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Westinghouse Energy Systems

WCAP-13101

TECHNICAL JUSTIFICATION FOR ELIMINATING PRESSURIZER SURGE LINE RUPTURE FROM THE STRUCTURAL DESIGN BASIS FOR COMANCHE PEAK UNIT 2

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SECTION 1.0
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.

1.2 Scope and Objective

The general purpose of this investigation is to demonstrate leak-before-break for the 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. A schematic drawing of the piping system is shown in Section 3.0. The recommendations and criteria proposed in NUREG 1061 Volume 3 (1-1) 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) Show that a through-wall crack will not result from fatigue crack growth.

4) 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. A margin of 10 is demonstrated between the calculated leak rate and the leak detection capability.

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

6) 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.

7) 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 analyses are 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 of 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 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). This operating history totals over 400 reactor-years, including five plants each having over 15 years of operation and 15 other plants each with over 10 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, 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. 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 lines since they are designed and operated to preclude the voiding condition in normally filled lines. 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 lines. Preoperational testing and operating experience have verified the Westinghouse approach. The operating transients of the RCS primary piping and connected surge 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 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. Recent field measurements on typical PWR plants 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 Potential Degradation During Service

There has never been any service cracking or wall thinning identified in the pressurizer surge lines 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 lines 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 used in the leak-before-break evaluation described in this report.

The 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 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 Material

The pipe material of the pressurizer surge line for Comanche Peak Unit 2 is SA376/TP316. This is a wrought product form of the type used for the primary loop piping of several PWR plants. 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 pipe or cast fitting. The welding processes used are gas tungsten arc (GTAW), and shielded metal arc (SMAW). Weld locations are identified in Figure 3-1.

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

3.2 Material Properties

Applicable material properties were developed from those in the Certified Materials Test Report as given in table 3-1. The ASME code minimum properties are given in table 3-2. It is seen that the measured properties well exceed those of the code. As seen later properties at [ ]^{a,c,e} and 653°F are required for the leak rate and stability analyses.

Industry data at 650°F were used as a basis for determining tensile properties at 653°F. Data for SA376 TP316 stainless steel pipe and welds are given in table 3-3 taken from reference 3-1. Data in table 3-3 are quite similar to the Comanche Peak Unit 2 piping data in table 3-1. By maintaining a constant ratio of properties at room temperature and 653°F, the 653°F properties for the surge line material were estimated. The properties at [ ]^{a,c,e} were obtained by maintaining the same ratio as those given in the ASME Code (reference 3-2). The modulus of elasticity at [ ]^{a,c,e} was obtained from reference 3-2. All the tensile properties are given in table 3-4. The properties at [ ]^{a,c,e} were obtained in a similar fashion to those above.

3.3 References

3-1 Witt, F. J. et. al., Integrity of the Primary Piping Systems of Westinghouse Nuclear Powe. Plants During Postulated Seismic Events, WCAP-9283, Westinghouse Electric Corp., March 1978, p 3-3.

3-2 ASME Boiler and Pressure Vessel Code Section III, Division 1, Appendices July 1, 1989.

TABLE 3-1

Room Temperature Mechanical Properties of the Pressurizer Surge Line Materials of Comanche Peak Unit 2

ID	MATERIAL	HEAT NO.	YIELD STRENGTH (psi)	ULTIMATE STRENGTH (psi)	ELONG. (%)	R/A (%)
1	SA376/TP316	J6565/28408	44,900	86,200	53.0	68.2
2	SA376/TP316	J6566/28400	47,700	87,800	52.5	68.2
3	SA376/TP316	J6565/28408	44,900	86,200	53.0	68.2
4	SA376/TP316	J6565/28409	46,100	86,600	52.6	66.9
5	SA376/TP316	J6566/28400	47,700	87,800	52.5	68.2

TABLE 3-2

Room Temperature ASME Code Minimum Properties

Material	Yield Stress (psi)	Ultimate Stress (psi)
SA376/TP316	30,000	75,000

TABLE 3-3

TYPICAL TENSILE PROPERTIES OF SA376 TP316 AND WELDS OF SUCH MATERIAL FOR REACTOR PRIMARY COOLANT SYSTEMS

Plant	Material	Test Temperature (°F)	Average Tensile Properties Yield (psi)	Ultimate (psi)
1	SA376 TP316	70	40,900 (48)[a]	83,200 (48)
		650	23,500 (19)	67,900 (19)
	E 308 Weld	70	63,900 (3)	87,600 (3)
2	SA376 TP316	70	47,100 (40)	88,300 (40)
		650	26,900 (22)	69,100 (25)
	E 308 Weld	70	59,900 (8)	87,200 (8)
		650	31,500 (1)	68,800 (1)
3	SA376 TP316	70	46,600 (36)	87,300 (36)
		650	24,200 (18)	66,800 (19)
	E 308 Weld	70	61,900 (4)	85,400 (4)

a. (___) indicates the number of test results averaged obtained from Certified Materials Test Report of the primary coolant system of a plant.

TABLE 3-4

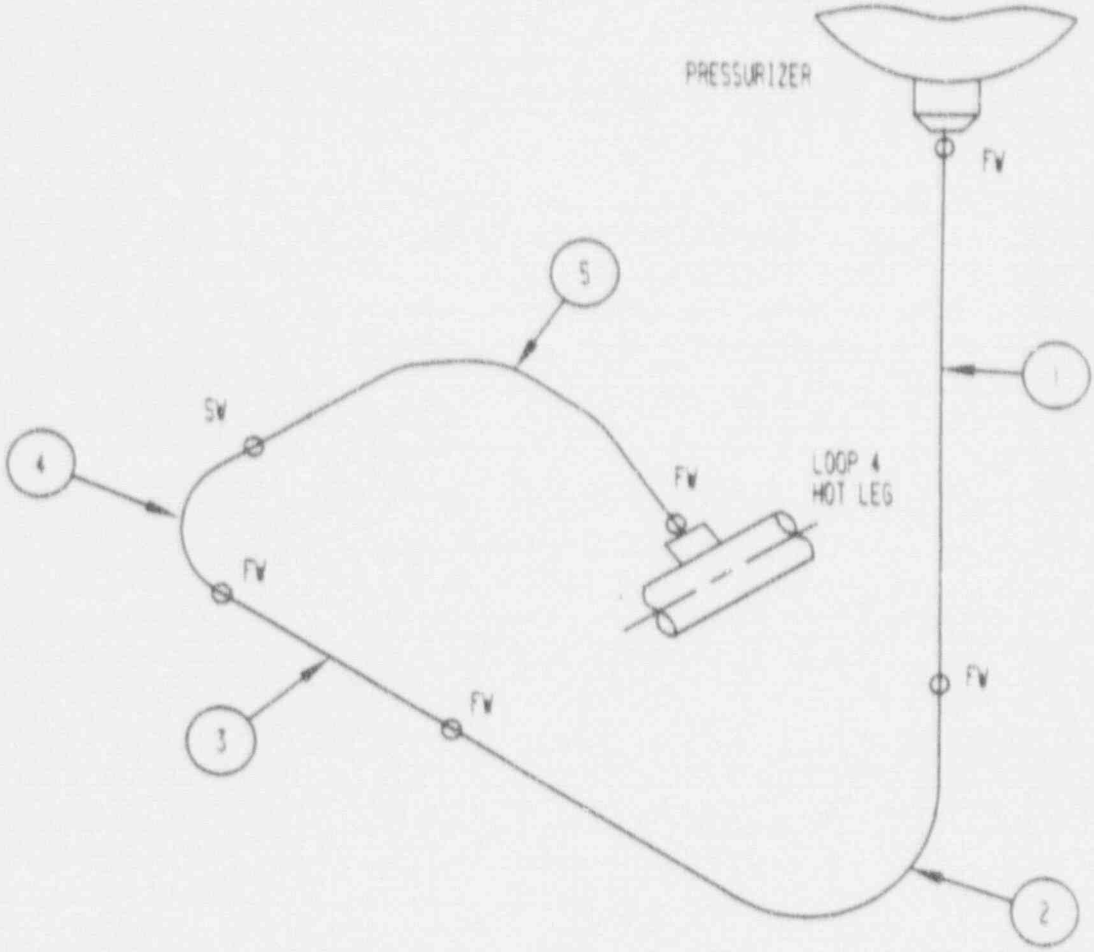
TENSILE PROPERTIES FOR THE SURGE LINE MATERIAL
AT [ ]a,c,e, [ ]a,c,e AND [ ]a,c,e

Temperature (°F)	Yield Stress (psi)		Ultimate Strength (psi)		Modulus of Elasticity
	Average	Minimum	Average	Minimum	($psi \times 10^6$)
70[a]	46,260	44,900	86,920	86,200	28.3

[

]a,c,e

a Minimum values from table 3-1.



"FW" means field welds.

"SW" means shop welds.

The numbers in the circles identify the materials.

(see the ID column of table 3-1)

Figure 3-1 Comanche Peak Unit 2 Surge Line Layout

SECTION 4.0

LOADS FOR FRACTURE MECHANICS ANALYSIS

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

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

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

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 computed by the methods to be 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:

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

DW = Deadweight

TH = Applicable thermal load (normal or stratified)

P = Load due to internal pressure

SSE = SSE loading including seismic anchor motion

4.2 Loads for Leak Rate Evaluation

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

$$F = F_{DW} + F_{TH} + F_p \qquad (4\text{-}6)$$
$$M_Y = (M_Y)_{DW} + (M_Y)_{TH} \qquad (4\text{-}7)$$
$$M_Z = (M_Z)_{DW} + (M_Z)_{TH} \qquad (4\text{-}8)$$

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

4.3 Loading Conditions

Because thermal stratification can cause large stresses at heatup and cooldown temperatures in the range of 455°F of the RCS fluid, 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} The logic for this ΔT [ ]^{a,c,e} is based on the following:

Actual practice, based on experience of other plants with this type of situation, indicates that the plant operators complete the cooldown as quickly as possible once a leak in the primary system is detected. Technical Specifications may require cold shutdown within 36 hours but actual practice is that the plant depressurizes the system as soon as possible once a primary system leak is detected. Therefore, the hot leg is generally on the warmer side of the limit (>200°F) when the pressurizer bubble is quenched. Once the bubble is quenched, the pressurizer is cooled down fairly quickly reducing the ΔT in the system.

4.4 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. A summary of the loads and stresses is given in table 4-4.

4.5 Governing Location

The welds at the Comanche Peak Unit 2 surge lines are fabricated using the GTAW and SMAW welding procedures. Node 1020 (which is at a GTAW weld) is the governing location, when the stress levels and the weld procedures are both taken into account for all the locations on the pressurizer surge line.

TABLE 4-1

Types of Loadings

Pressure (P)

Dead Weight (DW)

Normal Operating Thermal Expansion (TH)

Safe Shutdown Earthquake and Seismic Anchor Motion (SSE)[a]

[ ] a,c,e

[a] 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 an RCS temperature of 653°F consisting of the algebraic sum of the loading components due to P, DW and TH.

a,c,e

CASE D: This is the faulted operating case at an RCS temperature of 653°F consisting of the absolute sum (every component load is taken as positive) of P, DW, TH and SSE.

a,c,e

TABLE 4-2 (continued)

Normal and Faulted Loading Cases for Leak-Before-Break Evaluations

[ ] a,c,e

TABLE 4-3

Associated Load Cases for Analyses

A/D This is heretofore standard leak-before-break evaluation.

[ ]a,c,e

---

a These are judged to be low probability events.

TABLE 4-4

Summary of LBB Loads and Stresses by Case for Comanche Peak Unit 2

Node	Case	Axial Force F (lb)	Moment M (in-lb)	Axial Stress σ_F (psi)	Bending Stress σ_M (psi)	Total Stress σ_T (psi)	
1020	A	218262	1146697	3924	7118	11112	
1020							a,c,e
1020							
1020	D	235640	3030655	4236	18998	23234	
1020							a,c,e
1020							
1020							

Outside diameter is 14 in.

Wall thickness is 1.249 in.[a]

[a] Weld undercut is incorporated

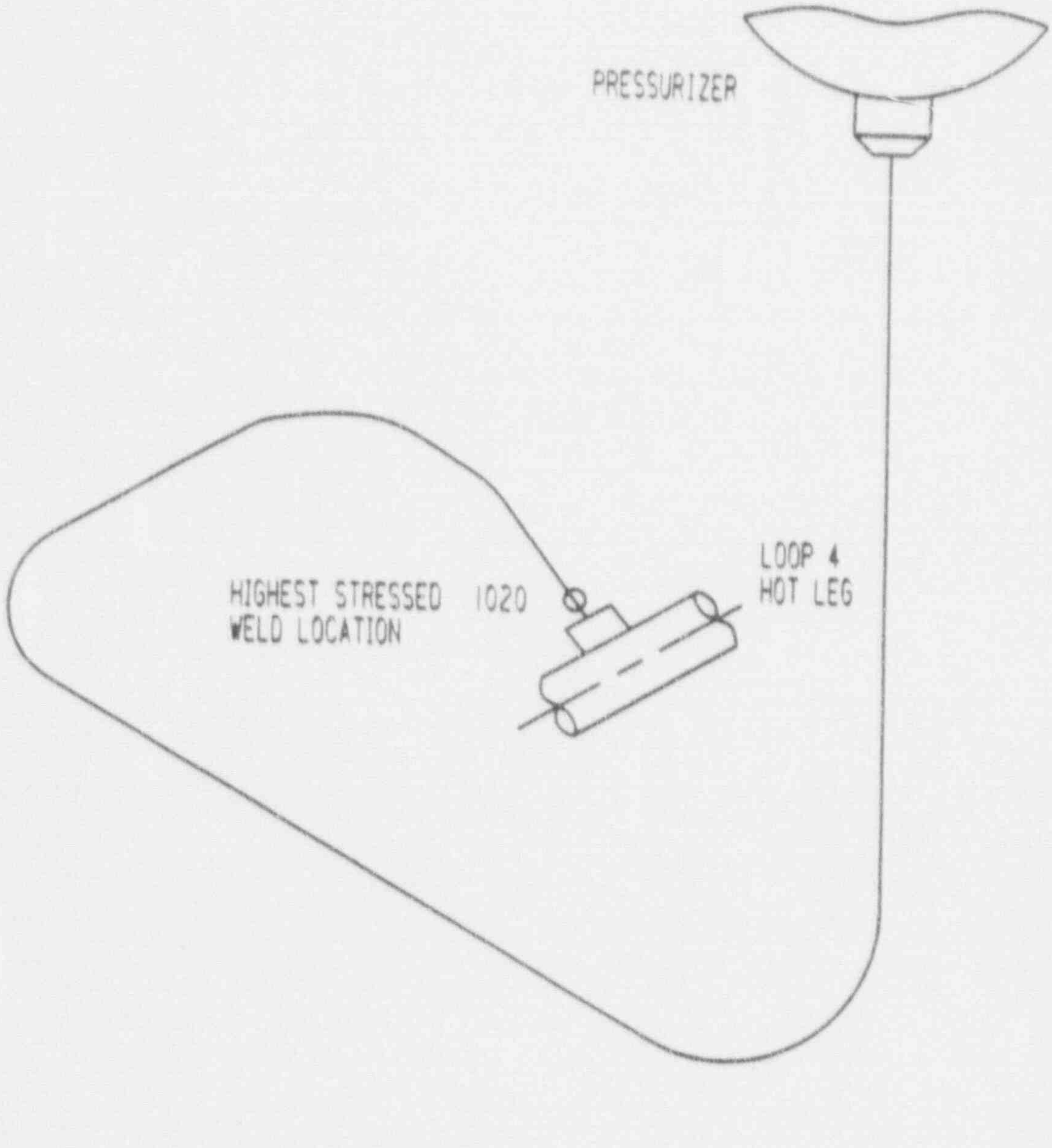


Figure 4-1 Comanche Peak Unit 2 Surge Line Showing Governing Location

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 sizes for the pressurizer surge line analysis cases. 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 surge line 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 down condition. The purpose of this section is to discuss the method which l 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

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

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 [

]^{a,c,e} to obtain the total pressure drop from the system under consideration to the atmosphere. Thus,

Absolute Pressure - 14.7 = [ ]^{a,c,e} (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 value of flow through the crack.

For the locations at the lower temperature, single phase calculations for the leak rate in gallons per minute (GPM) were performed, using an equation from reference 5-3 as follows:

$$\Big[\qquad\qquad \Big]^{a,c,e} \qquad (5\text{-}5)$$

where g: gravity acceleration (ft/sec^2)

Δp: pressure drop (lb/ft^2)

ρ: density at room temperature (lb/ft^3)

K: friction loss including passage loss, inlet and outlet of the through wall crack

A: crack opening area, (in^2)

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 5-4 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 node identified in section 4.0 as Node 1020. The crack lengths yielding a leak rate of 10 gpm (10 times the leak detection capability of 1.0 gpm) at this node are shown in table 5-1.

The Comanche Peak plant RCS pressure boundary leak detection system meets the intent of Regulatory Guide 1.45. Thus, to satisfy the margin of 10 on the

leak rate, the flaw sizes (leakage flaws) are determined which yield a leak rate of 10 gpm.

5.3 Stability Evaluation

A typical segment of the 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 5-9. 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. Table 5-2 shows a summary of the 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 "Thermal Engineering," C. C. Dillio and E.P. Nye, International Text Company, pp. 270-273, 1969.

5-4 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.

5-5 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 Comanche Peak Unit 2

Node Point	Load Case	Temperature (°F)	Crack Length (in.) (for 10 gpm leakage)

[ ] a,c,e

TABLE 5-2

Summary of Critical Flaw Size for Comanche Peak Unit 2

Node Point	Load Case	Temperature (°F)	Critical Flaw Size (in)

a,c,e

a,c,e

Figure 5-1 Fully Plastic Stress Distribution

a,c,e

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

[ ][a,c,e]

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

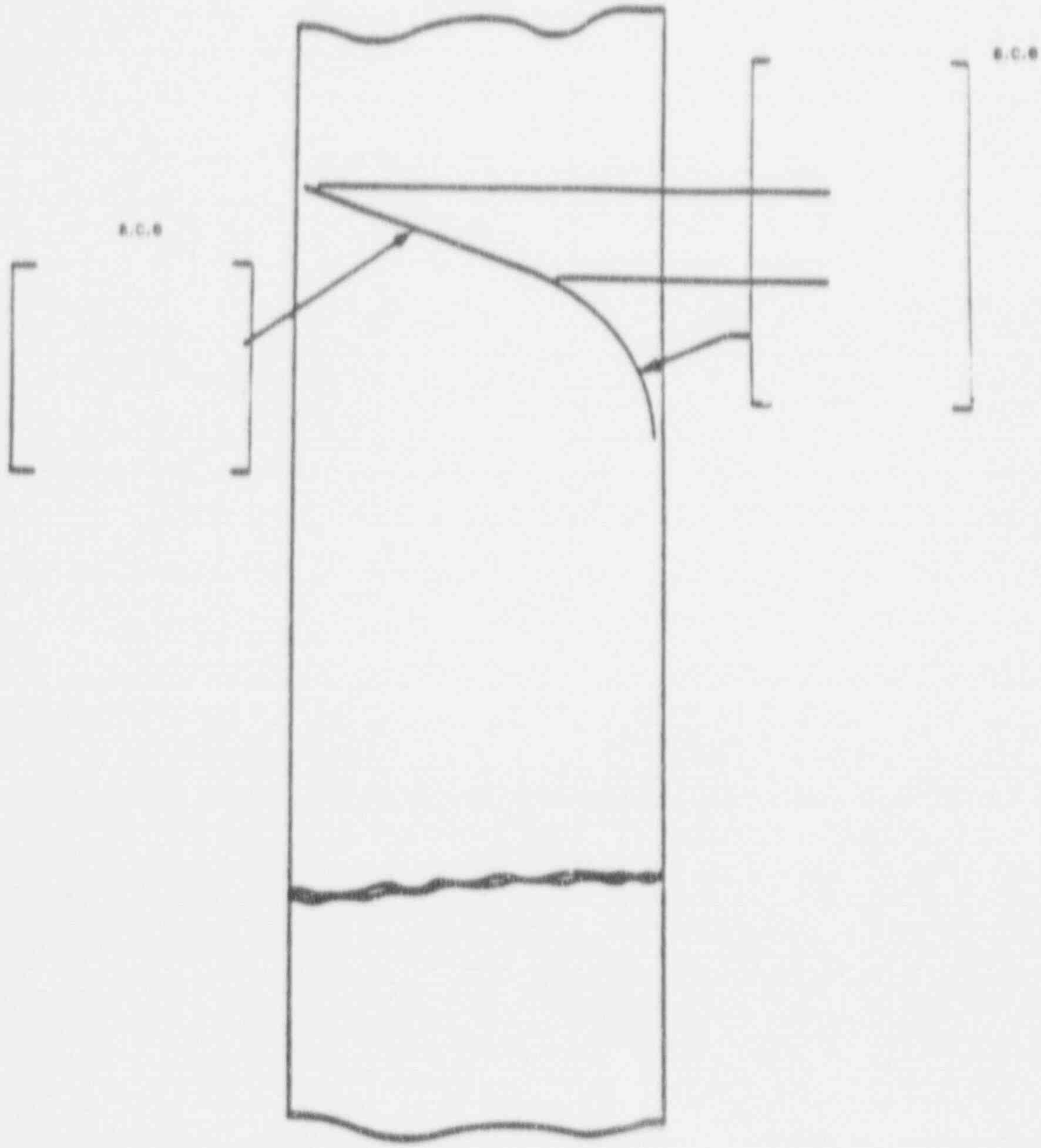


Figure 5-4. Idealized Pressure Drop Profile Through a Postulated Crack

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

LIMIT MOMENT (in-kips)

a,c,e

Figure 5-6. Critical Flaw Size Prediction for Comanche Peak Unit 2
Node 1020 Case D

LIMIT MOMENT (in-kips)

a,c,e

Figure 5-7. Critical Flaw Size Prediction for Comanche Peak Unit 2
Node 1020 Case E

a,c,e

LIMIT MOMENT (in-kips)

Figure 5-8 Critical Flaw Size Prediction for Comanche Peak Unit 2 Node 1020 Case F

LIMIT MOMENT (in-kips)

a,c,e

Figure 5-9 Critical Flaw Size Prediction for Comanche Peak Unit 2 Node 1020 Case G

SECTION 6.0
ASSESSMENT OF FATIGUE CRACK GROWTH

In WCAP-12248 (Reference 6-1) a detailed fatigue crack growth evaluation was performed for Comanche Peak Unit 1. That evaluation showed that the calculated flaw depth was far below the acceptable limit of 60% of wall thickness.

The normal loads plus transients are comparable between Comanche Peak Units 1 and 2. Also, the geometry and the piping layouts are similar. For both units, the outside diameter is 14 inches, and the minimum wall thickness is 1.25 in.

Based on the similarities between Units 1 and 2, it can be concluded that the fatigue crack growth for Unit 2 will also be well below the acceptable limit.

<u>Reference</u>

6-1 WCAP-12248, "Evaluation of Thermal Stratification for the Comanche Peak Unit 1 Pressurizer Surge Line, April 1989.

SECTION 7.0
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 the critical flaw sizes at the governing location are calculated. In Section 5.2 the leakage size flaws are calculated. These leakage flaws yield a leak rate of 10 gpm (10 times the leak detection capability of 1.0 gpm) for the critical locations. The leakage size flaws, the critical flaws, and margins are given in Table 7-1. The margins are the ratio of critical flaw size to leakage flaw size. 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} is about a factor of 2. As stated in Section 4.3, 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
for Comanche Peak Unit 2

Node	Load Case	Critical Flaw Size (in)	Leakage Flaw Size (in)	Margin

[a,c,e]

[a] These are judged to be low probability events

TABLE 7-2

LLB 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 Properties for Leakage
- Minimum Material Properties for Stability

SECTION 8.0
CONCLUSIONS

This report justifies the elimination of pressurizer surge line pipe breaks as the structural design basis for Comanche Peak Unit 2 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 lines) 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 assessed and shown acceptable. The effects of thermal stratification were evaluated and shown acceptable.

d. Adequate margin exists between the leak rate of small stable flaws and the capability of Comanche Peak Unit 2 reactor coolant system pressure boundary leakage detection system.

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

The postulated leakage flaws will be stable because of the margins in d and 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 Comanche Peak Unit 2.

APPENDIX A

LIMIT MOMENT

APPENDIX A

LIMIT MOMENT

[

]a,c,e

a,c,e

Figure A-1. Pipe With A Through-Wall Crack In Bending

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