Westinghouse Energy Systems

ENCLOSURE 2
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Westinghouse Energy Systems

WCAP-13166

TECHNICAL JUSTIFICATION FOR ELIMINATING RESIDUAL HEAT REMOVAL LINES RUPTURE AS THE STRUCTURAL DESIGN BASIS FOR COMANCHE PEAK NUCLEAR POWER PLANT - UNIT 2

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SECTION 1.0
INTRODUCTION

1.1 Background

The current structural design basis for the residual heat removal (RHR) lines in Comanche Peak Unit 2 requires postulating non-mechanistic circumferential and longitudinal pipe breaks. This results in additional plant hardware (e.g. pipe whip restraints and jet shields) which would mitigate the dynamic consequences of the pipe breaks. It is, therefore, highly desirable to be realistic in the postulation of pipe breaks for the RHR lines. Presented in this report are the descriptions of a mechanistic pipe break evaluation method and the analytical results that can be used for establishing that a circumferential type break will not occur within the Residual Heat Removal (RHR) Lines. This methodology used is generally referred to as leak-before-break (LBB). The evaluations considering circumferentially oriented flaws cover longitudinal cases.

The phenomena which could cause potential thermal cycling and stratification in the Comanche Peak Unit 1 RHR lines are described in WCAP-12258 (1-1) using the experience from the plants where cracking was observed due to thermal cycling and stratification in general. The cracking was observed in the RHR line of the Genkai Plant Unit 1 in particular. WCAP-12258 describes thermal loading, structural stability and the crack sizes corresponding to 10 GPM leakage for two cases: These are the case where the first valve from the primary loop is not leaking and the case where this valve is leaking. The RHR lines layout and geometries of Comanche Peak Unit 2 are same as those of Unit 1. The operating transients of both units are identical. Therefore the loadings obtained from the evaluation as described in WCAP-12258 were used in the LBB evaluations presented in this report.

1.2 Scope and Objectives

The general purpose of this investigation is to demonstrate leak-before-break (LBB) for the RHR lines. The scope of the report is limited to the high energy portion of the RHR lines (primary loop junction to the first valve).

The leak-before-break demonstrations for the RHR lines are considered for two cases: 1) when the leakage through the first valve from the primary loop would not occur and 2) the leakage through the first valve would occur. The first case corresponds to the non-stratification case and the second case includes the thermal stratification effects due to the valve leakage.

Schematic drawings of the piping systems are shown in Section 3.0. The recommendations and criteria proposed in NUREG 1061 Volume 3 (1-2) are used in this evaluation. The criteria and the resulting steps of the evaluation procedure can be briefly summarized as follows:

1) Calculate the applied loads. Identify the location at which the highest stress occurs.

2) Identify the materials and the associated material properties.

3) Postulate a through-wall flaw at the governing location. The size of the flaw should be large enough so that the leakage is assured of detection with margin using the installed leak detection equipment when the pipe is subjected to normal operating loads. A margin of 10 is demonstrated between the calculated leak rate and the leak detection capability.

4) Using maximum faulted loads, demonstrate that there is a margin of at least 2 between the leakage size flaw and the critical size flaw.

5) Review the operating history to ascertain that operating experience has indicated no particular susceptibility to failure from the effects of corrosion, water hammer or low and high cycle fatigue.

6) For the materials actually in the plant provide the material properties and justify that the properties used in the evaluation are representative of the plant specific material.

7) Demonstrate margin on applied load.

The flaw stability analyses are performed using the methodology described in SRP 3.6.3 (1-3).

The leak rates are calculated for the normal operating condition. The leak rate prediction model used in this evaluation is an [

]^{a,c,e}. The crack opening area required for calculating the leak rates is obtained by subjecting the postulated through-wall flaw to normal operating loads (1-4). Surface roughness is accounted for in determining the leak rate through the postulated flaw.

The computer codes used in this evaluation for leak rate and fracture mechanics calculations have been validated (bench marked).

1.3 References

1-1 WCAP-12258, Evaluation of Thermal Stratification for the Comanche Peak Unit 1 Residual Heat Removal Lines, April 1989 (Westinghouse Proprietary Class 2).

1-2 Report of the U.S. Nuclear Regulatory Commission Piping Review Committee - Evaluation of Potential for Pipe Breaks, NUREG 1061, Volume 3, November 1984.

1-3 Standard Review Plan; public comments solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday, August 28, 1987/Notices, pp. 32626-32633.

1-4 NUREG/CR-3464, 1983, "The Application of Fracture Proof Design Methods Using Tearing Instability Theory to Nuclear Piping Postulated Circumferential Through Wall Cracks."

SECTION 2.0

OPERATION AND STABILITY OF THE RHR LINE

2.1 Stress Corrosion Cracking

The Westinghouse reactor coolant system primary loop and connecting Class 1 lines have an operating history that demonstrates the inherent operating stability characteristics of the design. This includes a low susceptibility to cracking failure from the effects of corrosion (e.g., intergranular stress corrosion cracking). This operati history totals over 450 reactor-years, including five plants each having o..r 17 years of operation and 15 other plants each with over 12 years of operation.

In 1978, the United States Nuclear Regulatory Commission (USNRC) formed the second Pipe Crack Study Group. (The first Pipe Crack Study Group established in 1975 addressed cracking in boiling water reactors only.) One of the objectives of the second Pipe Crack Study Group (PCSG) was to include a review of the potential for stress corrosion cracking in Pressurized Water Reactors (PWR's). The results of the study performed by the PCSG were presented in NUREG-0531 (Reference 2-1) entitled "Investigation and Evaluation of Stress Corrosion Cracking in Piping of Light Water Reactor Plants." In that report the PCSG stated:

> "The PCSG has determined that the potential for stress-corrosion cracking in PWR primary system piping is extremely low because the ingredients that produce IGSCC are not all present. The use of hydrazine additives and a hydrogen overpressure limit the oxygen in the coolant to very low levels. Other impurities that might cause stress-corrosion cracking, such as halides or caustic, are also rigidly controlled. Only for brief periods during reactor shutdown when the coolant is exposed to the air and during the subsequent startup are conditions even marginally capable of producing stress-corrosion cracking in the primary systems of PWRs. Operating experience in PWRs supports this determination. To date, no stress-corrosion cracking has been reported in the primary piping or safe ends of any PWR."

During 1979, several instances of cracking in PWR feedwater piping led to the establishment of the third PCSG. The investigations of the PCSG reported in NUREG-0691 (Reference 2-2) further confirmed that no occurrences of IGSCC have been reported for PWR primary coolant systems.

As stated above, for the Westinghouse plants there is no history of cracking failure in the reactor coolant system loop or connecting Class 1 piping. The discussion below further qualifies the PCSG's findings.

For stress corrosion cracking (SCC) to occur in piping, the following three conditions must exist simultaneously: high tensile stresses, susceptible material, and a corrosive environment. Since some residual stresses and some degree of material susceptibility exist in any stainless steel piping, the potential for stress corrosion is minimized by properly selecting a material immune to SCC as well as preventing the occurrence of a corrosive environment. The material specifications consider compatibility with the system's operating environment (both internal and external) as well as other material in the system, applicable ASME Code rules, fracture toughness, welding, fabrication, and processing.

The elements of a water environment known to increase the susceptibility of austenitic stainless steel to stress corrosion are: oxygen, fluorides, chlorides, hydroxides, hydrogen peroxide, and reduced forms of sulfur (e.g., sulfides, sulfites, and thionates). Strict pipe cleaning standards prior to operation and careful control of water chemistry during plant operation are used to prevent the occurrence of a corrosive environment. Prior to being put into service, the piping is cleaned internally and externally. During flushes and preoperational testing, water chemistry is controlled in accordance with written specifications. Requirements on chlorides, fluorides, conductivity, and pH are included in the acceptance criteria for the piping.

During plant operation, the reactor coolant water chemistry is monitored and maintained within very specific limits. Contaminant concentrations are kept below the thresholds known to be conducive to stress corrosion cracking with the major water chemistry control standards being included in the plant operating procedures as a condition for plant operation. For example, during normal power operation, oxygen concentration in the RCS and connecting Class 1 lines is expected to be in the ppb range by controlling charging flow chemistry and maintaining hydrogen in the reactor coolant at specified concentrations. Halogen concentrations are also stringently controlled by maintaining concentrations of chlorides and fluorides within the specified limits. This is assured by controlling charging flow chemistry. Thus during plant operation, the likelihood of stress corrosion cracking is minimized.

2.2 Water Hammer

Overall, there is a low potential for water hammer in the RCS and connecting RHR line since they are designed and operated to preclude the voiding condition in normally filled lines. The RCS and connecting RHR line including piping and components, are designed for normal, upset, emergency, and faulted condition transients. The design requirements are conservative relative to both the number of transients and their severity. Relief valve actuation and the associated hydraulic transients following valve opening are considered in the system design. Other valve and pump actuations are relatively slow transients with no significant effect on the system dynamic loads. To ensure dynamic system stability, reactor coolant parameters are stringently controlled. Temperature during normal operation is maintained within a narrow range by control rod position; pressure is controlled by pressurizer heaters and pressurizer spray also within a narrow range for steady-state conditions. The flow characteristics of the system remain constant during a fuel cycle because the only governing parameters, namely system resistance and the reactor coolant pump characteristics are controlled in the design process. Additionally, Westinghouse has instrumented typical reactor coolant systems to verify the flow and vibration characteristics of the system. Preoperational testing and operating experience have verified the Westinghouse approach. The operating transients of the RCS primary piping and connected RHR lines are such that no significant water hammer can occur.

2.3 Low Cycle and High Cycle Fatigue

Low cycle fatigue considerations are accounted for in the design of the piping system through the fatigue usage factor evaluation to show compliance with the rules of Section III of the ASME Code.

Pump vibrations during operation would result in high cycle fatigue loads in the piping system. During operation, an alarm signals the exceedence of the RC pump shaft vibration limits. Field measurements have been made on the reactor coolant loop piping of a number of plants during hot functional testing. Stresses in the elbow below the RC pump have been found to be very small, between 2 and 3 ksi at the highest. Recent field measurements on typical PWR plants indicate vibration amplitudes less than 1 ksi. When translated to the connecting RHR lines, these stresses would be even lower, well below the fatigue endurance limit for the RHR line material and would result in an applied stress intensity factor below the threshold for fatigue crack growth.

2.4 Summary Evaluation of RHR Line for Potential Degradation During Service

In the Westinghouse PWR design only one incident of service cracking has been identified in the RHR piping. In that specific case the cracking was attributed to thermal cycling resulting from valve malfunction. The thermal cycling caused an initial surface flaw to grow and reveal its presence by leakage. The leakage was detected by the plant leak detection systems, the plant was shutdown and necessary repairs were completed. In addition only one incident of wall thinning has been identified in RHR lines of Westinghouse PWR design. However, this is of no concern in the present application as described later in this section. Sources of such degradation are mitigated by the design, construction, inspection, and operation of the RHR lines.

Based on a review of references 2-3 through 2-6 only one incident of water hammer has been reported in a PWR RHR system. This incident was a result of incorrect valve line-up preceding a pump start. The only damage sustained was to several pipe supports. Therefore it is concluded that water hammer in the

RHR system is unlikely to affect piping integrity or to cause pipe system degradation.

Wall thinning by erosion and erosion-corrosion effects will not occur in the RHR lines due to the low velocity, typically less than 10 ft/sec and the material, austenitic stainless steel, which is highly resistant to these degradation mechanisms. Per NUREG-0691 [2-2], a study of pipe cracking in PWR piping, only two incidents of wall thinning in stainless steel pipe were reported as noted earlier. One incident was related to the RHR system. However, this occurred in the pump recirculation path which has higher flow velocity and is more susceptible to other contributing factors such as cavitation, than the RHR piping near the primary loop. Therefore, wall thinning is not a significant concern in the portion of the system being addressed in this evaluation.

Flow stratification, where low flow conditions permit cold and hot water to separate into distinct layers, can cause significant thermal fatigue loadings. This was an important issue in PWR feedwater piping where temperature differences of 300°F were not uncommon under certain operational conditions. Stratification is believed to be important where low flow conditions and a temperature differential exist. This is not an issue in the RHR line, where typically there is no flow during normal plant operation. During RHR operation the flow causes sufficient mixing to eliminate stratification.

The maximum normal operating temperature of the RHR piping is about 618°F. This is well below the temperature which would cause any creep damage in stainless steel piping.

2.5 References

2-1 Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants, NUREG-0531, U.S. Nuclear Regulatory Commission, February 1979.

2-2 Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors, NUREG-0691, U.S. Nuclear Regulatory Commission, September 1980.

2-3 Utter, R. A., et. al., "Evaluation of Water Hammer Events in Light Water Reactor Plants," NUREG/CR-2781, published July 1982.

2-4 "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Other Dynamic Loads and Load Combinations," NUREG-1061 Volume 4, Published December 1984.

2-5 Chapman, R. L., et. al., "Compilation of Data Concerning Known and Suspected Water Hammer Events in Nuclear Power Plants, CY 1969-May 1981," NUREG/CR-2059, Published April 1982.

2-6 "Evaluation of Water Hammer Occurrence in Nuclear Power Plants," NUREG-0929 Revision 1, Published March 1984.

SECTION 3.0
MATERIAL CHARACTERIZATION

3.1 Pipe and Weld Materials

The pipe materials of the RHR lines for the Comanche Peak Unit 2 are A376/TP316, A403/WP316, A312/TP304 and A182/TP316. This portion of the RHR lines does not include any cast pipe or cast fittings. The welding processes used were GTAW, SMAW and SAW. Weld locations are identified in Figures 3-1 and 3-2. The two RHR lines of Comanche Peak Unit 2 are connected to loops 1 and 4.

In the following section the tensile properties of the materials are presented for use in the leak-before-break analyses.

3.2 Material Properties

The room temperature mechanical properties of the two RHR lines materials were obtained from the Certified Materials Test Reports and are given in Tables 3-1 and 3-2. The room temperature ASME Code Section III minimum properties (3-1) are given in Table 3-3. It is seen that the measured properties well exceed those of the Code. As mentioned in Section 1.2, LBB evaluations are performed for two cases, without and with leaking conditions in the first valve. The material properties for these evaluations are found separately for each case, since temperatures at the locations to analyze are different for each case. The minimum and average tensile properties at room temperature are given in Table 3-4.

The minimum and average tensile properties were calculated by using the ratio of the ASME Code Section III properties at the temperatures of interest alluded to above. The modulus of elasticity values were established at the temperatures of interest from the ASME Code Section III (Table 3-6). In the leak-before-break evaluation, the representative minimum properties (yield stress and ultimate strength) at the given temperature are used for the flaw stability evaluations and the representative average properties are used in the leak rate predictions. These properties are summarized in Table 3-5.

3.3 References

3-1 ASME Boiler and Pressure Vessel Code Section III, Division 1, Appendices July 1, 1989.

TABLE 3-1

ROOM TEMPERATURE MECHANICAL PROPERTIES OF
THE RHR LINES MATERIALS (LOOP 1 - RHR LINE)

ID	Heat/ Serial No.	Mat'l/ Type	Ultimate Strength	Yield Strength	Elong.	R/A
A[a]	CW2991-2-1	SA376/316	88,500	47,100	56	N/A[b]
B	C8456	SA403/WP316	83,600	40,300	56	N/A
C	CW2991-2-1	SA376/316	88,500	47,100	56	N/A
D	L6CUH	SA403/WP316	88,400	45,200	50	N/A
E	CW2991-2-1	SA376/316	88,500	47,100	56	N/A
F	CW1281-5	SA376/316	88,500	48,600	50	N/A
G	D4722	SA403/WP316	82,200	44,500	54	N/A
H	CW1281-5	SA376/316	88,500	47,100	56	N/A
I	49189	SA182/TP316	79,500	43,000	62	77.5
J	L4328	SA376/316	79.900	38,400	58.3	68.4
K	CW3020-4	SA376/316	83,100	49,300	49	N/A
L	F90746	SA312/TP304	89,100	49,400	55	N/A

[a]: shown in Figures 3-1 and 3-2
[b]: N/A means not available

TABLE 3-2

ROOM TEMPERATURE MECHANICAL PROPERTIES OF
THE RHR LINES MATERIALS (LOOP 4 - RHR LINE)

ID	Heat/ Serial No.	Mat'l/ Type	Ultimate Strength	Yield Strength	Elong.	R/A
A[a]	CW2991-2-1	SA376/TP316	88,500	47,100	56	N/A[b]
B	C8456	SA403/WP316	83,600	40,300	56	N/A
C	CW2991-2-1	SA376/TP316	88,500	47,100	56	N/A
D	L6CWH	SA403/WP316	86,800	45,300	53	N/A
E	CW2991-2-1	SA376/TP316	88,500	47,100	56	N/A
F	1281-14	SA376/TP316	86,500	42,350	56	N/A
G	D-5530	SA403/WP316	77,400	35,600	59.7	N/A
H	CW1281-2	SA376/TP316	85,500	48,900	43	N/A
I	52111	SA403/WP316	80,000	48,000	64	76.5
J	L4328	SA376/TP316	79,900	38,400	58.3	68.4
K	3055-1-4	SA376/TP316	85,000	43,100	56	N/A
L	39745	SA312/TP304	87,100	46,900	56	N/A

[a]: shown in Figures 3-1 and 3-2
[b]: N/A means not available

TABLE 3-3

Room Temperature ASME Code Minimum Properties

Material	Yield Stress (psi)	Ultimate Strength (psi)
A376/TP316	30,000	75,000
A403/WP316	30,000	75,000

TABLE 3-4

AVERAGE AND MINIMUM MECHANICAL PROPERTIES
INCLUDING LOOPS 1 AND 4 AT ROOM TEMPERATURE

	Material	Average[a]	Minimum[a]
Yield Stress	SA376 /TP316	45,625	38,400
	SA403 /WP316	42,743	35,600
Ultimate Strength	SA376 /TP316	86,279	79,900
	SA403 /WP316	83,143	77,400

[a] From Tables 3-1 and 3-2.

TABLE 3-5

REPRESENTATIVE TENSILE PROPERTIES FOR RHR LINES APPROPRIATE FOR WITHOUT AND WITH VALVE LEAKING CASES

	Temperature (°F)	Minimum Yield[a] (psi)	Average Yield[a] (psi)	Minimum Ultimate Strength[a] (psi)	Remark
SA376 /TP316	617	20,647	24,512	61,293	w/o leaking
SA403 /WP316	617	21,178	25,427	63,326	w/o leaking
SA376 /TP316	530[b]	22,506	26,741	61,702	w/ leaking
SA403 /WP316	585[c]	21,880	26,270	63,482	w/ leaking

TABLE 3-6

MODULUS OF ELASTICITY (E)

Temperature	E (ksi)
617°F	25,215
530°F	25,650
585°F	25,375

[a]: From Tables 3-1 and 3-2
[b]: Governing Location No. 4480 as discussed later in section 4.4.2
[c]: Governing Location No. 2152 as discussed later in section 4.4.2

Pipe - 12" Schedule 140
Minimum Wall Thickness: 1.0125 in.
FW - Field Weld
SW - Shop Weld

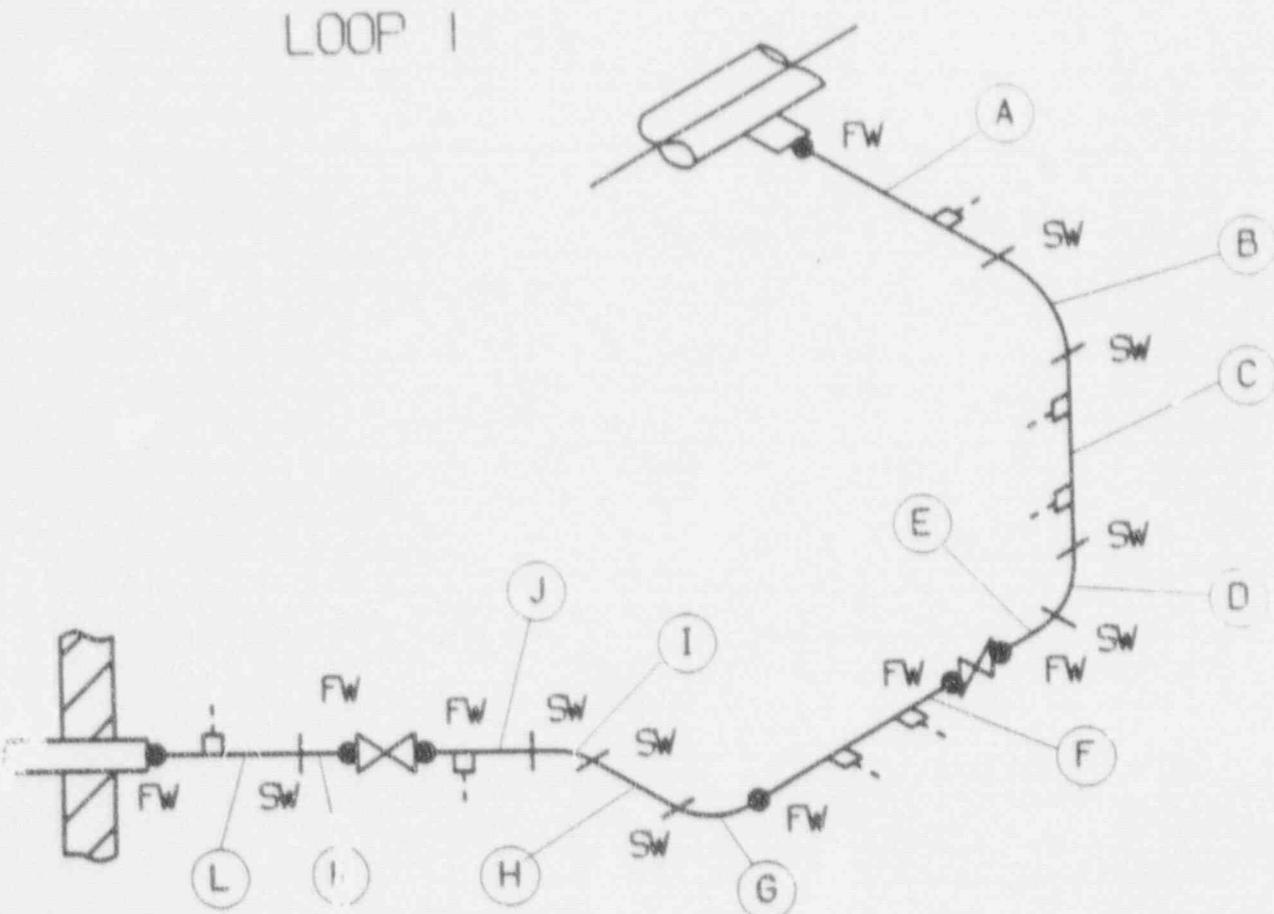

Figure 3-1
Comanche Peak RHR Lines Layout - Loop 1

Pipe - 12" Schedule 140
Minimum Wall Thickness: 1.0125 in.
FW - Field Weld
SW - Shop Weld

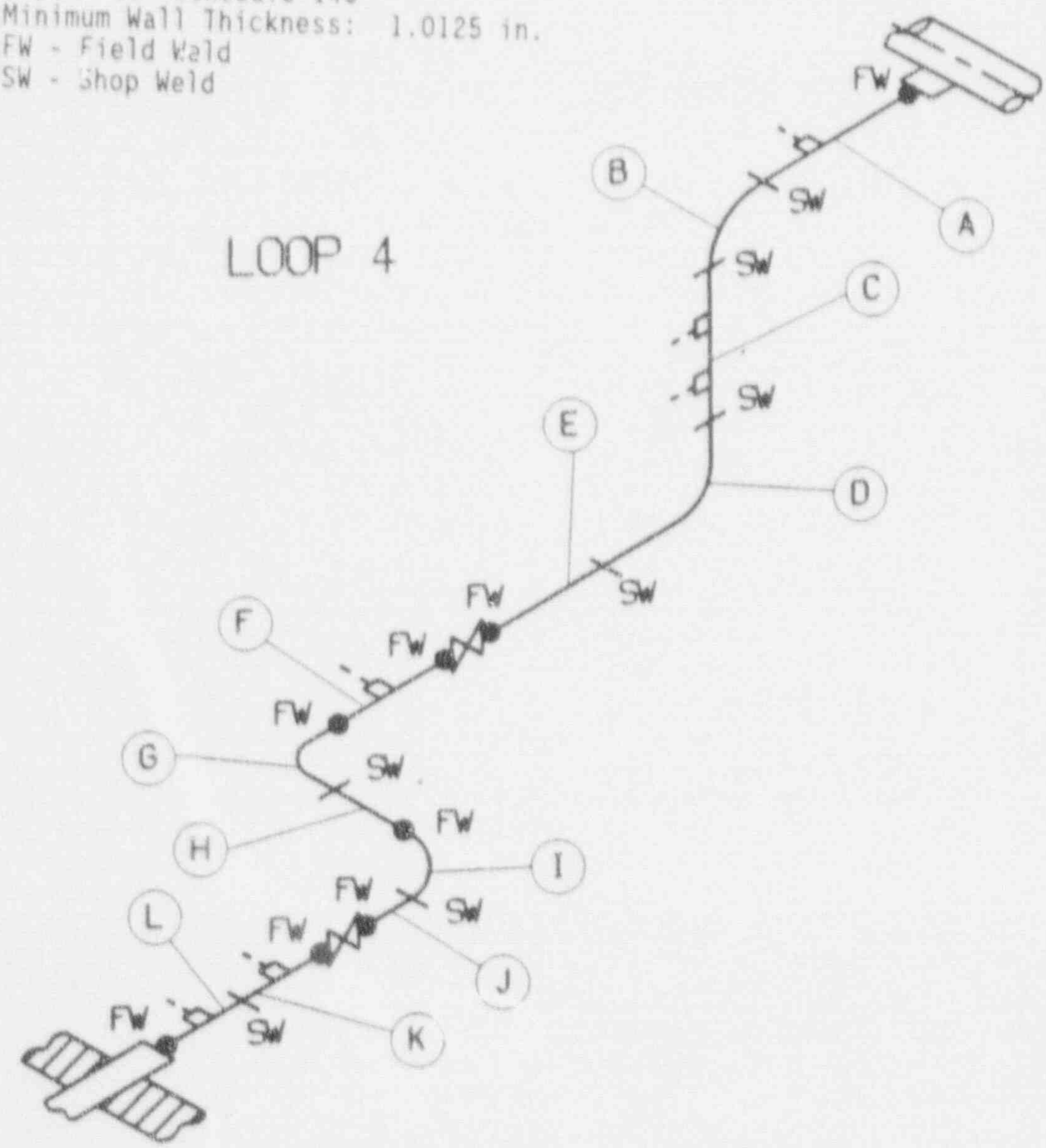

Figure 3-2
Comanche Peak RHR Lines Layout - Loop 4

SECTION 4.0

LOADS FOR FRACTURE MECHANICS ANALYSIS

4.1 Nature of the Loads

Under normal operating conditions, the RHR lines are subjected to axial and bending loads which arise from deadweight, pressure, and thermal expansion. Under faulted conditions, the loads caused by Safe Shutdown Earthquake (SSE) are superimposed on these normal operating loads.

Figures 3-1 and 3-2 show schematic layouts of the RHR lines for the Comanche Peak Unit 2 and identify the weld locations. Diameters and minimum wall thicknesses are shown in Figures 4-1 with governing locations.

The stresses due to these axial loads and bending moments were calculated by the following equation:

$$\sigma = \frac{F}{A} + \frac{M}{Z} \qquad (4\text{-}1)$$

where,

σ	=	stress
F	=	axial load
M	=	bending moment
A	=	metal cross-sectional area
Z	=	section modulus

The bending moments for the desired loading combinations were calculated by the following equation:

$$M_B = (M_Y^2 + M_Z^2)^{0.5} \qquad (4\text{-}2)$$

where,

M_B = bending moment for required loading
M_Y = Y component of bending moment
M_Z = Z component of bending moment

The axial load and bending moments for crack stability analysis and leak rate predictions are calculated by the methods to be explained in Sections 4.2 and 4.3 which follow.

4.2 Loads for Crack Stability Analysis

The faulted loads for the crack stability analysis by the absolute sum method are as follows:

$$F = |F_{DW}| + |F_{TH}| + |FP_{p}| + |F_{SSE}| \qquad (4\text{-}3)$$
$$M_Y = |(M_Y)_{DW}| + |M_{Y\ TH}| + |M_{Y\ SSE}| \qquad (4\text{-}4)$$
$$M_Z = |(M_Z)_{DW}| + |M_{Z\ TH}| + |M_{Z\ SSE}| \qquad (4\text{-}5)$$

DW = Deadweight
TH = Applicable thermal load (normal or stratified)
P = Load due to internal pressure
SSE = SSE loading including seismic anchor motion

The faulted loads for the crack stability analysis by the algebraic sum method are as follows:

$$F = |F_{DW} + F_{TH} + F_{P}| + |F_{SSE}| \qquad (4\text{-}6)$$
$$M_Y = |(M_Y)_{DW} + (M_Y)_{TH}| + |M_{Y\ SSE}| \qquad (4\text{-}7)$$
$$M_Z = |(M_Z)_{DW} + (M_Z)_{TH}| + |M_{Z\ SSE}| \qquad (4\text{-}8)$$

where the subscripts are the same as indicated above.

In this analysis, the absolute sum method was used.

4.3 Loads for Leak Rate Evaluation

The normal operating loads for leak rate predictions were calculated by the algebraic sum method as follows:

$$F = F_{DW} + F_{TH} + FP_{p} \quad (4\text{-}9)$$

$$M_Y = (M_Y)_{DW} + (M_Y)_{TH} \quad (4\text{-}10)$$

$$M_Z = (M_Z)_{DW} + (M_Z)_{TH} \quad (4\text{-}11)$$

The parameters and subscripts are the same as those explained in Section 4.1.

4.4 Loading Conditions for the RHR Lines

The normal operating loadings for the RHR line are pressure (P), deadweight (DW) and normal operating thermal expansion. The faulted loadings consist of normal operating loads plus Safe Shutdown Earthquake (SSE) loads including the Seismic Anchor Motion.

To evaluate the effect of thermal cycling, the loads resulting from thermal cycling were substituted for normal thermal expansion loads as applicable.

4.4.1 Summary of Loads and Geometry for the RHR Lines

The load combinations were evaluated at various weld locations. Normal loads and faulted loads were determined using the algebraic and absolute sum method respectively.

4.4.2 Governing Locations for the RHR Lines

Figures 3-1 and 3-2 show schematic layouts of the RHR lines for Comanche Peak Unit 2 and identify the weld locations. The diameter and minimum wall thickness is shown in Figure 4-1 with the governing locations, which are in the RHR line connected to loop 4.

All the welds at Comanche Peak RHR lines are fabricated using the GTAW, SMAW, and SAW procedures. The following governing locations were established based on the magnitude of total faulted stress for without and with valve leaking conditions. In all cases, the initial root passes are GTAW followed by either a SAW weld or a SMAW weld for the remaining, so that only the SAW and SMAW procedures are analyzed for in this report. As will be shown in Section 5.0, the SAW and SMAW analyses are more conservative than the GTAW analyses.

Crack stability calculations are performed for these two welding procedures and twice the crack length for the 10 GPM leak rate is demonstrated to be stable for SMAW and SAW.

The governing locations without the valve leaking condition are found to be:

SA376/TP316	Node 2030 (SMAW)
SA403/WP403	Node 2142 (SAW)

The governing locations with the valve leaking condition are found to be :

SA376/TP316	Node 4480 (SMAW)
SA403/WP316	Node 2152 (SAW)

The loads and stresses at these governing locations for the normal and the faulted loading combinations for the without and with valve leaking conditions are shown in Table 4-1 and 4-2 respectively. Figure 4-1 shows the governing locations.

TABLE 4-1

SUMMARY OF LBB LOADS AND STRESSES OF
RHR LINES WITHOUT VALVE LEAKING

Node	Case	Axial Force (lb)	Axial Stress (psi)	Bending Moment (in-lb)	Bending Stress (psi)	Total Stress (psi)
2030	Normal	186,921	5,007	618.468	6,087	11,094
	Faulted	209,853	5,621	1,327,137	13,062	18,683
2142	Normal	181,562	4,863	829,339	8,162	13,025
	Faulted	222,184	5,951	1,315,809	12,950	18,901

TABLE 4-2

SUMMARY OF LBB LOADS AND STRESS OF
RHR LINES WITH VALVE LEAKING

a,c,e

Pipe - 12" Schedule 140
Minimum Wall Thickness: 1.0125 in.
FW - Field Weld
SW - Shop Weld

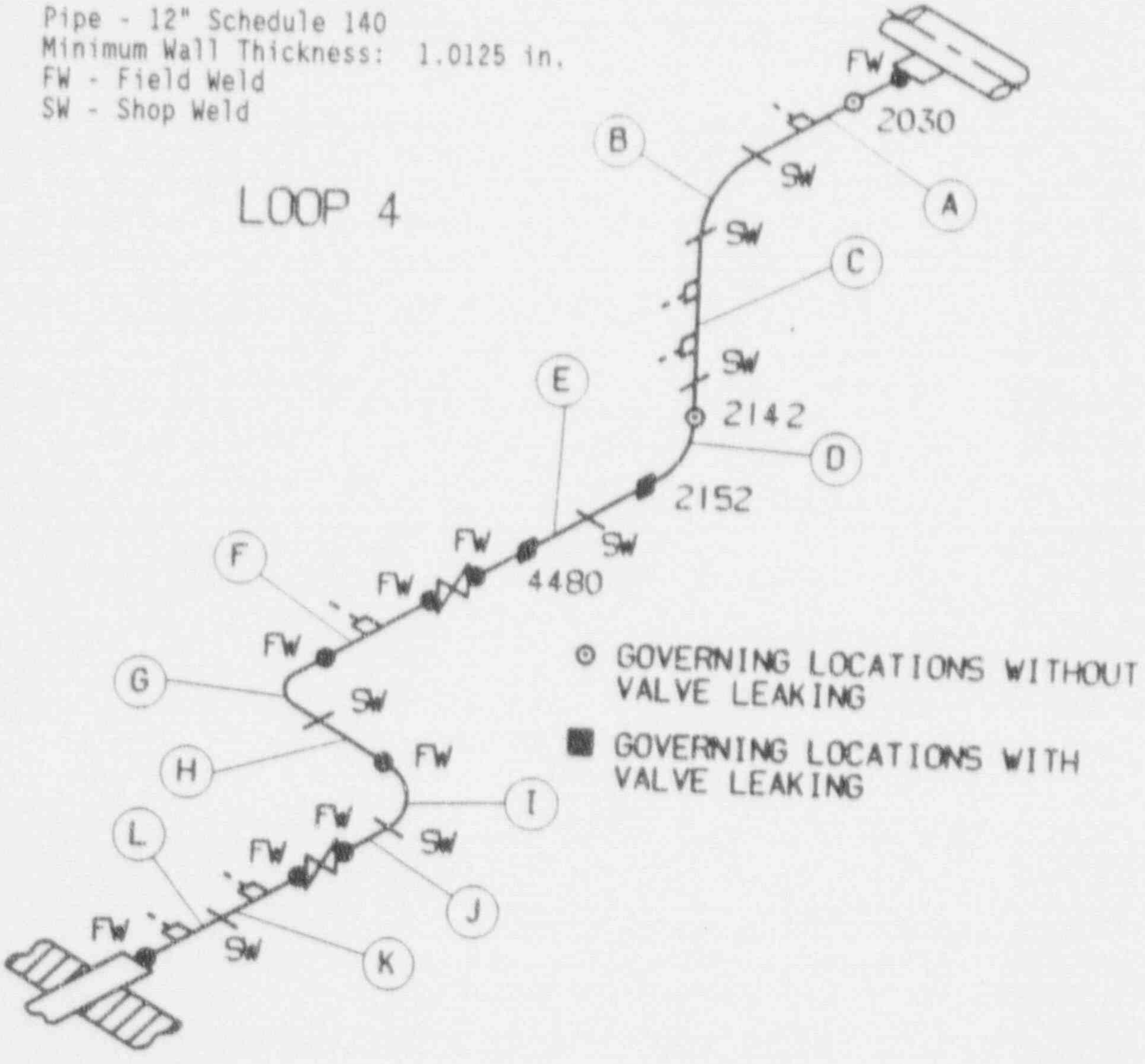

Figure 4-1

Comanche Peak RHR Lines Showing the Governing Locations

SECTION 5.0
FRACTURE MECHANICS EVALUATION

5.1 Global Failure Mechanism

Determination of the conditions which lead to failure in stainless steel should be done with plastic fracture methodology because of the large amount of deformation accompanying fracture. One method for predicting the failure of ductile material is the [ ]^{a,c,e} method, based on traditional plastic limit load concepts, but accounting for [ ]^{a,c,e} and taking into account the presence of a flaw. The flawed component is predicted to fail when the remaining net section reaches a stress level at which a plastic hinge is formed. The stress level at which this occurs is termed as the flow stress. [

.]^{a,c,e}

This methodology has been shown to be applicable to ductile piping through a large number of experiments and is used here to predict the critical flaw size in the RHR lines. The failure criterion has been obtained by requiring equilibrium of the section containing the flaw (Figure 5-1) when loads are applied. The detailed development is provided in Appendix A for a through-wall circumferential flaw in a pipe section with internal pressure, axial force, and imposed bending moments. The limit moment for such a pipe is given by:

[ ]^{a,c,e} (5-1)

where:

[

]^{a,c,e}

[

]^{a,c,e} (5-2)

The analytical model described above accurately accounts for the internal pressure as well as imposed axial force as they affect the limit moment. Good agreement was found between the analytical predictions and the experimental results (reference 5-1). Flaw stability evaluations, using this analytical model, are presented in section 5.3.

5.2 Leak Rate Predictions

Fracture mechanics analysis shows in general that postulated through-wall cracks in the RHR lines would remain stable and do not cause a gross failure of this component. However, if such a through-wall crack did exist, it would be desirable to detect the leakage such that the plant could be brought to a safe shutdown condition. The purpose of this section is to discuss the method which will be used to predict the flow through such a postulated crack and present the leak rate calculation results for through-wall circumferential cracks.

5.2.1 General Considerations

The flow of hot pressurized water through an opening to a lower back pressure (causing choking) is taken into account. For long channels where the ratio of the channel length, L, to hydraulic diameter, D_H, (L/D_H) is greater than [ ]^{a,c,e} both [ ]^{a,c,e} must be considered. In this situation the flow can be described as being single-phase through the channel until the local pressure equals the saturation pressure of the fluid. At this point, the flow begins to flash and choking occurs. Pressure losses due to momentum changes will dominate for [ ]^{a,c,e}. However, for large L/D_H values, the friction pressure drop will become important and must be considered along with the momentum losses due to flashing.

5.2.2 Calculational Method

In using the [

]^{a,c,e}

The flow rate through a crack was calculated in the following manner. Figure 5-2 from reference 5-2 was used to estimate the critical pressure, Pc, for the primary loop enthalpy condition and an assumed flow. Once Pc was found for a given mass flow, the [ ]^{a,c,e} was found from figure 5-3 taken from reference 5-2. For all cases considered, since [ ]^{a,c,e} Therefore, this method will yield the two-phase pressure drop due to momentum effects as illustrated in figure 5-4. Now using the assumed flow rate, G, the frictional pressure drop can be calculated using

$$\Delta P_f = [\qquad]^{a,c,e} \qquad (5\text{-}3)$$

where the friction factor f is determined using the [ ] The crack relative roughness, ε, was obtained from fatigue crack data on stainless steel samples. The relative roughness value used in these calculations was [ ]^{a,c,e} RMS.

The frictional pressure drop using Equation 5-3 is then calculated for the assumed flow and added to the [ ]^{a,c,e} to obtain the total pressure drop from the system under consideration to the atmosphere. Thus,

$$\text{Absolute Pressure} - 14.7 = [\qquad]^{a,c,e} \qquad (5\text{-}4)$$

for a given assumed flow G. If the right-hand side of equation 5-4 does not agree with the pressure difference between the piping under consideration and the atmosphere, then the procedure is repeated until equation 5-4 is satisfied to within an acceptable tolerance and this results in the flow value through the crack.

5.2.3 Leak Rate Calculations

Leak rate calculations were performed as a function of postulated through-wall crack length for the critical locations previously identified. The crack opening area was estimated using the method of reference 5-3 and the leak rates were calculated using the calculational methods described above. The leak rates were calculated using the normal operating loads at the governing locations identified in section 4.0. The crack lengths yielding a leak rate of 10 gpm (10 times the leak detection capability of 1.0 gpm) are shown in Table 5-1 for RHR lines without leaking at the first valve from the primary loop and in Table 5-2 for the RHR lines with leaking at the first valve from the primary loop .

The capabil·ty of each of the pressure boundary leak detection systems are given in Tatle 5.2-9 of the Comanche Peak FSAR.

5.3 Stability Evaluation

A typical segment of the pipe (at the governing location) under maximum loads of axial force F and bending moment M is schematically illustrated as shown in figure 5-5. In order to calculate the critical flaw size, plots of the limit moment ve s crack length were generated as shown in figures 5-6 to 5-9 (as recommended in reference 5-4) discussed below. The critical flaw size corresponds to the intersection of this curve and the maximum load line. In this evaluation the critical flaw size was calculated using the lower bound base metal tensile properties established in section 3.0.

As discussed earlier, the welds at the locations of interest (i.e. the governing locations) are SMAW and SAW. Therefore, "Z" factor corrections for SMAW and SAW welds were applied (references 5-5 and 5-6) as follows:

$$Z = 1.15 [1 + 0.013 (O.D. - 4)] \text{ (for SMAW)} \qquad (5\text{-}5)$$

$$Z = 1.30 [1 + 0.010 (O.D. - 4)] \text{ (for SAW)} \qquad (5\text{-}6)$$

where O.D. is the outer diameter in inches. Substituting O.D. = 14.00 inches, the Z factor was calculated to be 1.30 for SMAW and 1.43 for SAW. The Z factor for GTAW is 1.0. The applied loads were increased by the Z factors. The shop welding (SW) in the RHR lines is SAW and the field welding (FW) is a combination of GTAW and SMAW. Therefore the critical flaw lengths of the RHR lines are obtained using SMAW for FW and SAW for SW and leak-before-break (LBB) is demonstrated for the two welding procedures (SMAW and SAW). The critical flaw size without the valve leaking is given in Table 5-3 and Table 5-4 is the critical flaw size under the valve leaking condition. The plots of limit load versus crack length were generated as shown in Figures 5-6 and 5-7 for the condition of without leaking at the valve and in Figures 5-8 and 5-9 for the condition of with leaking at the valve.

5.4 References

5-1 Kanninen, M. F. et al., "Mechanical Fracture Predictions for Sensitized Stainless Steel Piping with Circumferential Cracks" EPRI NP-192, September 1976.

5-2 [

]^{a,c,e}

5-3 Tada, H., "The Effects of Shell Corrections on Stress Intensity Factors and the Crack Opening Area of Circumferential and a Longitudinal Through-Crack in a Pipe," Section II-1, NUREG/CR-3464, September 1983.

5-4 NRC letter from M. A. Miller to Georgia Power Company, J. P. O'Reilly, dated September 9, 1987.

5-5 ASME Code Section XI, Winter 1985 Addendum, Article IWB-3640.

5-6 Standard Review Plan; Public Comment Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday, August 28, 1987/Notices, pp. 32626-32633.

TABLE 5-1

LEAKAGE FLAW SIZE FOR THE RHR LINES - WITHOUT VALVE LEAKING

a,c,e

TABLE 5-2

LEAKAGE FLAW SIZE FOR THE RHR LINES WITH VALVE LEAKING

a,c,e

TABLE 5-3

SUMMARY OF CRITICAL FLAW SIZE FOR THE RHR LINES - WITHOUT VALVE LEAKING

a,c,e

TABLE 5-4

SUMMARY OF CRITICAL FLAW SIZE FOR THE RHR LINES - WITH VALVE LEAKING

a,c,e

a,c,e

Figure 5-1
Fully Plastic Stress Distribution

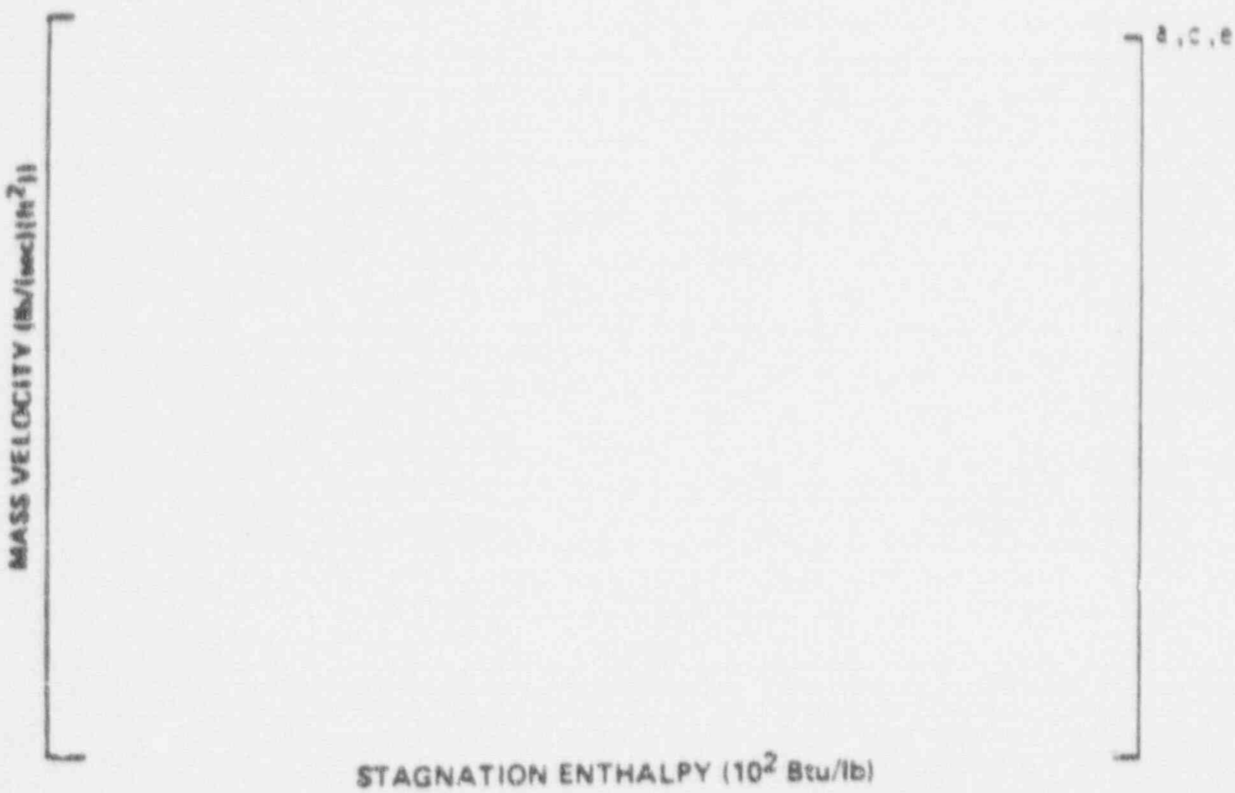

Figure 5-2
Analytical Predictions of Critical Flow Rates of
Steam-Water Mixtures

Figure 5-3
[ ] Pressure Ratio as a Function of L/D

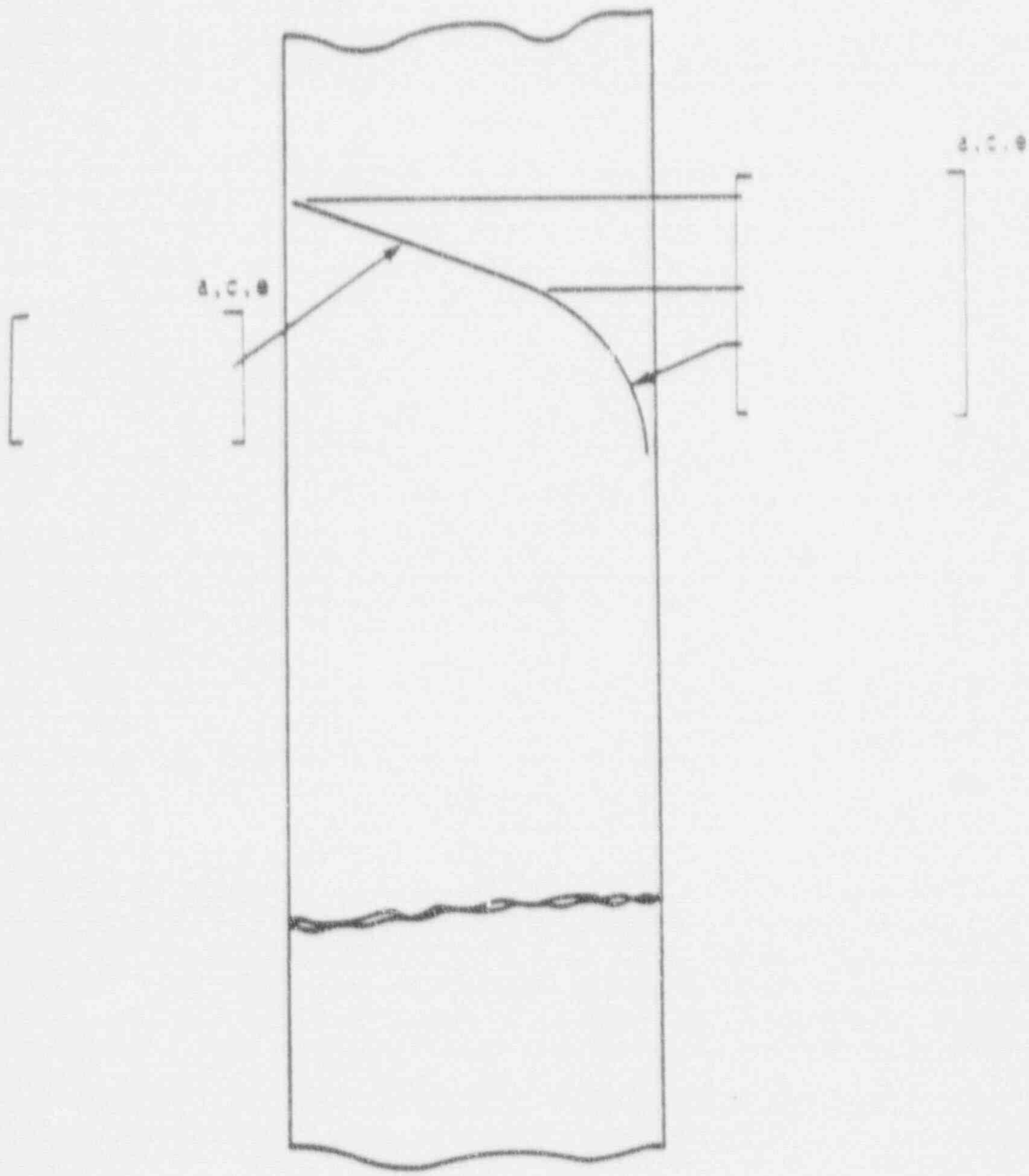

Figure 5-4
Idealized Pressure Drop Profile Through a Postulated Crack

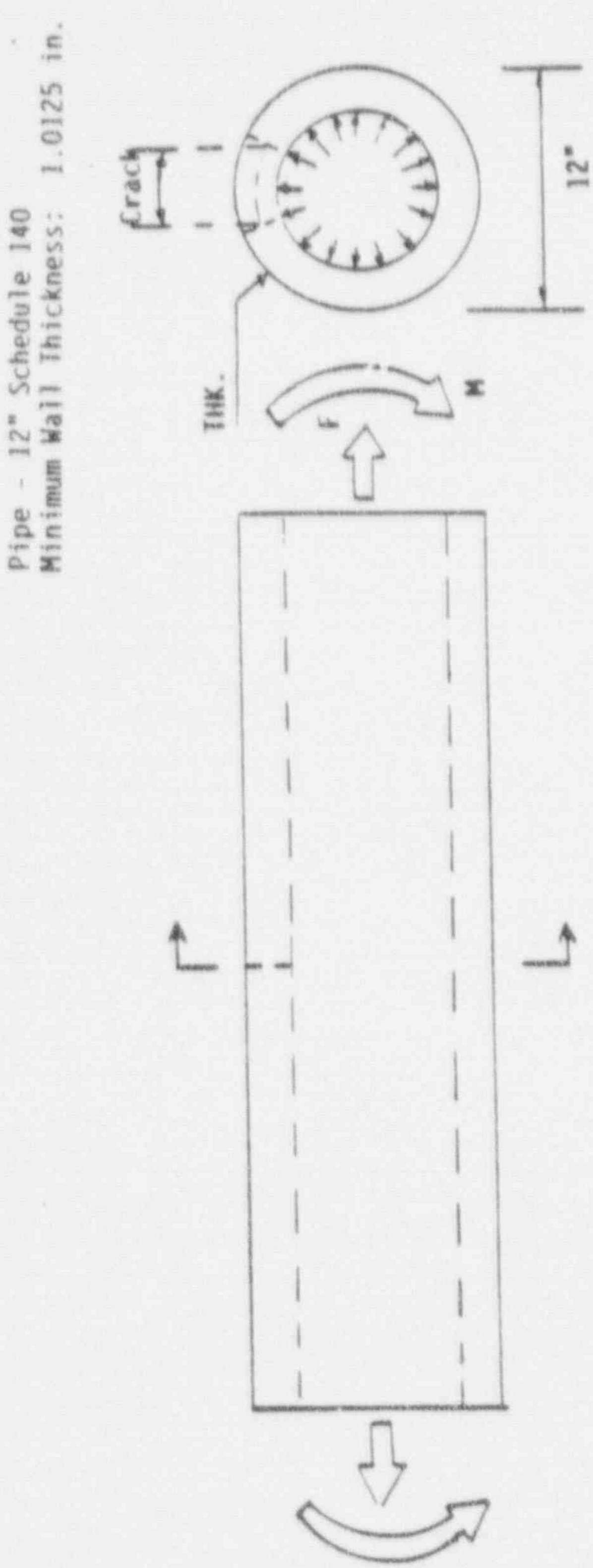

Figure 5-5
Loads Acting on the Model at the Governing Location

a,c,e

Figure 5-6

Critical Flaw Size Prediction for RHR Line
Node 2030 - Without Valve Leaking - SMAW

a,c,e

Figure 5-7

Critical Flaw Size Prediction for RHR Line
Node 2142 - Without Valve leaking - SAW

a,c,e

Figure 5-8

Critical Flaw Size Prediction for RHR Line
Node 2152 - With Valve Leaking - SAW

a,c,e

Figure 5-9

Critical Flaw Size Prediction for RHR Line
Node 4480 - With Valve Leaking - SMAW

SECTION 6.0
ASSESSMENT OF MARGINS FOR THE RHR LINES

In the preceding sections, the leak rate calculations, fracture mechanics analysis were performed. Margins at the critical locations are summarized below:

In Section 5.3 using the IWB-3640 approach (i.e. Z factor approach), the critical flaw sizes at the governing locations are calculated. In Section 5.2 the crack lengths yielding a leak rate of 10 gpm (10 times the leak detection capability of 1.0 gpm) for the critical locations are calculated.

The leakage size flaws, the instability flaws, and margins are shown in Tables 6-1 and 6-2 without and with valve leakage, respectively. Both the margins on flaw size and the margins on loads are seen to be met.

In this evaluation, the leak-before-break methodology is applied conservatively. The conservatisms used in the evaluation are summarized in Table 6-3.

TABLE 6-1

LEAKAGE FLAW SIZES, CRITICAL FLAW SIZES AND
MARGINS FOR RHR LINES- WITHOUT VALVE LEAKING

[ ] a,c,e

WPF1012J/122891:10

TABLE 6-2

LEAKAGE FLAW SIZES, CRITICAL FLAW SIZES AND MARGINS FOR RHR LINES - VALVE LEAKING

a,c,e

WPF1012J/122891:10

TABLE 6-3
LBB CONSERVATISMS

- o Factor of 10 on Leak Rate
- o Factor of 2 on Leakage Flaw for all cases
- o Algebraic Sum of Loads for Leakage
- o Absolute Sum of Loads for Stability
- o Average Material Properties for Leakage
- o Minimum Material Properties for Stability

SECTION 7.0
CONCLUSIONS

This report justifies the elimination of RHR lines pipe breaks as the structural design basis for the Comanche Peak Unit 2 nuclear plant as follows:

a. Stress corrosion cracking is precluded by use of fracture resistant materials in the piping system and controls on reactor coolant chemistry, temperature, pressure, and flow during normal operation.

b. Water hammer should not occur in the RCS piping (primary loop and the attached class 1 RHR lines) because of system design, testing, and operational considerations.

c. The effects of low and high cycle fatigue on the integrity of the RHR lines were evaluated and shown acceptable. The effects of thermal stratification were evaluated separately and shown acceptable.

d. Ample margin exists between the leak rates of small stable flaws and the capability of the Comanche Peak plant's reactor coolant system pressure boundary leakage detection system.

e. Ample margin exists between the small stable flaw sizes of item d and the critical flaw size.

f. With respect to stability of the critical flaw, ample margins exist between the maximum postulated loads and the plant specific maximum faulted loads.

The leakage size flaws will be stable because of the ample margins and will leak at a detectable rate which will assure a safe plant shutdown.

Based on the above, it is concluded that RHR lines pipe breaks should not be considered in the structural design basis of the Comanrhe Peak Unit 2 nuclear plant.

APPENDIX A
LIMIT MOMENT

[

]a,c,e

[ ]^{a,c,e}

Figure A-1

Pipe With A Through-Wall Crack In Bending