Westinghouse Energy Systems

WCAP-13077

TECHNICAL JUSTIFICATION FOR ELIMINATING 4" PRESSURIZER SPRAY LINES RUPTURE AS THE STRUCTURAL DESIGN BASIS FOR THE TROJAN NUCLEAR PLANT

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SECTION 1.0

INTRODUCTION

1.1 Background

The current structural design basis for the 4" pressurizer spray lines 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 these lines and thereby eliminate the need for some of the plant hardware. Presented in this report are the descriptions of a mechanistic pipe break evaluation method and the analytical results that are used for establishing that a circumferential type break will not occur using the methods of leak-before-break analysis. The evaluations considering circumferentially oriented flaws envelop longitudinal cases.

1.2 Scope and Objective

The purpose of this investigation is to demonstrate leak-before-break for the 4" pressurizer spray lines. The scope includes the 4" lines extending between the loop 2 cold leg anchor point and the loop 3 cold leg anchor point. For more detail, schematic drawings of the piping are shown in section 3.0. These comprise the 4" sections of the lines designated as WAB and WAC in the figures of section 3.0. The line above anchor point 13 of Figure 3-1 is not in the scope of this analysis. The recommendations and criteria proposed in NUREG 1061 Volume 3 [1-1]* are used in this evaluation. These criteria and 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.

* Bracketed numbers refer to the references given at the end of the section.

2) Identify the materials and the associated material properties.

3) Postulate a through-wall flaw at the governing location with the least favorable combination of stress and material properties. 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.

4) Using maximum [illegible] demonstrate that there is a margin of at least 2 between th[illegible] and the critical size flaw.

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

6) Justify that the material properties used in the evaluation are representative of the plant specific material. Evaluate long term effects such as thermal aging where applicable.

The flaw stability analysis is performed using the methodology described in SRP 3.6.3 [1-2].

The leak rates are calculated for the normal operating condition loads. 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 [1-3]. 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 Report of the U.S. Nuclear Regulatory Commission Piping Review Committee - Evaluation for Potential for Pipe Breaks, NUREG 1061, Volume 3, November 1984.

1-2 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.

1-3 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 4" PRESSURIZER SPRAY LINES AND THE REACTOR COOLANT SYSTEM

2.1 Stress Corrosion Cracking

The Westinghouse type 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 operating history totals over 450 reactor-years, including five plants each having over 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 type 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, sulphites, 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. 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 pressurizer spray lines since they are designed and operated to preclude the voiding condition in normally filled lines. The RCS and connecting pressurizer spray lines 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 RCS system. Preoperational testing and operating experience have verified the Westinghouse approach. The operating transients of the RCS primary piping and connected pressurizer spray 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.

High cycle fatigue loads in the system would result primarily from pump vibrations during operation. During operation, an alarm signals the exceedance 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. When translated to the connecting pressurizer spray lines, these stresses are even lower, well below the fatigue endurance limit for the pressurizer spray line material and would result in an applied stress intensity factor below the threshold for fatigue crack growth.

2.4 Potential Degradation During Service

Wall thinning by erosion and erosion-corrosion effects will not occur in the 4" pressurizer spray 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.

The Trojan 4" pressurizer spray line nozzles are forged product forms which are not susceptible to toughness degradation due to thermal aging. Finally, the maximum operating temperature of the 4" pressurizer spray line piping, which is about 560°F or below, 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.

SECTION 3.0
MATERIAL CHARACTERIZATION

3.1 Pipe and Weld Materials

The materials of the 4" pressurizer spray lines are A376/TP304 and A403/WP304. The 4" portion of the pressurizer spray line extends between the loop 2 cold leg anchor point and the loop 3 cold leg anchor point. The line above anchor point 13 of Figure 3-1 is not in the scope of this analysis. Note that the 4" pressurizer spray line system does not include any cast pipe or elbows. The welding processes used are shielded metal arc (SMAW) and gas tungsten arc (GTAW).

Weld locations are identified in Figures 3-1 and 3-2.

In the following sections the tensile properties of the materials are presented and criteria for use in the leak-before-break analyses are defined.

3.2 Material Properties

The room temperature mechanical properties of the Trojan Nuclear Power Plant 4" pressurizer spray line materials were obtained from the Certified Materials Test Reports and are provided in Table 3-1. The room temperature ASME Code (Reference 3-1) minimum properties are given in Table 3-2. It is seen that the measured properties well exceed those of the Code. The representative minimum and average tensile properties were established from the Certified Material Test Reports. The material properties at 552°F are required for the leak rate and stability analyses discussed later. The minimum and average tensile properties were calculated by using the ratio of the ASME Section III properties at room temperature and 552°F. Table 3-3 shows the tensile properties at 552°F for the 4" pressurizer spray line materials. The modulus of elasticity value was 25,540 ksi established at 552°F from the ASME BPVC Section III. In the leak-before-break evaluation, the representative minimum properties at temperature are used for the flaw stability evaluations and the representative average properties are used for the leak rate predictions. The minimum ultimate stresses are used for the flaw stability analyses. These properties are summarized in Table 3-3.

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 4" PRESSURIZER SPF AY LINE MATERIALS FOR THE TROJAN NUCLEAR POWER PLANT

ID	Heat No./ Serial No.	Material/Type	Yield Strength (psi)	Ultimate Strength (psi)	Elongation (%)	Area Reduction (%)
1	2P4795	A376/TP304	47,190	82,400	65	N/A*
2	J4MDH2	A403/WP304	32,200	82,300	70	79
3	2P4795	A376/TP304	47,190	82,400	65	N/A
4	2P4795	A376/TP304	47,190	82,400	65	N/A
5	J4MDH1	A376/TP304	32,200	82,300	70	79
6	2P4795	A376/TP304	47,190	82,400	65	N/A
7	J4MDH1	A403/WP304	32,200	82,300	70	79
8	2P4795	A376/TP304	47,190	82,400	65	N/A
9	C1725	A403/WP304	45,727	82,097	75	N/A
10	2P4895	A376/TP304	51,640	85,910	64	N/A
11	C1725	A403/WP304	45,727	82,097	75	N/A
12	2P4895	A376/TP304	51,640	85,910	64	N/A
13	J4MDH2	A403/WP304	32,200	82,300	70	79
14	2P4795	A376/TP304	47,190	87,400	65	N/A
15	2P4795	A376/TP304	47,190	82,400	65	N/A
16	2P4795	A376/TP304	47,190	82,400	65	N/A
17	2P4795	A376/TP304	47,190	82,400	65	N/A
18	S4HB-N	A403/WP304	35,900	81,200	67	78

TABLE 3-1

ROOM TEMPERATURE MECHANICAL PROPERTIES OF THE 4" PRESSURIZER SPRAY LINE MATERIALS FOR THE TROJAN NUCLEAR POWER PLANT

ID	Heat No./ Serial No.	Material/Type	Yield Strength (psi)	Ultimate Strength (psi)	Elongation (%)	Area Reduction (%)
19	2P4795	A376/TP304	47,190	82,400	65	N/A
20	C1772	A403/WP304	45,727	82,097	75	N/A
21	2P4795	A376/TP304	47,190	82,400	65	N/A
22	S4JZ-J	A403/WP304	35,500	82,900	65	N/A
23	2P4795	A376/TP304	47,190	82,400	65	N/A
24	C2491	A403/WP304	34,225	81,924	71	N/A
25	2P4795	A376/TP304	47,190	82,400	65	N/A
26	C1725	A403/WP304	45,727	82,097	75	N/A
27	2P4795	A376/TP304	47,190	82,400	65	N/A
28	C1725	A403/WP304	45,727	82,097	75	N/A
29	2P4795	A376/TP304	47,190	82,400	65	N/A
30	2P4795	A376/TP304	47,190	82,400	65	N/A
31	2P4795	A376/TP304	47,190	82,400	65	N/A
32	C1725	A403/WP304	45,727	82,097	75	N/A
33	2P4795	A376/TP304	47,190	82,400	65	N/A
34	J4MDH2	A403/WP304	32,200	82,300	70	79
35	2P4795	A376/TP304	47,190	82,400	65	N/A
36	J4MDH1	A403/WP304	32,200	82,300	70	79
37	2P4795	A376/TP304	47,190	82,400	65	N/A

TABLE 3-1

ROOM TEMPERATURE MECHANICAL PROPERTIES OF THE 4" PRESSURIZER SPRAY LINE MATERIALS FOR THE TROJAN NUCLEAR POWER PLANT

ID	Heat No./ Serial No.	Material/Type	Yield Strength (psi)	Ultimate Strength (psi)	Elongation (%)	Area Reduction (%)
38	2P4795	A376/TP304	47,190	82,400	65	N/A
39	2P4795	A376/TP304	47,190	82,400	65	N/A
40	C1725	A403/WP304	45,727	87,097	75	N/A
41	2P4795	A376/TP304	47,190	82,400	65	N/A
42	J4MDH1	A403/WP304	32,200	82,300	70	79
43	2P4795	A376/TP304	47,190	82,400	65	N/A
44	J4MDH2	A403/WP304	32,200	82,300	70	79
45	2P4795	A376/TP304	47,190	82,400	65	N/A
46	J4MDH	A403/WP304	32,200	82,300	70	79
47	2P4501	A376/TP304	48,480	83,650	70	N/A
48	2P4501	A376/TP304	48,480	83,650	70	N/A
49	2P4501	A376/TP304	48,480	83,650	70	N/A
50	J4MDH2	A403/WP304	32,200	82,300	70	79
51	2P4795	A376/TP304	47,190	82,400	65	N/A
52	2P4501	A376/TP304	48,480	83,650	70	N/A
53	J4MDH1	A403/WP304	32,200	82,300	70	79
54	2P4501	A376/TP304	48,480	83,650	70	N/A
55	2P4795	A376/TP304	47,190	82,400	65	N/A
56	J4MDH1	A403/WP304	32,200	82,300	70	79

TABLE 3-1

ROOM TEMPERATURE MECHANICAL PROPERTIES OF THE 4" PRESSURIZER SPRAY LINE MATERIALS FOR THE TROJAN NUCLEAR POWER PLANT

ID	Heat No./ Serial No.	Material/Type	Yield Strength (psi)	Ultimate Strength (psi)	Elongation (%)	Area Reduction (%)
57	2P4501	A376/TP304	48,480	82,650	70	N/A
58	J4MDH1	A403/WP304	32,200	82,300	70	79
59	2P4501	A376/TP304	48,480	83,650	70	N/A
60	JBQP	A403/WP304	33,568	79,299	71	N/A
61	2P4501	A376/TP304	48,480	83,650	70	N/A
62	JBQP	A403/WP304	33,568	79,299	71	N/A
63	2P4501	A376/TP304	48,480	83,650	47	N/A
64	J4MDH1	A403/WP304	32,200	82,300	70	79
65	2P4895	A376/TP304	47,750	85,860	64	N/A

*Not available

TABLE 3-2

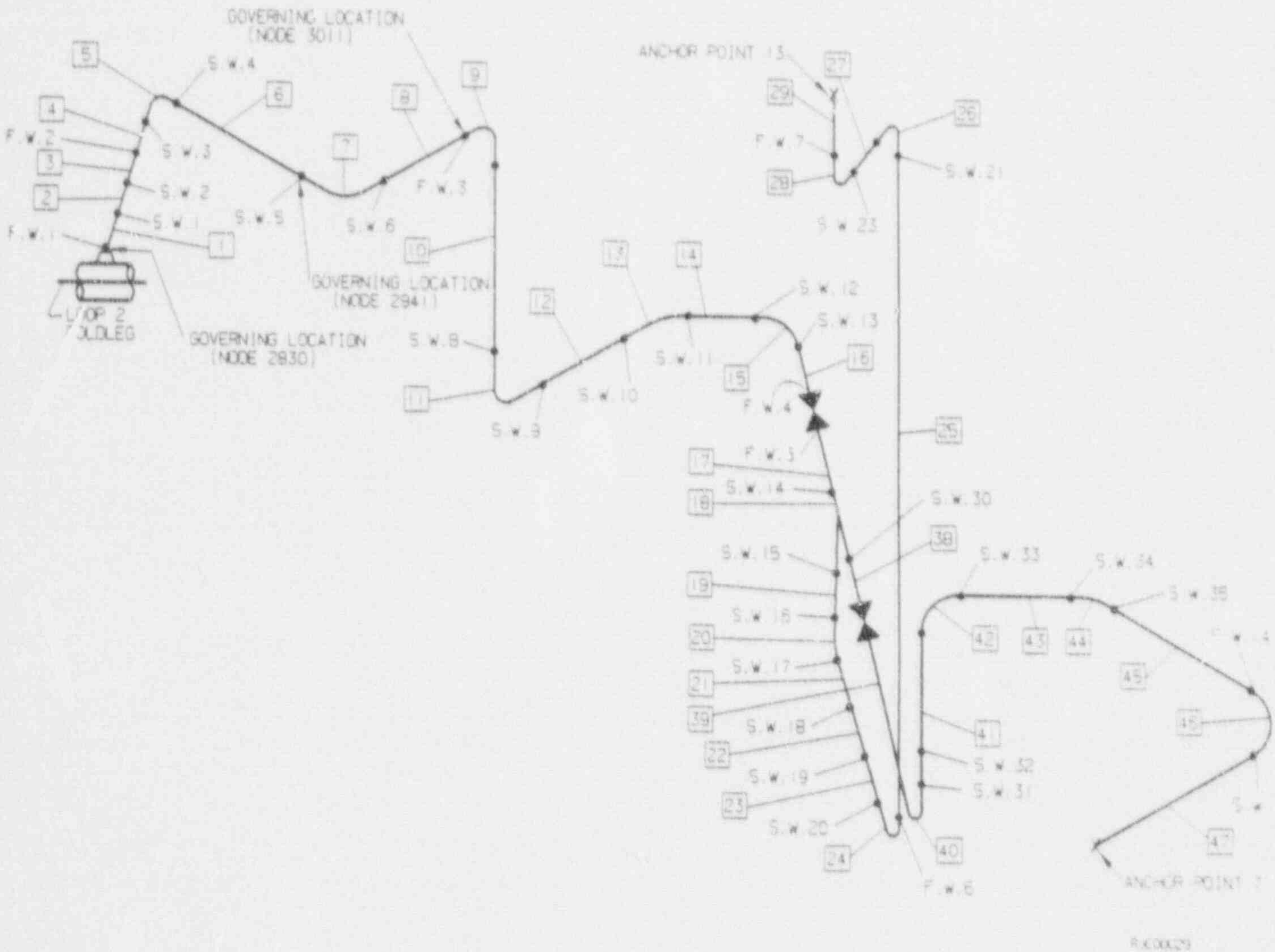
ROOM TEMPERATURE ASME CODE MINIMUM PROPERTIES

Material	Yield Stress (psi)	Ultimate Stress (psi)
A376/TP304	30,000	75,000
A403/WP304	30,000	75,000

TABLE 3-3

TENSILE PROPERTIES FOR THE TROJAN NUCLEAR POWER PLANT 4" PRESSURIZER SPRAY LINES AT 552°F

Material	Minimum Yield (psi)	Average Yield (psi)	Minimum Ultimate (psi)
A376/TP304	28,640	29,598	69,765
A403/WP304	20,158	22,823	67,140

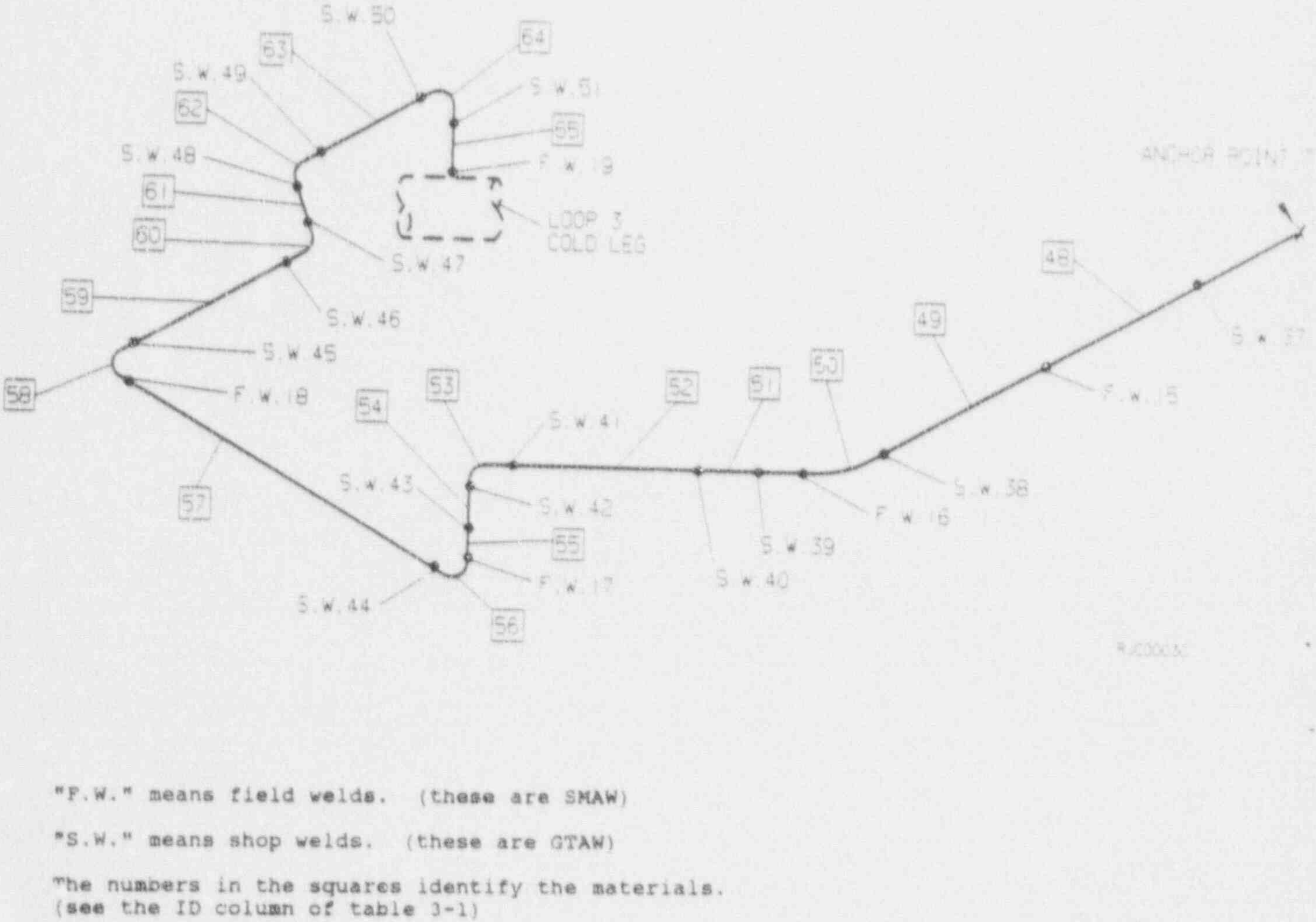


"F.W." means field welds. (these are SMAW)

"S.W." means shop welds. (these are GTAW)

The numbers in the squares identify the materials.
(see the ID column of table 3-1)

Figure 3-1. Layout of Section WAB of the 4" Pressurizer spray lines



"F.W." means field welds. (these are SMAW)

"S.W." means shop welds. (these are GTAW)

The numbers in the squares identify the materials.
(see the ID column of table 3-1)

Figure 3-2. Layout of Section WAC of the 4" Auxiliary Spray Lines

SECTION 4
LOADS FOR FRACTURE MECHANICS ANALYSIS

Figures 3-1 and 3-2 show the schematic layouts of the 4" pressurizer spray lines and identify the weld locations considered for the fracture mechanics analysis.

The stresses due to 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:

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

where,

M	=	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 were computed by the methods explained in Sections 4.1 and 4.2 which follow.

4.1 Loads for Crack Stability Analysis

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

$$F = |F_{DW}| + |F_{TH}| + |F_P| + |F_{SSE}| \quad (4\text{-}3)$$

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

$$M_Z = |(M_Z)_{DW}| + |(M_Z)_{TH}| + |(M_Z)_{SSE}| \quad (4\text{-}5)$$

Where, the subscripts of the above equations represent the following loading cases,

DW = deadweight
TH = normal thermal expansion
SSE = SSE loading including seismic anchor motion
P = load due to internal pressure

4.2 Loads for Leak Rate Evaluation

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

$$F = F_{DW} + F_{TH} + F_P \quad (4\text{-}6)$$

$$M_Y = (M_Y)_{DW} + (M_Y)_{TH} \quad (4\text{-}7)$$

$$M_Z = (M_Z)_{DW} + (M_Z)_{TH} \quad (4\text{-}8)$$

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

4.3 Summary of Loads and Geometry

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

4.4 Governing Locations

The governing locations were established on the basis of the strengths of the materials and the highest faulted stresses for the following weld types: GTAW weld locations in the A376/TP304 material and GTAW weld locations in the A403/WP304 material, as well as SMAW weld locations in the A376/TP304 material and SMAW weld locations in the A403/WP304 material. Both of the lines WAB and WAC were investigated and the following governing locations were established:

A376/TP304 Material

Nodes 2830 and 3011 (SMAW) and node 2941 (GTAW), Line WAB

A403/WP304 Material

Nodes 3011 (SMAW) and node 2941 (GTAW). Line WAB

The loads and stresses for the governing locations are shown in Table 4-1.

The governing locations have been indicated in the layout sketch of Figure 3-1.

TABLE 4-1

SUMMARY OF LBB LOADS AND STRESSES AT GOVERNING LOCATIONS

Node & Line	Load Case	Axial Force (lbs)	Axial Stress (psi)	Bending Moment (in-lbs)	Bending Stress (psi)	Total Stress (psi)
2830/WAB	Normal	21028	3481	36856	6695	10176
	Faulted	21644	3583	69789	12678	16261
2941/WAB	Normal	20154	3337	47963	8713	12050
	Faulted	22625	3746	71283	12949	16695
3011/WAB	Normal	20239	3350	33720	6126	9476
	Faulted	22290	3690	46861	8513	12472

SECTION 5.0
FRACTURE MECHANICS EVALUATION

5.1 Failure Mechanism

Determination of the conditions which lead to failure 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 [strain hardening]^{a,c,e} and taking into account the presence of a flaw. The flawed pipe 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 called 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 sizes in the 4" pressurizer spray 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 with internal pressure, axial force, and imposed bending moments. The limit moment for such a pipe is given by:

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

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

where:

[ ]^{a,c,e}

[

]a,c,e

The analytical model described above accurately accounts for the piping internal pressure as well as imposed axial torce 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

The purpose of this section is to discuss the method which will be used to predict the flow through a postulated crack and present the leak rate calculation results for postulated through-wall circumferential cracks in the pressurizer spray lines.

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 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, friction pressure drop will become important and must be considered along with the momentum losses due to flashing.

5.2.2 Calculation Method

Using an [

]^{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

$$\Delta P_f = \left[\qquad \right]^{a,c,e} \tag{5-3}$$

where the friction factor f is determined using the [ ]^{a,c,e} 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 momentum pressure drop calculated using the Fauske model to obtain the total pressure drop from the primary system to the atmosphere. Thus,

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

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 Calculation

Leak rate calculations were made as a function of postulated through-wall crack length for the three critical locations previously identified. The crack opening areas were estimated using the method of reference 5-3 and the leak rates were calculated using the calculational method described above. The leak rates were calculated using the normal operating loads at the governing nodes 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) for the governing locations at the Trojan pressurizer spray lines are shown in Table 5-1.

5.2.4 Leak Detection Capability

The Trojan Nuclear Power Plant leak detection system inside the containment can detect 1 gpm leak rates as required by Regulatory Guide 1.45. As seen above, a margin of 10 was applied to the leak rate to define the 4" pressurizer spray line leakage size flaws in accordance with NUREG 1061, Volume 3.

5.3 Stability Evaluation

A typical segment of a pipe 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 versus crack length are generated as shown in figures 5-6 to 5-10. Whenever the governing location lies between two different materials, two plots are provided, one for each of the materials. The critical flaw size corresponds to the intersection of this curve and the maximum load line. The critical flaw sizes are calculated using the lower bound base metal tensile properties established in section 3.0.

The "Z" factor correction for the SMAW welds was applied (references 5-4 and 5-5) as follows:

$$Z = 1.15 \left[1 + 0.013 \left(O.D. - 4\right)\right] \text{ (for SMAW)} \qquad (5\text{-}6)$$

where OD is the outer diameter in inches. Substituting OD - 4.5 inches, the Z factor was calculated to be 1.16 for SMAW welds. The Z factor for GTAW welds is 1.0. For SMAW welds, the applied loads at the SMAW locations were increased by 1.16 to generate the plots of limit load versus crack length. Table 5-2 shows the summary of critical flaw sizes for the Trojan nuclear power plant pressurizer spray lines.

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

5-5 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

LEAK RATE CRACK LENGTHS FOR THE 4" PRESSURIZER SPRAY LINES

Node Point	Material Size	Crack Length (in.) (for 10 gpm leakage)
[		] a,c,e
[		] a,c,e
[		] a,c,e

TABLE 5-2

SUMMARY OF CRITICAL FLAW SIZES

FOR THE TROJAN 4" PRESSURIZER SPRAY LINES

Node Point/Weld Type	Material Side	Trojan 4" Pressurizer Spray Critical Flaw Size (in)
		a,c,e
		a,c,e
		a,c,e

a, c, e

Figure 5-1. Fully Plastic Stress Distribution

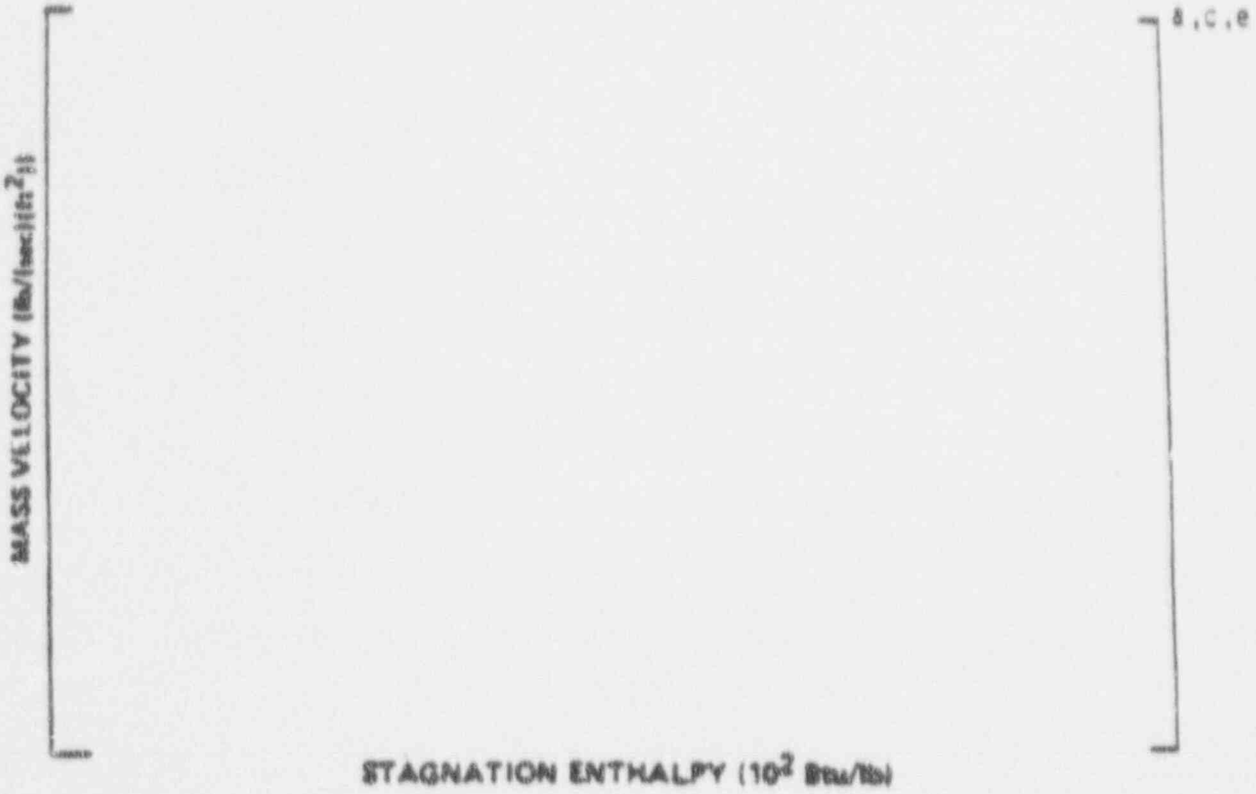


Figure 5-2. Analytical Predictions of Critical Flow Rates of Steam-Water Mixtures

Figure 5-3. [ ]a,c,e Pressure Ratio as a Function of L/D

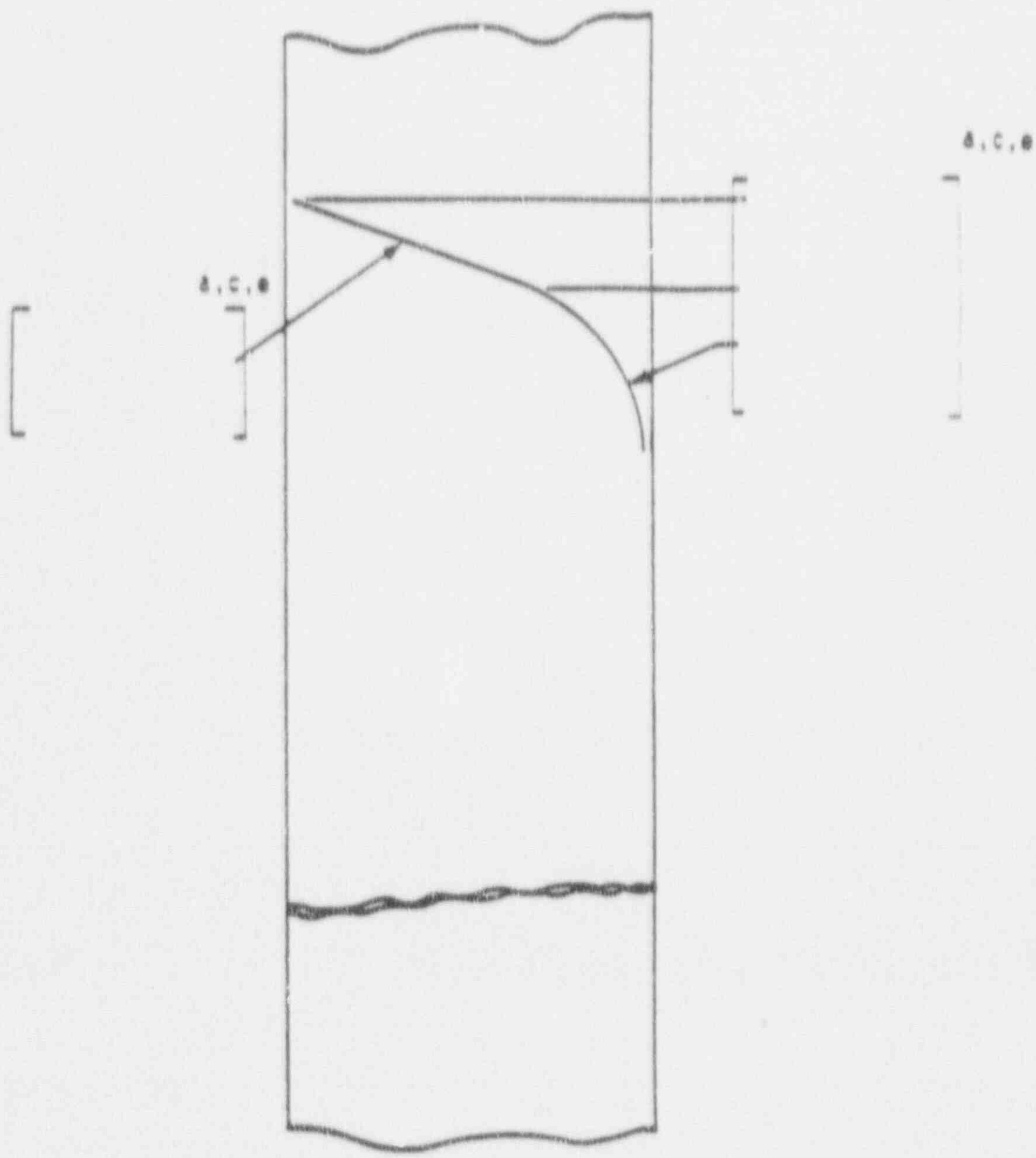


Figure 5-4. Idealized Pressure Drop Profile Through a Postulated Crack

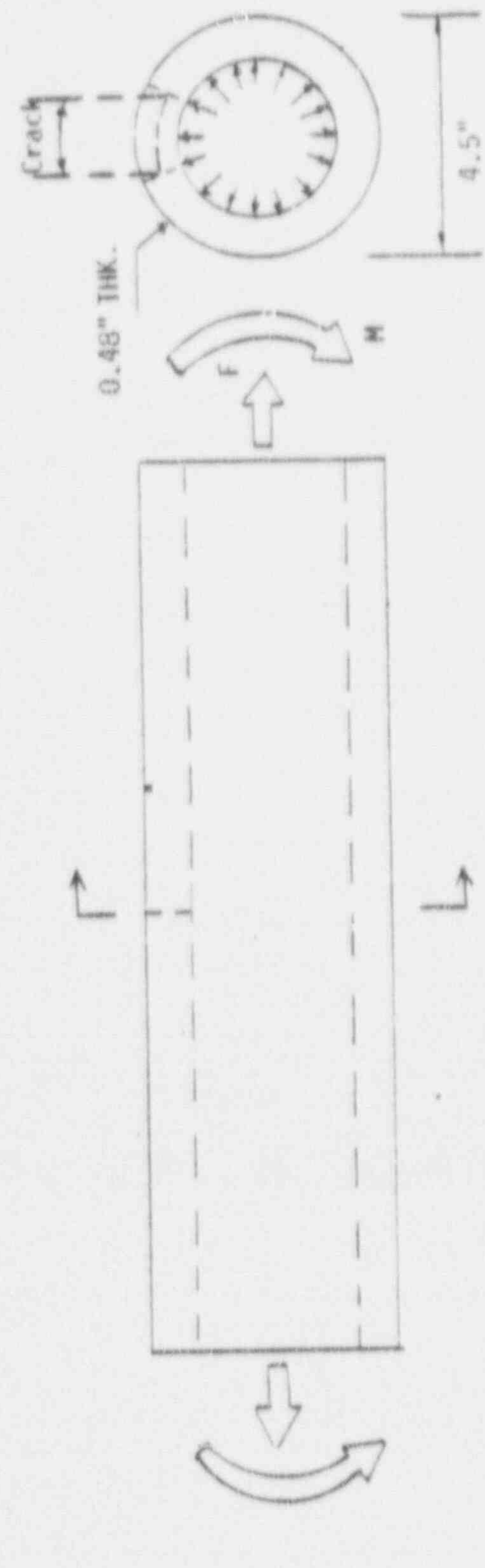


Figure 5-5. Loads Acting on the Model at a Governing Location

a,c,e

Figure 5-6. Critical Flaw Size Prediction for the Trojan Nuclear Power Plant (Node 2830 Line WAB). Material is A376/TP304 on both sides of the node.

a,c,e

Figure 5-7. Critical Flaw Size Predict on for the Trojan Nuclear Power Plant (Node 2941 Line WAB). Material this side is A376/TP304.

a,c,e

Figure 5-8. Critical Flaw Size Prediction for the Trojan Nuclear Power Plant (Node 2941 Line WAB). Material this side is A403/WP304.

a,c,e

Figure 5-9. Critical Flaw Size Prediction for the Trojan Nuclear Power Plant (Node 3011 Line WAB). Material this side is A376/TP304.

a,c,e

Figure 5-10. Critical Flaw Size Prediction for the Trojan Nuclear Power Plant (Node 3011 Line WAB). Material this side is A403/WP304.

SECTION 6.0
ASSESSMENT OF MARGINS

In the preceding sections, the leak rate calculations and fracture mechanics analyses were performed. Margins at the critical locations are summarized in Table 6-1. The table shows that [

$]^{a,c,e}$

In summary, relative to

1. Flaw Size

   A margin of about 2 exists between the critical flaws and the flaws yielding a leak a rate of 10 gpm.

2. Leak Rate

   For the reference flaw sizes a margin of 10 exists between the calculated leak rate and the 1 gpm leak detection criteria of Regulatory Guide 1.45.

In the evaluation, the leak-before-break methodology is applied conservatively. The conservatisms used in the evaluation are summarized in Table 6-2.

TABLE 6-1

LEAKAGE FLAW SIZES, CRITICAL FLAW SIZES, AND MARGINS

Node	Material Side	Critical Flaw Size (in.)	Leakage Flaw Size (in.)	Margin

a,c,e

a,c,e

TABLE 6-2

LBB CONSERVATISMS

- Factor of 10 on Leak Rate
- Factor of 2 on Leakage Flaw
- Algebraic Sum of Loads for Leakage
- Absolute Sum of Loads for Stability
- Average Material Strengths for Leakage[a]
- Minimum Material Strengths for Stability

---

[a]with exception noted

SECTION 7.0
CONCLUSIONS

This report justifies the elimination of pipe breaks from the structural design basis of the portion of the 4" pressurizer spray line located below anchor point 13 at the Trojan Nuclear Power Plant as follows:

a. Stress corrosion cracking is precluded by use of fracture resistant materials in the pipe 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 auxiliary lines) because of system design, testing, and operational considerations.

c. The effects of low and high cycle fatigue on the integrity of the 4" pressurizer spray lines piping are negligible.

d. Adequate margin exists between the leak rate of small stable flaws and the capability of the Trojan plant's reactor coolant system pressure boundary leakage detection system.

e. Adequate margin exists between the small stable flaw sizes of item d and the critical flaws.

The postulated reference flaws will be stable because of the margins in d and e and will leak at detectable rates which will assure a safe plant shutdown.

Based on the above, it is concluded that pipe breaks in the 4" pressurizer spray lines need not be considered in the structural design basis of the Trojan Nuclear Power Plant section below anchor point 13.

APPENDIX A
LIMIT MOMENT

]a,c,e

a,c,e

Figure A-1. Pipe with a Through-Wall Crack in Bending