ENCLOSURE 8

WESTINGHOUSE ELECTRIC CORPORATION, LLC.
WCAP-15380 (NON-PROPRIETARY)

TECHNICAL JUSTIFICATION FOR ELIMINATING PRESSURIZER SURGE LINE RUPTURE AS THE STRUCTURAL DESIGN BASIS FOR PRAIRIE ISLAND UNIT 2 NUCLEAR PLANT

46 pages follow

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WCAP-15380

Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for Prairie Island Unit 2 Nuclear Plant

Westinghouse Electric Company LLC

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1 INTRODUCTION

1.1 BACKGROUND

The current structural design basis for the pressurizer surge line 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 surge line. 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 pressurizer surge line. The evaluations considering circumferentially oriented flaws cover longitudinal cases. The pressurizer surge line is known to be subjected to thermal stratification and the effects of thermal stratification for Prairie Island Unit 2 surge Line have been evaluated and documented in WCAP-12639 (Reference 1-2) and WCAP-12639, Supplement 1 (Reference 1-3). The results of the stratification evaluation as described in WCAP-12639 and WCAP-12639, Supplement 1 have been used in the leak-before-break evaluation presented in this report.

1.2 SCOPE AND OBJECTIVE

The purpose of this investigation is to demonstrate leak-before-break for the Prairie Island Unit 2 pressurizer surge line. The scope of this work covers the entire pressurizer surge line from the primary loop nozzle junction to the pressurizer nozzle junction. Schematic drawing of the piping systems is shown in Section 3.0. The recommendations and criteria proposed in SRP 3.6.3 (Reference 1-4) 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 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 types used in the Plant provide representative material properties.

7. Perform an assessment of fatigue crack growth. Show that a through-wall crack will not result.

The leak rate is 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 (Reference 1-5). 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-7211, Revision 3, "Energy Systems Business Unit Policy and Procedures for Management, Classification, and Release of Information," March 1994.

1-2 WCAP-12639, "Westinghouse Owner's Group Pressurizer Surge Line Thermal Stratification Generic Detailed Analysis Program MUHP-1091 Summary Report ," June 1990.

1-3 WCAP-12639, Supplement 1, "Westinghouse Owner's Group Additional Information on Pressurizer Surge Line Stratification Detailed Analysis," November 1990.

1-4 Standard Review Plan; public comments solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday, February 28, 1987/Notices, pp. 32626-32633.

1-5 NUREG/CR-3464, 1983, "The Application of Fracture Proof Design Methods Using Tearing Instability Theory to Nuclear Piping Postulating Circumferential Through Wall Cracks."

2 OPERATION AND STABILITY OF THE PRESSURIZER SURGE LINE AND THE REACTOR COOLANT SYSTEM

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, IGSCC). This operating history totals over 900 reactor-years, including five Plants each having over 20 years of operation and 15 other Plants each with over 15 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 Plant." 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 Plant 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, sulfides, 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 Line 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 surge Line since they are designed and operated to preclude the voiding condition in normally filled Line. The RCS and connecting surge 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 and connecting surge Line. Preoperational testing and operating experience have verified the Westinghouse approach. The operating transients of the RCS primary piping and connected surge Line 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. A further assessment of the low cycle fatigue loading is discussed in Section 6.0 as part of this study in the form of a fatigue crack growth evaluation.

Pump vibrations during operation would result in high cycle fatigue loads in the piping system. 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 in 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. Field measurements on typical PWR Plant indicate vibration amplitudes less than 1 ksi. When translated to the connecting surge line, these stresses would be even lower, well below the fatigue endurance limit for the surge line material and would result in an applied stress intensity factor below the threshold for fatigue crack growth.

2.4 SUMMARY EVALUATION OF SURGE LINE FOR POTENTIAL DEGRADATION DURING SERVICE

There has never been any service cracking or wall thinning identified in the pressurizer surge Line of Westinghouse PWR design. Sources of such degradation are mitigated by the design, construction, inspection, and operation of the pressurizer surge piping.

There is no mechanism for water hammer in the pressurizer/surge system. The pressurizer safety and relief piping system which is connected to the top of the pressurizer could have loading from water hammer events. However, these loads are effectively mitigated by the pressurizer and have a negligible effect on the surge line.

Wall thinning by erosion and erosion-corrosion effects will not occur in the surge line due to the low velocity, typically less than 1.0 ft/sec and the material, austenitic stainless steel, which is highly resistant to these degradation mechanisms. Per

NUREG-0691, a study of pipe cracking in PWR piping, only two incidents of wall thinning in stainless steel pipe were reported and these were not in the surge line. Although it is not clear from the report, the cause of the wall thinning was related to the high water velocity and is therefore clearly not a mechanism which would affect the surge line.

It is well known that the pressurizer surge Line are subjected to thermal stratification and the effects of stratification are particularly significant during certain modes of heatup and cooldown operation. The effects of stratification have been evaluated for the Prairie Island Unit 2 surge Line and the loads, accounting for the stratification effects, have been derived in WCAP-12639 and WCAP-12639, Supplement 1(References 1-2 and 1-3). These loads are used in the leak-before-break evaluation described in this report.

The Prairie Island Unit 2 Nuclear Plant surge line piping and associated fittings are forged product forms (see Section 3) which are not susceptible to toughness degradation due to thermal aging.

Finally, the maximum operating temperature of the pressurizer surge piping, which is about 650°F, 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 Plant, NUREG-0531, U.S. Nuclear Regulatory Commission, March1979.

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

3 MATERIAL CHARACTERIZATION

3.1 PIPE, ELBOW MATERIALS AND WELDING PROCESS

The pipe material of the pressurizer surge line for the Prairie Island Unit 2 Nuclear Plant is A376/TP316. The elbow fitting material of the pressurizer surge line for the Prairie Island Unit 2 Nuclear Plant is A-403 WP-316. These are wrought product forms of the type used for the primary loop piping of several PWR Plant. The surge line is connected to the primary loop nozzle at one end and the other end of the surge line is connected to the pressurizer nozzle. The surge line system does not include any cast pipes or cast fittings. The welding processes used are Gas Tungten Arc Weld(GTAW) and Shielded Metal Arc Weld(SMAW). Figure 3-1 shows the schematic layout of the surge line and identify the weld locations by node points.

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

3.2 MATERIAL PROPERTIES

Prairie Island Unit 2 Plant specific data was used as a basis for determining tensile properties. The room temperature mechanical properties of the surge line material were obtained from the Certified Materials Test Reports and are given in Table 3-1. The representative minimum and average tensile properties were established (see Table 3-2). The material properties at temperatures (455°F and 653°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 Code Section III (Reference 3-1) properties at the temperatures of interest stated above. Table 3-2 shows the tensile properties at various temperatures. The modulus of elasticity values were established at various temperatures from the ASME Code Section III (see Table 3-3). In the leak-before-break evaluation, the representative minimum properties at temperature were used for the flaw stability evaluations and the representative average properties were used for the leak rate predictions. The minimum ultimate stresses were used for stability analyses. These properties are summarized in Table 3-2.

3.3 REFERENCES

3-1 ASME Boiler and Pressure Vessel Code Section III, "Rules for Construction of Nuclear Power Plant Components; Business Unit 2, Appendices", 1989 Edition, July 1, 1989.

Table 3-1 Room Temperature Mechanical Properties of the Pressurizer Surge Line Materials

Heat #	Material	Yield Strength (psi)	Ultimate Strength (psi)
54691 Serial # LR2504	A403 WP-316	37,500	81,700
54691 Serial # LR2505	A403 WP-316	37,500	81,700
52644 Serial # LR2503	A403 WP-316	37,800	84,300
54691 Serial # LR2506	A403 WP-316	37,500	81,700
54691 Serial # LR2302	A403 WP-316	37,500	81,700
54691 Serial # LR2507	A403 WP-316	37,500	81,700
J2471 Serial # CR1049	A403 WP-316	41,600	84,400
J3183 Serial # 8317 and 8314	A376/TP316	40,000	85,900
J3183 Serial # 8317 and 8314	A376/TP316	38,900	84,900
J3183 Serial # 8317 and 8314	A376/TP316	39,900	85,100
J2009 Serial # A5794	A376/TP316	41,900	86,400
J2009 Serial # A5794	A376/TP316	37,500	81,400

Table 3-2 Representative Tensile Properties				
Material	**Temperature (°F)**	**Minimum Yield (psi)**	**Average Yield (psi)**	**Minimum Ultimate (psi)**
A376/TP316 and A-403 WP-316	Room	37,500	38,758	81,400
	455	25,719	26,581	77,927
	653	23,095	23,869	77,927

Table 3-3 Modulus of Elasticity (E)	
Temperature (°F)	**E (ksi)**
Room	28,300
455	26,115
653	25,035

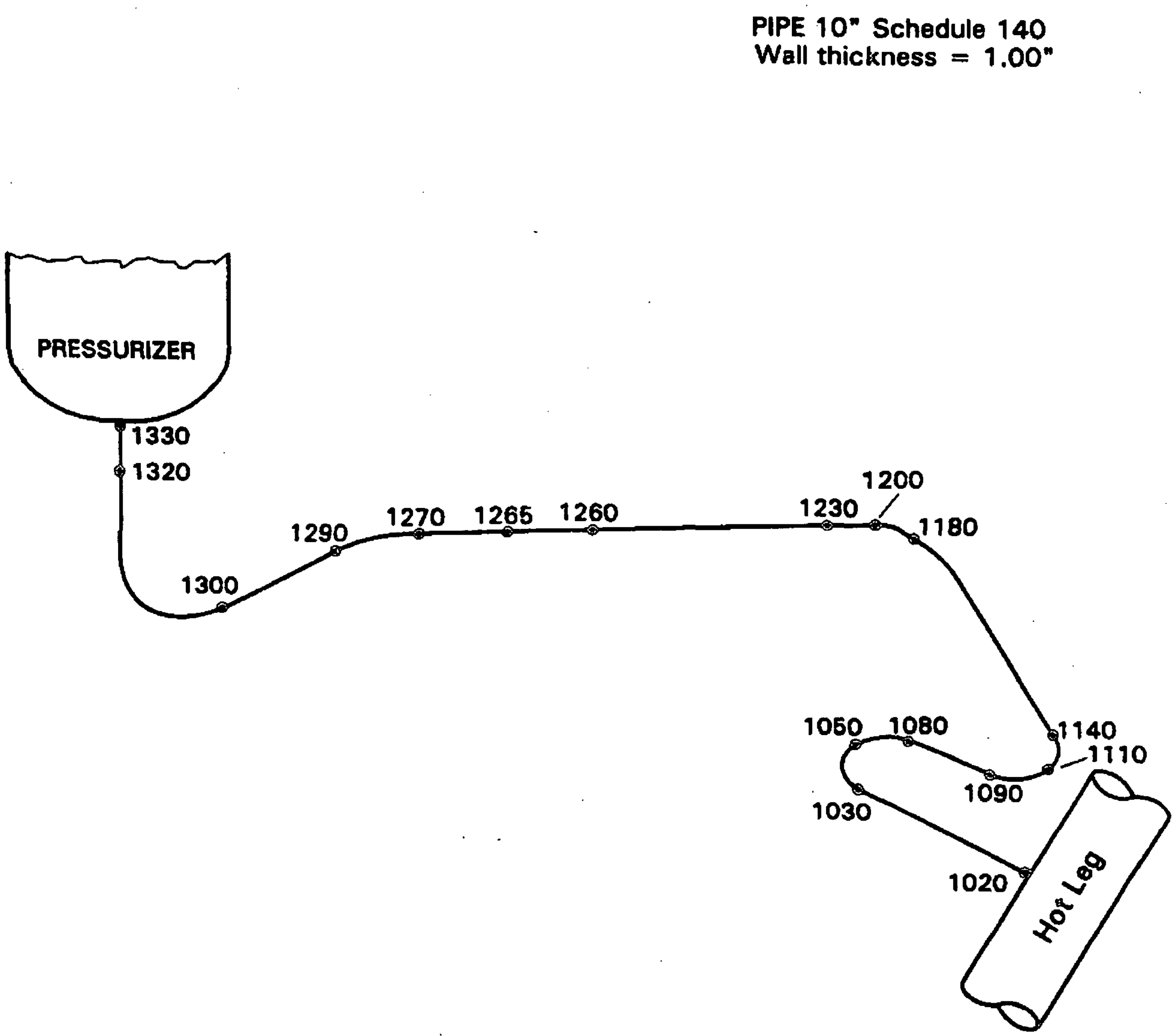


Figure 3-1 Prairie Island Unit 2 Surge Line Layout

4 LOADS FOR FRACTURE MECHANICS ANALYSIS

4.1 NATURE OF THE LOADS

Figure 3-1 shows schematic layout of the surge line for Prairie Island Unit 2 and identifies the weld locations.

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

$$\sigma = \frac{F}{A} + \frac{M}{Z} \tag{4-1}$$

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:

$$M_B = \left(M_Y^2 + M_Z^2\right)^{0.5} \tag{4-2}$$

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 computed 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 were calculated by the absolute sum method as follows:

$$F = |F_{DW}| + |F_{TH}| + |F_P| + |F_{SSE}| \tag{4-3}$$

$$M_Y = |M_{YDW}| + |M_{Y\,TH}| + |M_{Y\,SSE}| \tag{4-4}$$

$$M_Z = |M_{ZDW}| + |M_{Z\,TH}| + |M_{Z\,SSE}| \tag{4-5}$$

where

DW = Deadweight

TH = Applicable thermal load (normal or stratified)

P = Load due to internal pressure

SSE = SSE loading including seismic anchor motion

4.3 LOADS FOR LEAK RATE EVALUATION

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

$$F = F_{DW} + F_{TH} + F_P \tag{4-6}$$

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

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

The parameters and subscripts are the same as those explained in Sections 4.1 and 4.2.

4.4 LOADING CONDITIONS

Because thermal stratification can cause large stresses at heatup and cooldown temperatures in the range of 455°F, a review of stresses was used to identify the worst situations for LBB applications. The loading states so identified are given in Table 4-1.

Seven loading cases were identified for LBB evaluation as given in Table 4-2. Cases A, B, C are cases for leak rate calculations with the remaining cases being the corresponding faulted situations for stability evaluations.

The cases postulated for leak-before-break are summarized in Table 4-3. The cases of primary interest are the postulation of a detectable leak at normal power conditions

[

]a,c,e

The combination [

]a,c,e

The more realistic cases [

]a,c,e

[

]a,c,e

4.5 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.

4.6 GOVERNING LOCATION

All the welds at Prairie Island Unit 2 surge Line are fabricated using the GTAW and SMAW procedure. Node 1320 is the governing location, when the stress levels and the weld procedures are both taken into account for all the locations of Prairie Island Unit 2 pressurizer surge Line. Figure 4-1 shows the governing location. The loads and stresses at the governing location for all the loading combinations are shown in Tables 4-4.

Table 4-1 Types of Loadings
Pressure (P)
Dead Weight (DW)
Normal Operating Thermal Expansion (TH)
Safe Shutdown Earthquake and Seismic Anchor Motion (SSE)[1]
[ ]a,c,e
[ ]a,c,e
[ ]a,c,e

[1] SSE is used to refer to the absolute sum of these loadings.

Table 4-2 Normal and Faulted Loading Cases for Leak-Before-Break Evaluations	
CASE A	This is the normal operating case at 653°F consisting of the algebraic sum of the loading components due to P, DW and TH.
CASE B	[ ] a,c,e
CASE C	[ ] a,c,e
CASE D	This is the faulted operating case at 653°F consisting of the absolute sum (every component load is taken as positive) of P, DW, TH and SSE.
CASE E	[ ] a,c,e
CASE F	[ ] a,c,e
CASE G	[ ] a,c,e

Table 4-3 Associated Load Cases for Analyses	
A/D	This is heretofore standard leak-before-break evaluation.
A/F	[ ] a,c,e
B/E	[ ] a,c,e
B/F	[ ] a,c,e
B/G[1]	[ ] a,c,e
C/G[1]	[ ] a,c,e

[1] These are judged to be low probability events.

Table 4-4 Summary of LBB Loads and Stresses by Case for Governing Location						
Node	Case	F_X(lbs)	S_X(psi)	M_B(in-lb)	S_B(psi)	S_T(psi)
1320	A	132,521	4,330	298,625	4,360	8,690
1320	B	132,383	4,330	228,098	3,330	7,660
1320	C	21,470	700	1199,059	17,520	18,220
1320	D	137,572	4,500	414,954	6,060	10,560
1320	E	137,710	4,500	344,427	5,030	9,530
1320	F	30,384	990	1199,059	17,520	18,510
1320	G	31,547	1,030	1315,388	19,220	20,250

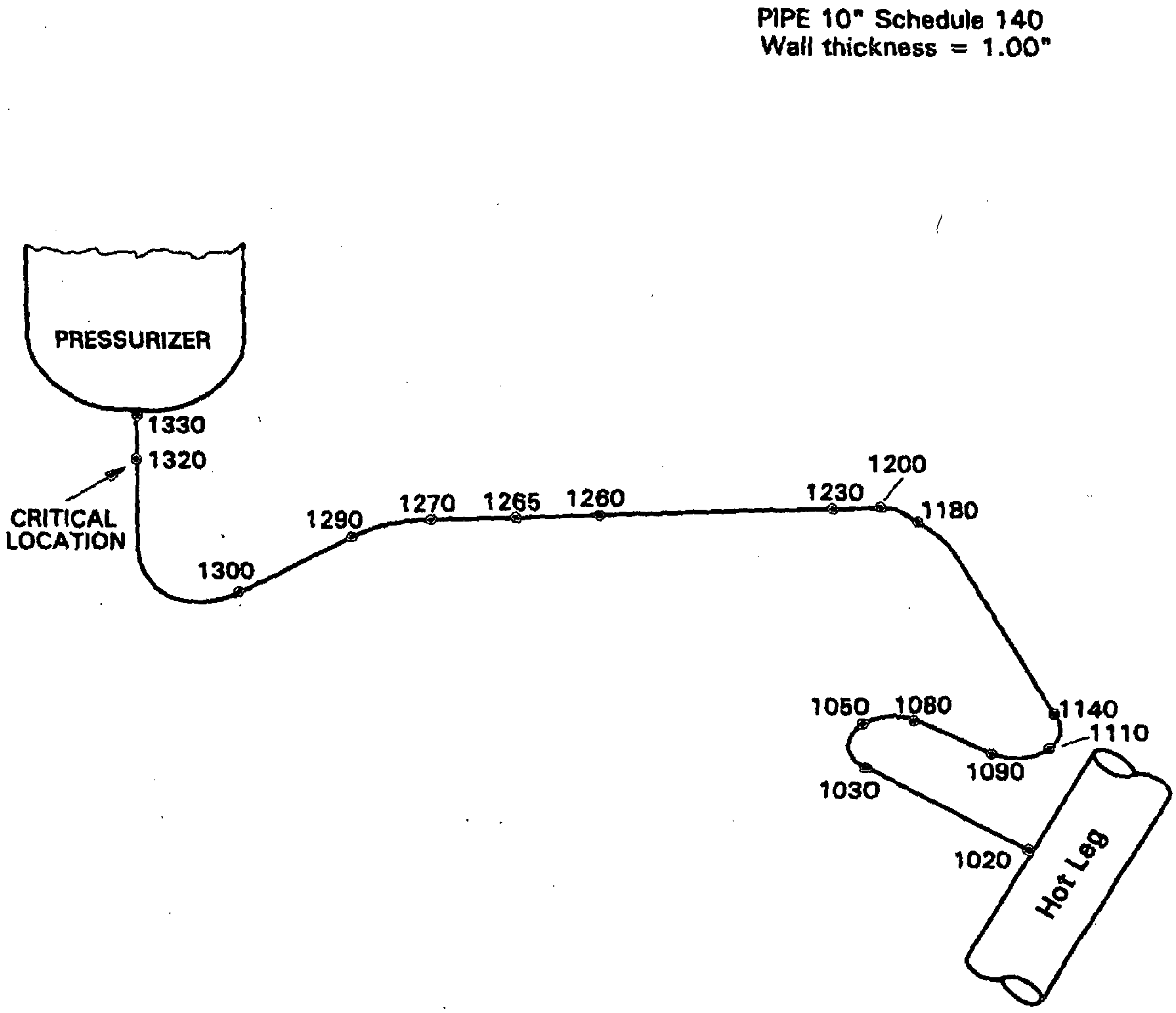


Figure 4-1 Prairie Island Unit 2 Surge Line Showing Governing Location

5 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 pressurizer surge line. 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} (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 that postulated through-wall cracks in the surge line would remain stable and would 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

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

where the friction factor f was 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 was 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,

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

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 Calculations

Leak rate calculations were performed as a function of postulated through-wall crack length for the critical location 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 location identified in Section 4.0. The crack lengths yielding a leak rate of 2 gpm (10 times the leak detection capability of 0.20 gpm) for critical location at the Prairie Island Unit 2 Nuclear Plant pressurizer surge Line are shown in Table 5-1.

The Prairie Island Plants RCS pressure boundary leak detection system has capability of detecting smaller than 0.2 gpm in one hour (Reference 5-4).

5.3 STABILITY EVALUATION

A typical segment of the pipe under maximum loads of axial force F and bending moment M is schematically illustrated 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 5-9. The critical flaw size corresponds to the intersection of this curve and the maximum load line. The critical flaw size is calculated using the lower bound base metal tensile properties established in Section 3.0.

The welds at the governing location are GTAW and SMAW. Therefore, the "Z" factor correction for the SMAW weld was applied (Reference 5-5) as follows:

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

where OD is the outer diameter in inches. Substituting OD = 10.75 inches, the Z factor was calculated to be 1.25 for SMAW. The applied loads were increased by the Z factors and the plots of limit load versus crack length were generated as shown in Figure 5-6 to 5-9. Table 5-2 shows the summary of critical flaw sizes.

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 Report to the United States Nuclear Regulatory Commission Division of Operating Reactors, Docket 50-282, 50-306, License No. DPR-42 and DPR-60, Coolant Leakage Detection System, Performance at the Prairie Island Nuclear Generating Plant. Dated March 31, 1976.

5-5 Standard Review Plan; Public Comment Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday, March28, 1987/Notices, pp. 32626-32633.

Table 5-1 Leakage Flaw Size			
Node Point	Load Case	Temperature (°F)	Leakage Flaw Size (in.) (for 2 gpm leakage)
1320	[		]a,c,e
1320	[		]a,c,e
1320	[		]a,c,e

Table 5-2 Summary of Critical Flaw Size			
Node Point	Load Case	Temperature (°F)	Critical Flaw Size (in)
1320	[		]a,c,e
1320	[		]a,c,e
1320	[		]a,c,e
1320	[		]a,c,e

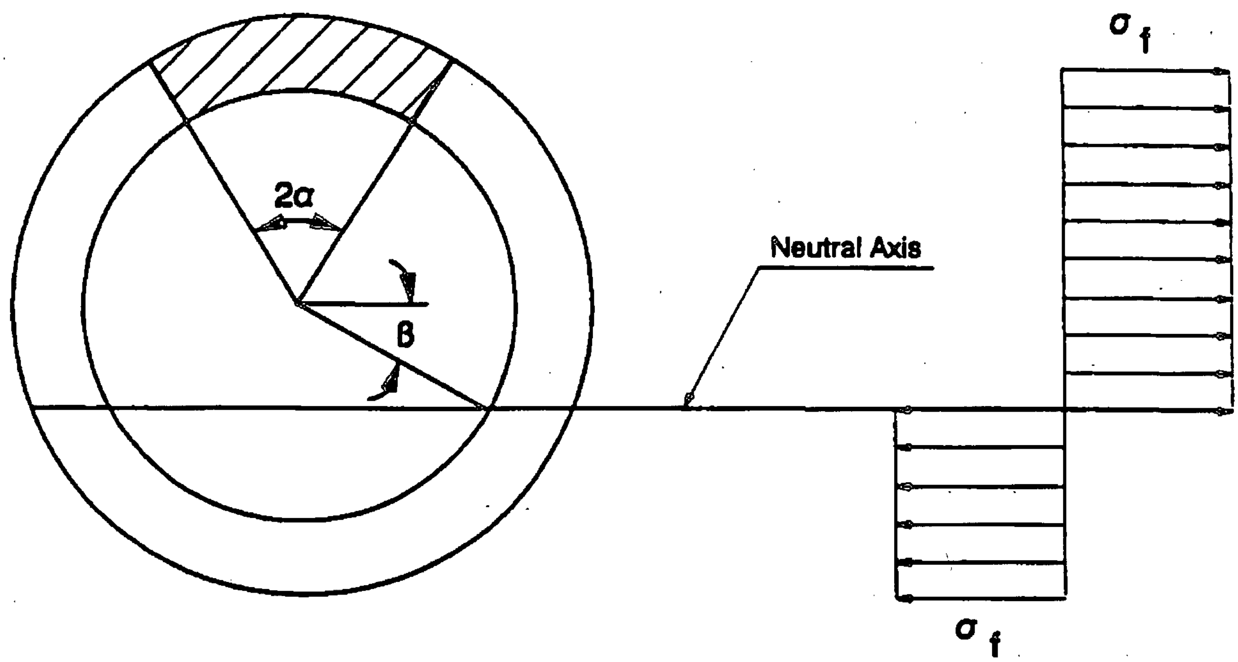


Figure 5-1 Fully Plastic Stress Distribution

a,c,e

MASS VELOCITY (lb/(sec)(ft^2))

STAGNATION ENTHALPY (10^2 Btu/lb)

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

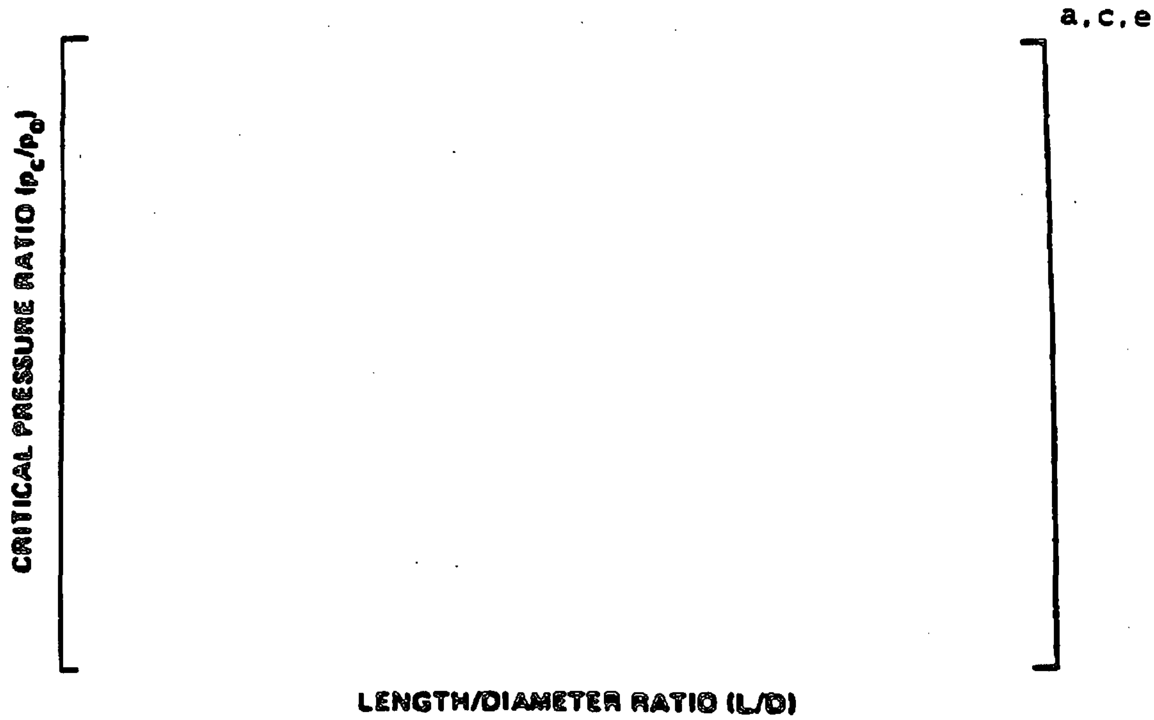


Figure 5-3 [ ]a,c,e Pressure Ratio as a Function of L/D

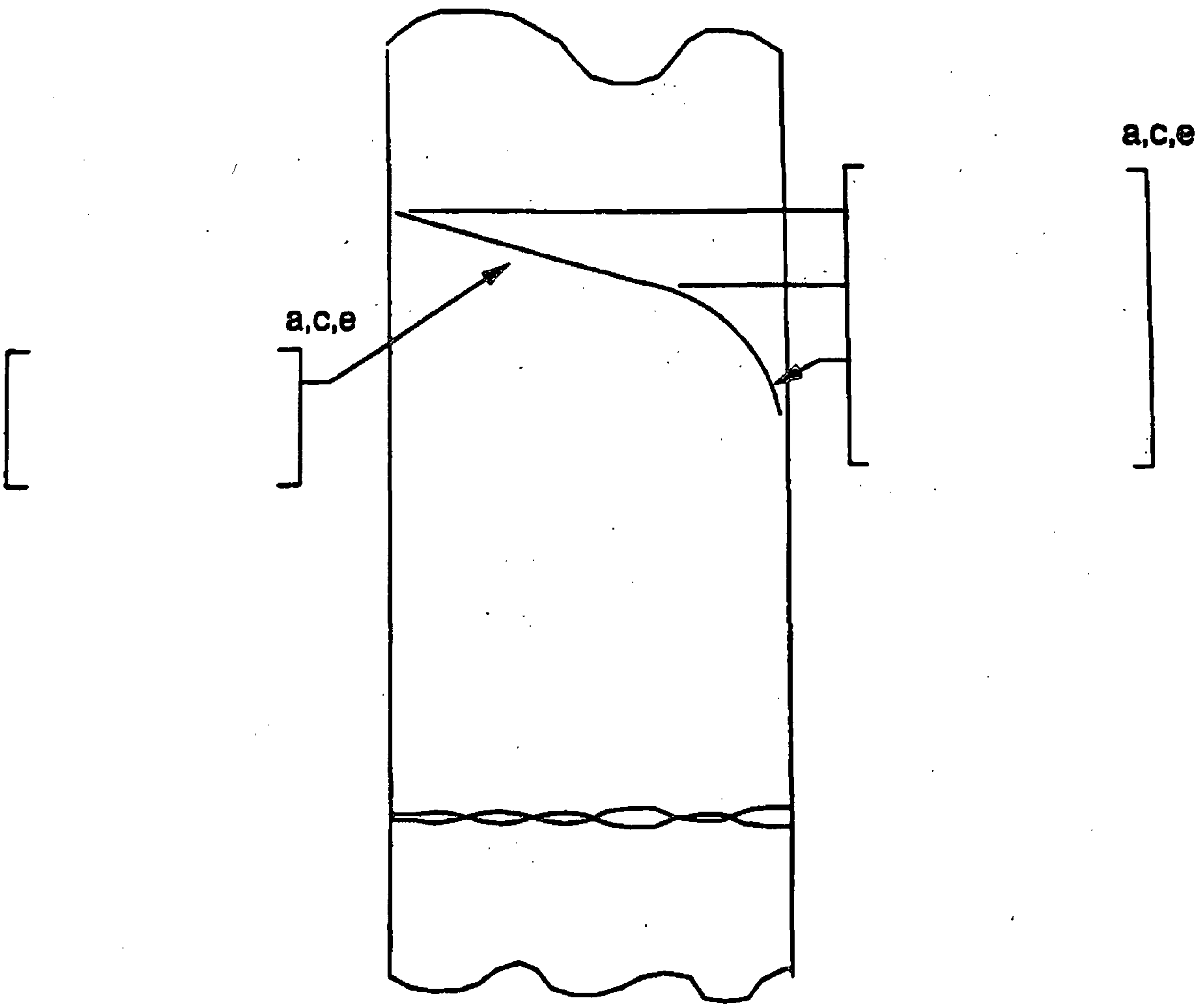
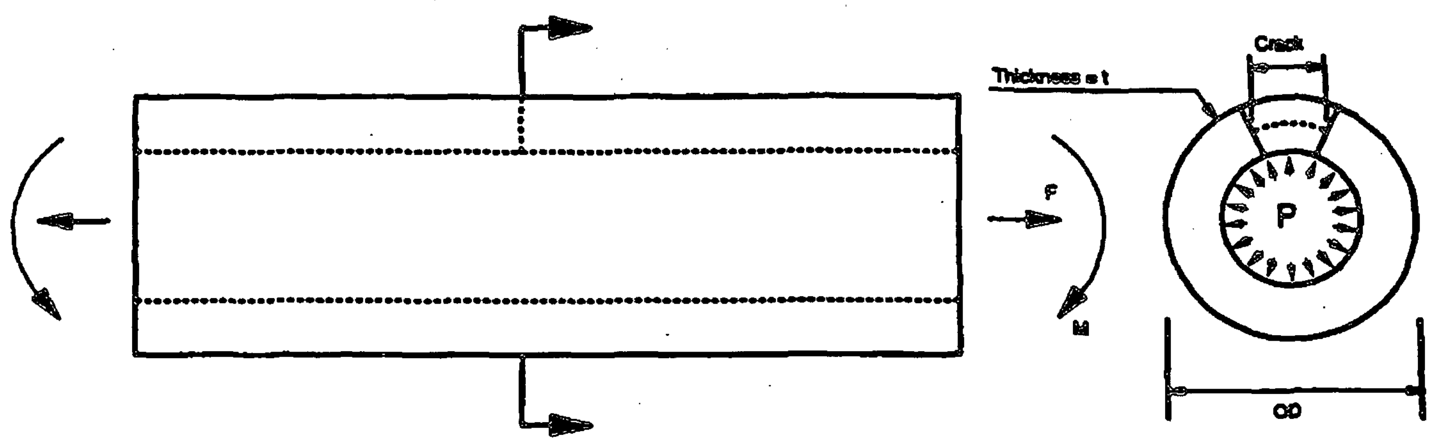


Figure 5-4 Idealized Pressure Drop Profile Through a Postulated Crack



OD (pipe outer diameter) = 10.75"
t = 1.00"

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

a,c,e

OD = 10.75 in. σ_y = 23.10 ksi F = 137.57 kips

t = 1.00 in. σ_u = 77.93 ksi M = 414.95 in-kips

A376-TP316/A403 WP-316 with SMAW weld

Figure 5-6 Critical Flaw Size Prediction for Node 1320 Case D

a,c,e

OD = 10.75 in. σ_y = 23.10 ksi F = 137.71 kips

t = 1.00 in. σ_u = 77.93 ksi M = 344.42 in-kips

A376-TP316/A403 WP-316 with SMAW weld

Figure 5-7 Critical Flaw Size Prediction for Node 1320 Case E

a,c,e

OD = 10.75 in. σ_y = 25.72 ksi F = 30.38 kips

t = 1.00 in. σ_u = 77.93 ksi M = 1199.06 in-kips

A376-TP316/A403 WP-316 with SMAW weld

Figure 5-8 Critical Flaw Size Prediction for Node 1320 Case F

a,c,e

OD = 10.75 in. σ_y = 25.72 ksi F = 31.55 kips

t = 1.00 in. σ_u = 77.93 ksi M = 1315.39 in-kips

A376-TP316/A403 WP-316 with SMAW weld

Figure 5-9 Critical Flaw Size Prediction for Node 1320 Case G

6 ASSESSMENT OF FATIGUE CRACK GROWTH

6.1 INTRODUCTION

To determine the sensitivity of the pressurizer surge line to the presence of small cracks when subjected to the various transients a fatigue crack growth analysis was performed for the Prairie Island Unit 1 and the results were documented in Section 6 of WCAP-12877 (Reference 6-1). Fatigue crack growth analysis was performed at two critical locations. Location 1 was near the reactor coolant loop nozzle and location 2 was near the pressurizer nozzle.

The results of the fatigue crack growth analysis obtained from Reference 6-1 is also presented in Table 6-1. Various initial surface flaws were assumed to exist. The flaws were assumed to be semi-elliptical with a six-to-one aspect ratio. The largest initial flaw assumed to exist was one with a depth equal to 10% of the minimum wall thickness, the maximum flaw size that could be found acceptable by Section XI of the ASME Code. The results show that the maximum fatigue crack growth was increased only by 1.2%, which is negligible. It was concluded that the fatigue crack growth is not a concern for the pressurizer surge line. Figure 6-1 shows the fatigue crack growth controlling positions of A, B, C and D at each location.

Since the Prairie Island Unit 2 pressurizer surge line pipe size, pipe schedule and pipe material are the same as those of Prairie Island Unit 2 and the design transients are identical, it is evident that the Prairie Island Unit 2 pressurizer surge line will have similar fatigue crack growth. Although there are some differences in the stratification transients between the Unit 1 and Unit 2 surge lines, those differences will have insignificant impact on the results of the fatigue crack growth and also as indicated above the fatigue crack growth is negligible. Therefore the results shown in Table 6-1 are also representative of the Prairie Island Unit 2 pressurizer surge line fatigue crack growth.

6.2 REFERENCES

6-1 WCAP-12877, " Technical Justification For Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for Prairie Island Unit 1," March 1991.

Table 6-1 Fatigue Crack Growth Results for 10% of Wall Initial Flaw Size					
Location	Position	Initial Size (in)	Initial (% Wall)	Final (40 year) Size (in)	Final Flaw (% Wall)
1	[				]a,c,e
1	[				]a,c,e
1	[				]a,c,e
1	[				]a,c,e
2	[				]a,c,e
2	[				]a,c,e
2	[				]a,c,e
2	[				]a,c,e

Note: Location 1 is near the reactor coolant loop nozzle and location 2 is near the pressurizer nozzle.

a,c,e

Figure 6-1 Fatigue Crack Growth Controlling Positions at Each Location (from Unit 1)

7 ASSESSMENT OF MARGINS

In the preceding sections, the leak rate calculations, fracture mechanics analysis and fatigue crack growth assessment were performed. Margins at the critical location are summarized below:

In Section 5.3 using the SRP 3.6.3 approach (i.e., "Z" factor approach), the "critical" flaw sizes at the governing location are calculated. In Section 5.2 the crack lengths yielding a leak rate of 2 gpm (10 times the leak detection capability of 0.2 gpm) for the critical location are calculated. The leakage size flaws, the instability flaws, and margins are given in Table 7-1. The margins are the ratio of instability flaw to leakage flaw. The margins for analysis combination cases A/D, [ ]a,c,e well exceed the factor of 2. The margin for the extremely low probability event defined by [ ]a,c,e also meets the LBB criteria. As stated in Section 4.4, the probability of simultaneous occurrence of SSE and maximum stratification due to shutdown because of leakage is estimated to be very low.

In this evaluation, the leak-before-break methodology is applied conservatively. The conservatisms used in the evaluation are summarized in Table 7-2.

Table 7-1 Leakage Flaw Sizes, Critical Flaw Sizes and Margins				
Node	Load Case	Critical Flaw Size (in)	Leakage Flaw Size (in)	Margin
1320	[			]a,c,e
	[			]a,c,e
	[			]a,c,e
	[			]a,c,e
	[[1]			]a,c,e
	[[1]			]a,c,e

1 These are judged to be low probability events

Table 7-2 LBB Conservatisms
Factor of 10 on Leak Rate
Factor of 2 on Leakage Flaw for all Cases
Algebraic Sum of Loads for Leakage
Absolute Sum of Loads for Stability
Average Material Properties for Leakage
Minimum Material Properties for Stability

8 CONCLUSIONS

This report justifies the elimination of pressurizer surge Line pipe breaks as the structural design basis for Prairie Island 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 auxiliary Line) because of system design, testing, and operational considerations.

c. The effects of low and high cycle fatigue on the integrity of the surge line were evaluated and shown acceptable. The effects of thermal stratification were evaluated and shown acceptable.

d. Ample margin exists between the leak rate of small stable flaws and the capability of Prairie Island Unit 2 reactor coolant system pressure boundary leakage detection system.

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

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

Based on the above, it is concluded that pressurizer surge line breaks should not be considered in the structural design basis of Prairie Island 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