling building is free of high energy lines, except for one 2 1/2-inch nominal OD control rod drive (CRD) line which conveys cold water at approximately 1900 psig. This line is prevented from damaging surrounding structures by means of closely spaced pipe supports and/or restraints of sufficient capacity. No equipment required for safe shutdown is located in the vicinity of the route of this line in the lowest elevation of the fuel handling building. The consequences of a postulated rupture of this line are limited to local flooding of the lowest elevation in the fuel handling building to a depth of approximately 6 inches.

c. Inside the Intermediate Building

The intermediate building contains no high energy piping.

Moderate energy lines whose failure could result in limited (less than 6 inches in depth) flooding of the lowest level of the intermediate building present no hazard to the operation of any systems essential to safe plant shutdown.

d. Inside the Auxiliary Building

The auxiliary building, excluding the structurally separated steam tunnel addressed in Item a, above, contains three sources of high energy pipe ruptures. The reactor water cleanup system piping and pumps are located in a compartment which is vented to a corridor containing safety related electrical cabling. Analysis of the conditions following the pipe rupture event indicate the safe shutdown capability of the plant is not jeopardized. The second source of high energy pipe rupture occurs in a main steam drain line routed through the same corridor which communicated with the RWCU pump room. The piping configuration of this drain line is such that the postulated break occurs within a guard pipe which vents also to the steam tunnel. Analysis of the effects of this event indicate the safe shutdown capability of the plant is not jeopardized. The third source of high energy pipe rupture is the auxiliary steam system.

The main auxiliary steam piping is routed over the auxiliary building roof, and enters through the steam tunnel roof.

Breaks in the 10-inch auxiliary steam main in the auxiliary building are confined to the steam tunnel. A four-inch test line to the RCIC turbine is normally isolated. Condensate lines through the auxiliary building are maintained below 275 psig and below 200°F by condensate coolers.

Two RCIC-RHR condensing cooling mode steam ''nes pressurized from the reactor vessel are located in two RHR h. exchanger rooms. However, included in the system design is an orifice which restricts the amount of energy escaping from a full rupture of these lines. The orifice is sized to assure that safe plant shutdown is not jeopardized by temperature or environmental effects.

e. Inside the Control Complex

The control complex is isolated from adjacent structures by 3-foot thick concrete walls and pressure tight doors where required.

A portion of moderate energy piping is concentrated in two areas of the control complex. One area, at elevation 599'-0", houses the nuclear closed cooling water (NCCW) heat exchangers served by service water piping. The piping and heat exchangers are in a single, enclosed room. Water flowing from a postulated leakage crack in NCCW or service water piping would either drain through sleeves in the floor to the next lower elevation at 574'-10" or discharge directly into that elevation. The area below the NCCW heat exchanger room at elevation 574'-10" houses the service and instrument air receiver tanks. Elsewhere at this elevation are essential shutdown systems. The water would drain to the floor of this space and from there to floor drain sumps equipped with safety related instrumentation that actuates alarms upon detection of high level.

The maximum leakage rate from a through-the-wall crack is calculated to be 930 gpm. Pipe size is 42 inches, nominal OD, with a wall thickness of 1/2 inch, and a system head of 140 feet of water. This leakage rate would flood elevation 574'-10" to a depth of 2-1/2 inches in 30 minutes at which time the leak would be isolated.

Equipment required for safe shutdown or for maintaining control room habitability is located at elevation 574'-10". This equipment includes three water chillers and the emergency closed cooling water pumps. This equipment is protected from flooding by mounting it on 6 inch foundation pads.

The area at elevation 679'-6" above the control room houses chilled water piping (CCCW) that provides cooling for the control room HVAC equipment. This is moderate energy piping. This area also houses a 140 kw electric boiler capable of producing 480 lb/hr of saturated steam at 5 psig. This boiler supplies a low-pressure humidification system, whose piping is defined as high energy piping.

An analysis of possible effects of jets and pipe whip due to humidification system breaks shows that safe shutdown is not jeopardized. The low power rating of the boiler and the small energy reservoir of the system preclude any rapid environmental effects. Redundant leak detection sensors are provided to assure that any failure is detected with ample time to shut off the boiler before environmental effects could compromise safe shutdown components.

The maximum leakage rate from a postulated moderate energy crack in the CCCW pipe is calculated to be 130 gpm. The pipe size is 10 inches, nominal OD, with a wall thickness of 0.365 inches, and a system pressure of 40 psig. The area at elevation 679'-6" is sealed off from the control room and is provided with completely embedded drain piping sized to carry water issuing from the design basis leakage crack to drain sumps outside the control complex.

f. Inside the Radwaste Building

The radwaste building contains a high energy steam line that supplies steam to the radwaste evaporators. This line is routed outside the radwaste building and enters the building, directly into the radwaste evaporator room, through the roof. Redundant, safety class, active valves with necessary instrumentation are provided to terminate steam flow following a postulated rupture of the steam line or any pressure retaining portion of the radwaste evaporator. Pressure tight doors are provided where required to isolate the affected area.

systems. An operational period is considered "short" if the total fraction of time that the system operates within the temperature and pressure conditions specified for a high energy fluid system is less than 2 percent of the total operating time for which the system was designed.

3.6.2.1.3 Postulated Pipe Breaks and Cracks

A postulated pipe break is defined as a sudden, gross failure of the pressure boundary either in the form of a complete circumferential severance (guillotine break) or as development of a sudden longitudinal, uncontrolled crack (longitudinal split) and is postulated for high energy fluid systems, only. For moderate energy fluid systems, pipe breaks are confined to postulated controlled cracks in piping and branch runs. These cracks affect surrounding environmental conditions, only, and do not result in whipping of the cracked pipe.

All high energy piping systems (or portions thereof) are considered as potential initiators for a postulated pipe break under normal plant conditions and are analyzed for potentially damaging dynamic effects.

Portions of piping systems isolated from the source of the high energy fluid under normal plant conditions are exempted from consideration of postulated pipe breaks. This exemption includes portions of piping systems beyond a normally closed valve. Pump and valve bodies are also exempted from consideration of pipe break because of the greater wall thickness of such components.

A high energy pipe break is not postulated to occur simultaneously with a moderate energy piping system crack nor is any pipe break or crack outside containment postulated to occur concurrently with a postulated pipe break inside containment.

3.6.2.1.4 Exemptions from Pipe Whip Protection Requirements

Protection from pipe whip is not provided where any one of the following conditions exist:

a. The postulated pipe break is in a moderate energy piping system.

b. The unrestrained movement of either end of the ruptured pipe in any feasible direction about a plastic hinge formed within the piping, following a single postulated pipe break, cannot impact any structure, system, or component required for safe shutdown.

c. Reaction forces on the broken pipe are insufficient to impart sufficient energy to the broken pipe to cause unacceptable damage to any structure, component or system required for safe shutdown. Any line restrictions (e.g., flow limiters) between the pressure source and the break location, and the effects of either a single ended or double ended flow condition may be considered in the determination of the reaction forces. The energy of the broken pipe is considered insufficient to cause unacceptable damage if any of the following criteria are met: -2, -17

1. The energy level in a whipping pipe is considered insufficient to damage another pipe of equal or greater nominal pipe size and equal or heavier wall thickness in accordance with NRC Branch Technical Position APCSB 3-1, item 2.b(2).

2. The reaction force, applied to the broken pipe, is insufficient to stress the piping to the elastic limit at any point, and the limits of deflection of the broken pipe, in any direction, do not allow impact of any structure, system or component required for safe shutdown. Cases where this criterion and method are used are listed in Table 3.6-17. -2, -17

3. The impacting energy of the broken pipe, determined by the strain energy method, does not impair the essential safety function of any impacted component.

These pipe whip analyses are further described in Section 3.6.2.2.2.e. Damage studies are described in Section 3.6.2.3.2. In all damage studies a rebound amplification factor of 1.2 is applied to the forcing function determined for the component, as required by Standard Review Plan 3.6.2, Section III.2.b.(2). Cases where this criterion and method are used are listed in Table 3.6-17. -2,-17

3.6.2.1.5 Locations for Postulated Pipe Breaks

Postulated pipe break locations are selected in accordance with NRC Branch Technical Position APCSB 3-1, Appendix B, and NRC Branch Technical Position MEB 3-1, November 24, 1975.

a. For piping systems classified as high energy, postulated break locations are as follows:

    1. The terminal ends of pressurized portions of the run. Terminal ends are the extremities of piping runs that connect to structures, equipment, or pipe anchors that are assumed to act as rigid constraints to free thermal expansion and any movement, from dynamic or static loading, of piping. -3

2. For ASME Code, Section III Class I, Seismic I piping, breaks are postulated to occur at intermediate locations between terminal ends whenever the following stress and fatigue limits are exceeded:

a) The maximum stress range between any two load sets (including the zero load set) shall be calculated according to equation (10) of article NB-3653 of the ASME Code, Section III for normal and upset plant conditions, including safety relief valve (SRV) loads, and an operating basis earthquake (OBE) event transient. If this value is less than 2.4 Sm, no break need be postulated.

-4,-5,-9,-31

b) If equation (10) exceeds 2.4 Sm but is less than 3.0 Sm and the cumulative usage factor U of article NB-3653.5 is less than 0.1, no break need be postulated.

c) If for a given load set, equation (10) exceeds 3.0 Sm, but the maximum stress ranges calculated according to equations (12) and (13) of article NB-3653.6 for that load set are each less than 2.4 Sm and the cumulative usage factor calculated according to article NB-3653.6 (using equation (14) or article NB-3653.3 for S_{alt}) does not exceed 0.1, no break need be postulated.

-5,-9,-31

In accordance with article NB-3653.6 and BTP-MEB 3.1, Section B.1.b.(1)(b), equations (12) and (13) need be evaluated only for those load sets for which equation (10) exceeds 3.0 Sm.

3. For ASME Code, Section III class 2 and 3 piping, breaks are postulated to occur at all locations where the sum of equations (9) and (10) of ASME Code Section III, article NC-3652, calculated under all normal and upset plant conditions, including safety relief valve (SRV) loads, and an independent operating basis earthquake (OBE) event transient, is greater than 0.8 (1.2 Sh + S_A), except that the more restrictive criteria of Sections 3.6.2.1.7 and 3.6.2.1.8 apply to containment penetration piping.

-4,-5,-6,-9,-31

4. Breaks are assumed in piping designed to the power piping code, ANSI B31.1, at each fitting-to-pipe weld, including welds to pumps, flanges, attachments and valves, except as provided in Section 3.6.2.1.7.1.

5. In the event that two or more intermediate locations cannot be determined by stress or usage factor limits, at least two intermediate locations are identified on a reasonable basis for each piping run or branch run, unless the piping is a straight run without fittings, attachments and valves, in which case only one intermediate location is chosen. A reasonable basis is one or more of the following:

   (a) Each fitting-to-pipe weld, including welds to pumps, flanges, attachments and valves.

   (b) Highest stress or usage factor locations.

Where more than two such intermediate locations are possible using the application of the above reasonable basis, those two locations having the greatest damage potential may be used. A break at each end of a fitting may be classified as two discrete break locations where the stress analysis is sufficiently detailed to differentiate stresses at each postulated break.

b. For piping systems which contain moderate energy fluids, through wall leakage cracks are postulated at locations as follows:

1. Locations that demonstrate the adequacy of separation or other means of protection from required structures, systems and components.

2. In moderate energy fluid system piping located within structures and compartments containing required systems and components. The through wall leakage cracks are postulated to occur individually at locations appropriate to form the basis for providing required protection from the hazards of fluid spraying, flooding, pressurization, and other environmental conditions.

3. Moderate energy fluid system piping or portions thereof that are located within a compartment or confined area containing a postulated break in high energy fluid system piping are considered acceptable without postulation of through wall leakage cracks, except where a postulated leakage crack in the moderate energy fluid system piping results in more severe environmental conditions than the break in the proximate high energy fluid piping system. In such cases the provisions of this section, Item b, apply.

c. Criteria for break locations in piping systems in the area of the containment isolation valves are provided in Sections 3.6.2.1.7. and 3.6.2.1.8.

3.6.2.1.6 Types of Breaks to Be Postulated in Fluid System Piping

The following types of breaks are postulated in high energy fluid system piping:

a. No breaks are postulated in piping having a nominal diameter less than or equal to one inch.

b. Circumferential breaks are postulated only in piping exceeding a one-inch nominal pipe diameter, except CRD insert lines (1-1/4 inch).

c. Longitudinal splits are postulated only in piping having a nominal diameter equal to or greater than four inches.

d. Circumferential breaks are assumed at all terminal ends and at intermediate locations identified by the criteria stated in Section 3.6.2.1.5. At each of the postulated break locations identified, in piping four inches nominal diameter or greater, either a circumferential or a longitudinal break, or both, is postulated according to the following criteria. "Maximum stress range" is calculated as described in Sections 3.6.2.1.5.a.2.a and 3.6.2.1.5.a.3.

1. If the maximum stress range exceeds the limit of Section 3.6.2.1.5.a.2.a or 3 and the maximum stress range in the longitudinal direction is greater than 1.5 times the maximum stress range in the circumferential direction, only the circumferential break need be postulated.

2. If the maximum stress range exceeds the limit of Section 3.6.2.1.5.a.2.a or 3 and the maximum stress range in the circumferential direction is greater than 1.5 times the maximum stress range in the longitudinal direction, only the longitudinal break need be postulated. - 31

3. If the maximum circumferential and longitudinal stress ranges are within a factor of 1.5 of each other, or if the analysis does not differentiate between circumferential and longitudinal stress ranges, then both types of breaks are postulated.

4. Circumferential breaks are postulated at fitting joints.

5. Longitudinal breaks are postulated in the center of the fitting at two diametrically opposed points (but not concurrently) located so that the reaction force is perpendicular to the plane of the piping and produces out-of-plane bending.

6. At intermediate locations where stress ranges are less than the criteria of Sections 3.6.2.1.5.a.2 and 3, and breaks are chosen to satisfy the criteria for a minimum number of break locations, only circumferential breaks are postulated in accordance with BTP-MEB 3.1, B.3.b(2)(b). - 31

7. At terminal ends without longitudinal welds, only circumferential breaks are postulated. At terminal ends with longitudinal welds, the criteria of paragraphs 1, 2 and 3 of Item d apply, according to BTP-MEB 3-1, Section B.3.b.(2)(a).

e. For design purposes, a longitudinal break area is assumed to be the equivalent of one circumferential pipe area.

f. For both longitudinal and circumferential breaks, after assessing the contribution of upstream piping flexibilities, pipe whipping is assumed to occur in the plane defined by the piping geometry and configuration for circumferential breaks and out of plane for longitudinal breaks, and to cause pipe movement in the direction of the jet reaction.

g. The dynamic force of the jet discharge at the break location is based upon the effective cross sectional flow area of the pipe and upon a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. Line restrictions, flow limiters, and the absence of energy reservoirs may be used, as applicable, in the reduction of the jet discharge. Blowdown calculation methods are described in section 3.6.2.2.1 below.

The following through wall leakage cracks are postulated in moderate energy fluid systems (or portions of systems):

a. Cracks are postulated in moderate energy fluid system piping and branch runs exceeding a nominal pipe size of one inch.

b. Crack openings are assumed as a circular orifice of cross sectional piping flow area equal to that of a rectangle of length equal to one-half pipe diameter and width of one-half pipe wall thickness.

c. Flow from the crack opening is assumed to result in an environment that wets all unprotected components within the structure or compartment, with consequent flooding in the compartment and communicating compartments, based upon a conservatively estimated time flow period to effect corrective actions.

d. Through wall leakage cracks instead of breaks are postulated in the piping of those fluid systems that qualify as high energy fluid systems for only short operational periods as defined in Section 3.6.2.1.2.

3.6.2.1.7 Criteria for High Energy Piping Systems in the Area of Containment Isolation Valves

3.6.2.1.7.1 Locations for Postulated Breaks

No pipe break is postulated in the portions of high-energy piping between primary containment isolation valves.

The containment penetration break exclusion region is defined as that section of piping between (1) the weld next outboard of the outboard torsional and moment restraint adjacent to the outboard containment isolation valve and (2) the weld next inboard of the inboard torsional and moment restraint, where such torsional and moment restraints are required to meet these break exclusion region stress criteria; or between (1) the outboard weld of the outboard containment isolation valve and (2) the inboard weld of the inboard containment isolation valve, on small piping which does not require torsional and moment restraints to meet the criteria below.

The break exclusion regions of major high-energy lines are indicated on Figures 3.6-5, 3.6-6 and 3.6-7. No safe-shutdown components other than containment isolation valves and their auxiliaries are located in break exclusion regions.

The containment penetration region of high energy piping meets the following criteria for break exclusion regions:

a. For High-Energy ASME Code, Section III, Class 1 Piping:

   1. Piping in this region shall meet the requirements of article NE-1120 of the Code.

   2. The stress criteria of Section 3.6.2.1.5.a.2, shall be met.

   3. The maximum stress in the break exclusion region due to a postulated rupture of the affected line outside the break exclusion region, as calculated by equation (9) of article NB-3653 of the Code, shall not

exceed 2.25 Sm, except that following a failure outside containment higher stresses are allowed between the outboard containment isolation valve and the outboard torsional and moment restraint, provided a plastic hinge is not formed and the operability of the valve is assured under this loading in accordance with Standard Review Plan 3.9.3. Primary loads for equation (9) include those loads which are deflection-limited by restraints.

Design specification criteria for Class 1 penetrations are presented in the Nutech Corporation design specification for Class 1 piping penetration assemblies.

b. For High-Energy ASME Code, Section III, Class 2 piping:

1. Piping in this region shall meet the requirements of article NE-1120 of the Code.

2. The stress criteria of Section 3.6.2.1.5.a.3 shall be met.

3. The maximum stress in the break exclusion region due to a postulated rupture of the affected line outside the break exclusion region, as calculated by equation (9) of Article NC-3652 of the Code, shall not exceed 1.8 Sh. The exceptions permitted for class 1 piping under Section 3.6.2.1.7.1.a.(3), above, may be applied to piping outboard of the outboard containment isolation valve, provided that any such piping between the valve and outboard torsional and moment restraint constructed to the ANSI B31.1 power piping code shall be provided with full radiography of all welds, both circumferential and longitudinal. Primary loads for equation (9) include those loads which are deflection-limited by restraints.

c. For both Class 1 and class 2 piping, the design and inspection requirements stated in Sections 3.6.2.1.7.2 through 3.6.2.1.7.5 are satisfied.

3.6.2.1.7.2 Welded Attachments to the Process Pipe

Welded attachments, for pipe supports or other purposes, to these portions of piping are designed by means of detailed stress analyses to demonstrate compliance with the limits of Section 3.6.2.1.5. A typical attachment is a -12
welded lug for torsional and moment restraints. In addition, the number of circumferential and longitudinal piping welds are minimized. There are no branch connections in these portions of the process pipe. Where guard pipes are used, the enclosed portion of fluid system piping is of seamless construction. The length of these portions of piping is the minimum practical. The analysis of the head fitting, including the welds to the main steam pipe and the guard pipe, is -12
in accordance with the draft GE report NEDO-23652, which has been reviewed by the NRC and IIT.

3.6.2.1.7.3 Design of Pipe Anchors and Restraints

Pipe anchors and restraints are designed to be 100 percent volumetrically examinable in service and a detailed stress analysis is performed to demonstrate compliance with the limits stated in Section 3.6.2.1.5.

3.6.2.1.7.4 Guard Pipe Design

Design criteria for guard pipe assembly are as follows:

a. Neither the guard pipe to head fitting circumferential weld nor any circumferential weld in the guard pipe is located in the annular space between the drywell wall and containment wall.

b. Construction requirements satisfy Subsection NE of Section III of the ASME Code.

c. The guard pipe is designed to a temperature and pressure equal to or greater than the normal operating temperature and pressure of the process pipe.

d. The guard pipe is pressure tested in accordance with SA-530-5 of the ASME Code, either by the material's manufacturer or the guard pipe fabricator. This test may be performed prior to fabrication of the complete assembly.

e. A 100 percent volumetric examination is performed in accordance with the requirements of the ASME Code, Section III, Subsection NE, for all longitudinal welds (Category A) and all circumferential welds (Category B) in the guard pipe.

f. A 100 percent volumetric examination is performed in accordance with the requirements of the ASME Code, Section III, Subsection NB or NC, depending upon class, for the head fitting to process pipe well as a full penetration Category C weld.

3.6.2.1.7.5 Augmented In-Service Inspection

Augmented inservice inspection for high energy piping systems in the area of containment isolation valves is described in Section 5.2.4.3.

3.6.2.1.8 Criteria for Moderate Energy Piping systems in the Area of Containment Isolation Valves

No pipe break is postulated in the portions of moderate-energy piping between containment isolation valves. The containment penetration break exclusion region is defined as that section of piping between the outboard weld or flange of the outboard containment isolation valve and the inboard weld or flange of the inboard containment isolation valve. Where an approved design allows placement of both isolation valves on the same side of containment, the boundary of the containment penetration break exclusion region extends to the weld defining the boundary between class 2 and class MC components.

Moderate-energy class 2 containment penetration break exclusion regions meet the following criteria: 10

a. The requirements of article NE-1120 of the ASME Code, Section III must be met.

b. The maximum stress range calculated as the sum of equations (9) and (10) of article NC-3652 of the ASME Code, section III, under normal and upset plant conditions including safety relief valve (SRV) discharge loads, and an operating base earthquake (OBE) event transient, may not exceed $0.4\ (1.2\ Sh + S_A)$.

3.6.2.2 <u>Analytical Methods to Define Blowdown Forcing Functions and Response Models</u>

3.6.2.2.1 Analytical Methods to Define Blowdown Forcing Functions

Rupture of a pressurized pipe causes the flow characteristics of the system to change, creating reaction forces which can dynamically excite the piping system. The reaction forces are a function of time and space and depend upon fluid state within the pipe prior to rupture, break flow area, frictional losses, plant system characteristics, piping system, and other parameters. A more detailed description of the analytical computer model used in defining the blowdown forces is presented in GAI Topical Report 104P[1].

Criteria used for calculation of fluid blowdown forcing functions include the following:

a. Circumferential breaks are assumed to result in pipe severance with a break area equivalent to the pipe's cross sectional area and separation amounting to at least one diameter lateral displacement of the ruptured piping sections. -13

   Longitudinal breaks are equal to a full circumferential break.

b. The dynamic force of the jet discharge at the break location is based upon the effective cross sectional flow area of the pipe and upon a calculated fluid pressure as modified by an analytically determined thrust coefficient. -13 Line restrictions, flow limiters, positive pump controlled flow, and the absence of energy reservoirs are taken into account in the reduction of jet discharge.

   Sections of broken piping without connecting pumps or tanks, containing only cold water or a negligible volume of steam or compressed water above 212°F, compared to the cross sectional area of the break, and separated from other pressurized sources by normally closed valves or check valves, are considered to contain insufficient energy to develop a jet. Frictional effects of piping, components, flow limiters, filters, and metering orifices and venturis may be included in determination of the steady-state portion of the blowdown, as described below. -8 Frictional effects and flow limiters are considered for all blowdown calculations, other than assumptions of 2.0 P_oA for cold water and 1.26 P_oA for steam and flashing water. No mechanical devices have been added solely to reduce jet discharge.

c. NSSS analyses assume instantaneous breaks. For the balance-of-plant analyses, a rise time of one millisecond is used for the initial pulse, except where longer crack propagation times or rupture opening times can be substantiated by experimental data or analytical theory. Break opening times greater than one millisecond are used only for main steam longitudinal breaks outside containment, as reflected in Table 3.6-12.

The break opening time for a longitudinal break of main steam lines was calculated using the BMI relationship [9]

$$A = 4.43\, t \left[\left(\frac{t^2 + 1.96}{1.213} \right)^{1/2} - 1.27 \right] ,$$

where:

A = opening area, in^2

t = time, milliseconds

Based on this equation, the longitudinal break in the 28 inch main steam line would reach one pipe flow area in 0.0118 seconds.

Blowdown forcing functions are determined by either of the two following methods:

a. Predicted blowdown forces on pipes fed by a pressure vessel can be described by transient and steady state forcing functions. The forcing functions used are based upon methods described in Reference 1. These may be described as follows:

1. The transient forcing functions at points along the pipe results from the propagation of waves (wave thrust) along the pipe and from the reaction force due to the momentum of the fluid leaving the end of the pipe (blowdown thrust).

3.6.2.2.2 Pipe Whip Dynamic Response Analyses

The prediction of time dependent and steady thrust reaction loads caused by blowdown of subcooled, saturated, and two-phase fluid from a ruptured pipe is used in design and evaluation of dynamic effects of pipe breaks. A detailed discussion of the analytical methods employed to compute these blowdown loads is presented in the following paragraphs.

Criteria used in performing pipe whip dynamic response analyses include the following:

a. A pipe whip analysis is performed for each postulated pipe break. However, a given analysis can be used for more than one postulated break location if the blowdown forcing function, piping and restraint system geometry, and piping and restraint system properties are conservative for other locations.

b. The analysis includes the dynamic response of the pipe in question and the pipe whip restraints which transmit loading to the structures.

c. The analytical model adequately represents the mass/inertia and stiffness properties of the system.

d. Pipe whipping is assumed to occur in the plane defined by the piping geometry and configuration and to cause pipe movement in the direction of the jet reaction.

e. Piping within the broken loop is no longer considered part of the reactor coolant pressure boundary. Plastic deformation in the pipe is considered as a potential energy absorber. The following strain limits are used:

   1. Fifty percent of the minimum actual ultimate uniform strain (at the maximum stress on an engineering stress-strain curve) based upon restraint material tests; or

- 1/91 - 17

2. One-half of minimum percent elongation as specified in the ASME Code, Section III, or ASTM Specifications, as applicable, when demonstrated to be as, or more, conservative than Item (1), above. -16, -17

These limits are the same as those imposed on the energy absorbing, plastically deforming pipe whip restraints (see Section 3.6.2.3.3.1.3.a.3).

Piping systems are designed so that plastic instability does not occur in the pipe from the design dynamic and static loads. Damage studies are performed to show that the broken pipe does not result in damage to any essential system or component. Damage studies are described in Sections 3.6.2.3.2 and 3.6.2.1.4. -15 and fluid blowdown forces were the same in both analyses. However, a linear approximation was made by NSC for the restraint load-deflection curve supplied by GE. This approximation is illustrated by Figure 3.6-48. The effect of this approximation is to give lower energy absorption of a given restraint deflection. Typically, this yields higher restraint deflections and lower restraint to structure loads than the GE analysis. The deflection limit used by NSC is the design deflection at one-half the ultimate uniform strain for the GE restraint design. The restraint properties used for both analyses are presented by Table 3.6-4. Break locations and restraints analyzed are shown by Figure 3.6-49.

A comparison of the NSC analysis with the PDA analysis, as presented by Table 3.6-5, shows that PDA predicts higher loads in 15 of the 18 restraints analyzed. This is due to the NSC model including energy absorbing effects in secondary pipe elements and structural members. However, PDA predicts higher restraint deflections in 50 percent of the restraints. The higher deflections predicted by NSC for the lower loads are caused by the linear approximation used for the force-deflection curve, rather than by differences in computer techniques. This comparison demonstrates that the simplified modeling system used in PDA is adequate for pipe rupture loading, restraint performance, and pipe movement predictions within the meaningful design requirements for these low probability postulated accidents.

3.6.2.3 Dynamic Analysis Methods to Verify Integrity and Operability

3.6.2.3.1 Jet Impingement Analyses and Effects on Safety Related Components

Potential jet impingement effects are identified by comprehensive reviews of all areas of the plant which contain high energy piping. Potential spray and environmental effects of cracks in moderate energy piping are identified by comprehensive reviews of plant areas containing moderate energy piping. Jets from all postulated breaks are identified. All structures, systems and components that can possibly be struck by each jet are reviewed to determine which are safety related.

Criteria used for evaluating the effects of fluid jets on safety related structures, systems, and components are as follows:

a. Safety related structures, systems, and components should not be so impaired as to preclude essential functions.

b. Safety related structures, systems, and components which are not necessary to safely shut down the plant for a given postulated pipe break need not be protected from the consequences of the fluid jet.

c. Safe shutdown of the plant following postulated pipe rupture in the reactor coolant pressure boundary must not be aggravated by sequential failures of safety related piping and required ECCS performance must be maintained.

d. Offsite dose limits specified in 10 CFR 100 must be met.

e. Postulated design basis breaks resulting in jet impingement loads are assumed to occur in high energy lines at full (100 percent) power operation of the plant.

f. Through wall leakage cracks are postulated to occur in moderate energy lines and are assumed to result in wetting and spraying of safety related structures, systems, and components.

g. Reflected jets are considered only when there is an obvious reflecting surface (such as flat plate) which directs the jet onto a safety related target. Only the first reflection is considered in evaluating potential targets.

The analytical methods used to determine which targets will be impinged upon by a fluid jet and the corresponding jet impingement load include the following:

a. The direction of the fluid jet for purposes of determining impingement loads is based upon the position of the pipe during steady state blowdown. Intermediate positions are also reviewed to identify all impacted structures, systems and components.

b. The impinging jet proceeds along a straight path.

c. The total impingement force acting upon any cross sectional area of the jet is time and distance invariant, with a total magnitude equivalent to the fluid blowdown force as defined below.

d. The jet impingement force is uniformly distributed across the cross sectional area of the jet and only the portion intercepted by the target is considered.

e. The break opening may be assumed to be a circular orifice of cross sectional flow area equal to the effective flow area of the break.

f. The jet impingement force is equal to the steady state value of the fluid blowdown force as calculated using the methods described in Section 3.6.2.2.1.

g. The distance of jet travel is divided into two or three regions. Region 1 (see Figure 3.6-50) extends from the break to the asymptotic area. Within this region the discharging fluid flashes and undergoes expansion from the break area pressure to atmospheric pressure. In region 2 the jet remains at a constant diameter. For partial separation circumferential breaks the area increases as the jet expands. Therefore, it is assumed that region 3 never occurs. In region 3 (except in partial separation circumferential breaks) interaction with the surrounding environment is assumed to start and the jet expands at a half angle of 10 degrees.

h. Moody(6) has developed a simple analytical model for estimating the asymptotic jet area for steam, saturated water and steam-water blowdown conditions. For fluids discharging from a break and which are below the saturation temperature at the corresponding room pressure or have a pressure at the break area equal to the room pressure, expansion does not occur.

i. The distance downstream from the break where the asymptotic area is reached (region 1) has been found by Moody (for circumferential and longitudinal breaks) to be approximately equal to five pipe diameters. Assuming a linear expansion from the break area to the asymptotic area, the jet shape can be defined for region 1, as well as for regions 2 and 3. Figure 3.6-51 is used to determine the asymptotic area.

(a) Flat Surface

For a case where a target with physical area, A_t, is oriented at an angle, $\emptyset$, with respect to the jet axis and with no flow reversal, the effective target area, A_{te}, is as follows:

$$A_{te} = A_t \sin \emptyset$$

(b) Pipe Surface

As the jet hits the convex surface of the pipe, the forward momentum of the jet is decreased rather than stopped. The jet impingement load on the impacted area is therefore reduced. The analytically determined shape factor for a cylindrical surface is 0.5. The effective target area, A_{te}, is as follows:

$$A_{te} = D_A(D)$$

Where:

D_A = Diameter of the jet at the target interface.
D = OD of the target pipe for a fully submerged pipe.

When the target pipe is larger than the area of the jet, the effective target area equals the expanded jet area:

$$A_{te} = A_x$$

(c) For all cases the jet area, A_x, is assumed to be uniform and the load is uniformly distributed on the impinged target area, A_{te}.

m. For the partial separation circumferential break described in Item k, above, the target loads are calculated in a similar manner, except that the jet cross section appears as shown by part (B) of Figure 3.6-50 and A_R equals A_x and D_A equals M and is calculated in accordance with Item k.7, above.

Evaluation of the ability of potential targets to withstand the jet impingement loads is performed using the following methods:

a. Evaluation of Piping Systems under Jet Impingement Loads

1. General Electric piping:

(a) Jet impingements on piping are considered faulted loads; therefore, only primary stresses are considered when comparing to ASME Code Section III service level D allowables.

(b) Motions of piping due to jet impingement loads due to yielding are limited by structural steel, pipe whip restraints, snubbers or other equipment capable of carrying the jet load. These motions are self-limiting; therefore, they are considered secondary stresses and are not included in the primary stress calculations.

2. Balance-of-plant piping:

Jet impingement loads on piping are considered emergency loads and are evaluated as primary stresses in ASME Code Section III piping analyses. Emergency service limits are used. Upset service limits may be used for piping whose function is required for the given event.

3. Each jet impingement load is applied independently to the piping system and the load which supplies the largest bending moment for each particular component is used for the evaluations of the pressure retaining capability or functionality of that component.

4. Jet impingement load can be characterized as a two part load application on a piping system as follows:

   (a) Dynamic Load Portion

   Where static analysis methods are used, a suitable dynamic load factor (DLF) is applied to the static load. Dynamic load factors are conservatively estimated using Reference 10. Snubbers are assumed to be activated and the calculated moments or stresses are combined with concurrent vibratory dynamic load cases by the square root of the sum of the squares (SRSS) method.

   (b) Static Load Portion

   Where a steady state static load is being applied to the piping system, snubbers are not activated and the calculated moments or stresses are combined with other simultaneous loads using the absolute sum method.

b. Evaluation of Structural Components under Jet Impingement Loads

1. Each jet impingement load is applied independently to the structure and the load which results in the largest internal stress is used for evaluation of the structural component.

2. Specifically designed jet impingement barriers, wherever installed, are considered structural components.

3. Jet impingement load can be characterized as a two part load application as follows:

   (a) Dynamic Load Portion

   Where static analysis methods are used, a suitable DLF is applied to the static load. Dynamic load factors are conservatively estimated using Reference 10. The ratio of the duration of the applied load and the period of the structure in the direction of the applied load is used in the appropriate response curves shown in this reference. Stresses are combined with concurrent vibratory dynamic load cases by the SRSS method.

(b) Static Load Portion

The static load portion of the impingement is combined arithmetically with other simultaneous loads by the absolute sum method.

c. Evaluation of Jet Impingement Loads on Mechanical System Components

1. The physical configuration of valves, pumps, etc., is approximated by rectangular and cylindrical solid shapes enveloping the component elements for the purposes of determining angular deflection coefficients and shape factors.

2. Loads are considered to be part of piping or structural loads due to jet impingement, according to the physical arrangement of the target. Moments are included in the piping loads for jet impingement on valve operators where a component of the valve loading is normal to the pipe axis.

d. Evaluation of Jet Impingement Loads on Electrical Cable Trays

The only safety related electrical cable trays subject to impingement by high energy jets are located in the RWCU heat exchanger rooms. These trays are fully protected by jet shields.

e. Evaluation of Jet Impingement Loads on Electrical Conduit and Instrumentation Impulse Lines

The design criteria for routing of electrical conduit and instrument impulse lines is intended to ensure that impingement by high energy jets does not occur. However, this is not always feasible. Support of conduit and impulse lines subject to jet impingement is established by adjustments to spacing criteria which assure conduit integrity under governing load conditions. Design of special supports for rigid conduit and routings of flexible connections to equipment consider individual load conditions from impacting jets. Jet shields are used to protect against jet impingement if protection by support design and by routing is not feasible.

f. Evaluation of Jet Impingement Loads on Instrument Racks and Panels

No safety related instrumentation racks or panels are subject to impingement by high energy jets.

3.6.2.3.2 Pipe Whip Effects on Safety Related Components

Potential pipe whip effects are identified by comprehensive reviews of all areas of the plant which contain high energy piping. Potential whips due to each postulated circumferential break, and displacement due to each longitudinal break, are evaluated. All structures, systems and components that can possibly be struck by each whipping or displacing pipe are reviewed to determine which are safety related. -15

Pipe whip (displacement) effects on safety related structures, systems, and components can be placed into two categories: pipe displacement effects on components (nozzles, valves, tees, etc.) located in the piping run in which the break occurred; and pipe whip or controlled displacements onto external components, such as building structure or other piping systems.

a. Pipe Displacement Effects on Components in the Same Piping Run

1. Criteria used for determining the effects of pipe displacements on in-line components are as follows:

   (a) Components, such as vessel safe ends and valves which are attached to the broken piping system and do not serve a safety function or the failure of which would not further escalate the consequences of the accident, need not be designed to satisfy ASME Code, Section III, imposed limits for essential components under emergency loading.

   (b) If these components are required for safe shutdown or serve a safety function to protect the structural integrity of an essential component, limits to satisfy ASME Code requirements for component emergency conditions and limits to ensure operability, if required, are satisfied.

2. Methods used to calculate pipe whip loads on piping components located in the piping run where the break is postulated to occur are described in Section 3.6.2.2.2.

b. Pipe Displacement Effects on Structures, Other Systems, and Components

1. The criteria used to assure the mitigation of the effects of high energy pipe whip on structures, systems, and components require that the arrangement of pipe whip restraints, supporting structures, and piping system components preclude impact of whipping pipe on any structure, system, or component essential to the safe shutdown of the plant in the event of occurrence of a given postulated pipe rupture. In exceptional cases a damage study as described in section 3.6.2.1.4.c above is used to show that pipe whip impact on any structures, systems or components essential to safe shutdown does not compromise the safe shutdown function of those structures, systems or components. -2,-17

2. Methods used to calculate the pipe whip dynamic response from high energy pipe rupture are given by the piping configuration and are discussed in Section 3.6.2.2.2. Loading combinations and design criteria used to assure the physical limits of motion of ruptured pipe are presented in Section 3.6.2.3.3.1.

3.6.2.3.3 Loading Combinations and Design Criteria for Pipe Whip Restraints

Pipe whip restraints (i.e., those devices which serve only to control movement of a ruptured pipe following gross failure), and torsional and moment restraints, as differentiated from simple piping supports, are designed to function and carry load for an extremely low probability gross failure in a piping system containing high energy fluid. Piping integrity does not depend upon the pipe whip restraints for any loading combination. Piping integrity in high energy containment penetration regions is assured by torsional and moment restraints. When piping integrity is lost as a result of the occurrence of a postulated break, the pipe whip restraints act to limit movement of the broken pipe to an acceptable distance. Pipe whip restraints are subject to once in a lifetime loading. For purposes of design the pipe break event is considered to be a faulted plant condition and the pipe, associated restraints, and structures to which restraints are attached are analyzed and designed accordingly. Pipe whip restraints are designed to strain limits as described below in this section. -23,-24 -23,-24 -23,-24

Some piping supports are also designed to accommodate pipe rupture loads. Torsional and moment restraints are also designed for piping support loads. -23,-24

3.6.2.3.3.1 Main Steam Pipe Whip Restraints - Inside Containment

Pipe whip restraints designed, tested, fabricated, and installed by GE for the main steam piping use energy absorbing U-rods to attenuate the kinetic energy of a ruptured pipe. A typical pipe whip restraint is illustrated by Figure 3.6-54. A principal feature of these restraints is that they are installed with several inches of annular clearance between the restraint and the process pipe. This arrangement allows for installation of normal piping insulation and unrestricted thermal movement of the piping. All supports and pipe whip restraints are monitored during pre-operational testing to provide verification of adequate clearances prior to plant operation.

3.6.2.3.3.1.1 Restraint Design Objectives

Specific design objectives for the restraints are as follows:

a. The restraints must in no way increase reactor coolant pressure boundary stresses during any normal mode of reactor operation or condition.

b. The restraint system must function to stop movement of a failed pipe (gross loss of piping integrity) without allowing damage to critical components or missile development.

c. Restraints should produce minimum hindrance to performance of inservice inspection of process piping.

3.6.2.3.3.1.2 Restraint Dynamic Loads

For purposes of design the pipe whip restraints are designed for the following dynamic loads:

a. Blowdown thrust of the pipe section that impacts the restraint.

b. Dynamic inertia loads of the moving pipe section which is accelerated by the blowdown thrust and subsequent impact on the restraint.

Design characteristics of the pipe whip restraints are included and verified by the pipe whip dynamic analysis described in Section 3.6.2.2.2. Since the pipe whip restraints are not contacted during normal plant operation, the postulated pipe rupture event is the only design loading condition.

3.6.2.3.3.1.3 Restraint Components

The main steam pipe whip restraints are composed of several components, each of which performs a different function. These components are categorized as Types I, II, III, and IV, as described below:

a. Type I - Restraint Energy Absorption Members

   Restraint energy absorption members, under the influence of impacting pipe (pipe whip), absorb energy by significant plastic deformation (e.g., U-rods).

b. Type II - Restraint Connecting Members

   Restraint connecting members are those components which form a direct link between the restraint plastic members and the structure (e.g., clevises, brackets, pins).

c. Type III - Restraint Connecting Member Structural Attachments

   Restraint connecting member structural attachments are those fasteners which provide the method of securing the restraint connecting members to the structure (e.g., weld attachments, bolts).

d. Type IV - Structural and Civil Components

   Structural and civil components are the steel and concrete structures which ultimately must carry the restraint load (e.g., biological shield, trusses).

Each of the types of components (I through IV) is typically constructed of a different material, with a different design objective, to perform the overall design function. Therefore, the material and inspection requirements and design limits for each type of component are somewhat different. The requirements for each type of component are as follows:

(d) Procedures and welders will be qualified in accordance with the latest AWS Code for welding in building structures.

d. Type IV Restraint

Material, inspection, and design requirements for structural and civil components are provided by industry standards, such as AISC, ACI, and ASME (ASME Code, Section III, Division II), along with appropriate requirements imposed for similar loading events. These components are also designed for other operational and accident loadings, seismic loadings, wind loadings, and tornado loadings.

The design basis approach of categorizing components is consistent in allowing less stringent requirements for those components subject to lower stresses. Considerable strength margins exist in Type II through IV components even to the limit of load capacity (fracture) of a Type I component. Impact properties in all components are considered since brittle type failures could reduce restraint system effectiveness.

3.6.2.3.3.1.4 Restraint Material Allowables

In addition to the design considerations discussed above, strain rate effects and other material property variations have been considered in the design of the pipe whip restraints. The material properties used in design have included one or more of the following methods:

a. Code minimum or specification yield and ultimate strength values for the affected components and structures are used for both the dynamic and steady state events; or

b. Not more than a 10 percent increase in code or specification values is used when designing Type IV components or structures for the dynamic event. Code minimum or specification yield and ultimate strength values are used for the steady state loads; or

c. Representative or actual test data values are used in the design of components and structures; or 2-2

b. Control Rod Drive Bundles at Reactor Pressure Vessel Azimuths 106° and 254°

The CRD bundles are not essential, in themselves, and could be overstressed by the impact of a main steam jet. Only open withdraw lines are essential for safe shutdown since lines may be broken or damaged, but not crimped closed. Figure 3.6-82 shows the physical arrangement and impact loads.

The main steam jet impact loads noted in Items a and b, above, are resolved as follows:

a. High Pressure Core Spray Injection Pipe

The impact on the HPCS pipe of a jet from the ruptured main steam line was analyzed as both an impact load and as a steady state load in combination with thermal, deadweight, and seismic loads acting on the pipe simultaneously. At no time was the maximum allowable stress exceeded at any point in the impacted HPCS piping run.

b. Control Rod Drive Bundles at Reactor Pressure Vessel Azimuths 106° and 254°

The impact of a jet from the ruptured main steam line was analyzed as described in Items a and b, above, for the HPCS and LPCI A piping for both pipe stress and support adequacy. The entire bundle of CRD insert and withdraw lines is supported by a single set of supports. These supports adequately withstand the jet load in combination with thermal, deadweight, and seismic loads. However, the individual withdraw lines would be over stressed if impacted by the full force of the jet from the postulated main steam line rupture; there are no reliable analytical models which could be used to show the line overstresses could not result in crimping. Therefore, a jet shield is provided above the CRD bundle arrangement to prevent overstressing of individual pipes as a result of this postulated event (see Figure 3.6-83).

3.6.2.5.3.2 Jet Effects for Postulated Ruptures of Recirculation Piping System - Inside Containment

Fluid jet thrusts for each of the postulated break locations in the recirculation piping system are listed in Table 3.6-13. Structures, systems, or components essential to safe plant shutdown in the case of a particular pipe break and subject to impact by the steam jet from the particular break are discussed in the following paragraphs:

a. High Pressure Core Spray Piping

   Figure 3.6-84 illustrates the physical arrangement and impact loads.

b. Control Rod Drive Bundles - Longitudinal Break

   Similar jets from four different postulated recirculation line breaks could impact any of the four CRD bundles at reactor pressure vessel azimuths 74°, 106°, 254° or 286°. Figure 3.6-85 illustrates the physical arrangement and impact loads.

The recirculation line jet impact loads noted in Items a and b, above, are resolved as follows:

a. High Pressure Core Spray Piping

   The impact on HPCS pipe of a jet from a ruptured recirculation discharge header was analyzed as both an impact load and as a steady state load in combination with thermal, deadweight, and seismic loads acting on the pipe simultaneously. At no time was the maximum allowable stress exceeded at any point in the impacted HPCS piping run.

b. Control Rod Drive Bundles - Circumferential Break

   A circumferential break in the recirculation discharge header connection at any of four locations, 60°, 120°, 240°, or 300°, was found to result in a jet impact that caused overstress of individual withdraw lines and exceeded the design capacity of the entire impacted CRD tube bundle supports if a jet shield were supported from the bundle. Therefore, a jet shield is provided around the CRD bundle arrangement to prevent over stress of individual withdraw pipes or tube bundle supports as a result of the postulated event (see Figure 3.6-86).

3.6.2.5.3.3 Jet Effects for Postulated Ruptures of Feedwater Piping System - Inside Containment

Fluid jet thrusts for each of the postulated break locations in the feedwater piping system are listed in Table 3.6-14. Structures, systems, or components essential to safe shutdown of the plant in the case of a particular pipe break and which are jeopardized by the jet resulting from a particular break are discussed in the following paragraphs:

a. Control Rod Drive Bundles at Reactor Pressure Vessel Azimuths 74° and 286°

   Figure 3.6-87 illustrates the physical arrangement and impact loads for loop A piping. Loop B piping is a mirror image and affects the CRD bundle at azimuth 286°.

b. Low Pressure Core Injection B Piping

   Figure 3.6-88 illustrates the physical arrangement and impact loads.

The feedwater line jet impact loads noted in Items a and b, above, are resolved as follows:

a. Control Rod Drive Bundles at Reactor Pressure Vessel Azimuths 74° and 286°

   The full circumferential break in the 20-inch feedwater header results in a jet which was found to overstress individual withdraw lines and exceed the design capacity of the entire impacted CRD tube bundle supports if a jet shield were supported from the bundle. A jet shield is provided around the CRD bundle arrangement to prevent overstress of individual withdraw lines or tube bundle supports as a result of the postulated event (see

9. Eiber, R.J., et. al., "Investigation of the Initiation and Extent of Ductile Pipe Rupture - Phase 1 Final Report - Task 17," Battelle Memorial Institute, BMI-1866, July 1969.

10. Biggs, J.M., "Introduction to Structural Dynamics," McGraw-Hill, 1964.

TABLE 3.6-1

HIGH ENERGY LINES[1]

System Number	System Designation
B-21	Main Steam - inside containment
N-11	Main Steam - outside containment
N-27	Feedwater System
B-33	Recirculation System
N-22	Main Steam System Drains - including RCIC steam drain
E-51	Reactor Core Isolation Cooling System - steam supply from main steam line "A" out to E51-MOF045 and E12-MOF052
E-51	RCIC Head Spray - from RPV to E51-AOF066
G-33	Reactor Water Cleanup System
G-36	RWCU Filter/Demineralizer System
E-12	Low Pressure Core Injection Loops "A", "B" and "C" (RHR) - from RPV to E12-F041A, B & C
E-21	Low Pressure Core Spray - from RPV to E21-F005
E-22	High Pressure Core Spray - from RPV to E22-F005
C-11	Control Rod Drive Hydraulic System
C-41	Standby Liquid Control Supply Line - from RPV to C41-F007
B-21	RPV Head Vent to Main Steam Line "A"
P-61	Auxiliary Steam System
M-29	Control and Computer Room Humidification System

NOTE:

1. Fluid systems that, during normal plant conditions, are either in operation or maintained pressurized under conditions where either or both of the following are met:

    a. maximum operating temperature exceeds 200°F, or

    b. maximum operating pressure exceeds 275 psig

TABLE 3.6-2

MODERATE ENERGY LINES[1]

System Number	System Designation
P-43	Nuclear Closed Cooling System
P-50	Containment Vessel Chilled Water System
P-54	Fire Protection System
P-11	Condensate Transfer - Storage System
P-46	Turbine Building Chilled Water
P-47	Control Complex Chilled Water
R-44	Diesel Generator Starting Air - from receiver tank to start air admission valves
N-71	Circulating Water System
N-26	Low Pressure Heater Drain System
N-23	Condensate Filtration System
N-24	Condensate Demineralizer System
G-50	Liquid Radwaste System
G-41	Fuel Pool Cooling and Cleanup System
P-71	Potable Water System
P-41	Service Water System
P-20	Make-up Water System
E-12	Residual Heat Removal System - except high energy (Table 3.6-1)
P-55	Building Heating Hot Water System
N-21	Condensate System
N-11	Condenser Air Ejector Steam System
P-12	Condensate Seal Water System
P-21	Two-Bed Demineralizer Water System
P-22	Mixed Bed Demineralizer Water System
E-32	MSIV Leakage Control System Downstream of Normally Closed Isolation Valves
E-51	RCIC - Except High Energy Lines (Table 3.6-1)

NOTE:

1. Fluid systems that, during normal plant conditions, are either in operation or maintained pressurized (above atmospheric pressure) under conditions where both of the following are met:

   a. maximum operating temperature is 200°F or less, and
   b. maximum operating pressure is 275 psig or less

TABLE 3.6-7
SUMMARY OF MAIN STEAM PIPING SYSTEM OPERATING STRESSES AT BREAK LOCATIONS(1)

a. Main Steam "A" Piping:

BREAK I.D. NO:	NODE NO.	G.E. REF: 22A7134 PAGE:(2)	STRESS RATIOS EQ. (10) Sn/Sm	STRESS RATIOS EQ. (12) Sn/Sm	STRESS RATIOS EQ. (13) Sn/Sm	USAGE FACTOR U	BREAK TYPE	BREAK BASIS SECTION NO.
SA1	001	86, 88	2.187	1.146	1.338	0.0192	Circ.	Terminal End
SA2	002	93, 95	3.621	2.718	0.987	0.0389	Circ.	3.6.2.1.5.a.2.(c)
SA2LL	002	93, 95	3.621	2.718	0.987	0.0389	Long.	3.6.2.1.5.a.2.(c)
SA3C	029	298, 300	2.181	0.798	1.392	0.2579	Circ.	Terminal End
SA3A	021	284, 286	2.199	1.140	1.008	0.0055	Circ.	Terminal End
SA4LL	030	173, 175	3.084	0.585	1.899	0.0927	Long.	3.6.2.1.5.a.2.(b)
		CRITERIA:	2.4 or 3.0	2.4, and (10) > 3.0	2.4, and (10) > 3.0	0.10, and (10) ≥ 2.4		

b. Main Steam "C" Piping ("B" is a mirror image of Main Steam "C"):

BREAK I.D. NO:	NODE NO.	G.E. REF: 22A7135 PAGE:(3)	STRESS RATIOS EQ. (10) Sn/Sm	STRESS RATIOS EQ. (12) Sn/Sm	STRESS RATIOS EQ. (13) Sn/Sm	USAGE FACTOR U	BREAK TYPE	BREAK BASIS SECTION NO.
SC1	001	105, 107	2.082	0.957	1.350	0.0154	Circ.	Terminal End
SC2	002	112, 114	3.135	2.175	0.984	0.0141	Circ.	3.6.2.1.5.a.2.(c)
SC4LL	030	194, 196	3.018	0.621	1.962	0.1005	Long.	3.6.2.1.5.1.2.(b)
SC3C	023	427, 429	2.751	0.543	1.227	0.4003	Circ.	3.6.2.1.5.a.2.(b)
SC5LL	046	227, 229	2.961	0.618	1.986	0.0959	Long.	3.6.2.1.5.a.2.(b)
SC3A	020	413, 415	2.049	0.972	1.053	0.0042	Circ.	Terminal End
		CRITERIA:	2.4 or 3.0	2.4, and (10) > 3.0	2.4, and (10) > 3.0	0.10, and (10) ≥ 2.4		

TABLE 3.6-7 (Continued)

c. Main Steam "D" Piping:

BREAK I.D. NO:	NODE NO.	G.E. REF: 22A7136(4) PAGE:	STRESS RATIOS EQ. (10) Sn/Sm	EQ. (12) Sn/Sm	EQ. (13) Sn/Sm	USAGE FACTOR U	BREAK TYPE	BREAK BASIS SECTION NO.
SD1	001	88, 90	2.178	1.140	1.365	0.0187	Circ.	Terminal End
SD2A	002	95, 97	3.570	2.709	1.023	0.0364	Circ.	3.6.2.1.5.a.2.(c)
SD2LL	002	95, 97	3.570	2.709	1.023	0.0364	Long.	3.6.2.1.5.a.2.(c)
SD3C	024	312, 314	2.367	0.804	1.782	0.6153	Circ.	3.6.2.1.5.a.2.(b)
SD4LL	030	153, 155	3.054	0.588	2.040	0.0977	Long.	3.6.2.1.5.a.2.(b)
SD5LL	046	228, 230	2.937	0.465	2.031	0.1028	Long.	3.6.2.1.5.a.2.(b)
SD6LL	062	263, 265	2.862	0.417	2.064	0.1020	Long.	3.6.2.1.5.a.2.(b)
SD7LL	090	193, 195	2.988	0.522	2.034	0.0968	Long.	3.6.2.1.5.a.2.(b)
SD3A	021	298, 300	2.238	1.152	1.101	0.0060	Circ.	Terminal End
		CRITERIA:	2.4 or 3.0	2.4, and (10) > 3.0	2.4, and (10) > 3.0	0.10, and (10) $\geq$ 2.4		

NOTES:

1. Underlining (_____) indicates determinative criteria for postulating a break. Partial underlining (_ _ _) indicates a break is postulated even though the value is slightly less than the criterion.

2. Reference: General Electric Document 22A7134 Rev. 2, "Design Report - ASME Boiler and Pressure Vessel Code, Section III: Line A Steam Piping. Project: Perry Nuclear Power Plant 1 and 2."

3. Reference: General Electric Document 22A7135 Rev. 2, "Design Report - ASME Boiler and Pressure Vessel Code, Section III: Line C Steam Piping. Project: Perry Power Plant 1 and 2."

4. Reference: General Electric Document 22A7136 Rev. 2, "Design Report - ASME Boiler and Pressure Vessel Code, Section III: Line D Steam Piping. Project: Perry Power Plant 1 and 2."

TABLE 3-6-9

SUMMARY OF RECIRCULATION PIPING SYSTEM OPERATING STRESSES AT BREAK LOCATIONS[1]

BREAK I.D. NO:	NODE NO.	G.E. REF: 22A7140 PAGE:[2]	STRESS RATIOS EQ. (10) Sn/Sm	EQ. (12) Sn/Sm	EQ. (13) Sn/Sm	USAGE FACTOR U	BREAK TYPE	BREAK BASIS SECTION NO.
RS1	100	183	1.869	0.525	1.509	0.0	Circ.	Terminal End
RD1	202	206, 307	2.568	1.188	1.629	0.0006	Circ.	Terminal End
RD2	183	311	2.001	0.843	1.602	0.0	Circ.	Terminal End
RD3	151	315	1.893	0.471	1.671	0.0	Circ.	Terminal End
RD4	184	319	2.130	0.705	1.614	0.0	Circ.	Terminal End
RD5	203	323, 324	2.721	1.263	1.626	0.0008	Circ.	Terminal End
RD6LL	169	336, 338	4.188	2.178	2.253	0.3754	Long.	3.6.2.1.5.a.2.(b)
RD7LL	141	350, 352	3.981	1.911	2.442	0.2298	Long.	3.6.2.1.5.a.2.(b)(c)
RD8LL	139	364, 366	3.585	1.788	2.442	0.0671	Long.	3.6.2.1.5.a.2.(c)
RD9LL	166	378, 380	4.170	2.118	2.289	0.3510	Long.	3.6.2.1.5.a.2.(b)
RD10	116	435, 436	4.230	2.235	2.397	0.0375	Circ.	3.6.2.1.5.a.2.(c)
		CRITERIA:	2.4 or 3.0	2.4, and (10) > 3.0	2.4, and (10) > 3.0	0.10, and (10) $\geq$ 2.4		

NOTES:

1. Underlining indicates determinative criteria for postulating a break. Partial underlining(_ _ _) indicates a break is postulated even though the value is slightly less than the criterion.

2. Reference: General Electric Document 22A7140 Rev. 0, "Design Report - ASME Boiler and Pressure Vessel Code, Section III, Recirculation Piping, Volume 1. Project: Perry Nuclear Power Plant - Units 1 and 2."

TABLE 3.6-12

FLUID BLOWDOWN THRUST TIME HISTORIES FOR MAIN STEAM PIPING SYSTEM

a. Line A - Inside Containment (for NSSS Design and Analysis):[1]

BREAK LOCATION[1]	TYPE OF BREAK	SIDE OF BREAK	F_o (kips)	F_{int} (kips)	F_{ss} (kips)	t_1 (sec)	t_2 (sec)
SA1	Circ.	Turbine	446	312	208	.0037	.0988
SA2A	Circ.	Vessel	446	446	497	.00187	.01227
SA2A	Circ.	Turbine	446	312	208	.0037	.0988
SA2LL	Longit.	-	446	446	519	.00122	.00263
SA3C	Circ.	Vessel	446	312	461	.01425	.05736
SA3C	Circ.	Turbine	446	312	208	.00086	.03323
SA3A	Circ.	Vessel	446	312	461	.01425	.05736
SA3A	Circ.	Turbine	446	312	208	.00086	.03323
SA4LL	Longit.	-	83	83	94	.00113	.00209

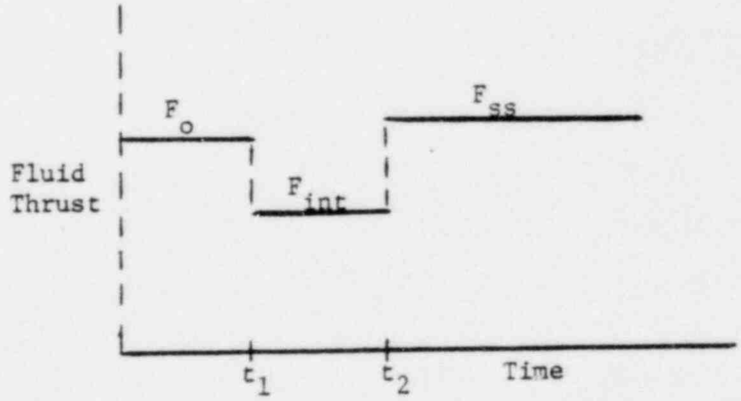


TABLE 3.6-12 (Continued)

b. Lines B, C - Inside Containment (For NSSS Design and Analysis):(1)

BREAK LOCATION(1)	TYPE OF BREAK	SIDE OF BREAK	F_o (kips)	F_{int} (kips)	F_{ss} (kips)	t_1 (sec)	t_2 (sec)
SC1	Circ.	Turbine	446	312	208	.0037	.0948
SC2A	Circ.	Vessel	446	446	506	.00187	.01227
SC2A	Circ.	Turbine	446	312	208	.0037	.0948
SC2LL	Longit.	-	446	446	521	.00122	.00274
SC3A	Circ.	Vessel	446	312	486	.01971	.06609
SC3A	Circ.	Turbine	446	312	208	.00094	.01523
SC3C	Circ.	Vessel	446	312	486	.01971	.06609
SC3C	Circ.	Turbine	446	312	208	.00094	.01523
SC4LL	Longit.	-	83	83	94	.00113	.00209
SC5LL	Longit.	-	83	83	94	.00113	.00209

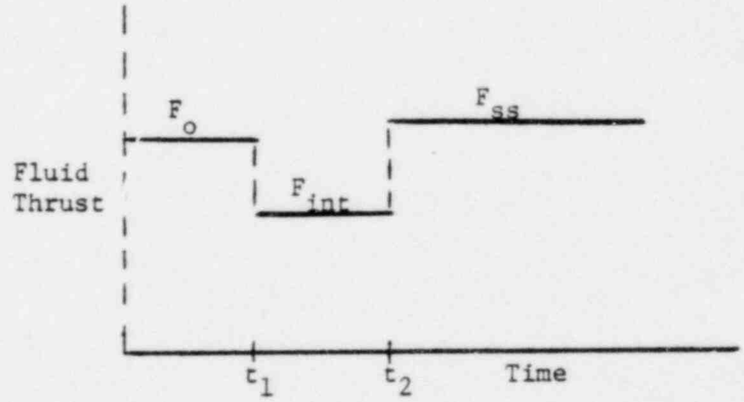


TABLE 3.6-12 (Continued)

c. Line D - Inside Containment (for NSSS Design and Analysis):[1]

BREAK LOCATION[1]	TYPE OF BREAK	SIDE OF BREAK	F_o (kips)	F_{int} (kips)	F_{ss} (kips)	t_1 (sec)	t_2 (sec)
SD1	Circ.	Turbine	446	312	208	.0037	.0988
SD2A	Circ.	Vessel	446	446	497	.00187	.01227
SD2A	Circ.	Turbine	446	312	208	.0037	.0988
SD2LL	Longit.	-	446	446	519	.00122	.00263
SD3A	Circ.	Vessel	446	312	461	.01425	.05736
SD3A	Circ.	Turbine	446	312	208	.00086	.03323
SD3C	Circ.	Vessel	446	312	461	.01425	.05736
SD3C	Circ.	Turbine	446	312	208	.00086	.03323
SD4LL	Longit.	-	83	83	94	.00113	.00209
SD5LL	Longit.	-	83	83	94	.00113	.00209
SD6LL	Longit.	-	83	83	94	.00113	.00209
SD7LL	Longit.	-	83	83	94	.00113	.00209

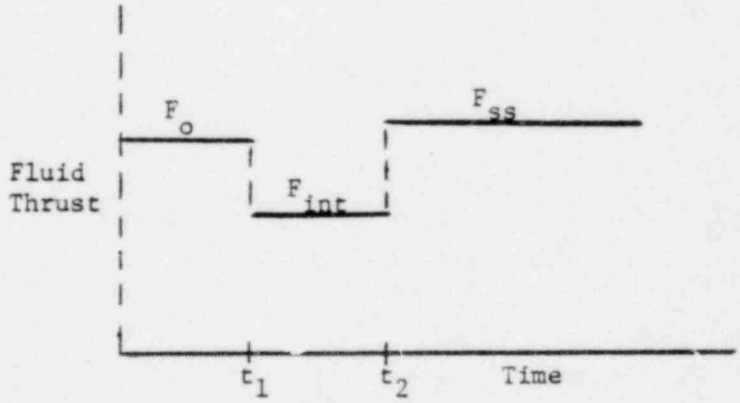


TABLE 3.6-12 (Continued)

d. 26" Breaks - Inside Containment (for BOP Design and Analysis):(1)

Flow Element Side of Break (2)

Time (seconds)	Thrust (Kips)
0-.001	0-450.
.001-.003	450.
.003-.009	315.
.009-∞	186.

Unrestricted Side of Break(2)

Time (seconds)	Thrust (Kips)
0-.001	0-450.
.001-.091(3)	305.

e. 28" Breaks - Outside Containment

Break Designation(4)	Time (sec)	Thrust (Kips)
Break No. 1		
Double-Ended Break		
Reactor Side (23.19" I.D.)	0. - .001	0 - 433.
	.001 - .21	433.
	.21 - ∞	366.
Turbine Side (25.15" I.D.)	0. - .001	0 - 509.
	.001 - .31	509.
	.31 - ∞	366.
Longitudinal Break (25.15" I.D.)	0. - .0118	0 - 509.
	.0118 - .21	509.
	.21 - ∞	454.
Break No. 2		
Longitudinal Break (25.15" I.D.)	0. - .0118	0 - 509.
	.0118 - .24	509.
	.24 - ∞	439.
Break No. 3		
Double-Ended Break		
Reactor Side (25.15" I.D.)	0. - .001	0 - 509.
	.001 - .31	509.
	.31 - ∞	425.

NOTES:

1. See Figure 3.6-65 for identification of postulated break locations.
2. I.D. of piping is 23.65". Credit is taken for the main steam flow element (I.D. = 12.125") on one side of full circumferential breaks.
3. Will decrease after this time.
4. Break designations are as follows:
   a. Break No. 1 - 3 feet outboard of column line AXA in the Turbine Building steam tunnel - Double-ended and longitudinal breaks (SA1 on Figure 3.6-75).
   b. Break No. 2 - 46 feet outboard of column line AXA in the turbine building steam tunnel - longitudinal break only (SA2 on Figure 3.6-75).
   c. Break No. 3 - 18 feet of piping from Break #2 toward the turbine - Double ended reactor side (SA4 on Figure 3.6-75).

TABLE 3.6-13

FLUID BLOWDOWN THRUST TIME HISTORIES FOR RECIRCULATION PIPING SYSTEM

BREAK LOCATION(1)	TYPE OF BREAK	SIDE OF BREAK	F_o (kips)	F_{int} (kips)	F_{ss} (kips)	t_1 (sec)	t_2 (sec)
RD1	Circ.	Pump	135	115	128	.00183	.02733
RD2	Circ.	Pump	135	115	128	.00183	.02733
RD3	Circ.	Pump	135	115	128	.00183	.02733
RD4	Circ.	Pump	135	115	128	.00183	.02733
RD5	Circ.	Pump	135	115	128	.00183	.02733
RD6LL	Longit.	-	214	189	196	.00036	.03348
RD7LL	Longit.	-	214	189	196	.00036	.03348
RD8LL	Longit.	-	214	189	196	.00036	.03348
RD9LL	Longit.	-	214	189	196	.00036	.03348
RD10	Circ.	Jet Pump	135	93	52	.00212	.01798
RD10	Circ.	Pump	135	101	115	.00036	.02386
RS1	Circ.	Pump	323	285	149	.00178	.08104

NOTE:

1. See Figure 3.6-66 for identification of postulated break locations.

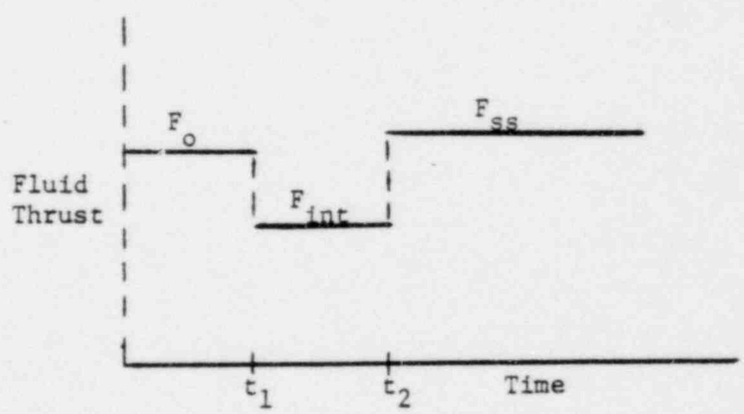


TABLE 3.6-14

FLUID BLOWDOWN THRUST TIME HISTORIES FOR FEEDWATER PIPING SYSTEM

This information is presented by Figures 3.6-95 through 3.6-98.

TABLE 3.6-15

FLUID BLOWDOWN THRUST TIME HISTORIES FOR EMERGENCY CORE COOLING PIPING SYSTEM

Time (sec.)	Thrust (kips)
0-0.001	0 - 128
0.001-∞	128

TABLE 3.6-16

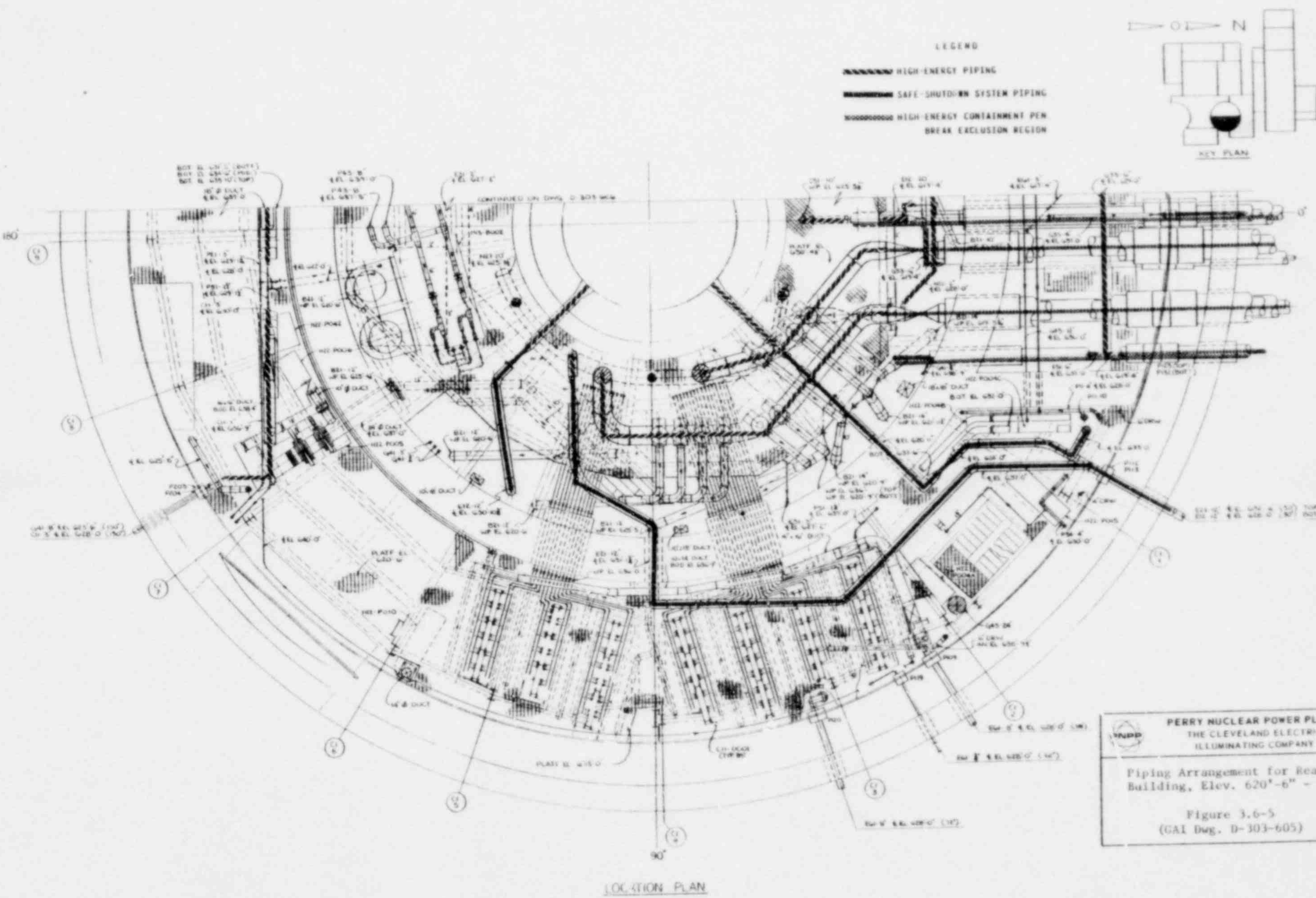
BLOWDOWN THRUSTS - HIGH ENERGY PIPE BREAKS OUTSIDE CONTAINMENT

System	Line Size	Break Type	Initial Blowdown Thrust lbs	Steady State Thrust lbs	Remarks
Main Steam	28"	Circumf.	433,000	386,000	Reactor side (1)
Main Steam	28"	Circumf.	509,000	366,000	Turbine side (1)
Main Steam	28"	Longit.	509,000	454,000	(1)
Main Steam	28"	Longit.	509,000	439,000	(2)
Main Steam	28"	Circumf.	509,000	425,000	Reactor side (3)
Feedwater	20"	Circumf.	265,000	99,000	Pump side (4)
Feedwater	20"	Longit.	265,000	99,400	(5)
Feedwater	36"	Longit.	859,700	310,000	(6)
Main Steam Drains	1-1/2"	Circumf.	1,720	(7)	
Main Steam Drains	2"	Circumf.	2,740	(7)	
Main Steam Drains	3"	Circumf.	6,610	(7)	
RWCU	4"	Circumf.	14,000	(7)	
RWCU	4"	Longit.	14,000	(7)	
RWCU	6"	Circumf.	32,250	(7)	
RWCU	6"	Longit.	32,250	(7)	
CRD Supply	2-1/2"	Circumf.	16,100	7,400	
Auxiliary Steam	1-1/2"	Circumf.	530	(7)	
Auxiliary Steam	2"	Circumf.	590	(7)	
Auxiliary Steam	2-1/2"	Circumf.	960	(7)	
Auxiliary Steam	3"	Circumf.	2,500	(7)	
Auxiliary Steam	4"	Circumf.	2,730	(7)	
Auxiliary Steam	4"	Longit.	2,730	(7)	
Auxiliary Steam	6"	Circumf.	5,500	(7)	
Auxiliary Steam	6"	Longit.	5,500	(7)	
Auxiliary Steam	10"	Circumf.	16,900	(7)	
Auxiliary Steam	10"	Longit.	16,900	(7)	

TABLE 3.6-17

PIPE WHIP ANALYSIS EMPLOYING STRAIN ENERGY METHODS OR COMPONENT DAMAGE STUDIES

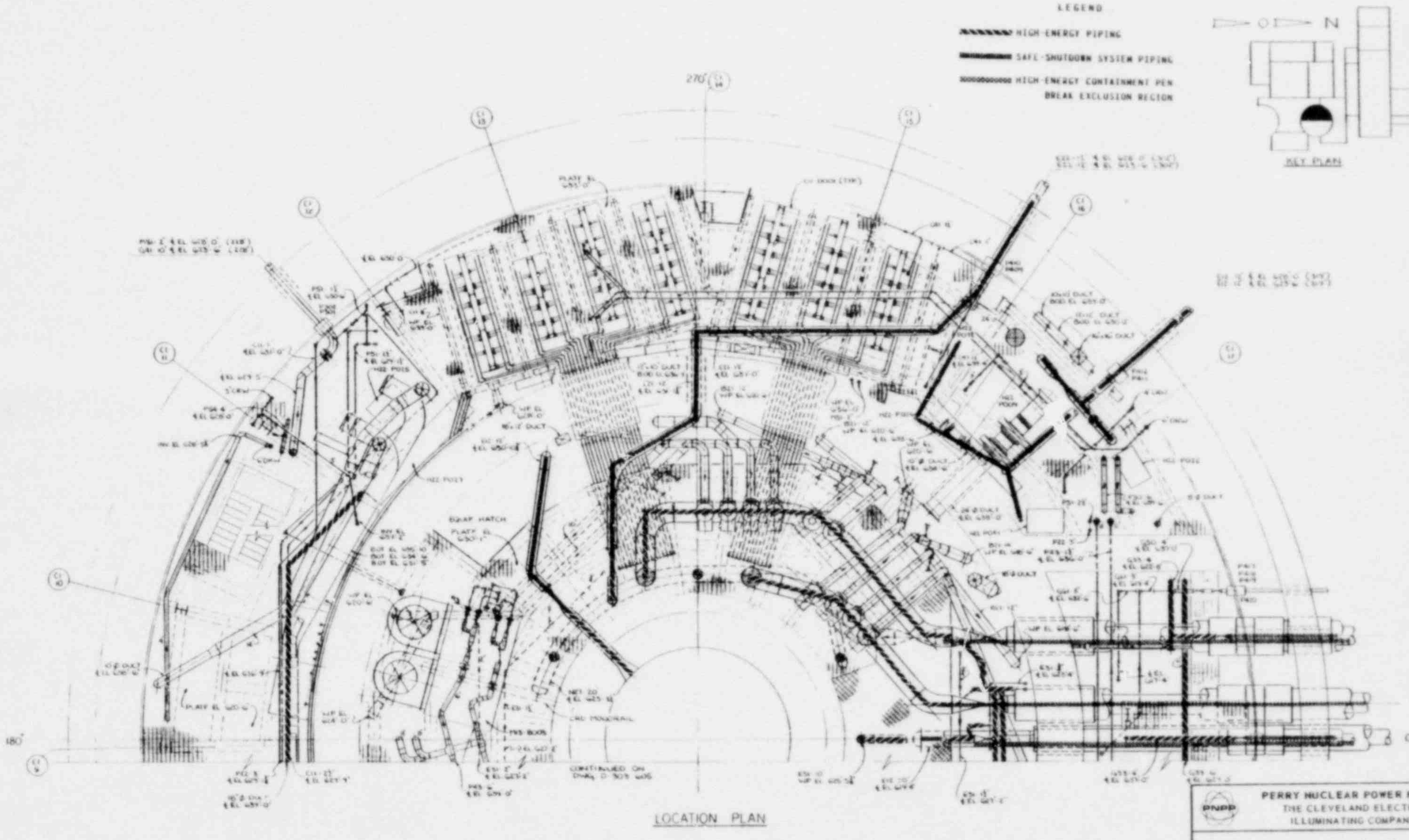
Impacting System	Impacted System or Component	Description of Protection or Qualification
B-21 Main Steam	Restraints	Energy-absorbing U-bolt restraints inside containment - see Section 3.6.2.3.3.1
B-33 Recirculation	Restraints	Energy-absorbing U-bolt restraints inside containment - see Section 3.6.2.3.3.2
N-27 Feedwater	Restraints	Energy-absorbing U-bolt restraints inside containment - see Section 3.6.2.3.3.3
E-12, 21, 22 ECCS Lines	Restraints	Energy-absorbing U-bolt restraints inside containment - see Section 3.6.2.3.3.4
E-51 RCIC Steam	Restraints	Energy-absorbing U-bolt restraints inside containment - see Section 3.6.2.3.3.5
Other High-Energy Lines	Restraints	Energy-absorbing U-bolt restraints inside containment - see Section 3.6.2.3.3.4, 5, 6
M-29 Control Rm. & Computer Humidifier	M-23 HVAC Duct M-25 HVAC Duct	Low pressure steam line impacts safety-related duct with negligible energy.
N-11 Main Steam in Turbine Building Steam Tunnel	Non-Safety Steam Tunnel	No impacted Safe-shutdown Components
N-27 Feedwater in Turbine Building Steam Tunnel	Non-Safety Steam Tunnel	No impacted Safe-shutdown Components



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Piping Arrangement for Reactor Building, Elev. 620'-6" - East

Figure 3.6-5
(GAI Dwg. D-303-605)

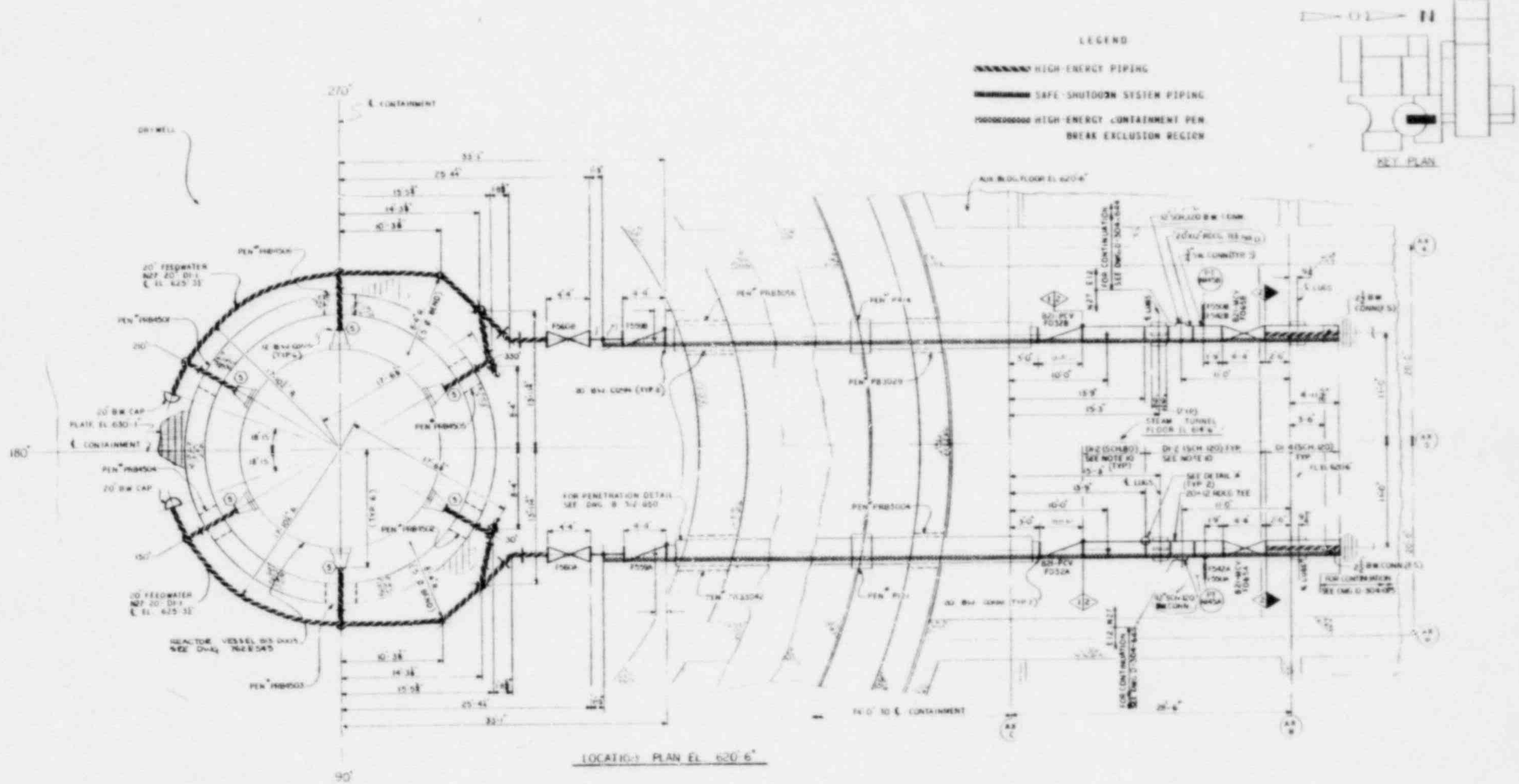


PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Piping Arrangement for Reactor Building, Elev. 620'-6" - West

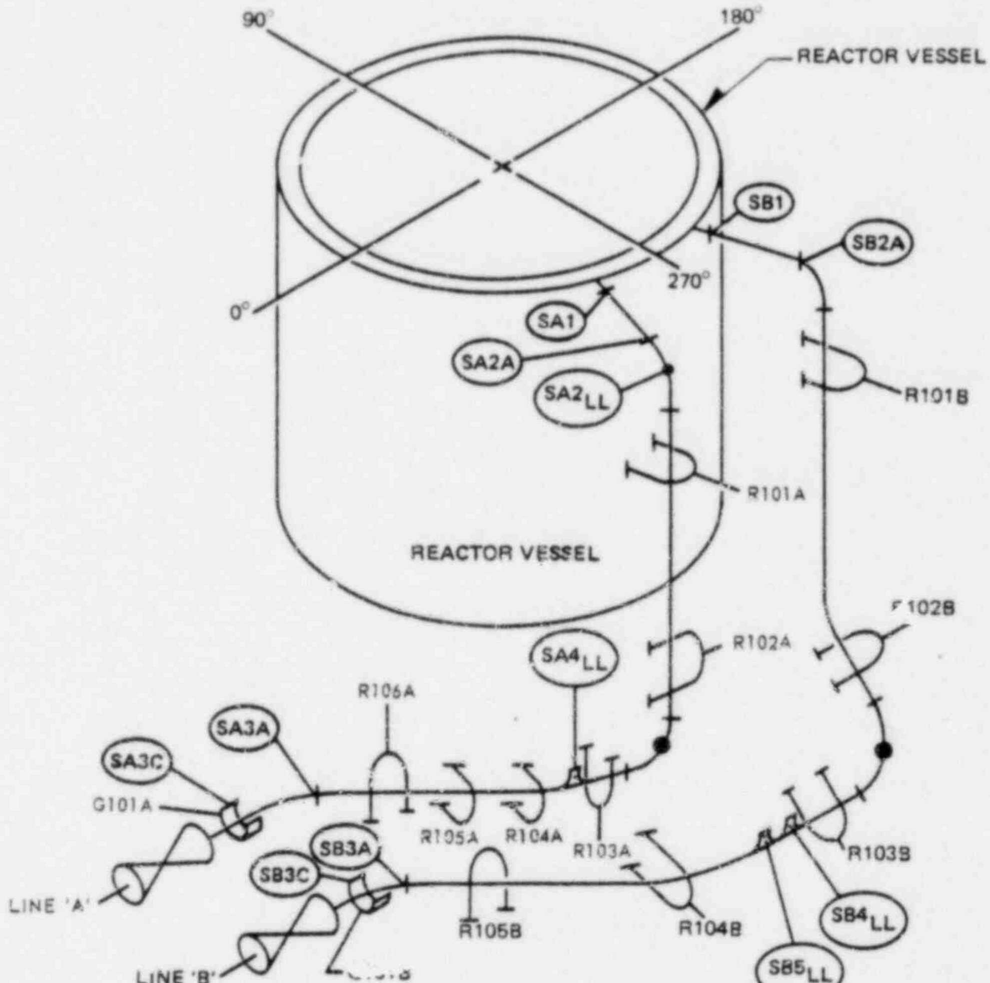
Figure 2

(GAI Dwg. D-3..-606)



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Piping Arrangement for Feedwater in Reactor Building and Steam Tunnel
Figure 3.6-7
(GAI Dwg. D-304-078)



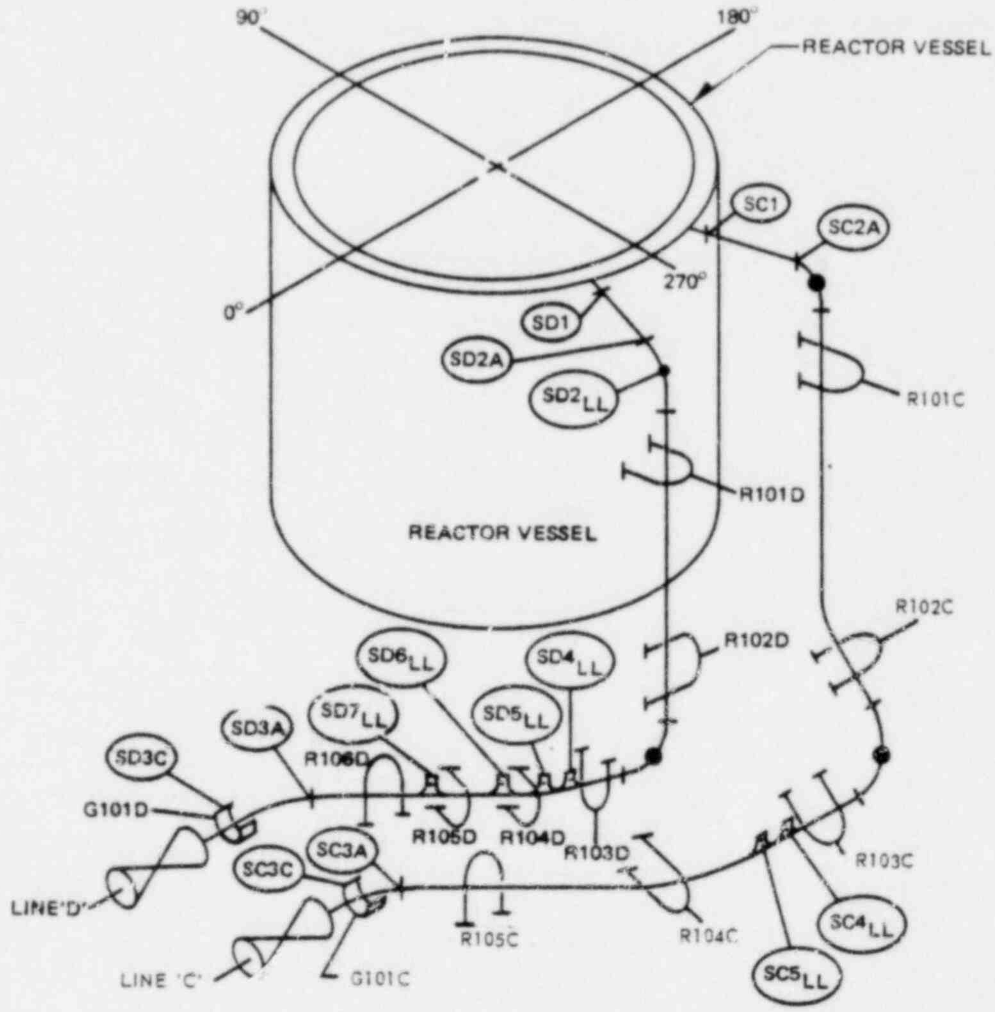
STEAM LINES B & A SHOWN
(LINE 'C' IS A MIRROR IMAGE OF LINE 'B')

PNPP

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Main Steam System Piping
Postulated Break Locations
and Restraint Locations

Figure 3.6-65 (Sheet 1 of 2)

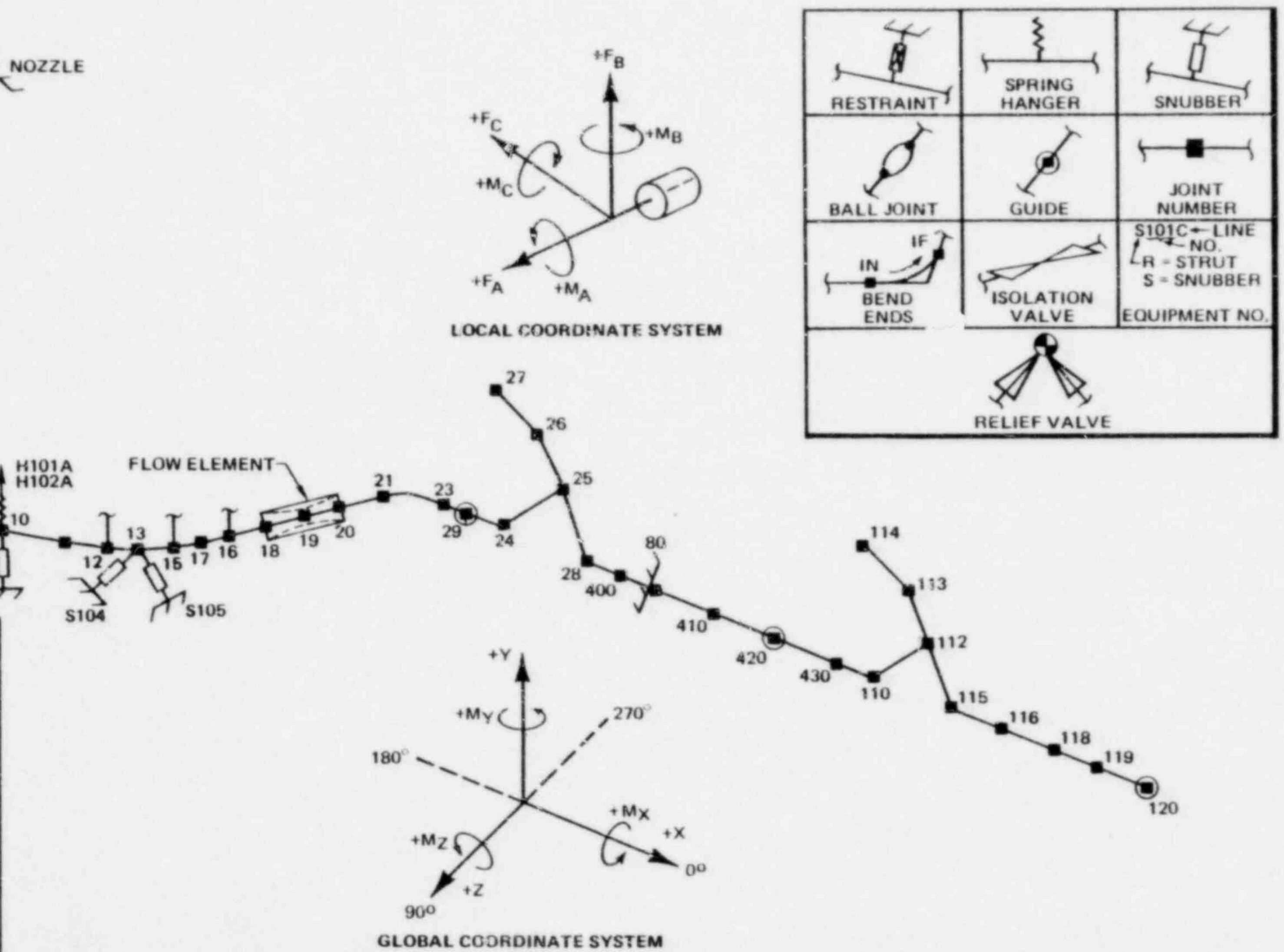


STEAM LINES C & D SHOWN
(LINE 'B' IS A MIRROR IMAGE OF LINE 'C')

PNPP PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Main Steam System Piping Postulated Break Locations and Restraint Locations

Figure 3.6-65 (Sheet 2 of 2)

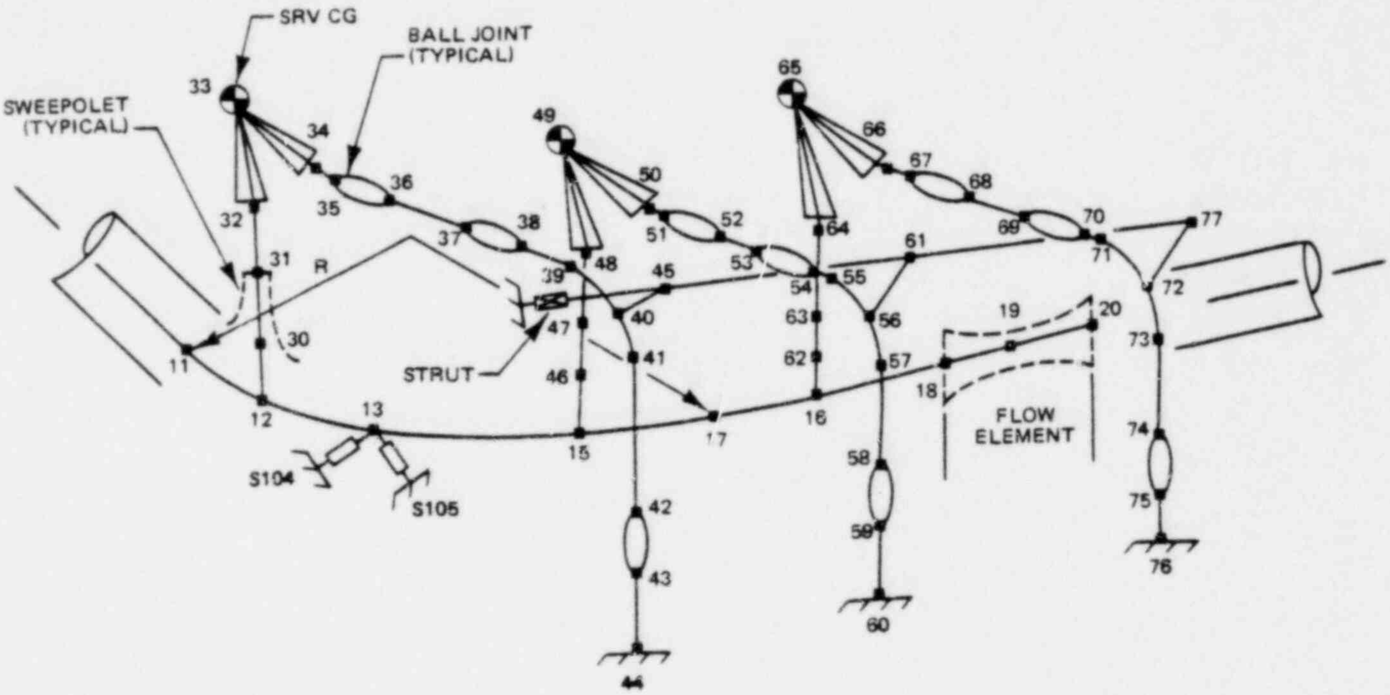


PNPP

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Line A - Main Steam (With RCIC) Piping Stress Node Locations

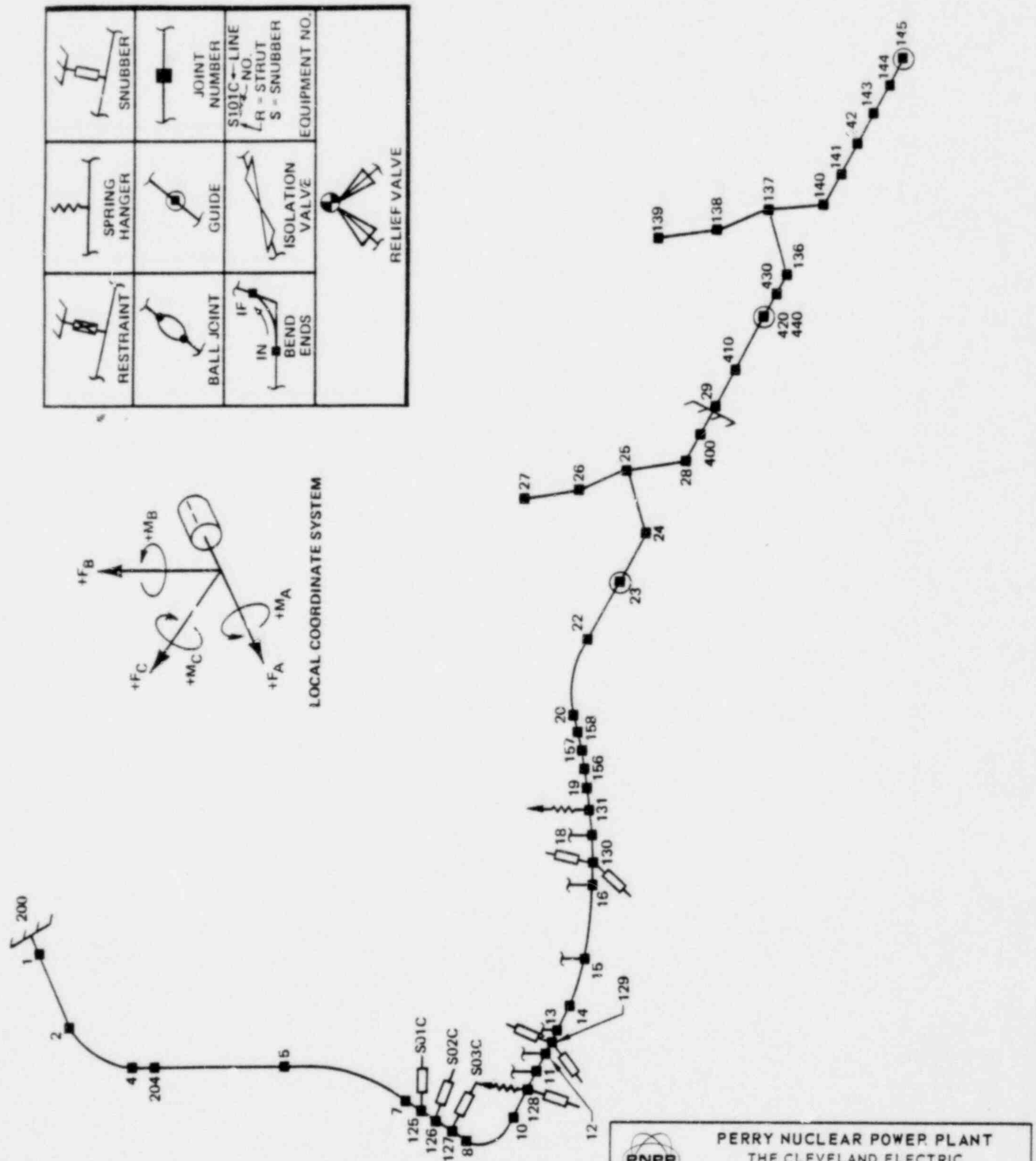
Figure 3.6-65a



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Line A - Main Steam Piping
Stress Node Locations (Sweepolet)

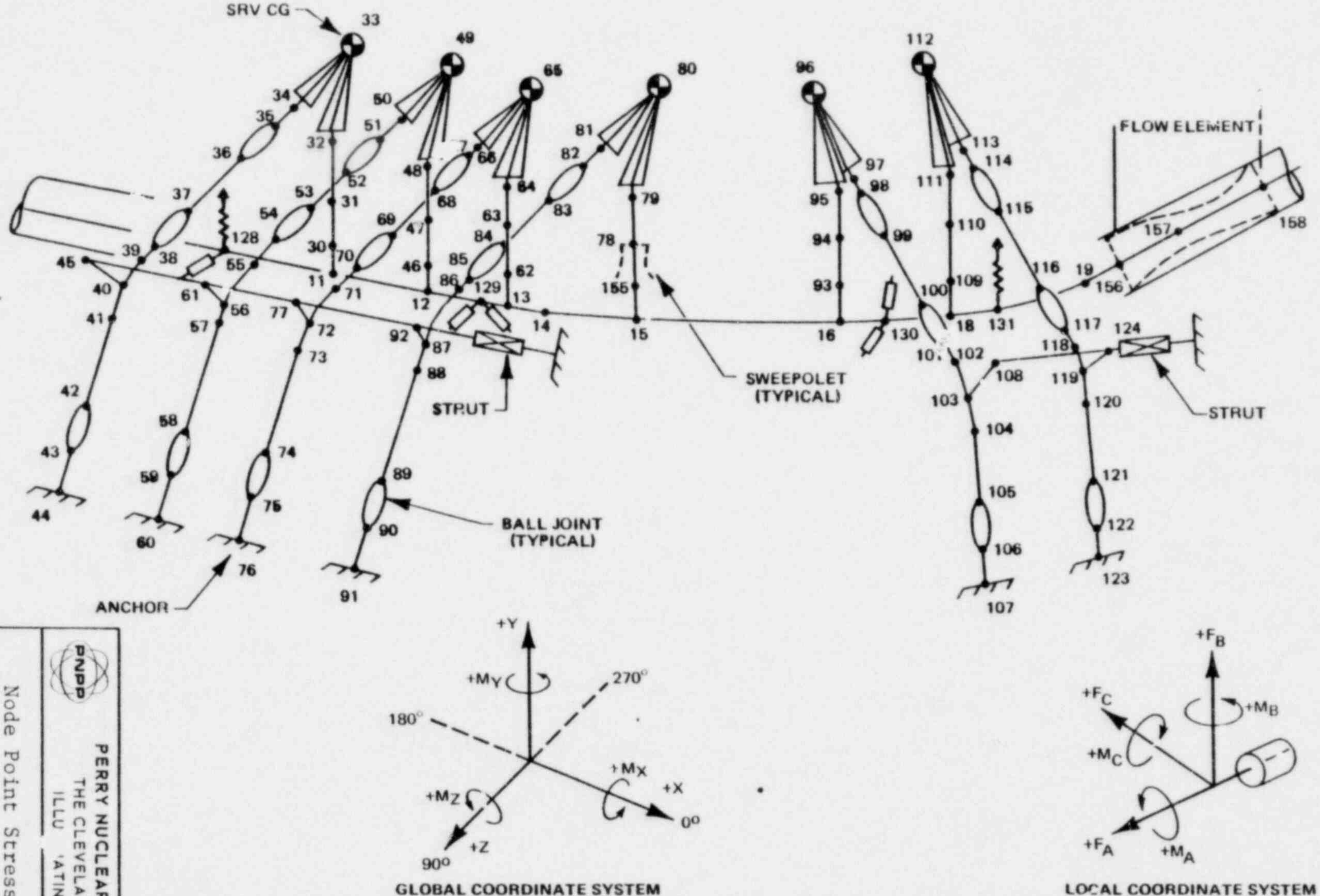
Figure 3.6-65b



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Node Point Stress Analysis
Diagram - Main Steam
Lines C and B

Figure 3.6-65c

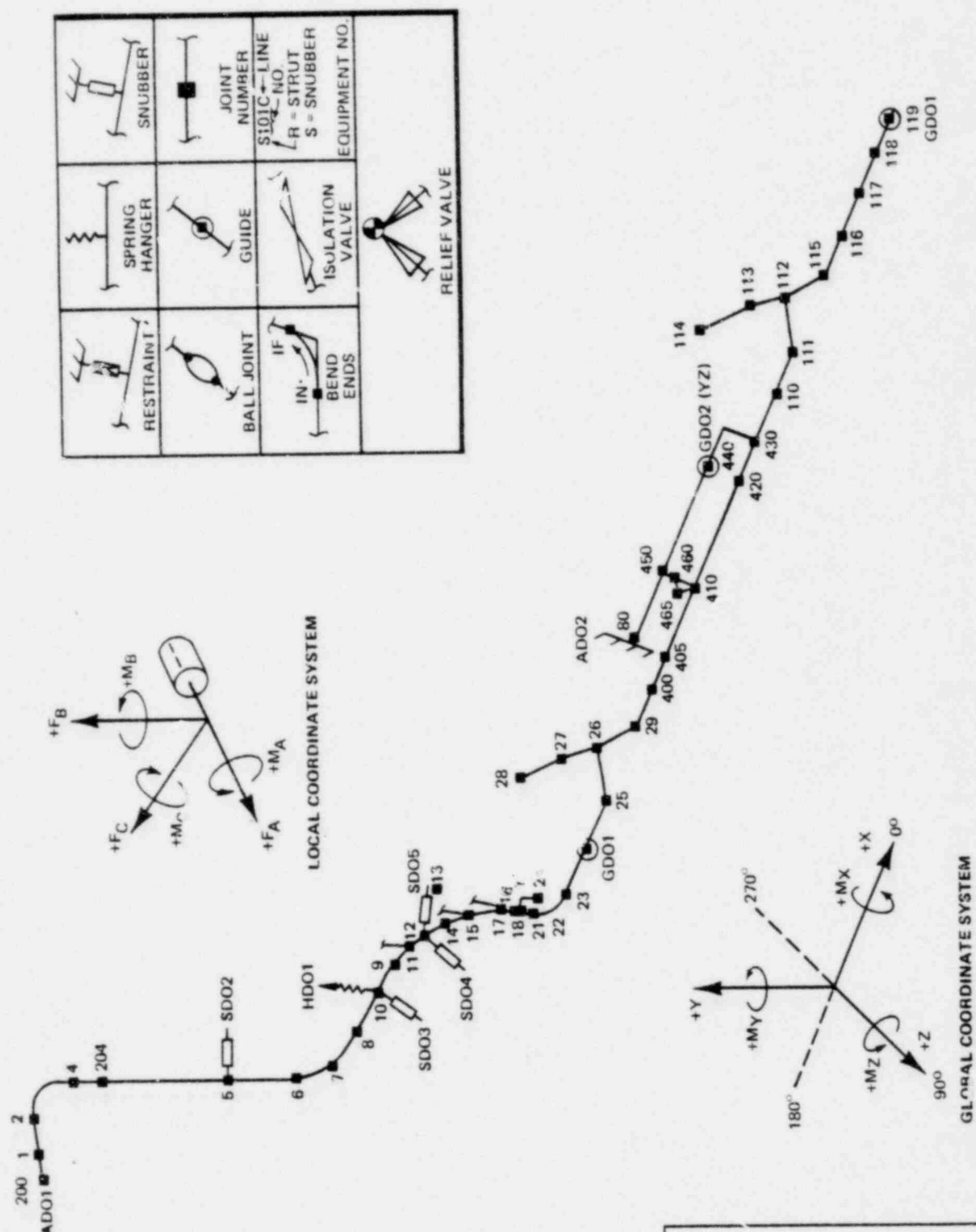


PNPP

PERRY NUCLEAR POWER PLANT
THE CLEVELAN[illegible] ECTRIC
ILLU[illegible]ATING [illegible]ANY

Node Point Stress Analysis Diagram — Main Steam Lines [illegible] and [illegible] (Sweepolet)

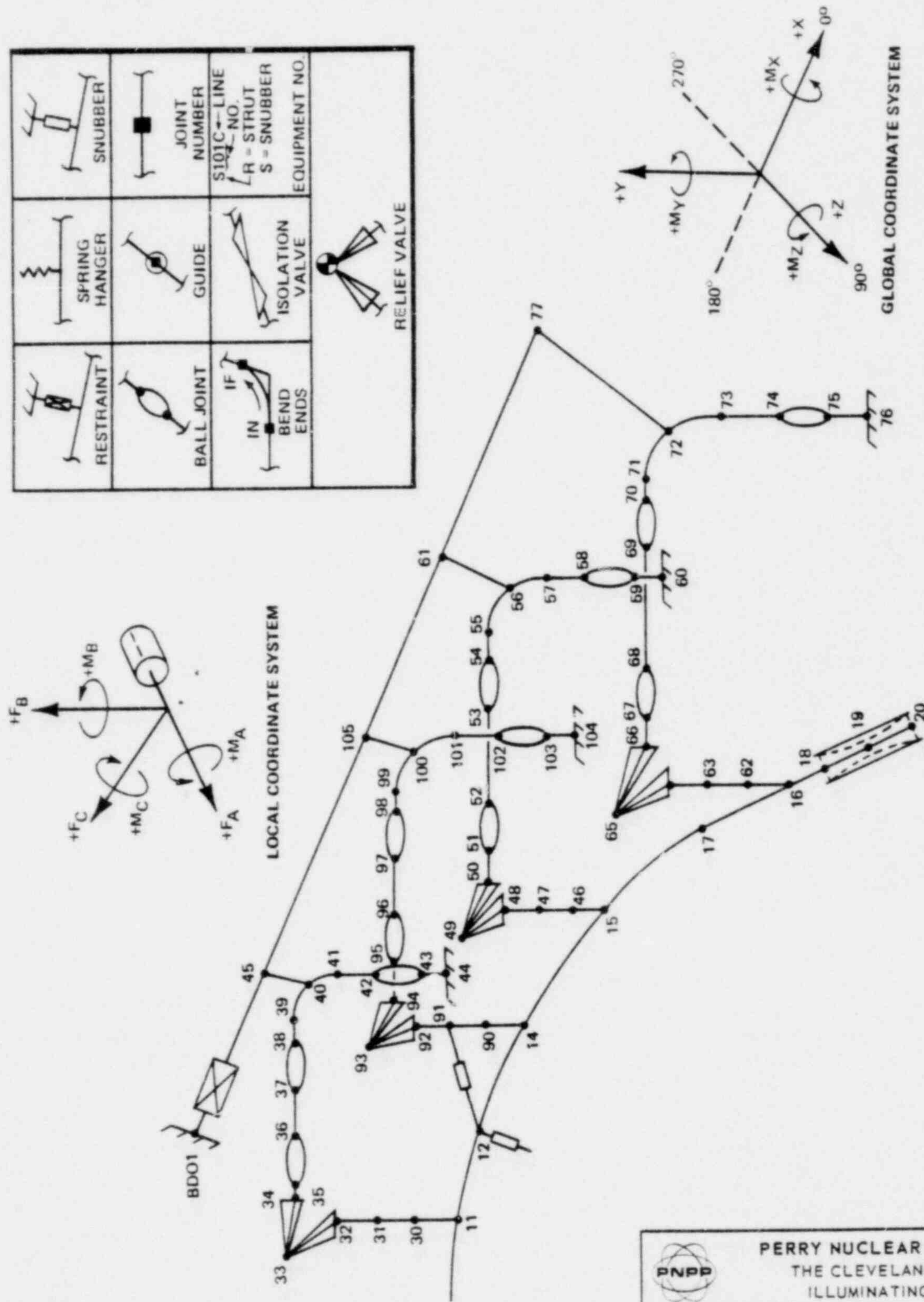
Figure 3.6-65d



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Node Point Stress Analysis
Diagram - Main Steam Line D

Figure 3.6-65e



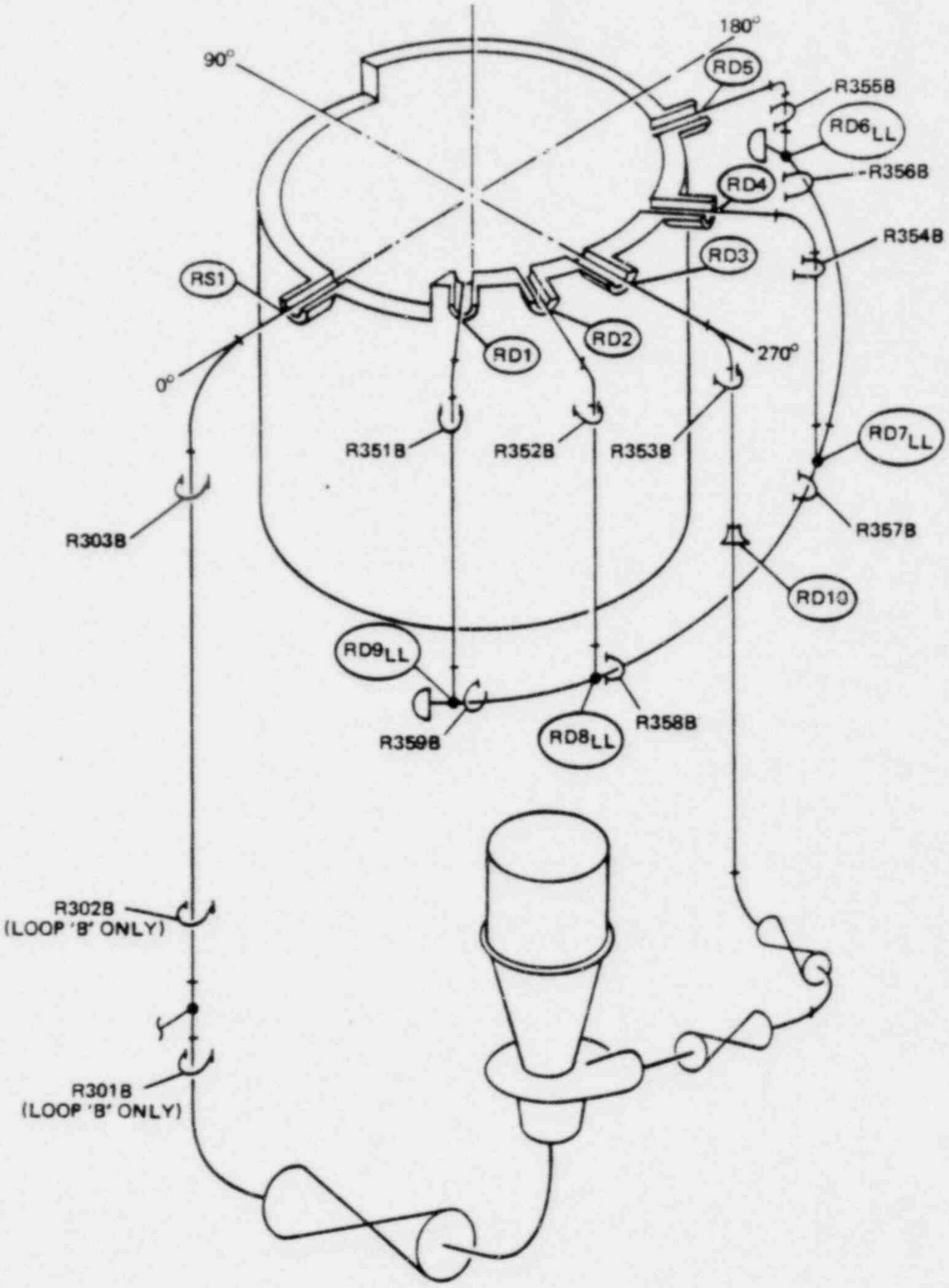
PNPP

PERRY NUCLEAR POWER ..NT

THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Node int Stress Analysis
am - Main Steam
ne D (Sweepolet)

Figure 3.6-65f

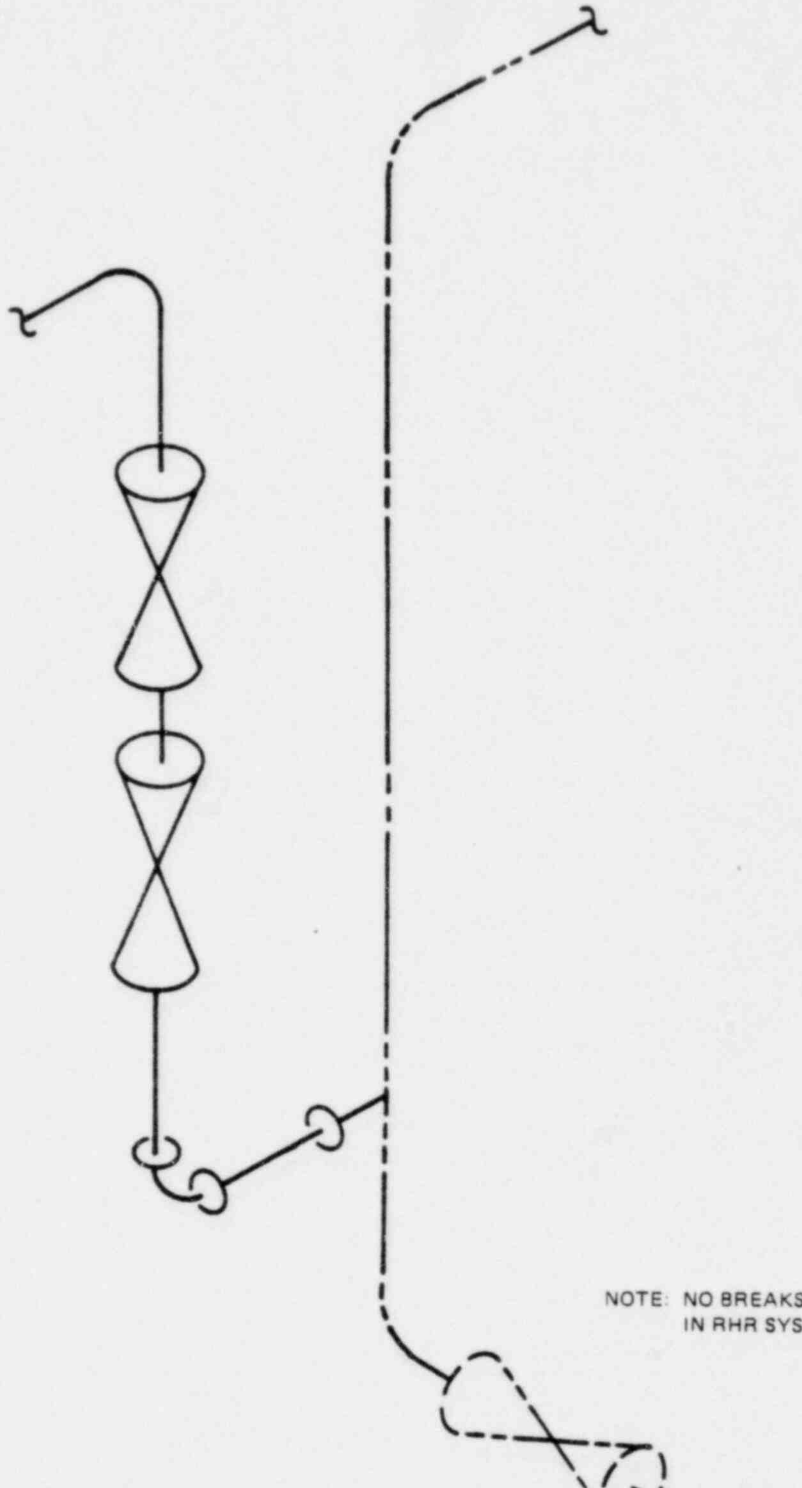


THIS IS REPRESENTATIVE OF LOOP 'B'
LOOP 'A' SAME AS LOOP 'B' (EXCEPT FOR RHR SUCTION)
BREAKS ARE POSTULATED ONLY
AT NUMBERED LOCATIONS SHOWN

PNPP
PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Recirculation System Piping
Postulated Break Locations
and Restraint Locations

Figure 3.6-66



NOTE: NO BREAKS POSTULATED
IN RHR SYSTEM.

PNPP

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

RHR System Piping Postulated
Break Locations
and Restraint Locations

Figure 3.6-66a

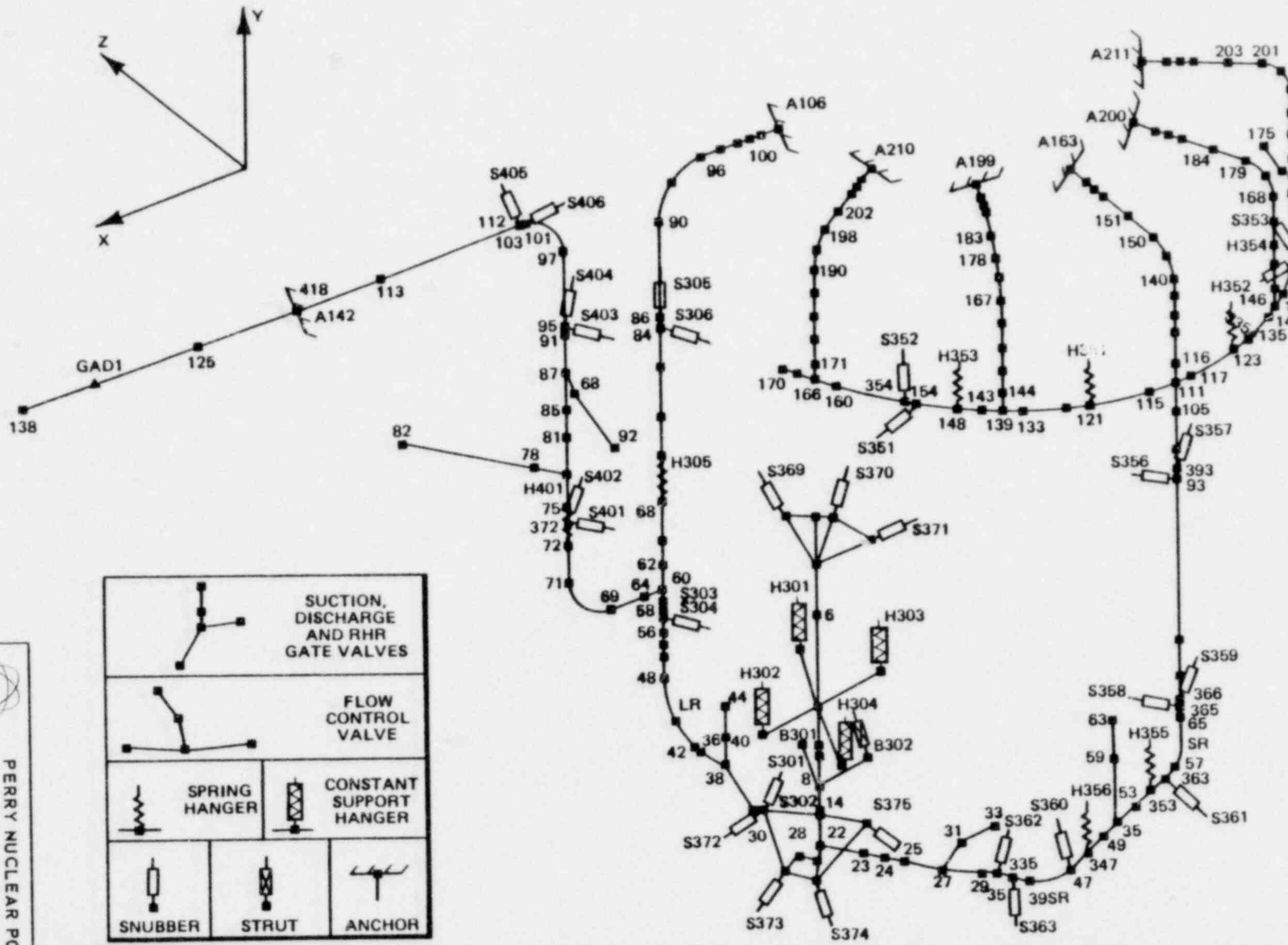
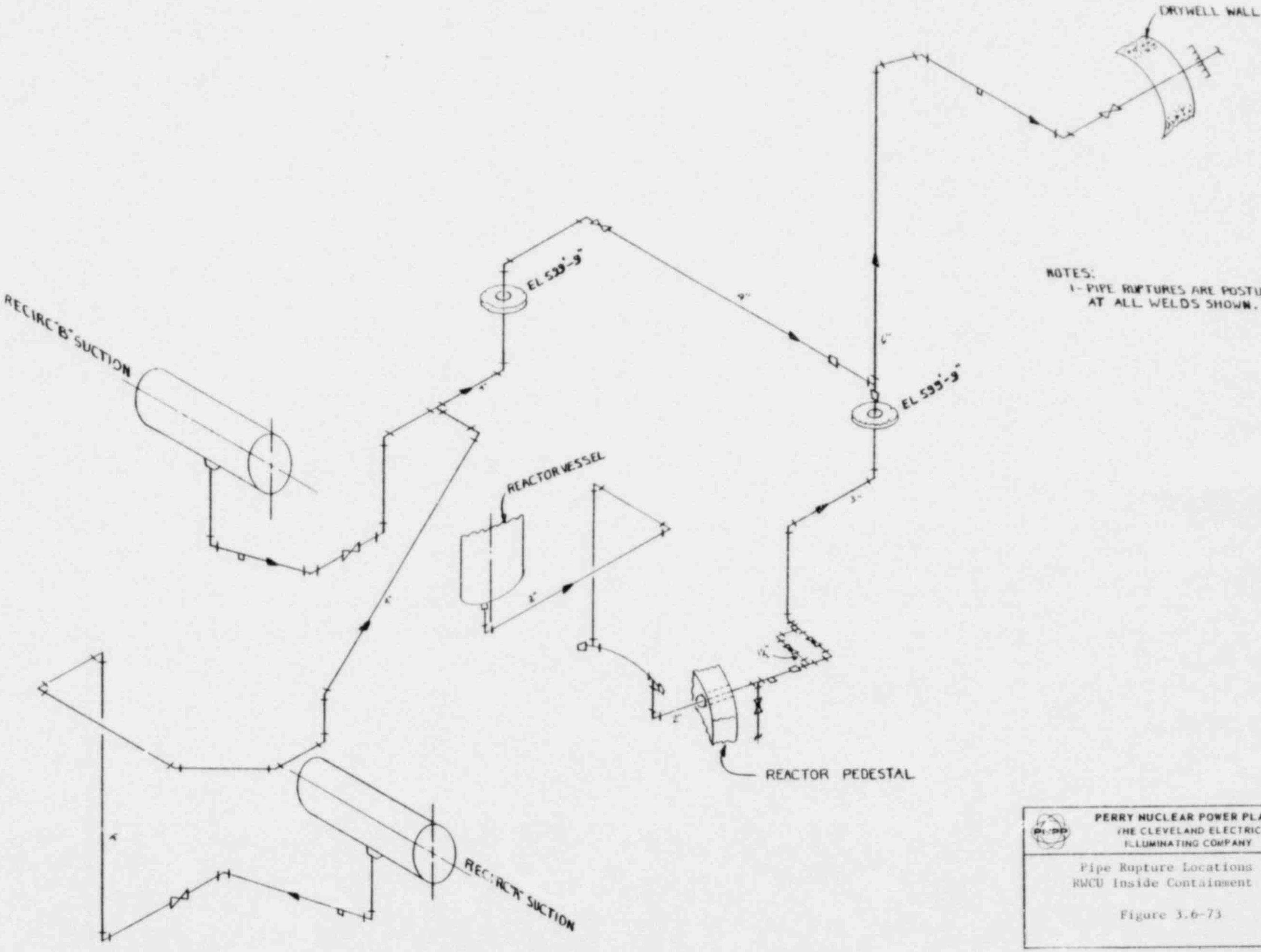


Figure 3.6-4c. RECIRCULATION SYSTEM NODE DIAGRAM (LOOP A AND LOOP B)

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Recirculation System Node Diagram (Loop A and B)

Figure 3.6-66b

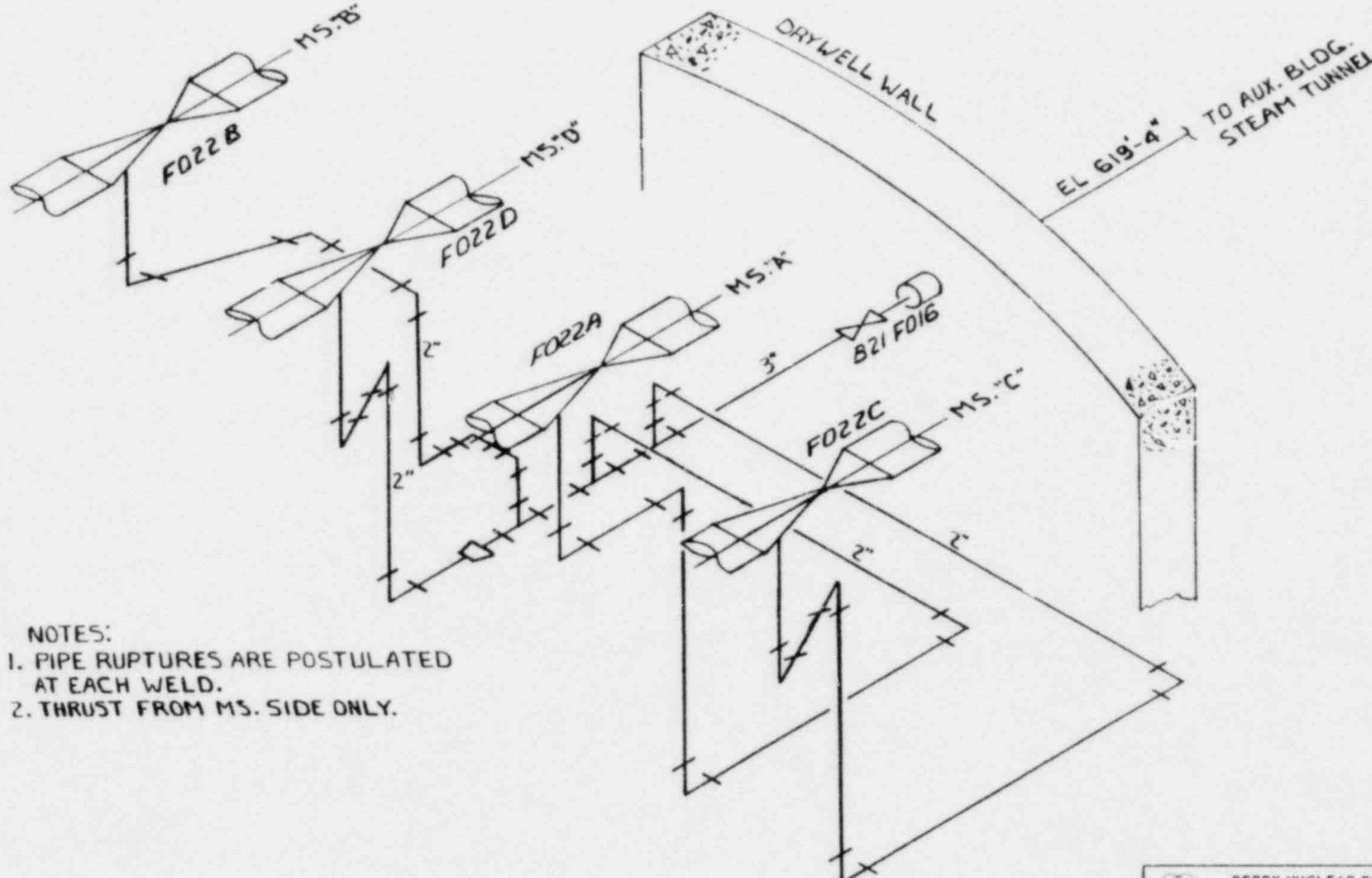


NOTES:
1- PIPE RUPTURES ARE POSTULATED AT ALL WELDS SHOWN.

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Pipe Rupture Locations
RWCU Inside Containment

Figure 3.6-73

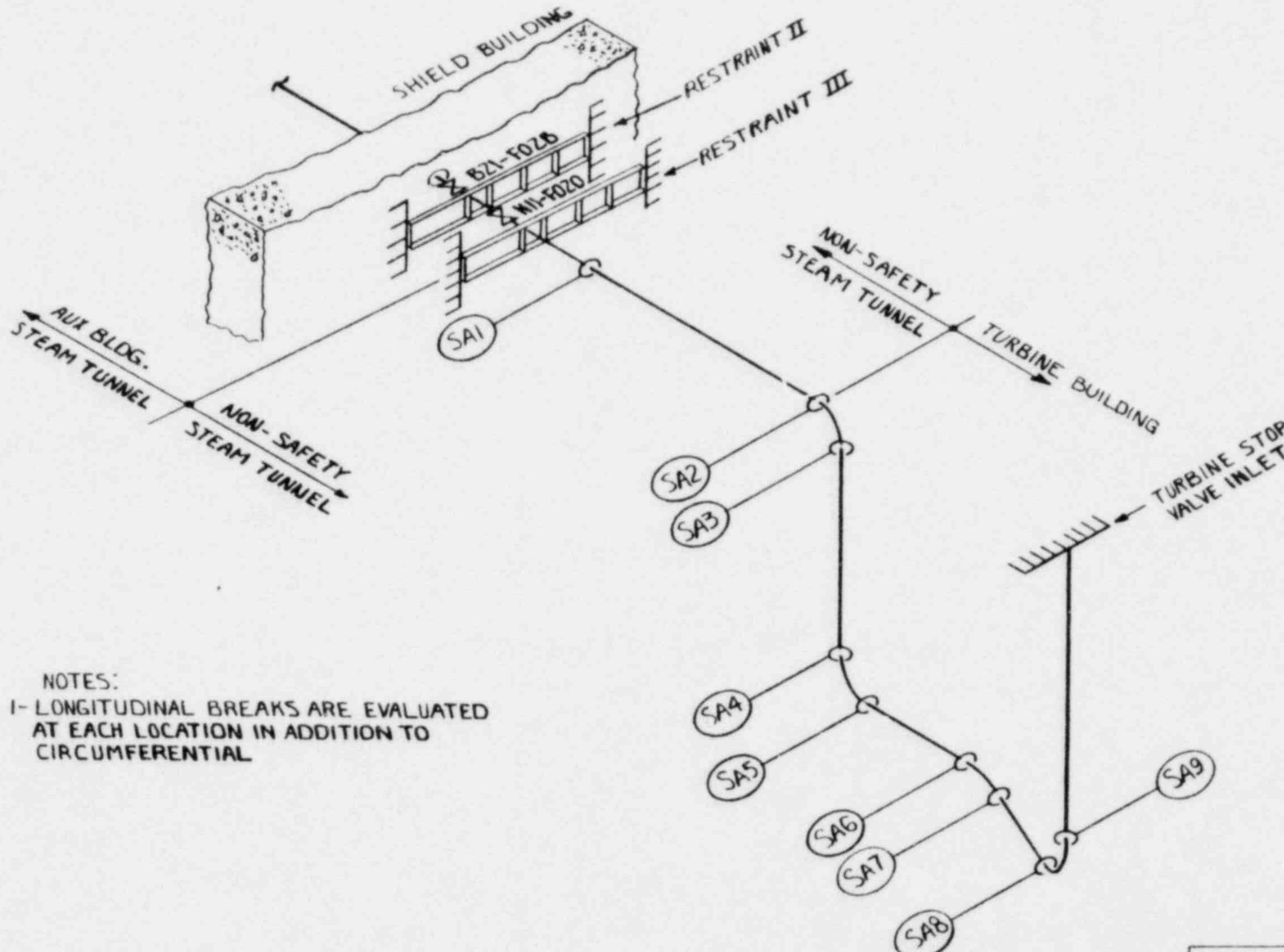


NOTES:
1. PIPE RUPTURES ARE POSTULATED AT EACH WELD.
2. THRUST FROM MS. SIDE ONLY.

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Pipe Rupture Locations
MS Drain Inside Containment

Figure 3.6-74

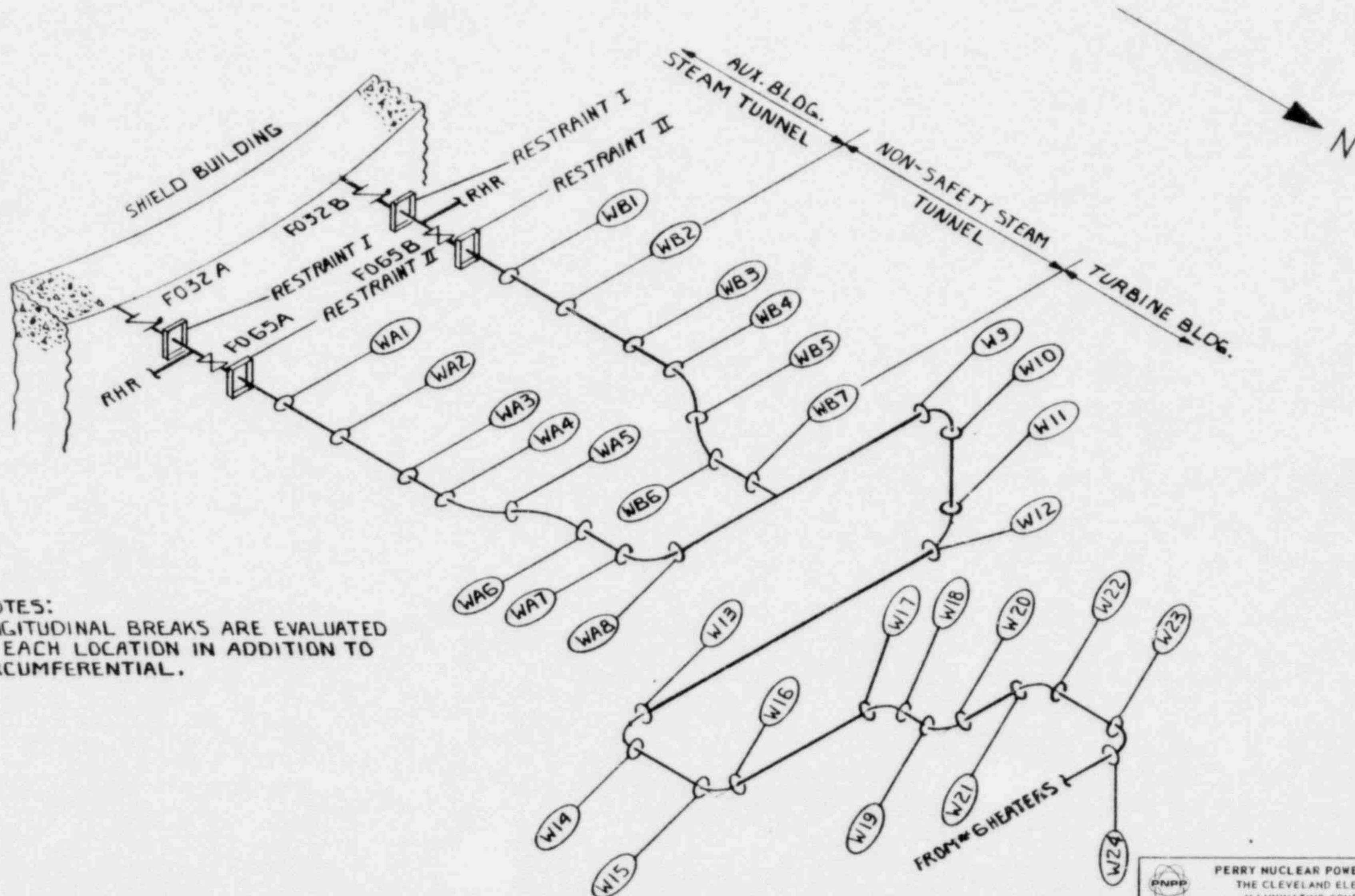


NOTES:
1- LONGITUDINAL BREAKS ARE EVALUATED AT EACH LOCATION IN ADDITION TO CIRCUMFERENTIAL

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Pipe Rupture Locations
MS Outside Containment

Figure 3.6-75

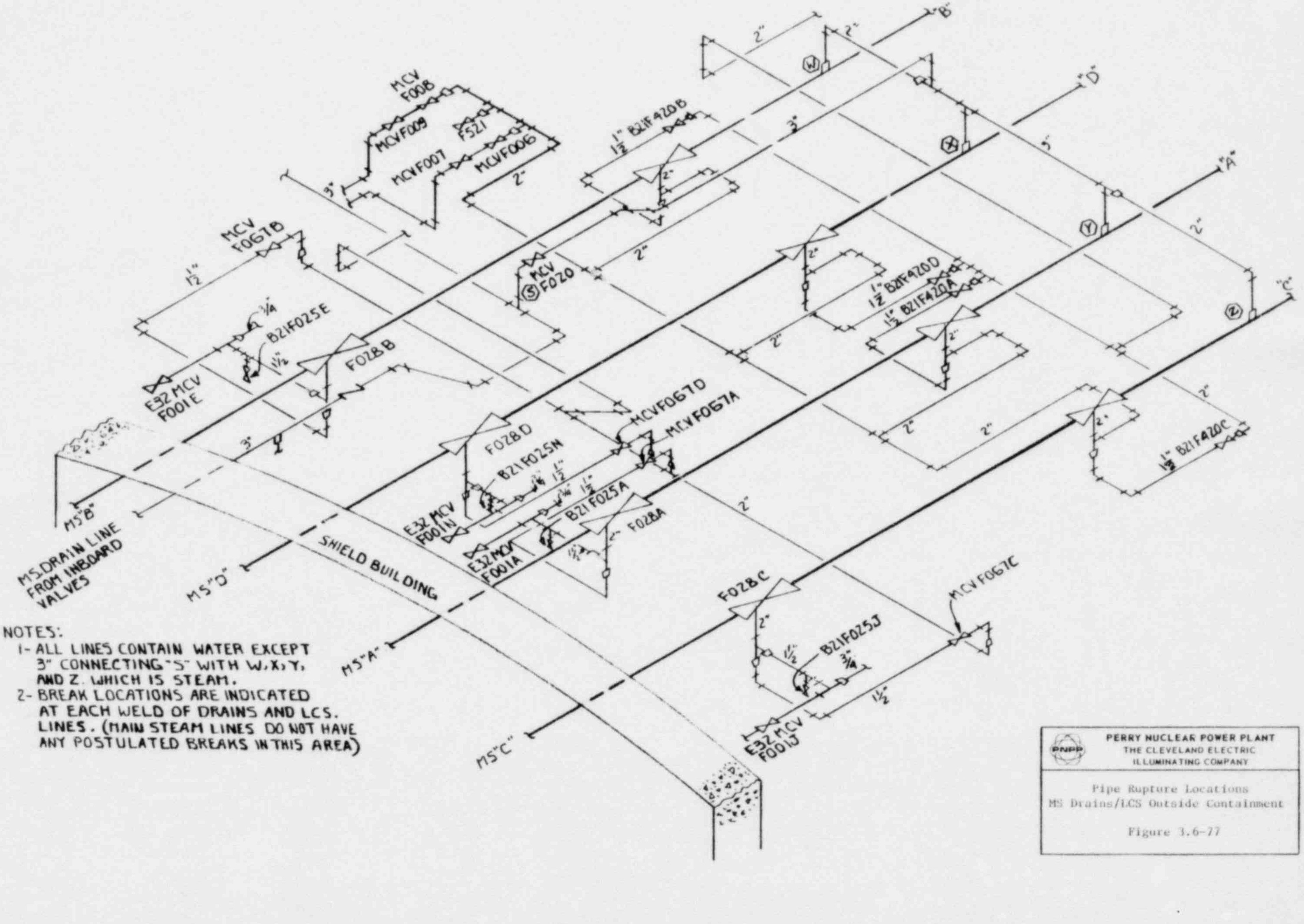


NOTES:
1-LONGITUDINAL BREAKS ARE EVALUATED AT EACH LOCATION IN ADDITION TO CIRCUMFERENTIAL.

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Pipe Rupture Locations
FW Outside Containment

Figure 3.6-76



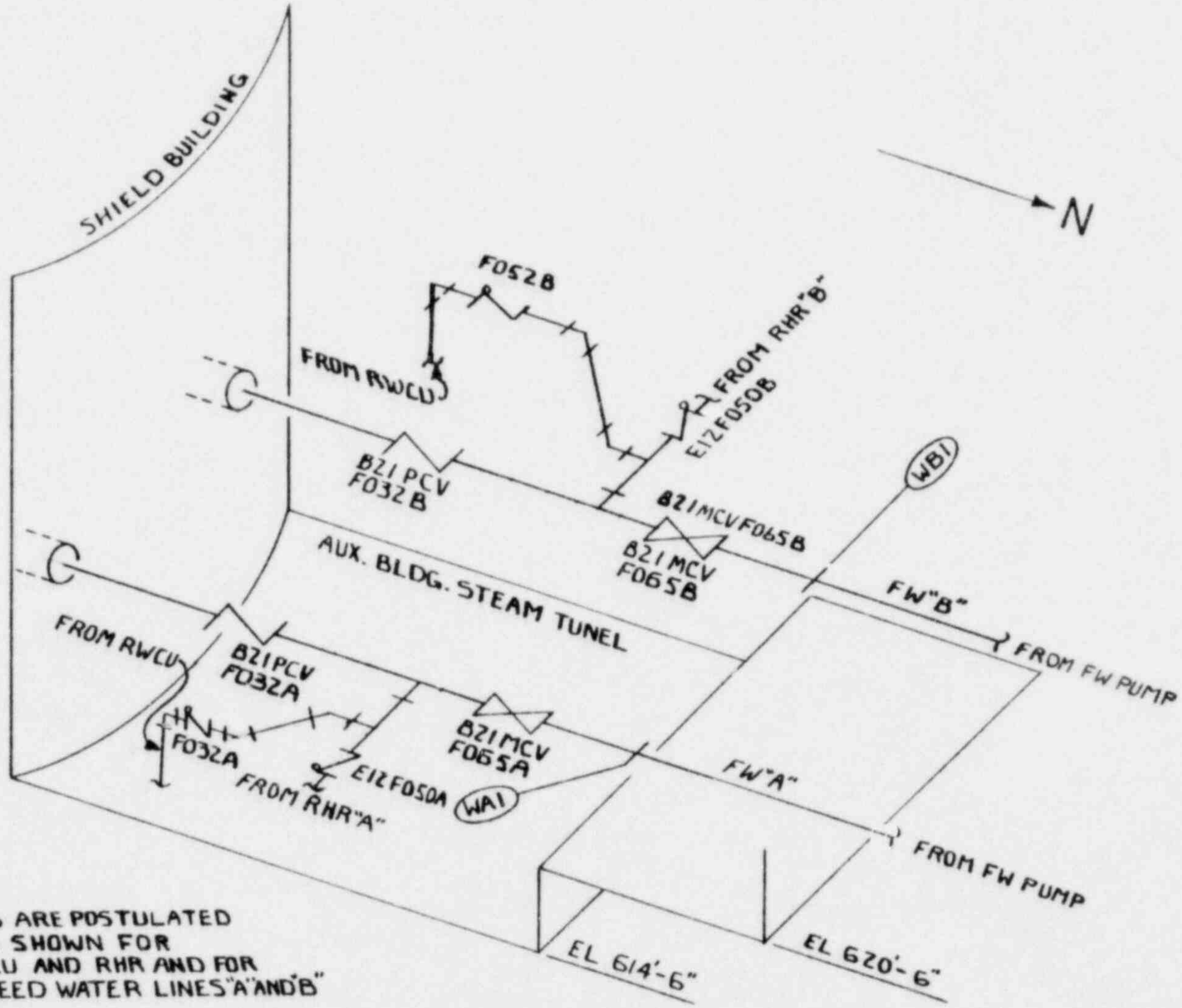
NOTES:

1- ALL LINES CONTAIN WATER EXCEPT 3" CONNECTING "S" WITH W, X, Y, AND Z. WHICH IS STEAM.
2- BREAK LOCATIONS ARE INDICATED AT EACH WELD OF DRAINS AND LCS. LINES. (MAIN STEAM LINES DO NOT HAVE ANY POSTULATED BREAKS IN THIS AREA)

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Pipe Rupture Locations
MS Drains/LCS Outside Containment

Figure 3.6-77

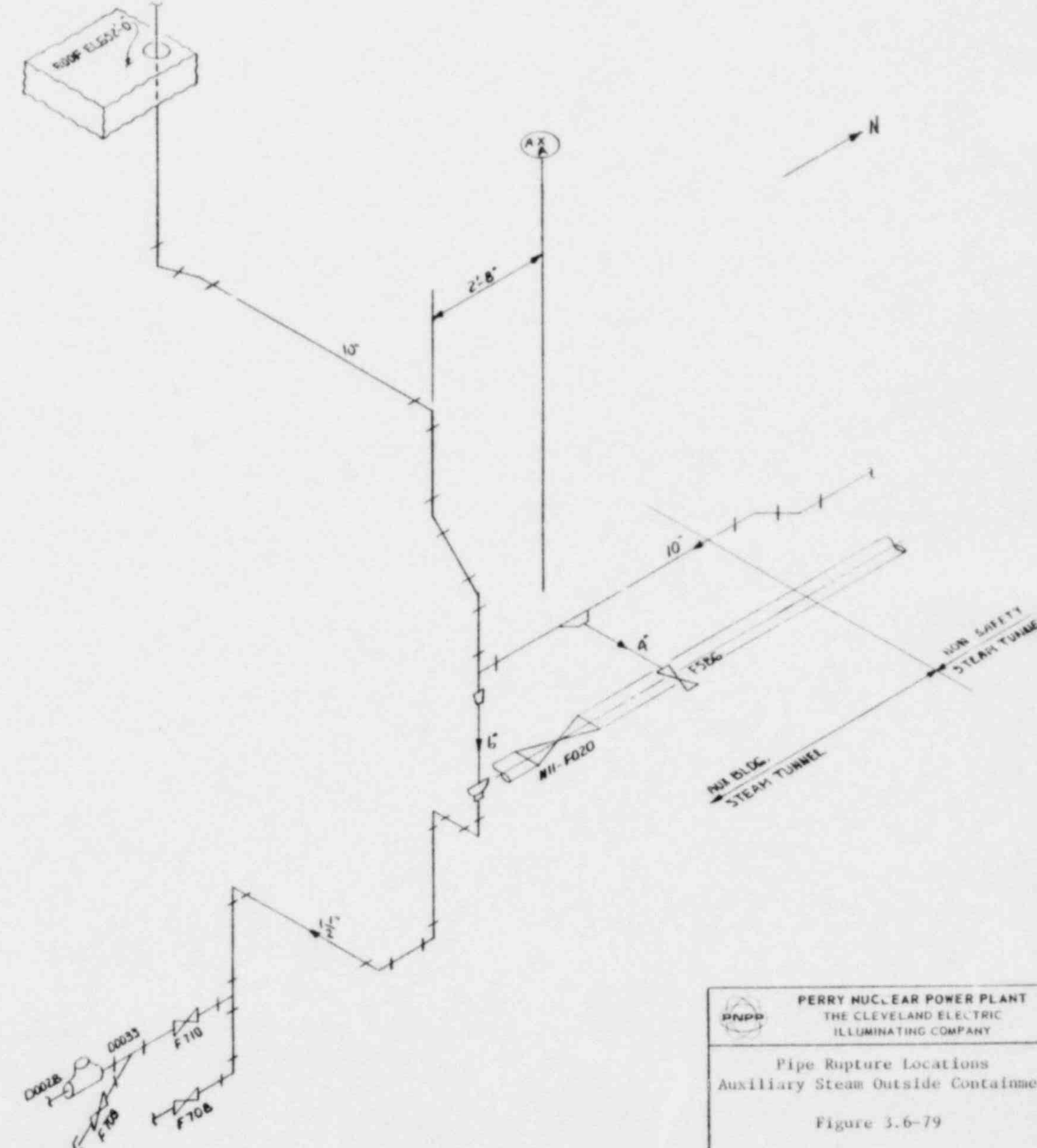


NOTES:
1- PIPE RUPTURES ARE POSTULATED AT ALL WELDS SHOWN FOR LINES FROM RWCU AND RHR AND FOR WA1 AND WB1 FEED WATER LINES "A" AND "B"

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Pipe Rupture Locations
RWCU/RHR to FW Outside Containment

Figure 3.6-78



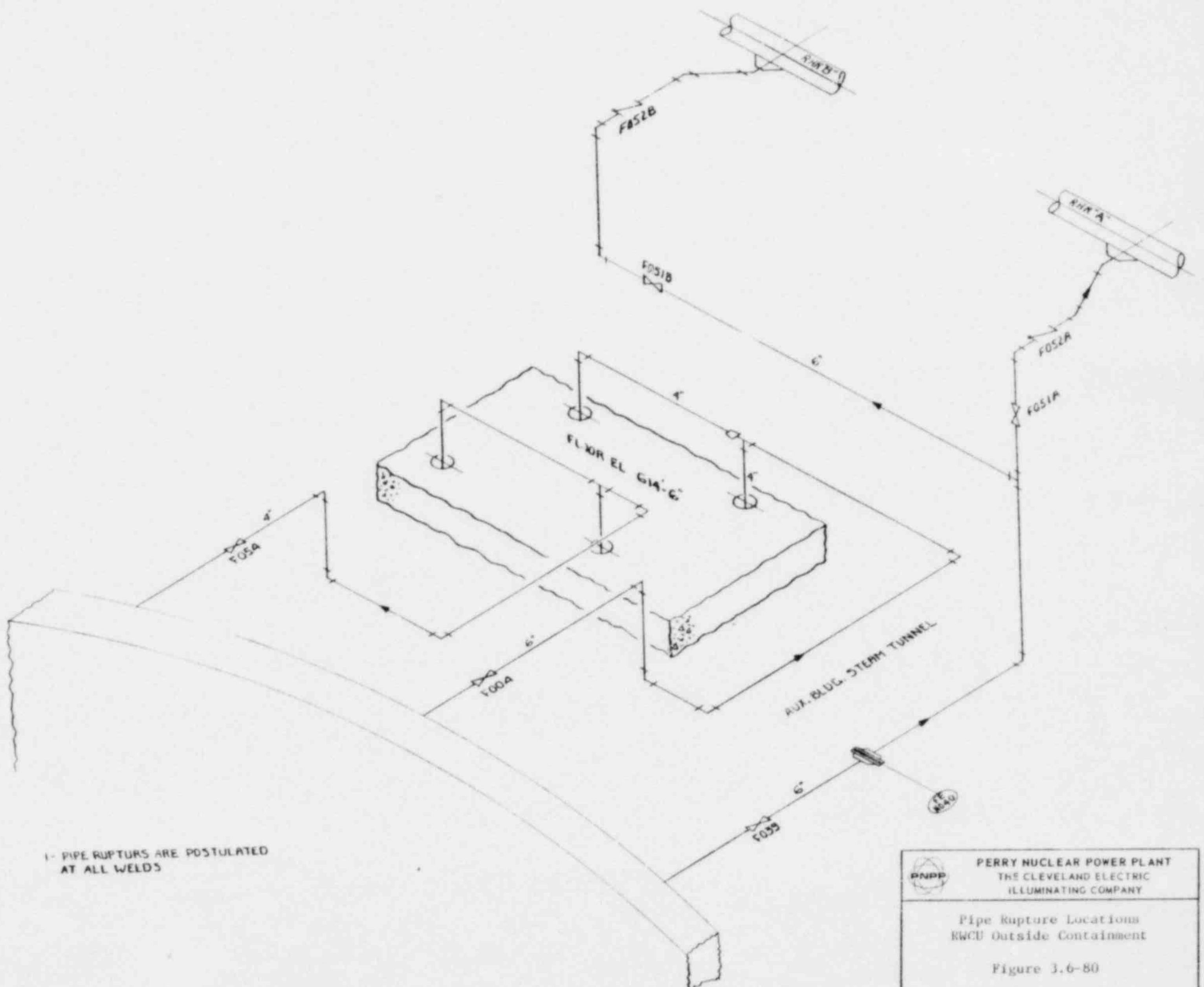
NOTES:
1. PIPE RUPTURES ARE POSTULATED AT EACH WELD.
2. THRUST FROM BOILER SIDE ONLY.

PNPP

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Pipe Rupture Locations
Auxiliary Steam Outside Containment

Figure 3.6-79



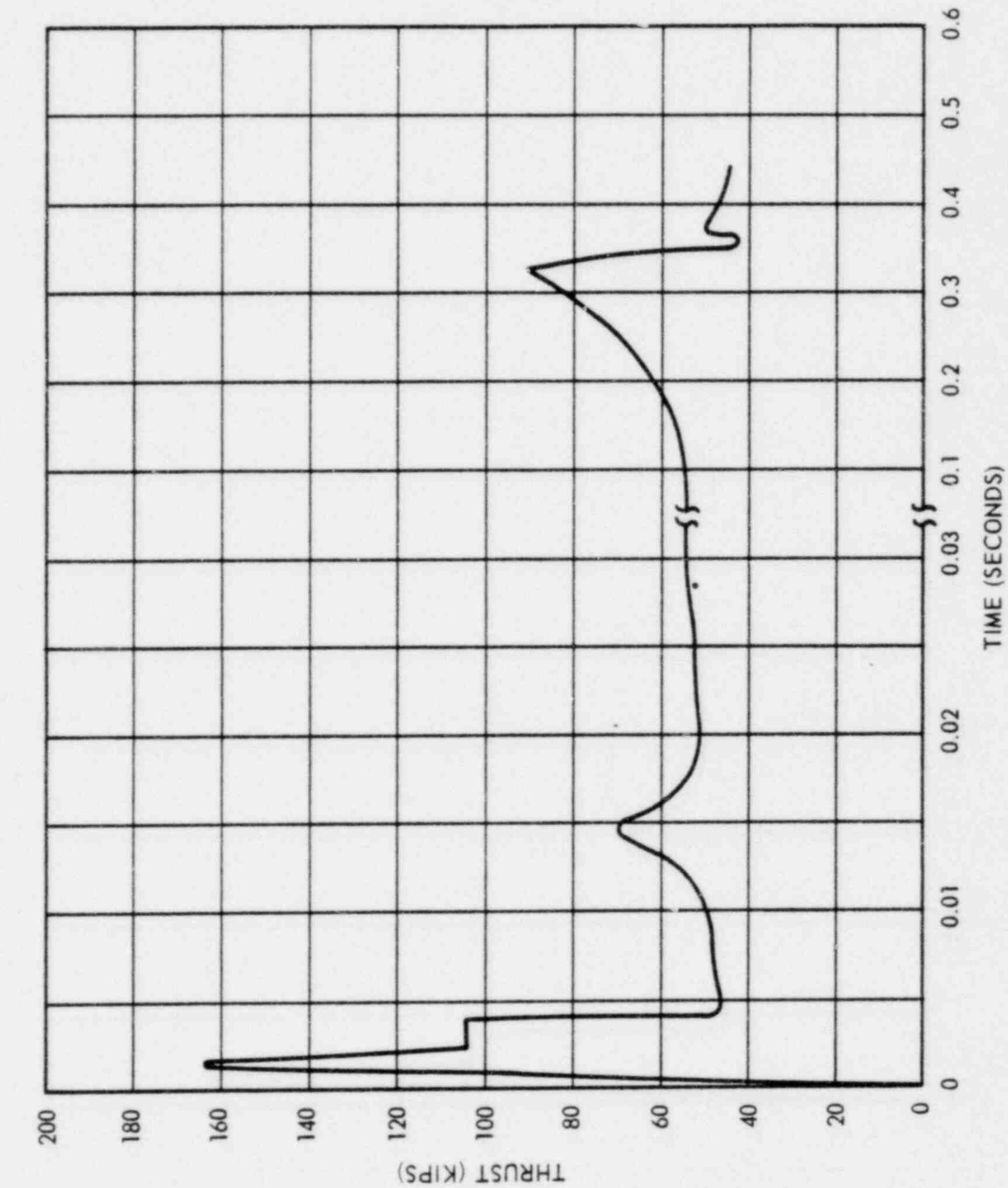
1- PIPE RUPTURS ARE POSTULATED AT ALL WELDS

PNPP

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Pipe Rupture Locations
RWCU Outside Containment

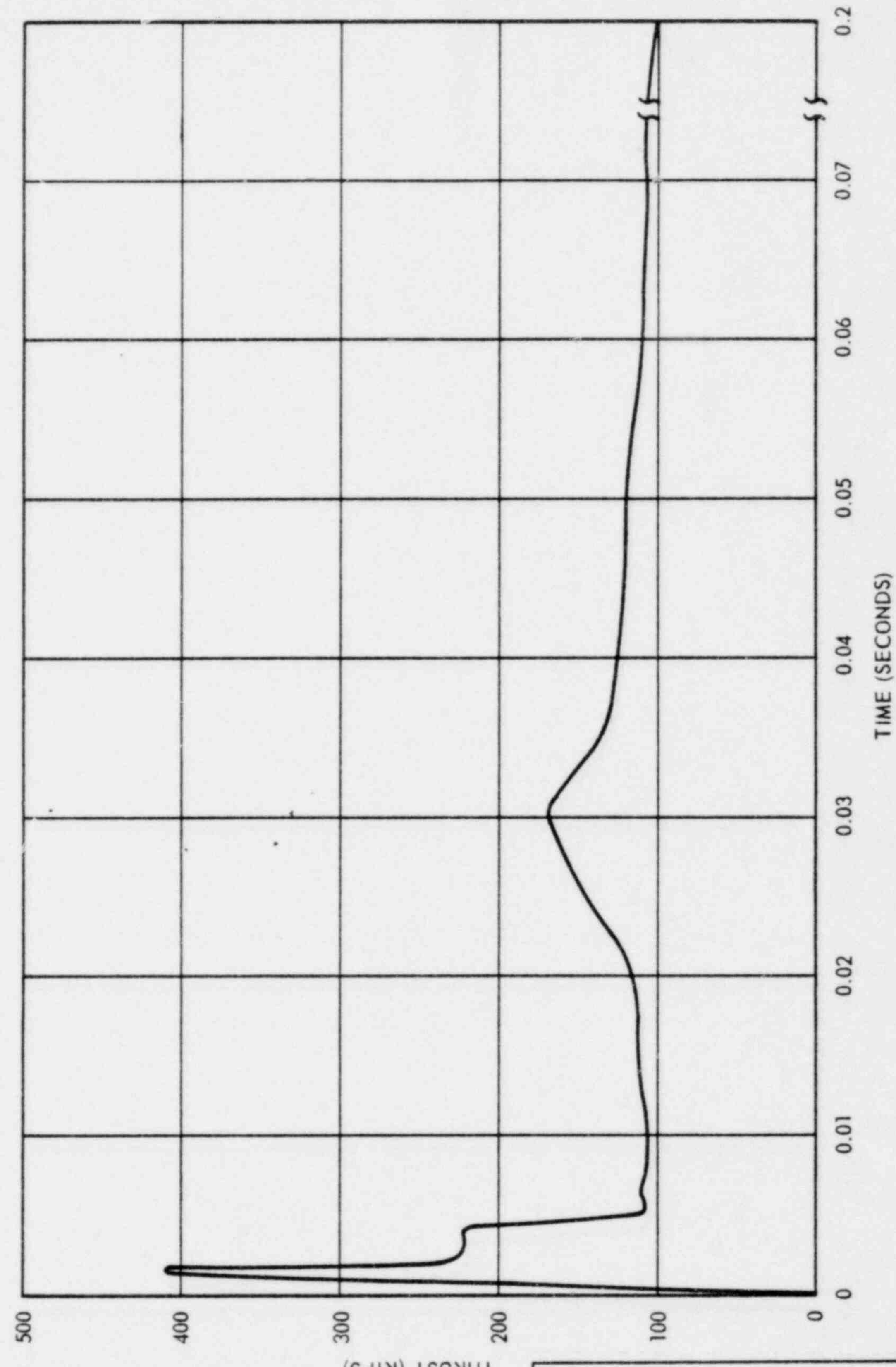
Figure 3.6-80



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Feedwater Break WB9C -
Reactor Vessel Side
(14" line)

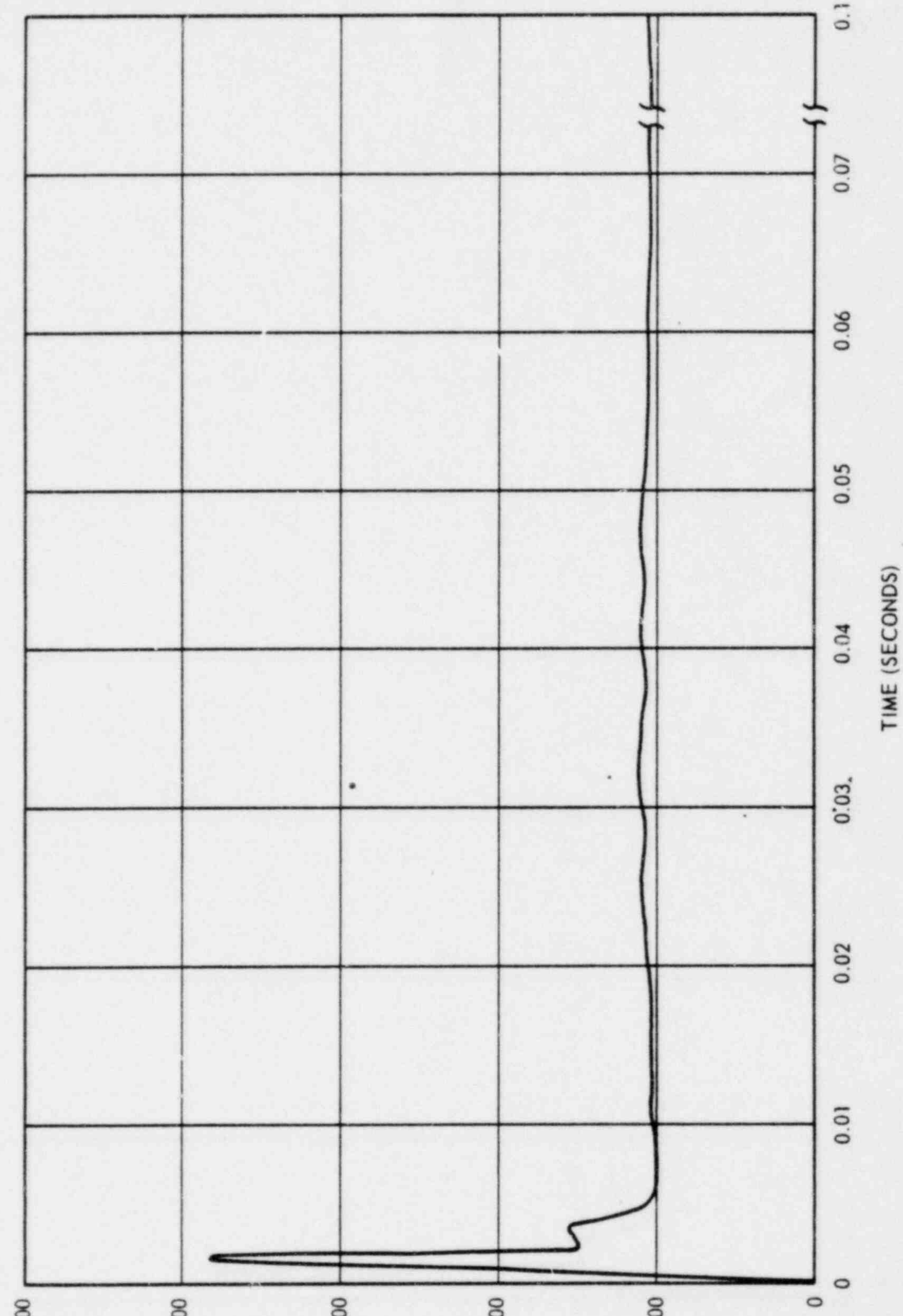
Figure 3.6-95



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Feedwater Break WB12C - Reactor Vessel Side (20" line)

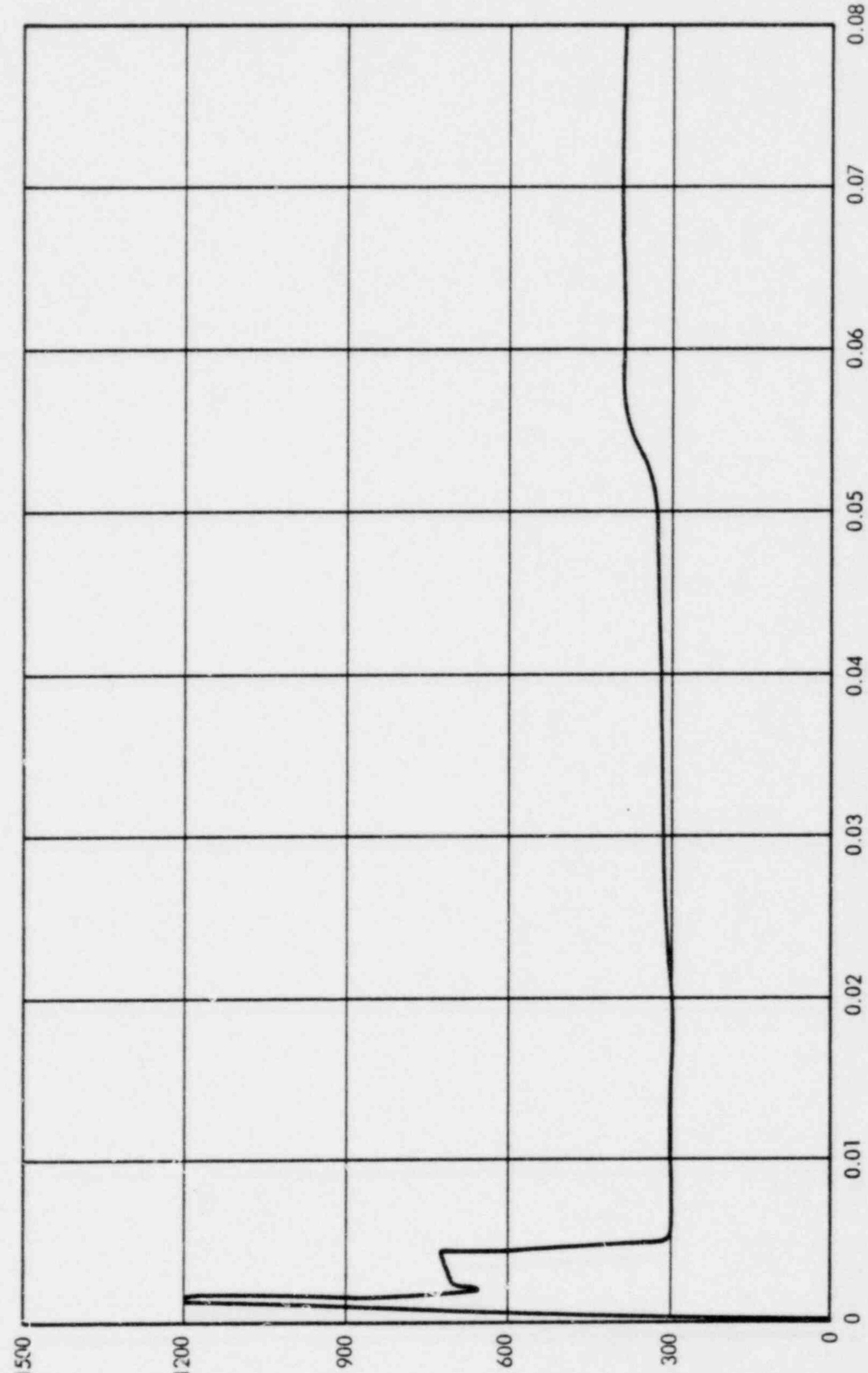
Figure 3.6-96



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Feedwater Break WB11 -
End Cap Break (20" line)

Figure 3.6-97



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Feedwater Longitudinal Break
at Node 46 In
Turbine Building

Figure 3.6-98

DSER 3.7-1 (3.7.2.1.2.5, Page 3.7-11)

The discussion on "Different Seismic Movement of Interconnected Components" requires some clarification. "The stresses thus obtained for each natural mode are then superimposed for all modal displacements of the structure by the SRSS (square root sum of the squares) method." Provide an example of what was done here.

Response

Response to this question is provided in revised Section 3.7.2.1.2.5.

DSER 3.7-2 (3.7.2.1.2.5, Page 3.7-11)   What criteria was used to determine whether or not a mode was significant?

Response

All modes within a frequency range of interest are included in the dynamic analysis. Generally, the number of modes considered for the analysis of any given system is dependent on the system characteristics and the amplitude/frequency content of the input forcing functions. The criterion is to choose the number of modes to cover the peak responses of the applicable loads to totally represent the actual system responses at the peak response frequency ranges. In accordance with SRP 3.7-2, Section II.1.a(5), the participation of all significant modes is assured when the inclusion of additional modes does not result in more than a 10 percent increase in responses.

DSER 3.7-3 (3.7.2.1.2.5, Page 3.7-11)	"When a component is covered by the ASME Boiler and Pressure Vessel Code, the stresses due to relative displacement as obtained above are treated as secondary stresses." Does this statement pertain to piping or supports?

<u>Response</u>

This statement pertains to piping only.

DSER 3.7-4
(3.7.3.1.1, Page 3.7-20)

"Seismic analyses were performed for those subsystems that could be modeled to correctly predict the seismic response." What procedure was used for the other systems? Provide an example of some of those systems.

Response

When satisfactory mathematical models of subsystems cannot be developed, seismic qualification is achieved by using conservative static criteria, such as designing to floor response spectra or by dynamic testing. As an example, a structural item which may be designed using the floor response spectra is electrical cable tray supports. Electrical cabinets and consoles, often with attached equipment, are qualified by dynamic shaking table testing.

DSER 3.7-5
(3.7.3.1.1, Page 3.7-21)

What is meant by "Closely spaced in phase modes"?

Response

Closely spaced modes are defined in accordance with Regulatory Guide 1.92.

DSER 3.7-6

(3.7.3.2.1, Page 3.7-21)

How many stress cycles are used in the BOP design?

<u>Response</u>

The response to this question is provided in revised Section 3.7.3.2.1.

DSER 3.7-7 Question deleted.

DSER 3.7-8 (3.7.3.3.2.1, Page 3.7-23)

Part (a) discussing decoupling of main steam and branch line is not a criteria.

Response

The response to this question is provided in revised Section 3.7.3.3.2.1.

DSER 3.7-9 (3.7.3.3.2.2, Page 3.7-24) Mention is made of using 33 hertz as a frequency cutoff for seismic analysis. At some point in the FSAR the applicant must address the frequencies of 50 to 60 hertz and greater than some from the suppression pool hydrodynamics.

Response

Frequencies higher than 33 Hz have been considered in the New Load analysis documented in Section 3.9.

DSER 3.7-10 (3.7.3.5, Page 3.7-25)	"For flexible equipment, the equivalent static load is taken as the product of 1.5 times the equipment mass and the peak floor response spectrum value." Regulatory Guide 1.100 allows the use of the 1.5 factor for verifying the integrity of frame type structures. For equipment having configurations other than a frame type structure, justification is required for use of the 1.5 factor.

Response

The response to this question is provided in revised Section 3.7.3.5.

DSER 3.7-11
(3.7.3.7.1, Page 3.7-26)

What procedure is used for combining closely spaced modes of systems in the BOP scope?

Response

The grouping method of Regulatory Guide 1.92 is used.

DSER 3.7-12 (3.7.3.7.2, Page 3.7-26) The referenced equation should be as follows:

$$P = \left[\sum_{K=1}^{N} \sum_{S=1}^{N} \left| R_K R_S \right| \varepsilon_{KS} \right]^{1/2}$$

Response

The response to this question is provided in revised Section 3.7.3.7.2.

DSER 3.7-13 (3.7.3.8.1, Page 3.7-28)	Justification must be provided that the modeling of valves with offset motor operators is detailed enough to provide acceleration values to be used for valve qualification.

Response

A valve with an offset operator is modelled either as a lump mass representing the valve/operator assembly or as lump masses of the valve and operator. The member between the pipe centerline and the center of gravity of the valve/operator assembly is modelled to represent the stiffness of the valve structure. The dynamic characteristics of the assembly are thus represented to account for the offset effects of the valve and operator.

DSER 3.7-14 (3.7.3.8.1, Page 3.7-28)

"In addition, the effects of the modes not included are added to the SRSS response as one term, using the acceleration at the highest frequency from the SRSS response under 33 hertz to obtain the total response." Provide an example of what was done here.

<u>Response</u>

For this example the 33 Hertz value referenced should be replaced with 62.5 Hertz since it involved higher frequency hydrodynamic caused response spectra. However, the principle applies equally as well to those systems seeing only the earthquake response spectra where the 33 Hertz is applicable.

One piping analysis has 27 modes calculated, of which the first 16 are under 62.5 Hertz. The dynamic mass associated with modes 17 to 27 was combined and assigned a spectral acceleration equal to that determined for the 16th mode. This combined equivalent mode was treated as mode number 17. The 17 modes were then combined per Regulatory Guide 1.92. This analysis was run two ways: first as described above and the second with all 27 modes considered individually. The results indicate that the two approaches provide very similar values of forces, stresses, etc. Since the first method provided the higher values, it demonstrated that the approach is conservative. Details of this problem are on file at GAI.

DSER 3.7-15 Provide a detailed explanation of the information in this table.
(Table 3.7-11,
Page 3.7-54)

Response

The response to this question is provided in revised Section 3.7.3.2.2 and Table 3.7-11.

The relative displacement between supports is determined from the dynamic analysis of the structure. The relative support point displacements are used for a static analysis to determine the additional stresses due to support displacements. Further details are given in Section 3.7.2.1.2.5.

3.7.2.1.2.5 Differential Seismic Movement of Interconnected Components

The procedure for considering differential displacements for equipment anchored and supported at points with different displacement excitation is discussed in the paragraphs that follow.

The relative displacements between the supporting points induce additional stresses in the equipment supported at these points. From the dynamic analysis of the complete structure, the time history of displacement at each supporting point is available. These displacements are used to calculate stresses.

In the static calculation of the stresses due to relative displacements in the response spectrum method, the maximum value of the modal displacement is used. Therefore, the mathematical model of the equipment is subjected to a maximum displacement at its supporting points obtained from the modal displacements. This procedure is repeated for the significant modes (modes contributing most to the total displacement response at the supporting point) of the structure. The total stresses due to relative displacement are obtained by combining the modal results using the SRSS method. Since the maximum displacement for different modes do not occur at the same time, the SRSS method is a realistic and practical method.

When a component is covered by the ASME Boiler and Pressure Vessel Code, the stresses due to relative displacement as obtained above are treated as secondary stresses.

response using the square root of the sum of the squares method. The absolute sum of the responses is considered for closely spaced, in phase modes as set forth in Section 3.7.3.7. In cases for which some dynamic degrees of freedom do not contribute to the total response, kinematic condensation was employed in the analysis.

3.7.3.1.2 Components and Equipment Provided by the NSSS Vendor

Seismic analysis methods for subsystems within General Electric scope of responsibility are given in Section 3.7.2.1.2.

3.7.3.2 Determination of Number of Earthquake Cycles

3.7.3.2.1 Balance of Plant

In accordance with Section III of the ASME Code[4], the effects of cyclic loadings are considered for Class 1 piping and Class MC components. During the plant life one safe shutdown earthquake (SSE) and five operating basis earthquakes (OBE) are considered with 10 maximum stress cycles per earthquake. Two hundred cycles were considered in the fatigue analysis of equipment as follows:

a. When the dominant frequency of the equipment fell on or within the widened peak of the floor response spectrum, the initial 100 cycles were taken at the maximum amplitude loading, and the remaining cycles were considered at one-half the maximum amplitude loading.

b. When the dominant frequency of the equipment fell outside the widened peak of the floor response spectrum, the initial 30 cycles were taken at the maximum amplitude loading, and the remaining cycles were considered at one-half the maximum amplitude loading.

These criteria provide a conservative fatigue evaluation of BOP equipment and are applicable to OBE and SSE. 6-1

3.7.3.2.2 Components and Equipment Provided by the NSSS Vendor

To evaluate the number of cycles which exist within a given earthquake, a typical boiling water reactor building-reactor dynamic model was excited by three different recorded time histories: May 18, 1940, El Centro NS component 29.4 sec; 1952, Taft N 69° W component, 30 sec; and March 1957, Golden Gate S 80° E component, 13.2 sec. The modal response was truncated such that the response of three different frequency bandwidths could be studied: 0-10, 10-20, and 20-50 Hertz. This was done to give a good approximation to the cyclic behavior expected from structures with different frequency content.

Enveloping the results from the three earthquakes and averaging the results from several different points of the dynamic model, the cyclic behavior given in Table 3.7-11 was formed.

Independent of earthquake or component frequency, 99.5 percent of the stress reversals occur below 75 percent of the maximum stress level, and 95 percent of the reversals lie below 50 percent of the maximum stress level. This relationship is graphically shown in Figure 3.7-15.

In summary, the cyclic behavior number of fatigue cycles of a component during an earthquake is found in the following manner:

a. The fundamental frequency and peak seismic loads are found by a standard seismic analysis.

b. The number of cycles which the component experiences are found from Table 3.7-11 according to the frequency range within which the fundamental frequency lies.

c. For fatigue evaluation, 1/2 percent (0.005) of these cycles are conservatively assumed to be at the peak load, 4.5 percent (0.045) at or above three-quarter peak. The remainder of the cycles will have negligible contribution to fatigue usage. -15

The safe shutdown earthquake has the highest level of response. However, the encounter probability of the SSE is so small that it is not necessary to postulate the possibility of more than one SSE during the 40-year life of a plant. Fatigue evaluation due to the SSE is not necessary since it is a faulted condition and thus not required by ASME Section III.

The OBE is an upset condition and therefore, must be included in fatigue evaluations according to ASME Section III. Investigation of seismic histories for many plants shows that during a 40-year life it is probable that five earhtquakes with intensities one-tenth of the SSE intensity, and one earthquake approximately 20 percent of the proposed SSE intensity, will occur. Therefore, the probability of even one OBE is extremely low. To cover the combined effects of these earthquakes and the cumulative effects of even lesser earthquakes, one OBE intensity earthquake with 10 peak stress cycles is postulated for fatigue evaluation.

-15

3.7.3.3 Procedure Used for Modeling

3.7.3.3.1 Balance of Plant

Equipment within the balance of plant scope is modeled as a series of discrete mass points, connected by mass free members, having sufficient mass points to ensure adequate representation of dynamic behavior. Detailed modeling of piping systems is described in Section 3.7.3.8.

3.7.3.3.2 Equipment and Components Provided by the NSSS Vendor

3.7.3.3.2.1 Modeling of Piping Systems

The continuous piping system is modeled as an assemblage of three dimensional straight or curved pipe elements. The mass of each pipe element is lumped at the nodes connected by weightless elastic members, representing the physical properties of each segment. The pipe lengths between mass points is no greater than the length which would have a natural frequency of 33 Hertz when calculated as a simply supported beam. In addition, mass points are located at all points on the piping system where concentrated weight such as valves, motors, etc. are located and also at points of significant change in the geometry of the system. All concentrated weights on the piping system such as main valves, relief valves, pumps, and motors are modeled as lumped masses. The torsional effects of the valve operators and other equipment with offset center of gravity with respect to center line of the pipe is included in the analytical model. If the torsional effect is expected to cause pipe stresses less than 500 psi, this effect may be neglected.

The criteria employed for decoupling the main steam and recirculation piping systems for establishing the analytical models to perform seismic analysis is given below:

a. The small branch lines (6-inch diameter and less) decoupled from the main steam and recirculation piping systems are analyzed separately, because the dynamic interaction is insignificant due to the disparity in the moment of inertia of the two lines.

b. The stiffness of all the anchors and their supporting steel is large enough to effectively decouple the piping on either side of the anchor for analytic and code jurisdictional boundary purposes. The RPV is very stiff compared to the piping system and thus during normal operating conditions the RPV is also assumed to act as an anchor. Penetration assemblies (head fittings) are also very stiff compared to the piping

3.7.3.4 Basis for Selection of Frequencies

3.7.3.4.1 Balance of Plant

The selection of a "rigid" frequency to preclude resonance is based on the floor response curves. This "rigid" frequency is the one beyond which no secondary peak is present and is related to the frequency value at which the ground design spectrum approaches maximum ground acceleration (33 Hertz) beyond which there is no significant structural mode. Hence the "rigid" frequency for equipment setting inside a building can be specified as 33 Hertz.

3.7.3.4.2 Equipment and Components Provided by the NSSS Vendor

All frequencies in the range of 0.25 to 33 Hertz are considered in the analysis and testing of structures, systems and components. These frequencies would cover the natural frequencies of most of the components and structures under consideration. If the fundamental frequency of a component is greater than or equal to 33 Hertz, it is treated as rigid and analyzed accordingly. Frequencies less than 0.25 Hertz are not considered as they represent very flexible structures which are not encountered in this plant.

The frequency range of 0.25 to 33 Hertz covers the range of the broad band response spectrum used in the design.

3.7.3.5 Use of Equivalent Static Load Method of Analysis

The equivalent static loads, for flexible equipment having simple frame type structural configurations, are taken as 1.5 times the product of the equipment masses and the peak spectral accelerations of the applicable floor response spectra.

When static coefficient analyses are performed for equipment with other more complex configurations, justification based on the equipment eigenvalue and frequency/amplification content of the applied loads will be provided for the value of the static coefficient used.

3.7.3.6 Three Components of Earthquake Motion

Responses to the two horizontal and the vertical component seismic inputs are calculated separately for the entire subsystem. The maximum value of a particular response due to simultaneous action of three components of earthquake were obtained by taking the square root of the sum of the squares of corresponding maximum response values to each of the three components calculated separately. This procedure is in conformance with the guidance of Regulatory Guide 1.92.

3.7.3.7 Combination of Modal Responses

3.7.3.7.1 Balance of Plant

Modal responses and spatial components of earthquake in seismic response analysis were combined in accordance with the guidance of Regulatory Guide 1.92.

3.7.3.7.2 NSSS

In a response spectrum modal dynamic analysis, if the modes are not closely spaced (i.e., if the frequencies differ from each other by more than 10 percent of the lower frequency), the modal responses are combined by the square root of the sum of the squares (SRSS) method as described in Section 3.7.3.7.2.1. If some or all of the modes are closely spaced, a double sum method, as described in Section 3.7.3.7.2.2, is used to evaluate the combined response. In a time-history method of dynamic analysis, a vector sum at every step is used to calculate the combined response. The use of the time-history analysis method precludes the need to consider closely spaced modes.

3.7.3.7.2.1 Square Root of the Sum of the Squares Method

Mathematically, this SRSS method is expressed as:

$$R = \left(\sum_{i=1}^{n} (R_i)^2 \right)^{1/2}$$

where:

R = Combined Response,

R_i = Response in the i^{th} mode, and

n = Number of modes considered in the analysis -12

3.7.3.7.2.2 Double Sum Method

This method is defined mathematically as:

$$R = \left[\sum_{k=1}^{N} \sum_{s=1}^{N} \left| R_k R_s \right| \varepsilon_{ks} \right]^{1/2}$$

TABLE 3.7-11

NUMBER OF DYNAMIC RESPONSE CYCLES EXPECTED DURING A SEISMIC EVENT

	Frequency Bandwidth (Hertz)		
	0 - 10	10 - 20	20 - 50
Total Number of Seismic Cycles	168	359	643
No. of Seismic Cycles (0.5% of Total) between 75% and 100% of Peak Loads	0.8	1.8	3.2
No. of Seismic Cycles (4.5% of Total) between 50% and 75% of Peak Loads	7.5	16.2	28.9

DSER 3.9-1 (3.9, Page 3.9-1) Any ref.rences to the ASME Boiler and Pressure Vessel Code should indicate what part is being referenced.

Response

T'e response to this question is provided in revised Section 3.9.1.1.

DSER 3.9-2 (3.9.1.2, Page 3.9-1)	Methods of verification are required for all NSSS computer codes used in the analysis.

Response

Response to this question is presently being prepared.

DSER 3.9-3 (3.9.1.2.6, Page 3.9-16)	All computer programs used in the design and analysis of systems and components within the BOP scope must be listed. Methods of verification are required for all BOP programs.

Response

A list of computer programs used in the design and analysis of BOP systems and components is being prepared and will be forthcoming.

DSER 3.9-4 (3.9.1.4.12, Page 3.9-26)	It is stated that elastic-plastic methods of analysis may be used for some components. We would like to review the analysis procedures that would be used if an elastic-plastic analysis was done.

Response

The procedures used for elastic-plastic analysis are described in Section 3.6.2.2.2.

DSER 3.9-5 (3.9.2, Page 3.9-27)	More detail is needed for the NSSS and BOP preoperational vibration testing program. What locations will be monitored? What types of instrumentation will be used? What are the actual values that will be used for deflection and stress limits?

The staff's position is that acceptance limits for vibration should be based on half of the endurance limit as defined by the ASME Code at 10^6 cycles. We will require a copy of your results from your preoperational vibration testing program.

Response

As explained in Sections 3.9.2.1.1.2 through 3.9.2.1.1.4, the main steam and recirculation piping are instrumented with transducers to measure temperature, thermal movement and vibration deflections. As explained in Section 3.9.2.1.1.1, during preoperational vibration testing of recirculation piping, visual observation and manual measurements by hand-held vibrograph are made to supplement the remote measurements.

The piping systems are tested for both steady-state vibration and vibration due to operating transients. A different acceptance criterion is established for each type of vibration.

For steady-state vibration the piping peak stress (zero to peak) due to vibration only (neglecting pressure) will not exceed 10,000 psi for Level 1 criteria and 5,000 psi for Level 2 criteria. These limits are below the piping material fatigue endurance limits as defined in Design Fatigue Curves in Appendix I of ASME Code for 10^6 cycles.

For operating transient vibration the piping bending stress (zero to peak) due to operating transient only will not exceed 1.2 Sm or pipe support loads will not exceed the Service Level D ratings for Level 1 criteria. The $1.2S_m$ limit insures that the total primary stress including pressure and dead weight will not exceed $1.8S_m$, the new Code Service Level B limit. Level 2 criteria are based on pipe stresses and support loads not to exceed design basis predictions. Design basis criteria require that operating transient stresses and loads not to exceed any of the Service Level B limits including primary stress limits, fatigue usage factor limits, and allowable loads on snubbers.

If all Level 2 criteria are satisfied for both steady-state vibration and operating transient vibration there will be no fatigue damage to the piping system due to steady-state vibration and all operating transient vibrations are less than the calculated values in the stress report. If any Level 2 limits aren't satisfied, detailed engineering evaluation is needed to develop corrective action or to show that the measured results are acceptable. Any resolution must be properly documented and approved as described in Section 3.9.2.1.4.2.

The response to this question is provided in revised Sections 3.9.2.1.4.1 and 3.9.2.1.4.2.

DSER 3.9-6 (3.9.2.1.2, Page 3.9-29)	"The piping system does 'shakedown' after a few thermal expansion cycles." Provide an explanation of this statement.

<u>Response</u>

All components, after a few thermal cycles, will settle down to their stablized positions.

This above quoted sentence is deleted, since the implication is already in Section 3.9.2.1.2a.

DSER 3.9-7 (3.9.2, Page 3.9-65)	"In addition to the above components, vibration measurements of the core spray sparger will be measured during preoperational testing of that system at the designated prototype 251 BWR/6 plant (Grand Gulf)." Show how this is applicable to Perry.

Response

Even though Grand Gulf and Perry are of different sizes, the flow condition and the core spray sparger design are similar. The pre-op testing of this system in Grand Gulf will provide additional data to verify the predicted performance of a similar system under a similar flow condition.

DSER 3.9-8 (3.9.2.4.1, Page 3.9-66)	Provide a commitment that Perry will be in compliance with Regulatory Guide 1.20 for prototype reactors.

Response

The response to this question is provided in revised Section 3.9.2.4.1.

DSER 3.9-9 (3.9.2.5, Page 3.9-67) "These periods will be determined from a comprehensive dynamic model of the RPV and internals with 12 degrees of freedom." It is not clear what is actually done here. How can a model be comprehensive and have only 12 degrees of freedom?

Response

The response to this question is provided in revised Section 3.9.2.5.

DSER 3.9-10 (3.9.2.6, Page 3.9-68)	It appears that some results from Grand Gulf will be used in the evaluation and qualification of the reactor internals at Perry. Show that the similarity between the two sets of internals is sufficient to allow direct comparisons.

Response

The response to this question is provided in revised Section 3.9.2.6.

DSER 3.9-11 (3.9.3, Page 3.9-68)	Several references are made throughout this section to allowable stresses for bolting. Specifically, what allowable stress limits are used for bolting for (a) equipment anchorage, (b) component supports, and (c) flanged connections? Where are these limits defined?

<u>Response</u>

Response to this question is presently being prepared.

DSER 3.9-12 (3.9.3.1.2, Page 3.9-78)

Are there any Class 1 systems in the BOP scope of responsibility?

Response

Yes. See revised Section 3.9.3.1.2.

DSER 3.9-13 (3.9.3.4.1, Page 3.9-107)	"For the NSSS scope of supply, all valve operators which are mounted on Class 1 piping will not be used as attachment points for component supports." What about Class 2 and 3 piping? This question also applies to the BOP scope of responsibility.

Response

Valve operators are not used as attachment points for component supports. See revised Section 3.9.3.4.1.

DSER 3.9-14
(3.9.3.4.1,
Page 3.9-109)

Provide more detail on the testing done on snubbers.

<u>Response</u>

Two snubbers of each size and each model were tested under upset and faulted loads in the manner described below:

a. Snubbers were tested dynamically to ensure that they could perform as required under upset loading conditions in the following manner:

   1. The snubbers were subjected to a force that varied approximately as a sine wave.

   2. The frequency (Hz) of the input force was in increments of 5 Hz within the range of 3 to 33 Hz.

   3. The test was conducted with the snubber at room temperature and at 200° F.

   4. The peak load in both tension and compression was equal to or higher than the rated load of the snubbers.

   5. The duration of the test at each frequency was 10 seconds or more.

b. Snubbers were tested dynamically to ensure that they could perform as required under emergency and faulted loading conditions in the following manner:

   1. The snubbers were subject to forces that varied approximately as a sine wave.

   2. The test was conducted with the snubbers at room temperature.

Question 3.9-14 (Cont'd)

3. The peak load in both tension and compression was equal to 1.5 times the rated load of the snubbers.

4. The duration of the test was 10 seconds.

Snubbers are qualified for service by General Electric by testing for bleed rate, lockup rate, drag or friction force and for response to dynamic loading. The dynamic loading test is accomplished by subjecting the snubber to a sinusoidal force that is equal to the rated load of the snubber. The force is applied at frequencies that are at 5 Hz increments within the range of 3 Hz to 33 Hz. The dynamic load tests are conducted with the snubber at both room temperature and at 200° F.

The snubbers are modeled as linear elastic springs in the dynamic analysis of the piping system. The vast majority of all dynamic loadings occur with frequencies ranging from 3 Hz to 33 Hz. By using the results of the dynamic testing, spring constants are calculated. These constants increase with higher frequencies. The average spring constant, including all lost motions (dead band, etc.) of the snubber, is then used by General Electric in the analytical model of the snubber.

In addition to the testing of the snubbers by themselves, General Electric has subjected our safety relief valve piping to safety relief valve discharge while monitoring the piping system for stresses. The safety relief valve discharge creates acoustic waves that propagate through the safety relief valve piping and impose momentary forces on the pipe at each change in direction. The results of this testing of the piping system show a satisfactory correlction between actual stresses and predicted stresses in the pipe. Since the analytical model of the piping system uses the spring constants obtained from the aforementioned snubber test, this correlation serves as a calibration of the snubber spring constant.

Although the frequencies induced on the snubbers have not bee measured directly, the vibration frequencies of the piping have been obtained for the main steam line during the Monticello and Match 2 tests and the frequencies of the main steam and recirculation piping have been measured during the Caorso test. The frequencies induced on the snubbers may be assumed to be closely related to the piping vibrating frequencies.

Although frequencies were measured from 5 Hz to 50 Hz, the dominant frequencies varied from 12 Hz to 36 Hz.

The tests at Monticello and Hatch 2 measured responses of the piping system to the acoustic wave in the SRV discharge piping following SRV opening. These loads are a function of the configuration of the discharge piping and are not related to the configuration of the containment. The Caorso tests measured the responses of the piping system to the acoustic wave in the discharge piping and also measured the responses of the piping to the hydrodynamic load associated with the SRV discharge. Although the Mk II and Mk III containments are different, the Caorso tests show that the structural responses (accelerations) to the hydrodynamic load in the suppression pool was much smaller than the calculated values at design conditions. It is expected that the load for Mk III containment due to SRV will also be much smaller than the calculated values. The frequencies at which the piping responds to the hydrodynamic loads are not expected to differ significantly in the Mk III containment.

The stresses in the main steam branch pipe of a BWR due to safety/relief valve blowdown were measured from an insitu piping system test (Hatch). The test results were compared with analytical results. The calculated stresses for SRV discharge piping response loads were found to be conservative when compared to measured stress values. The ratios of measured to calculated stress value are as follows:

Question 3.9-14 (Cont'd)

Strain Gage Data Set	Stress Ratio (Measured/Calculated)
1	0.53
2	0.55
3	0.73
4	0.48
5	0.59
6	0.55
7	0.57
8	0.70

Therefore, assuming linearity with piping stresses, it is expected that:

$$\frac{\text{Predicted snubber load}}{\text{Calculated snubber load}} = 0.67$$

Since the calculaed snubber load is less than the snubber capacity, it can be expected that the predicted snubber load is less than 67% of the snubber capacity.

DSER 3.9-15	What elastic-plastic analysis has been done on supports?
(3.9.3.4.4,	Provide an example of this analysis.
Page 3.9-112)	

Response

The response to this question is provided in revised Section 3.6.2.3.

DSER 3.9-16 (3.9.4.3, Page 3.9-114)	Reference is made to allowable deformation in the title of this section but there is no discussion of allowable deformations in the text.

Response

The acceptance of the drive is based on the stress analysis reported in Table 3.9-3. The deformation is not a controlling factor. Accordingly, Section 3.9.4.3 has been revised.

DSER 3.9-17 (3.9.5.1.1.8, Page 3.9-120)	Recently, cracking has been observed in BWR jet pump holddown beams. The resolution of this problem may affect the design or testing of the Perry jet pumps (see I&E Bulletin 80-07).

<u>Response</u>

CEI is currently evaluating the two alternates whether to de-tension the existing beams or to replace the existing beams by new beams with the updated design.

DSER 3.9-18 (3.9.5.1.1.10, Page 3.9-121) What feedwater sparger design and Control Rod Drive Return Line Modification is used at Perry? Provide a commitment to NUREG-0619.

Response

Perry has committed to install the non-cladded feedwater nozzles and triple-sleeve with dual piston ring type feedwater spargers. Perry's current design already has RWCU flow routing through the feedwater lines. The CRD return line has been removed and the nozzle capped. These features meet the recommendations given in NUREG-0619.

DSER 3.9-19 (3.9.5.3.3, Page 3.9-129)	Have the reactor internals placed in the "other internals" category been seismically analyzed to show that they will not compromise the integrity of seismically qualified reactor internals?

Response

Response to this question is presently being prepared.

DSER 3.9-20 (3.9.6, Page 3.9-131) There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to pretect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure systems. The leak-tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems thus causing an intersystem LOCA.

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

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

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

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

Leak rates higher than 1 GPM will be considered if the leak rate changes are below 1 GPM above the previous test leak rate or system design precludes measuring 1 GPM with sufficient accuracy. These items will be reviewed on a case by case basis.

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

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

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

Response

CEI will supply details of the leak testing program at a later date.

DSER 3.9-21   Does this table apply to Perry?
(Table 3.9-1,
Page 3.9-134)

Response

Yes; however, further clarification is provided in revised Table 3.9-1.

DSER 3.9-22 What does "1*****" refer to?
(Table 3.9-1,
Page 3.9-135)

Response

Response to this question is provided in revised Table 3.9-1.

DSER 3.9-23 How many ADS cycles are included in the design of Perry?
(Table 3.9-1,
Page 3.9-135)

<u>Response</u>

Analysis shows that the probability of an inadvertent ADS actuation is 10^{-4} to 10^{-2} per year. Therefore, one inadvertent ADS cycle for the 40-year plant life is conservative.

DSER 3.9-24
(Table 3.9-1,
Page 3.9-136)

Standard Review Plan 3.9 requires 5 OBE's of 10 cycles each. If fewer cycles are used, justification must be provided.

<u>Response</u>

A GE generic study serves as the basis for one OBE with 10 cycles. This study included a large sample which bounds the Perry plant. In summary, the probability of one OBE intensity earthquake (50% of SSE intensity) is extremely low. The assumption that OBE envelopes the cumulative fatigue damage of 20 quarter-SSE's is more realistic. In addition, the use of 10 peak magnitude cycles is conservative as compared to 3 or 4 peak cycles shown by this study. Section 3.7.3.2.2 (see response to question DSER 3.7-15) has been revised to briefly describe this study. The following table shows a comparison between Perry and the actual earthquakes used in the base study.

	Earthquake	Component	Duration (sec)	Max Acceleration (g)
1.	El Centro, California (May 18, 1940)	NS	29.4 [3,5]	0.3[illegible] [1]
2.	Taft, California July 21, 1952)	S69E	30.0 [4,5]	0.18 [1]
3.	Golden Gate Park March 22, 1957)	S80E	13.24 [5]	0.13 [1]
4.	Perry Site	Horiz.	-	0.007 [6]
5.	Perry Site (Design Basis)	Horiz. (SSE)	10.0 [2]	0.15 [2]

NOTES:

1. Reference - "Earthquake Engineering", Robert L. Wiegel, Coordinating Editor, Prentice-Hall, Inc., 1970 - Table 4.8, page 83.

2. Reference - "Perry Nuclear Power Plant - Units 1 and 2 FSAR", page 3.7-2 and Figures 3.7-1 through 3.7-10.

3. Reference - "Earthquake Engineering", Robert L. Wiegel, Coordinating Editor, Prentice-Hall, Inc., 1970 - Figure 6.7, page 113.

4. Reference - "Earthquake Engineering", Robert L. Wiegel, Coordinating Editor, Prentice-Hall, Inc., 1970 - Figure 6.2, page 109.

5. Reference - General Electric Company internal study, "Seismic Safety of Nuclear Power Plants", Tables 1, 2, and 3, on pages 6.2, 6.4 and 6.5, respectively.

6. Reference - General Electric Company internal study, "Ratio of Ground Acceleration to SSE Accelerations", Table 2.

DSER 3.9-25 (Table 3.9-3, Page 3.9-141)	The acceptance criteria should reference the ASME Code Service Limits. A similar table is needed for the BOP.

Response

The response to this question is provided in revised Table 3.9-3 which refers to the ASME Code Service Limits.

A similar table for BOP is attached.

DSER 3.9-26 (Table 3.9-3a, Page 3.9-143)	"The results of stress and fatigue usage analysis are in detail in the vessel manufacturer's stress report and in new loads evaluation by GE within the code limits." Provide clarification of this statement.

Response

The response to this question is provided in revised Table 3.9-3a.

DSER 3.9-27 Some values in these tables are missing. Provide a schedule for
(Table 3.9-3m, their completion.
-3o, -3q, -3h)

Response

Response to this question is presently being prepared.

DSER 3.9-28 (Table 3.9-3s, Page 3.9-225) Provide an explanation for the results in this table.

Response

The response to this question is provided in revised Table 3.9-3a.

DSER 3.9-29 (Table 3.9-28, Page 3.9-282)	Where are the loads used in this table defined? How are these loads combined?

Response

The loads are defined at the end of the table under "LOADING LEGEND".

The loads are combined by vector summation.

DSER 3.9-30 (Table 3.9-32, Page 3.9-297)

Has Eq. b) been used? If so, provide the supporting data. If not, delete the equation from the table.

Response

Response to this question is presently being prepared.

DSER 3.9-31 (Table 3.9-33, Page 3.9-298)

Have Eqs. e), f), or g) been used? If so, provide the supporting data. If not, delete these equations from the table.

Response

Response to this question is presently being prepared.

DSER 3.9-32 (Table 3.9-34, Page 3.9-301)

Has Eq. c) been used. If so, provide the supporting data. If not, delete the equation from the table.

Response

Response to this question is presently being prepared.

DSER 3.9-33

(3.9.1.1.1, Page 3.9-1)

How many cycles due to SRV discharge are included in the analysis?

Response

There are 1800 actuations/40 years on Lo-Lo set mechanism.

DSER 3.9-34 (3.9.2.5, Page 3.9-67) Previous analyses for other nuclear plants have shown that certain reactor system components and their supports may be subjected to previously underestimated asymmetric loads under the conditions that result from the postulation of ruptures of the reactor coolant piping at various locations.

The applicant has described the design of the reactor internals for blowdown loads only. The applicant should also provide information on asymmetric loads. It is, therefore, necessary to reassess the capability of these reactor system components to assure that the calculated dynamic asymmetric loads resulting from these postulated pipe ruptures will be within the bounds necessary to provide high assurance that the reactor can be brought safely to a cold shutdown condition. The reactor system components that require reassessment shall include:

a. Reactor pressure vessel

b. Core supports and other reactor internals

c. Control rod drives

d. ECCS piping that is attached to the primary coolant piping

e. Primary coolant piping

f. Reactor vessel supports

The following information should be included in the FSAR about the effects of postulated asymmetric LOCA loads on the above mentioned reactor system components and the various cavity structures:

1. Provide arrangement drawings of the reactor vessel support systems in sufficient detail to show the geometry of all principal elements and materials of construction.

2. If a plant-specific analysis will not be submitted for your plant, provide supporting information to demonstrate that the generic plant analysis under consideration adequately bounds the postulated accidents at your facility. Include a comparison of the geometric, structural, mechanical, and thermal-hydraulic similarities between your facility and the case analyzed. Discuss the effects of any differences.

3. Consider all postulated breaks in the reactor coolant piping system, including the following locations:

   a. Steam line nozzles to piping terminal ends.

   b. Feedwater nozzle to piping terminal ends.

   c. Recirculation inlet and outlet nozzles to recirculation piping terminal ends.

   Provide an assessment of the effects of asymmetric pressure differentials* on the systems and components listed above in combination with all external loadings including safe shutdown earthquake loads and other faulted condition loads for the postulated breaks described above. This assessment may utilize the following mechanistic effects as applicable:

   a. Limited displacement -- break areas

   b. Fluid-structure interaction

   c. Actual time-dependent forcing function

   d. Reactor support stiffness

   e. Break opening times

*Blowdown jet forces at the location of the rupture (reaction forces), transient differential pressures in the annular region between the component and the wall, and transient differential pressures across the core barrel within the reactor vessel.

4. If the results of the assessment of item 3 above indicate loads leading to inelastic action of these systems or displacement exceeding previous design limits, provide an evaluation of the inelastic behavior (including strain hardening) of the material used in the system design and the effect of the load transmitted to the backup structures to which these systems are attached.

5. For all analyses performed, include the method of analysis, the structural and hydraulic computer codes employed, drawings of the models employed and comparisons of the calculated to allowable stresses and strains or deflections with a basis for the allowable values.

6. Demonstrate that safety-related components will retain their structural integrity when subjected to the combined loads resulting from the loss-of-coolant accident and the safe shutdown earthquake.

7. Demonstrate the functional capability of any essential piping when subjected to the combined loads resulting from the loss-of-coolant accident and the safe shutdown earthquake.

The applicant has outlined his approach for determining the forcing functions considered in the system and component dynamic analyses of reactor structures for normal operation and anticipated transients. These methods are a combination of analytical methods and predictions based on data from previously tested reactor internals of a similar design. The forcing function information is combined with dynamic modal analysis to form a basis for interpretation of the pre-operational and initial startup test results. Modal stresses are calculated and relationships are obtained between sensor responses and peak component stresses for each of the lower modes.

Response

Response to this question is presently being prepared.

DSER 3.9-35 (3.9.3.3.2, Page 3.9-106)

Provide justification for using a modified static analysis on the safety relief valve piping in the suppression pool and explain what is used for the "conservative dynamic load factor" in the analysis.

Provide the time-history transient forces resulting from the SRV actuation used in the SRV piping and support design including the loads developed from the discharging water slug.

Discuss the types of supports used on the SRV piping in both the drywell and suppression pool and provide drawings of the supports.

Provide the type of safety relief valves used in the plant, the valve opening time, and the sequences of valve actuation used in the analysis.

Response

The modified static analysis is accepted by the industry as an alternative to dynamic analysis for the purposes of piping stress analysis and support design. A dynamic load factor of 2 was used. This methodology agrees with Regulatory Guide 1.67 position C.4 and our experience shows it to be conservative.

The SRV piping system does not use a water seal and is sloped to avoid any water or condensation accumulation in the pipe. The only water slug in the submerged portion of the discharge pipe tends to dampen the peak of the unbalanced force exerted on the pipe sections during the discharge flow transient.

Although the static analysis is considered conservative, GAI is in the process of refining the analysis by using analysis codes such as RELAPS/TPIPE to perform complete thermohydraulic/structural time domain dynamic analyses for some of the lines in order to obtain more realistic design data. The effects of the water column in the submerged pipe on the transients also will be included.

The SRV piping uses anchors, guides, springs, snubbers (mechanical and hydraulic), and struts to obtain a rigidly restrained dynamic arrangement while allowing the needed flexibility for minimization of constraint to the thermal expansion of the piping.

The support and piping isometric drawings are attached for one typical SRV piping.

The SRV valves are made by Dikkers. They are described in Section 5.2. The valve set pressures are also included in Section 5.2. The valve opening time is .020 seconds (minimum).

DSER 3.9-36 (3.9.3.4.6, Page 3.9-113) Are the stress due to differential anchor movements considered as primary or secondary stresses for BOP supports?

Response

Stresses due to differential anchor movements are considered as secondary stresses in BOP supports. However, for some BOP supports, differential anchor movements are considered as primary stresses for the sake of simplicity.

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9.1.1 Design Transients

A summary of design transients used in the design and/or fatigue analysis of a typical plant are listed in Table 3.9-1. The number of cycles or events associated with each transient are included. Transients or combinations of transients are classified with respect to the plant and system operating condition categories identified as "Normal", "Upset", "Emergency", "Faulted" (service levels A, B, C, D respectively) or "Testing" in the ASME Code Section III, Division I, NA-2140, as applicable.

3.9.1.1.1 Architect/Engineer Defined Component Transients

The ASME Code Class 1 components not supplied by the NSSS vendor are comprised of piping, valves, containment penetrations, and pipe supports. These components have been specified and designed in accordance with the system design transients listed in Table 3.9-2.

3.9.1.2 Computer Programs Used in Analyses

The following sections discuss computer programs used in the analysis of the major safety related components. Computer programs were not used in the analysis of all components, thus, not all components are listed; e.g., main steam isolation and safety/relief valves and recirculation gate valves.

The GE computer programs are maintained either by General Electric or by outside computer program developers. In either case, the quality of the programs and the computed results are controlled. For each program, one or more engineers are assigned, whose duties are:

a. To keep abreast of the capability, the software contents, and the theory of the program.

3.9.2.1.2 Thermal Expansion Testing of Main Steam and Recirculation Piping

A thermal expansion preoperational and startup testing program, performed through the use of potentiometer sensors, has been established to verify that normal thermal movement occurs in the piping systems. The main purpose of this program is to ensure the following:

a. The piping system during system heatup and cooldown is free to expand, contract, and move without unplanned obstruction or restraint in the x, y, and z directions.

b. The piping system is working in a manner consistent with the assumption of the NSSS stress analysis.

c. There is adequate agreement between calculated values of displacements and measured value of displacement.

d. There is consistency and repeatability in thermal displacements during heatup and cooldown of the NSSS systems.

Limits of thermal expansion displacements would be established prior to start of piping testing to which the actual measured displacements can be compared. to determine acceptability of the actual motion. If the measured displacement does not vary from the acceptance limits values by more than the specified tolerance, the piping system is responding in a manner consistent with predictions and is therefore acceptable. Two levels of limits of displacements would be established to check the systems as explained in Section 3.9.2.1.4.

Based on the above criteria, Level 1 displacement limits are established for all instrumented points in the piping system. These limits shall be compared with the field measured piping displacements. Method of acceptance shall be as explained in the following section.

3.9.2.1.4 Test Evaluation and Acceptance Criteria for Main Steam and Recirculation Piping

The piping response to test conditions shall be considered acceptable if the organization responsible for the stress report reviews the test results and determines that the tests verify that the piping responded in a manner consistent with the predictions of the stress report and/or that the tests verify that piping stresses are within code limits (ASME Code, Section III, NB-3600). Acceptable deflection and acceleration limits are determined after the completion of piping systems stress analysis and are provided in the startup test specifications. To insure test data integrity and test safety, criteria have been established to facilitate assessment of the test while it is in progress. These criteria, designated Level 1 and 2, are described in the following paragraphs.

3.9.2.1.4.1 Level 1 Criterion

Level 1 establishes the maximum limits for the level of pipe motion which, if exceeded, makes a test hold or termination mandatory.

If the Level 1 limit is exceeded, the plant will be placed in a satisfactory hold condition, and the responsible piping design engineer will be advised. Following resolution, applicable tests must be repeated to verify that the requirements of the Level 1 limits are satisfied.

3.9.2.1.4.2 Level 2 Criterion

Level 2 specifies the level of pipe motion which, if exceeded, requires that the responsible piping design engineer be advised. If the Level 2 limit is not satisfied, plant operating and startup testing plans would not necessarily be altered. Investigations of the measurements, criteria, and calculations used to generate the pipe motion limits would be initiated. An acceptable resolution must be reached by all appropriate and involved parties, including the responsible piping design engineer. Depending upon the nature of such resolution, the applicable tests may or may not have to be repeated. 5

the component dynamic analyses. Comparison of measured vibration amplitudes to be predicted and allowable amplitudes will then be 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 head and the shroud head will be removed, the vessel will be drained, and major components will be inspected on a selected basis. The inspections will cover the shroud, shroud head, and core support structures, the jet pumps, and the peripheral control rod drive and incore guide tubes. Access will be provided to the reactor lower plenum for these inspections.

3.9.2.4.1 Compliance With Regulatory Guide 1.20

PNPP is committed to comply with Regulatory Guide 1.20.

Regulatory Guide 1.20 describes a comprehensive vibration assessment program for reactor internals during preoperational and initial startup testing. The vibration assessment program meets the requirements of Criterion 1, "Quality Standards and Records", of Appendix A to 10CFR50 and Section 50.34, "Contents of Applications: Technical Information", of 10CFR50. 8

Vibration testing of reactor internals is performed on all GE-BWR plants. Perry, being the first BWR/6 238 plant, will be considered a prototype and will be instrumented and subjected to preoperational and startup flow testing to demonstrate that flow-induced vibrations similar to those expected during operation will not cause damage.

General Electric is committed to confirm satisfactory vibration performance of internals in these plants through preoperational flow testing followed by inspection for evidence of excessive vibration. Extensive vibration measurements in prototype plants together with satisfactory operating experience in all eleven BWR/4 plants have established the adequacy of BWR/6 reactor internal designs. General Electric will continue these test programs for the BWR/6 238 plants to verify structural integrity and to establish the margin of safety.

3.9.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

In order to assure that no significant dynamic amplification of load occurs as a result of the oscillatory nature of the blowdown forces (Figures 3.9-16 and 3.9-17), a comparison will be made of the periods of the applied forces and the natural periods of the core support structures acted upon by the applied forces. These periods will be determined from a 12 node vertical dynamic model of the BWR6-238 RPV and internals. Only motion in the vertical direction will be considered here; hence, each structural member (between two mass points) can only have an axial load. Besides the real masses of the RPV and core support structures, account will be made for the water inside the RPV.

Typical curves of the variation of pressures during a steam line break are shown in Figures 3.9-16 and 3.9-17. The accident analysis method is described in Section 3.9.5.2.

The time varying pressures are applied to the dynamic model of the reactor internals described above. Except for the nature and locations of the forcing functions and the dynamic model, the dynamic analysis method is identical to that described for seismic analysis and is detailed in Section 3.7.2.1. The dynamic components of forces from these loads will be combined with dynamic force components from other dynamic loads (including seismic and hydrodynamic), all acting in the same direction, by the square root of the sum of the squares (SRSS) method. This resultant force will then be combined with other steady state and static loads on an absolute sum basis to determine the design load in a given direction.

A summary of the results of the dynamic analysis of the reactor internals is given in Table 3.9-3b.

3.9.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

BWR 4 and 5 reactors now in service provide the basis for analytical prediction of vibrational behavior during the component design stage. GE Licensing Topical Report NEDE-24057-P presented to the NRC for Susquehanna SER contains results of such tests and measurement. However, the BWR 4 and 5 operational experience has not been used in lieu of a vibration measurement for the Perry reactor internals. Perry's component design adequacy for flow-induced vibration is confirmed through actual in-reactor measurements. Additionally, Grand Gulf, with similar flow characteristics and internal design as Perry, is also the BWR/6 prototype for Perry.

Prior to initiation of the instrumented vibration measurement program for the prototype plant, extensive dynamic analyses of the reactor and internals are performed. The results of these analyses are used to generate the allowable vibration levels during the vibration test. The vibration data obtained during the test are analyzed in detail. The results such as vibration amplitudes, natural frequencies, and mode shapes, are then compared to those obtained from the dynamic model for seismic and LOCA analyses.

3.9.3 ASME CODE CLASS 1, 2, AND 3 COMPONENTS, COMPONENT SUPPORTS, AND CORE SUPPORT STRUCTURES

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

This section delineates the criteria for selection and definition of design limits and loading combinations associated with normal operation, postulated accidents, and specified seismic and hydrodynamic events for the design of safety related ASME Code components (except containment components) which are discussed in Section 3.8.

Table 3.9-3c shows the calculated stress values and allowable stress limits for the heat exchangers.

3.9.3.1.2 Loading Combinations, Design Transients, and Stress Limits for Balance of Plant Components and Supports

Balance of plant systems and components are identified in accordance with ASME Code Class and Safety Class as discussed in Section 3.2. Design limits and loading combinations for Seismic Category I fluid system components are in compliance with the recommendations of Regulatory Guide 1.48. ASME Code Class 1 components are discussed in Section 3.9.1.1.1. ASME Code Class 2 and 3 systems and components are designed to operate under the following plant conditions:

a. ASME Code Class 2 and 3 systems and components are designed to operate under anticipated environmental conditions, such as pressure, temperature, irradiation, etc., that may occur during normal plant operations and transients, including startup, power generation, relief valve operation and shutdown.

b. Components of essential systems, required to function during and/or after any of the abnormal events identified in Section 3.9.1 are designed to function under environmental conditions that would occur during and after such events. Section 3.11 describes environmental design conditions associated with such abnormal events.

The plant conditions postulated to occur during the life of the plant are identified in Section 3.9.3.1.1.1.

Loadings considered in component design are those effects derived from plant and system conditions of operation, natural phenomena, and site related hazards. These loadings include, but are not limited to, loading effects resulting from:

a. Internal or external pressure.

The design criteria and dynamic testing requirements for component supports are given below:

a. Component Supports

All component supports are designed, fabricated, and assembled so that they cannot become disengaged by the movement of the supported pipe or equipment after they have been installed. All component supports are designed in accordance with the rules of Subsection NF of the ASME Code. For the NSSS scope of supply, valve operators which are mounted on all safety-graded piping systems are not used as component supports. -13

b. Hangers

The design load on hangers is the load caused by dead weight. The hangers are calibrated to ensure that they support the design load at both their hot and cold load settings. Hangers provide a specified down travel and up travel in excess of the specified thermal movement.

c. Snubbers

1. Required Load Capacity and Snubber Location

The entire piping system including valves and suspension system between anchor points is mathematically modeled for complete structural analysis. In the mathematical model, the snubbers are modeled as a spring with a given spring stiffness depending on the snubber size. The analysis determines the forces and moments acting on each piping component and the forces acting on the snubbers due to all dynamic loading conditions defined in the piping design specification. The design load on snubbers includes loads caused by seismic forces including hydrodynamic forces (operating basis earthquake and safe shutdown earthquake), system anchor movements, reaction forces caused by relief valve discharge and turbine stop valve closure.

c. Hydraulic power supply (pumps),

d. Interconnecting piping,

e. Flow and pressure and isolation valves and,

f. Instrumentation and electrical controls.

Quality group classification is not applicable to the CRD.

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

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

Pertinent aspects of the design and qualification of the CRD components are discussed in the following locations: transients in Section 3.9.1.1, faulted conditions in Section 3.9.1.4, and seismic testing in Section 3.9.2.2.

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformation

The ASME Code components of the CRD system have been evaluated analytically and the design loading conditions, stress criteria and calculated stresses, and the allowable stresses are summarized in Table 3.9-3. For the noncode components, experimental testing was used to determine the CRD performance under all possible conditions as described in Section 3.9.4.4. Deformation has been compared with the allowables and is not a limiting factor in the analysis of the CRD components since the stresses are in the elastic region.

3.9.4.4 CRD Performance Assurance Program

The CRD test program consists of the following:

a. Development tests

b. Factory quality control tests

c. Five year maintenance life tests

d. 1.5X design life tests

TABLE 3.9-1

PLANT EVENTS

a. (Pressure/Temperature/Flow Transients) Normal, Upset, and Testing Conditions	No. of Cycles
1. Bolt up (1)	123
2. Design Hydrostatic Test	40
a. Leak Checks at 400 psig prior to power operation, 3 cycles/ startup	
3. Startup (100°F/hr Heatup Rate) (2)	120
4. Daily Reduction to 75% Power (1)	10,000
5. Weekly Reduction 50% Power (1)	2,000
6. Control Rod Pattern Change (1)	400
7. Loss of Feedwater Heaters (80 Cycles Total)	80
8. OBE at Rated Operating Conditons	10 (4)
9. Scram:	
a. Turbine Generator Trip, Feedwater On, Isolation Valves Stay Open	[illegible]
b. Other Scrams	14[illegible]
c. Loss of Feedwater Pumps, Isolation Valves Closed	10
d. Turbine Bypass, Single Safety or Relief Valve Blowdown	8
10. Reduction to 0% Power, Hot Standby, Shutdown (100°F/hr Cooldown Rate) (2)	111
11. Unbolt (1)	123

TABLE 3.9-1 (Continued)

(Pressure/Temperature/Flow Transients) Emergency Conditions	No. of Cycles
b.	
1. Scram:	
a. Reactor Overpressure with Delayed Scram, Feedwater Stays on, Isolation Valves Stay Open	1[3]
b. Automatic Blowdown (ADS)	1[3]
2. Improper Start of Cold Recirculation Loop	1[3]
3. Sudden Start of Pump in Cold Recirculation Loop	1[3]
4. Hot Standby with Reactor Drain Shut Off Followed by Pump Restart	1[3]
c. Faulted Condition	
1. Pipe Rupture and Blowdown	1[3]
2. Safe Shutdown Earthquake during Refueling	1[3]

NOTES:

1. Applies to reactor pressure vessel only.
2. Bulk average vessel coolant temperature change in any 1-hour period.
3. The annual encounter probability of the one cycle event is $<10^{-2}$ for emergency and $<10^{-4}$ for faulted events.
4. Includes 10 maximum loading cycles per event.

TABLE 3.9-1 (Continued)

In addition to the above temperature/pressure/flow transients the following dynamic load transients have been considered to the design and/or fatigue evaluation of a typical BWR 6 standard plant:

d. Dynamic/Transient Load

	Dynamic/Transient Load	Category	Cycles/Events
1.	Operating Basis Earthquake (OBE)(5)	Upset	10 cycles
2.	Safe Shutdown Earthquake (SSE)(6)	Faulted	1 cycle
3.	Turbine Stop Valve Closure (TSV)(7)	Upset	690 cycles
4.	Safety Relief Valve Actuation (Accoustic wave)(7)	Upset	5460 cycles
5.	Safety Relief Valve Actuation (Structural Feedback)(8)	Upset	660 full range cycles 880 half range cycles
6.	Loss of Collant Accident (LOCA):		
	Small break LOCA	Emergency/faulted	1 event
	Intermediate break LOCA	Faulted	1 event
	Large break LOCA	Faulted	1 event

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5. One 50% SSE event includes 10 maximum load cycles.
6. One stress reversal cycle of maximum seismic amplitude.
7. Applicable to main steam piping system only.
8. Applicable to main steam and recirculation piping systems only; however for core support and reactor internals components, the cycles used are 2,600 based on distributed SRV loadamplitudes.

TABLE 3.9-3

DESIGN LOADING COMBINATIONS OF ASME CODE CLASS 1, 2 AND 3 COMPONENTS

a. NSSS

Load Case[1]	N	SRV_x[4]	SRV_{ADS}	OBE	SSE	SBA/IBA[3]	DBA	ASME Code Service Limit
1	X	X						B
2	X	X		X				B[5]
3	X	X			X			D[2]
4	X		X			X		D[2]
5	X		X	X		X		D[2]
6	X		X		X	X		D[2]
7	X				X		X	D[2]
8	X							A
9	X			X				B

b. BOP

Load Case[1]	N	SRV_x[4]	SRV_{ADS}	OBE	SSE	SBA/IBA[3]	DBA	ASME Code Service Limit
1	X	X						B
2	X			X				B
3	X				X			C
4	X	X		X				P
5	X	X			X			C
6	X	X				X		B
7	X		X			X		B
8	X		X	X		X		C
9	X		X		X	X		C
10	X				X		X	C

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NOTES:

1. See legend for definition of terms.

2. All ASME Code Class 1, 2, and 3 piping systems which are required to function for safe shutdown under the postulated events shall meet the requirements of NRC's "Interim Technical Position-Function Capability of passive components" - by MEB.

3. SBA or IBA whichever is greater.

4. SRV_{ALL} or SRV, - whichever is controlling will be used.

5. For load case 2, all ASME code service level B requirements are to be met, excluding fatigue evaluation.

TABLE 3.9-3a

REACTOR PRESSURE VESSEL AND SHROUD SUPPORT ASSEMBLY[1]:

The reactor vessel is designed and analyzed to comply with ASME Code, Section III (NB-3200). The results of stress and fatigue usage analysis are given in detail in the stress report. They are within the code allowables, as demonstrated by the following tabulation.

The shroud support is designed and analyzed to comply with the ASME Code Section III, subsection NG. Stress and fatigue analysis results are completed by GE and all results are within the Code limits.

NOTE:

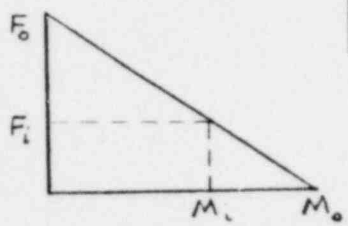
1. The vessel, support skirt, and shroud support, including legs, cylinder, and plate, are furnished as a completed assembly by the vessel manufacturer.

TABLE 3.9-3o - (Continued)

Loading/Component	Criteria	Allowable Nozzle Forces and Moments Force in lb., Moment in ft-lb	Actual Nozzle Loads
3. Nozzle Loads: Faulted Design Pressure and Temperature Dead Weight, Thermal Expansion, Safe Shutdown Earthquake,	The maximum moments due to pipe reaction and the maximum forces shall not exceed the allowable limits. Primary Stress Smaller of $0.70\ S_u$ or 2.4S per ASME Section III allowable.	See below (a) (b)	(By 2-28-82)

(a) The following expression relates the allowable combination of forces and moments:

$$\left[\frac{Fi}{Fo}\right] + \left[\frac{Mi}{Mo}\right] \leq 1$$



Where:

Fi - The largest of any of the three actual external orthogonal forces (Fx, Fy, and Fz)
Mi = The largest of any of the three actual external orthogonal moments (Mx, My, and Mz)
Fo = The allowable value of Fi when all moments are zero
Mo = The allowable value of Mi when all forces are zero

NOTES: One coordinate axis must be the nozzle centerline. Another coordinate axis must be parallel to the heat exchanger centerline except where the heat exchanger centerline is parallel to the nozzle centerline. In this case, the coordinate axis must be orthogonal to the nozzle centerline and at 0° - 180° or 90° - 270° azimuth.

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TABLE 3.9-3a - (Continued)

Fuel Assembly (Including Channel)

Acceptance Criteria	Loading	Primary Load	Combined[1] Acceleration	Design Basis[1] Acceleration
Based on Methodology contained in GE Document Number NEDE-21175-P				
	Normal & Upset Condition: 1. Peak pressure 2. Operating Basis Earthquake 3. Safety Relief Valve	Acceleration profile	0.64g	2.6g
	Emergency/Faulted Condition: 1. Peak pressure 2. Safe Shutdown Earthquake 3. Annulus pressurization	Acceleration profile	1.13g	3.9g

NOTES:

1. The design assessment show that the appropriately combined accelerations, as per Table 3.9-3, are lower than the design basis accelerations.

2. The analysis results of the limiting vertical load cases, i.e., OBE+SRV+SCRAM for normal/upset events and SSE+SRV+SCRAM for emergency/faulted events, indicate that the fuel lift does not occur.

3. The fatigue analysis indicates that the fuel asembly has adequate fatigue capability to withstand the loadings resulting from multiple SRV actuations and the OBE+SRV event.

TABLE 3.9-28

SUMMARY OF RESULTS
ASSURANCE OF OPERABILITY ANALYSIS
CONTROL COMPLEX CHILLED WATER PUMPS & EMERGENCY CLOSED COOLING PUMPS

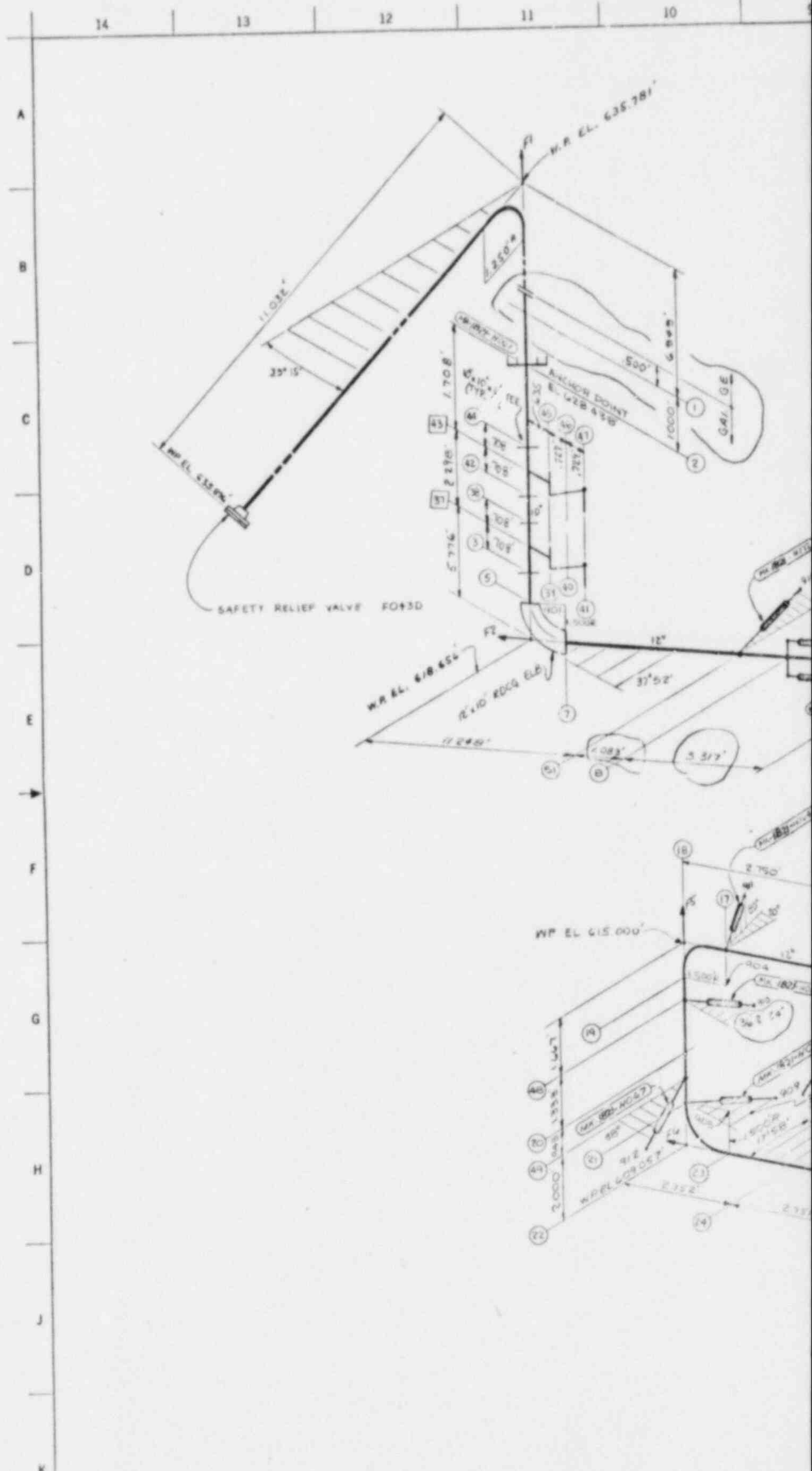
Component	Combined Stress Designation	Loads	Calculated Stress, psi	Allowable Stress, psi
Casing	Bending	1	10396	14000
Flange Bo'ting	Direct	1	15139	25000
Suction Flange	Longitudinal	1 + 4	5154	21000
	Radial	1 + 4	1400	21000
	Tangential	1 + 4	1901	21000
	Longitudinal	1 + 5	5733	31500
	Radial	1 + 5	1564	31500
	Tangential	1 + 5	2123	31500
Discharge Flange	Longitudinal	1 + 4	8959	21000
	Radial	1 + 4	3231	21000
	Tangential	1 + 4	3278	21000
	Longitudinal	1 + 5	9742	31500
	Radial	1 + 5	3528	31500
	Tangential	1 + 5	3580	31500
Mounting Foot	Direct + Bending	2 + 3 + 4	4227	14000
	Shear	2 + 3 + 4	945	8400
	Direct + Bending	2 + 3 + 5 + 6	5.18	21000
	Shear	2 + 3 + 5 + 6	1144	12600
Feet Bolts	Normal	2 + 3 + 4	35140	35640
	Normal	2 + 3 + 5 + 6	42590	53460

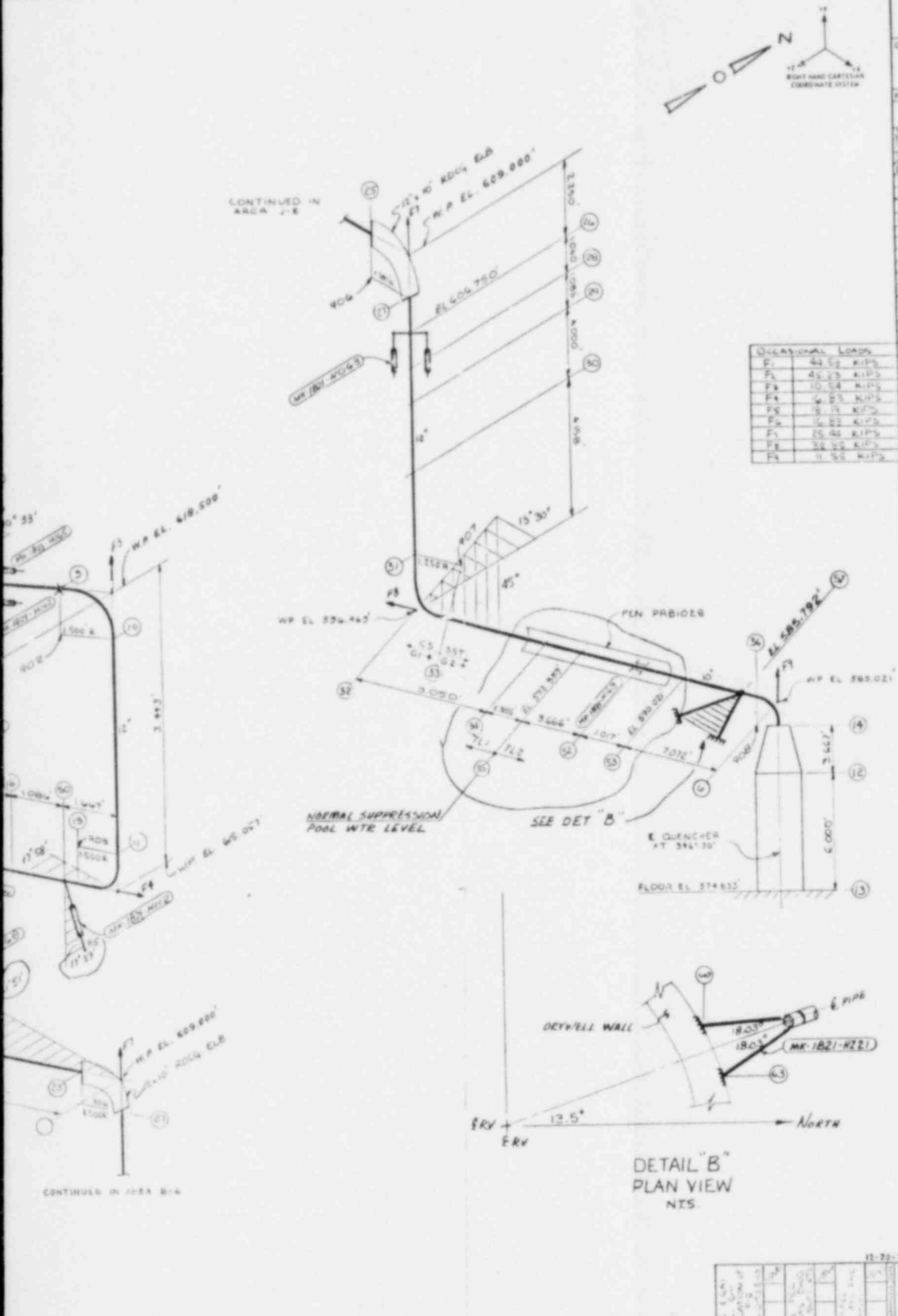
TABLE 3.9-28 (Continued)

Component	Combined Stress Designation	Loads	Calculated Stress, psi	Allowable Stress, psi
Feet Shear Pins	Shear	2 + 3 + 4	15352	16800
	Shear	2 + 3 + 5 + 6	18464	25200
Mounting Pad	Direct + Bending	2 + 3 + 4	2547	18000
Weld	Shear	2 + 3 + 4	837	12000
	Direct + Bending	2 + 3 + 5 + 6	3083	24000
	Shear	2 + 3 + 5 + 6	1006	16000
Top-plate	Bending	2 + 4	13192	18000
	Bending	2 + 5 + 6	17317	24000
Foundation Bolts	Normal	2 + 3 + 4	11286	18410
	Shear	2 + 3 + 4	5990	10800
	Normal	2 + 3 + 5 + 6	14224	27615
	Shear	2 + 3 + 5 + 6	7403	16200
Stuffing Box Bolts	Normal	1	6786	25000
Shaft	Shear	3	1885	7500

LOADING LEGEND

1. Design Pressure
2. Deadweight
3. Shaft Torque
4. Normal or steady state nozzle loads
5. O.B.E./S.S.E. Nozzle Loads
6. Safe Shutdown Earthquake


W.P. EL. 635.781'
11.032'
35° 15'
WP EL. 632.696'
SAFETY RELIEF VALVE FO43D
ANCHOR POINT
W.P. EL. 618.656'
10"x10" RDCG ELB
12"
37° 52'
3.317'
WP EL 615.000'
2.750'
2.352'



Occasional Loads	
F_1	44.50 KIPS
F_2	45.23 KIPS
F_3	10.54 KIPS
F_4	6.83 KIPS
F_5	18.19 KIPS
F_6	16.83 KIPS
F_7	25.44 KIPS
F_8	38.45 KIPS
F_9	11.56 KIPS

DESIGN CONDITIONS: 500 PSIG 470°F

OPERATING CONDITIONS: 500 PSIG 500° F TL1, 350° F TL2

PIPING CODE ANSI B31.1

PIPE MATERIAL	LINE SPEC	LINE SIZE	REF TEMP °F	LIN EXP	Ex	Ec
SEAMLESS [illegible]	G1-4	10	70			
[illegible]	G1-4	12	70			
[illegible]	G2-4	10	70			
[illegible]	G2-4	30	70			
[illegible]	G1-4	6	70			

PIPE SIZE NOM	O.D.	WALL THK.	WEIGHT LB. LINEAR FT. PIPE	CONTENT	INSULATION
10	10.750	.365	40.5		
12	12.750	.406	53.5		
6	6.625	.280	18.98		
30	30.000	1.563	475.26		

ANCHOR BRANCH & TERMINAL POINTS	COORDINATES X	Y	Z

SYMBOLS

- RIGID HANGER Y+X
- VARIABLE SPRING HGR
- CONSTANT LOAD HGR
- GUIDE
- HYDRAULIC RESTRAINT
- RIGID RESTRAINT
- ANCHOR
- CENTER OF GRAVITY
- MASS POINT
- U-BOLT
- WELDOLET
- FISHMOUTH
- SWEEPOLET
- TEE

NOTES:

1. DIMENSIONS ARE IN FEET AND DECIMALS OF A FOOT. DECIMALS SHALL BE CARRIED TO 3 PLACES.
2. ~~ALL VALVES AND EQUIPMENT ASSUMED INFLEXIBLE.~~

FOR REFERENCE ONLY

B21G08

REFERENCE DWG

TITLE	NUMBER	REV
M.S. SAFETY RELIEF PIPING	[illegible]	A1
M.S. SAFETY RELIEF PIPING	D-[illegible]	A1

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

PERRY NUCLEAR POWER PLANT UNIT 1

PIPING SYS N 11 ANALYSIS DIAGRAM

MAIN STEAM SAFETY RELIEF VALVE

[illegible] DISCHARGE TO SUPP POOL

GILBERT ASSOCIATES, INC.
ENGINEERS AND CONSULTANTS

12-20-74

REVISIONS

04 4549 D-314-011 12 3