Enclosure 5

TOPICAL REPORT WCAP-15550-NP, REV. 2

Virginia Electric and Power Company
(Dominion Energy Virginia)
Surry Power Station Units 1 and 2

WCAP-15550-NP
Revision 2

March 2019

Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Surry Units 1 and 2 Nuclear Power Plants for the Subsequent License Renewal Program (80 Years) Leak-Before-Break Evaluation

Westinghouse

WCAP-15550-NP
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Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Surry Units 1 and 2 Nuclear Power Plants for the Subsequent License Renewal Program (80 Years) Leak-Before-Break Evaluation

March 2019

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RECORD OF REVISIONS

Rev	Date	Revision Description
0	August 2000	Original Issue (WCAP-15550)
1-A (Draft for customer review)	March 2017	Revised to include the LBB results from the Measured Uncertainty Recapture (MUR) Program. This WCAP revision also includes LBB evaluation for subsequent license renewal for 80 years of operation for Surry Units 1 and 2.
1	June 2017	Finalized Revision 1-A and incorporated customer's comments.
2	March 2019	Revised to address three chemical content errors in Tables 4-6 and 4-7 and to address the updated CMTR data from Surry Unit 1 that have not been considered in Revision 1. The updated CMTR data is provided in Table 4-6. As shown in the revised Tables 4-6 and 4-7, the corrections of CMTR errors are shown in bold font and updates due to additional CMTRs are marked by grey-shaded color. Changes in the text of the document are shown with revision bars in the right margin.

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EXECUTIVE SUMMARY

The original structural design basis of the reactor coolant system for the Surry Units 1 and 2 Nuclear Power Plants required consideration of dynamic effects resulting from pipe break and that protective measures for such breaks be incorporated into the design. Subsequent to the original Surry design, an additional concern of asymmetric blowdown loads was raised as described in Unresolved Safety Issue A-2 (Asymmetric Blowdown Loads on the Reactor Coolant System). Surry Units 1 and 2 Nuclear Power Plants were part of the utilities which sponsored Westinghouse to resolve the A-2 issue. Generic analyses by Westinghouse to resolve the A-2 issue was approved by the NRC and documented in Generic Letter 84-04 (Reference 1-1). Generic Letter 84-04 served as the original basis for the elimination of large primary loop pipe rupture from the structural design basis for Surry Units 1 and 2. As identified in Generic Letter 84-04, the primary technical references supporting the NRC's safety evaluation of eliminating postulated pipe breaks are documented in WCAP-9558 (Reference 1-9) and WCAP-9787 (Reference 1-10).

Research by the NRC and industry coupled with operating experience determined that safety could be negatively impacted by placement of pipe whip restraints on certain systems. As a result, NRC and industry initiatives resulted in demonstrating that Leak-before-break (LBB) criteria can be applied to reactor coolant system piping based on fracture mechanics technology and material toughness.

Subsequently, the NRC modified 10CFR50 General Design Criterion 4, and published in the Federal Register (Vol. 52, No. 207) on October 27, 1987 its final rule, "Modification of General Design Criterion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures," (Reference 1-2). This change to the rule allows use of leak-before-break technology for excluding from the design basis the dynamic effects of postulated ruptures in primary coolant loop piping in pressurized water reactors (PWRs).

The LBB evaluation is performed based on loading, pipe geometry and fracture toughness considerations, enveloping critical locations were determined at which leak-before-break crack stability evaluations were made. Through-wall flaw sizes were found which would cause a leak at a rate of ten (10) times the leakage detection system capability of the plant. Large margins for such flaw sizes were demonstrated against flaw instability. Finally, fatigue crack growth was shown not to be an issue for the primary loops.

Revision 0 of this report had demonstrated compliance with LBB technology for the Surry reactor coolant system piping for the 60 year plant life based on a plant specific analysis. Subsequently, an LBB evaluation was performed for the MUR (Measurement Uncertainty Program), the results of that particular analysis are also incorporated in Revision 1 of this report. Lastly, based on the LBB evaluation in Revision 1 of this report herein, it also demonstrated that dynamic effects of reactor coolant system primary loop pipe breaks need not be considered in the structural design basis of the Surry Units 1 and 2 Nuclear Power Plants for the 80 year plant life (Subsequent License Renewal Program). The technical evaluations utilized in Revision 0 through Revision 2 of this report are consistent with the methodology and principles of WCAP-9558 and WCAP-9787. Therefore, the justifications demonstrated herein are compliant with the original conclusions of Generic Letter 84-04.

The report documents the plant specific geometry, loading, and material properties used in the fracture mechanics evaluation. Mechanical properties were determined at operating temperatures.

Since the piping systems include cast austenitic stainless steel, fracture toughness considering thermal aging was determined for each heat of material. Fully aged fracture toughness properties were used for the LBB evaluation. The full aged condition is applicable for plants operating at beyond 15 EFPY (Effective Full Power Years) for the CF8M materials (elbows for Surry Units 1 and 2). As of January 2017, Surry Units 1 and 2 are operating at 33.78 and 33.69 EFPY, respectively. Thus, the LBB evaluation in this report has been demonstrated for the primary loops at Surry Units 1 and 2 for 80 years of plant operation.

1.0 INTRODUCTION

1.1 PURPOSE

This report applies to the Surry Units 1 and 2 Reactor Coolant System (RCS) primary loop piping. It is intended to demonstrate that for the specific parameters of the Surry Units 1 and 2 Nuclear Power Plants, RCS primary loop pipe breaks need not be considered in the structural design basis for the 80 year plant life (Subsequent License Renewal Program). This report also includes the LBB evaluation results based on the Measurement Uncertainty Recapture (MUR) Program.

1.2 BACKGROUND INFORMATION

Westinghouse has performed considerable testing and analysis to demonstrate that RCS primary loop pipe breaks can be eliminated from the structural design basis of all Westinghouse plants. The concept of eliminating pipe breaks in the RCS primary loop was first presented to the NRC in 1978 in WCAP-9283 (Reference 1-3). That topical report employed a deterministic fracture mechanics evaluation and a probabilistic analysis to support the elimination of RCS primary loop pipe breaks. That approach was then used as a means of addressing Generic Issue A-2 and Asymmetric LOCA Loads.

Westinghouse performed additional testing and analysis to justify the elimination of RCS primary loop pipe breaks. This material was provided to the NRC along with Letter Report NS-EPR-2519 (Reference 1-4).

The NRC funded research through Lawrence Livermore National Laboratory (LLNL) to address this same issue using a probabilistic approach. As part of the LLNL research effort, Westinghouse performed extensive evaluations of specific plant loads, material properties, transients, and system geometries to demonstrate that the analysis and testing previously performed by Westinghouse and the research performed by LLNL applied to all Westinghouse plants (References 1-5 and 1-6). The results from the LLNL study were released at a March 28, 1983, ACRS Subcommittee meeting. These studies, which are applicable to all Westinghouse plants east of the Rocky Mountains, determined the mean probability of a direct LOCA (RCS primary loop pipe break) to be 4.4×10^{-12} per reactor year and the mean probability of an indirect LOCA to be 10^{-7} per reactor year. Thus, the results previously obtained by Westinghouse (Reference 1-3) were confirmed by an independent NRC research study.

Based on the studies by Westinghouse, LLNL, the ACRS, and the AIF, the NRC completed a safety review of the Westinghouse reports submitted to address asymmetric blowdown loads that result from a number of discrete break locations on the PWR primary systems. The NRC Staff evaluation (Reference 1-1) concludes that an acceptable technical basis has been provided so that asymmetric blowdown loads need not be considered for those plants that can demonstrate the applicability of the modeling and conclusions contained in the Westinghouse response or can provide an equivalent fracture mechanics demonstration of the primary coolant loop integrity. In a more formal recognition of Leak-Before-Break (LBB) methodology applicability for PWRs, the NRC appropriately modified 10 CFR 50, General Design Criterion 4, "Requirements for Protection Against Dynamic Effects for Postulated Pipe Rupture" (Reference 1-2).

1.3 SCOPE AND OBJECTIVES

The general purpose of this investigation is to demonstrate leak-before-break for the primary loops in Surry Units 1 and 2 on a plant specific basis for the 80 year plant life. The recommendations and criteria proposed in References 1-7 and 1-8 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 locations at which the highest stress occurs.

2. Identify the materials and the associated material properties.

3. Postulate a surface flaw at the governing locations. Determine fatigue crack growth. Show that a through-wall crack will not result.

4. Postulate a through-wall flaw at the governing locations. 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 faulted loads, demonstrate that there is a margin of 2 between the leakage flaw size and the critical flaw size.

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. For the materials actually used in the plant provide the properties including toughness and tensile test data. Evaluate long term effects such as thermal aging.

8. Demonstrate margin on applied load.

This report provides a fracture mechanics demonstration of primary loop integrity for the Surry Units 1 and 2 plants consistent with the NRC position for exemption from consideration of dynamic effects.

It should be noted that the terms “flaw” and “crack” have the same meaning and are used interchangeably. “Governing location” and “critical location” are also used interchangeably throughout the report.

The computer codes used in this evaluation for leak rate and fracture mechanics calculations have been validated and used for all the LBB applications by Westinghouse.

1.4 REFERENCES

1-1 USNRC Generic Letter 84-04, Subject: "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops," February 1, 1984.

1-2 Nuclear Regulatory Commission, 10 CFR 50, Modification of General Design Criteria 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures, Final Rule, Federal Register/Vol. 52, No. 207/Tuesday, October 27, 1987/Rules and Regulations, pp. 41288-41295.

1-3 WCAP-9283, "The Integrity of Primary Piping Systems of Westinghouse Nuclear Power Plants During Postulated Seismic Events," March, 1978.

1-4 Letter Report NS-EPR-2519, Westinghouse (E. P. Rahe) to NRC (D. G. Eisenhut), Westinghouse Proprietary Class 2, November 10, 1981.

1-5 Letter from Westinghouse (E. P. Rahe) to NRC (W. V. Johnston) dated April 25, 1983.

1-6 Letter from Westinghouse (E. P. Rahe) to NRC (W. V. Johnston) dated July 25, 1983.

1-7 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.

1-8 NUREG-0800 Revision 1, March 2007, Standard Review Plan: 3.6.3 Leak-Before-Break Evaluation Procedures.

1-9 WCAP-9558, Revision 2, "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Throughwall Crack," May, 1981.

1-10 WCAP-9787, "Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation," May, 1981.

2.0 OPERATION AND STABILITY OF THE REACTOR COOLANT SYSTEM

2.1 STRESS CORROSION CRACKING

The Westinghouse reactor coolant system primary loops 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 1400 reactor-years, including 16 plants each having over 30 years of operation, 10 other plants each with over 25 years of operation, 11 plants each with over 20 years of operation, and 12 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 (PCSG) established in 1975, addressed cracking in boiling water reactors only.) One of the objectives of the second 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. The discussion below further qualifies the PCSG's findings.

For stress corrosion cracking (SCC) to occur in piping, the following three conditions must exist simultaneously: high tensile stresses, susceptible material, and a corrosive environment. Since some residual stresses and some degree of material susceptibility exist in any stainless steel piping, the potential for stress corrosion is minimized by properly selecting a material immune to SCC as well as preventing the occurrence of a

corrosive environment. The material specifications consider compatibility with the system's operating environment (both internal and external) as well as other material in the system, applicable ASME Code rules, fracture toughness, welding, fabrication, and processing.

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

During plant operation, the reactor coolant water chemistry is monitored and maintained within very specific limits. Contaminant concentrations are kept below the thresholds known to be conducive to stress corrosion cracking with the major water chemistry control standards being included in the plant operating procedures as a condition for plant operation. For example, during normal power operation, oxygen concentration in the RCS 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.

It should be noted that there are no primary water stress corrosion cracking material such as Alloy 82/182 in the dissimilar metal welds in the Surry Units 1 and 2 Reactor Coolant System (RCS) primary loop piping.

2.2 WATER HAMMER

Overall, there is a low potential for water hammer in the RCS since it is designed and operated to preclude the voiding condition in normally filled lines. The reactor coolant system, including piping and primary components, is 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; pressure is controlled by pressurizer heaters and pressurizer spray also within a narrow range for steady-state conditions. The flow characteristics of the system remain constant during a fuel cycle because the only governing parameters, namely system resistance and the reactor coolant pump characteristics, are controlled in the design process. Additionally, Westinghouse has instrumented typical reactor coolant systems to verify the flow and vibration characteristics of the system. Preoperational

testing and operating experience have verified the Westinghouse approach. The operating transients of the RCS primary piping are such that no significant water hammer can occur.

2.3 LOW CYCLE AND HIGH CYCLE FATIGUE

An assessment of the low cycle fatigue loadings was carried out as part of this study in the form of a fatigue crack growth analysis, as discussed in Section 8.0.

High cycle fatigue loads in the system would result primarily from pump vibrations. These are minimized by restrictions placed on shaft vibrations during hot functional testing and operation. During operation, an alarm signals the exceedance of the vibration limits. Field measurements have been made on a number of plants during hot functional testing, including plants similar to Surry Units 1 and 2. Stresses in the elbow below the reactor coolant pump resulting from system vibration have been found to be very small, between 2 and 3 ksi at the highest. These stresses are well below the fatigue endurance limit for the material and would also result in an applied stress intensity factor below the threshold for fatigue crack growth.

2.4 WALL THINNING, CREEP, AND CLEAVAGE

Wall thinning by erosion and erosion-corrosion effects should not occur in the primary loop piping due to the low velocity, typically less than 1.0 ft/sec and the stainless steel material, which is highly resistant to these degradation mechanisms. The cause of wall thinning is related to high water velocity and is therefore clearly not a mechanism that would affect the primary loop piping.

Creep is typical experienced for temperatures over 700°F for stainless steel material, and the maximum operating temperature of the primary loop piping is well below this temperature value; therefore, there would be no significant mechanical creep damage in stainless steel piping.

Cleavage type failures are not a concern for the operating temperatures and the stainless steel material used in the primary loop 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.

3.0 PIPE GEOMETRY AND LOADING

3.1 INTRODUCTION TO METHODOLOGY

The general approach is discussed first. As an example a segment of the primary coolant hot leg pipe is shown in Figure 3-1. The as-built outside diameter and minimum wall thickness of the pipe are 34.00 in. and 2.395 in., respectively, as shown in the figure. The normal stresses at the weld locations are from the load combination procedure discussed in Section 3.3 whereas the faulted loads are as described in Section 3.4. The components for normal loads are pressure, dead weight and thermal expansion. An additional component, Safe Shutdown Earthquake (SSE), is considered for faulted loads. Tables 3-1 and 3-2 show the enveloping loads for Surry Units 1 and 2; these loads were determined as part of the MUR project. As seen from Table 3-2, the highest stressed location in the entire loop is at Location 1 at the reactor vessel outlet nozzle to pipe weld. This is one of the locations at which, as an enveloping location, leak-before-break is to be established. Essentially a circumferential flaw is postulated to exist at this location which is subjected to both the normal loads and faulted loads to assess leakage and stability, respectively. The loads (developed below) at this location are also given in Figure 3-1.

Since the elbows are made of different materials than the pipe, locations other than the highest stressed pipe location were examined taking into consideration both fracture toughness and stress. The four most critical locations among the entire primary loop are identified after the full analysis is completed. Once loads (this section) and fracture toughnesses (Section 4.0) are obtained, the critical locations are determined (Section 5.0). At these locations, leak rate evaluations (Section 6.0) and fracture mechanics evaluations (Section 7.0) are performed per the guidance of References 3-1 and 3-2. Fatigue crack growth (Section 8.0) assessment and stability margins are also evaluated (Section 9.0). All the weld locations considered for the LBB evaluation are those shown in Figure 3-2.

Please note that the piping loads and stresses based on the MUR Program were considered in the LBB evaluation as part of Revision 1 of this WCAP report.

3.2 CALCULATION OF LOADS AND STRESSES

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

$$\sigma = \frac{F}{A} + \frac{M}{Z} \qquad (3\text{-}1)$$

where,

σ	=	stress, ksi
F	=	axial load, kips
M	=	bending moment, in-kips
A	=	pipe cross-sectional area, in^2
Z	=	section modulus, in^3

The total moments for the desired loading combinations are calculated by the following equation:

$$M = \sqrt{M_X^2 + M_Y^2 + M_Z^2} \qquad (3\text{-}2)$$

where,

M	=	total moment for required loading
M_X	=	X component of moment (torsion)
M_Y	=	Y component of bending moment
M_Z	=	Z component of bending moment

NOTE: X-axis is along the center line of the pipe.

The axial load and bending moments for leak rate predictions and crack stability analyses are computed by the methods to be explained in Sections 3.3 and 3.4.

3.3 LOADS FOR LEAK RATE EVALUATION

The normal operating loads for leak rate predictions are calculated by the following equations:

$$F = F_{DW} + F_{TH} + F_P \qquad (3\text{-}3)$$

$$M_X = (M_X)_{DW} + (M_X)_{TH} \qquad (3\text{-}4)$$

$$M_Y = (M_Y)_{DW} + (M_Y)_{TH} \qquad (3\text{-}5)$$

$$M_Z = (M_Z)_{DW} + (M_Z)_{TH} \qquad (3\text{-}6)$$

The subscripts of the above equations represent the following loading cases:

DW	=	deadweight
TH	=	normal thermal expansion
P	=	load due to internal pressure

This method of combining loads is often referred to as the algebraic sum method (References 3-1 and 3-2).

The loads based on this method of combination are provided in Table 3-1 at all the weld locations identified in Figure 3-2. The as-built dimensions are also given in Table 3-1.

3.4 LOAD COMBINATION FOR CRACK STABILITY ANALYSES

In accordance with Standard Review Plan 3.6.3 (References 3-1 and 3-2), the margin in terms of applied loads needs to be demonstrated by crack stability analysis. Margin on loads of 1.4 ($\sqrt{2}$) can be demonstrated if normal plus Safe Shutdown Earthquake (SSE) are applied. The 1.4 ($\sqrt{2}$) margin should be reduced to 1.0 if the deadweight, thermal expansion, internal pressure, pressure expansion, SSEINERTIA and seismic anchor motion (SAM) loads are combined based on individual absolute values as shown below.

The absolute sum of loading components is used for the LBB analysis which results in higher magnitude of combined loads and thus satisfies a margin on loads of 1.0. The absolute summation of loads is shown in the following equations:

$$F = |F_{DW}| + |F_{TH}| + |F_P| + |F_{SSEINERTIA}| + |F_{SSEAM}| \quad (3\text{-}7)$$

$$M_X = |(M_X)_{DW}| + |(M_X)_{TH}| + |(M_X)_{SSEINERTIA}| + |(M_X)_{SSEAM}| \quad (3\text{-}8)$$

$$M_Y = |(M_Y)_{DW}| + |(M_Y)_{TH}| + |(M_Y)_{SSEINERTIA}| + |(M_Y)_{SSEAM}| \quad (3\text{-}9)$$

$$M_Z = |(M_Z)_{DW}| + |(M_Z)_{TH}| + |(M_Z)_{SSEINERTIA}| + |(M_Z)_{SSEAM}| \quad (3\text{-}10)$$

where subscript SSEINERTIA refers to safe shutdown earthquake inertia, SSEAM is safe shutdown earthquake anchor motion, respectively.

The loads so determined are used in the fracture mechanics evaluations (Section 7.0) to demonstrate the LBB margins at the locations established to be the governing locations. These loads at all the weld locations (see Figure 3-2) are given in Table 3-2.

3.5 REFERENCES

3-1 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.

3-2 NUREG-0800 Revision 1, March 2007, Standard Review Plan: 3.6.3 Leak-Before-Break Evaluation Procedures.

Table 3-1 Dimensions, Normal Loads and Stresses for Surry Units 1 and 2					
Location[a]	Outside Diameter (in)	Minimum Thickness (in)	Axial Load[b] (kips)	Moment (in-kips)	Total Stress (ksi)
1	34.00	2.395	1482	19815	17.51
2	34.00	2.395	1482	837	6.71
3	34.00	2.395	1482	9661	11.73
4	37.75	3.270	1614	14956	9.87
5	37.625	3.208	1628	7852	7.55
6	36.32	2.555	1605	6815	9.11
7	36.32	2.555	1599	6786	9.07
8	36.32	2.555	1709	1071	6.81
9	37.625	3.208	1709	2666	5.90
10	37.625	3.208	1844	9466	8.76
11	32.26	2.270	1365	2588	8.11
12	32.26	2.270	1365	2927	8.33
13	32.26	2.270	1366	2373	7.97
14	33.60	2.940	1366	3636	6.64
15	33.60	2.940	1363	4955	7.29

Notes:

a. See Figure 3-2

b. Included Pressure

Table 3-2 Faulted Loads and Stresses for Surry Units 1 and 2

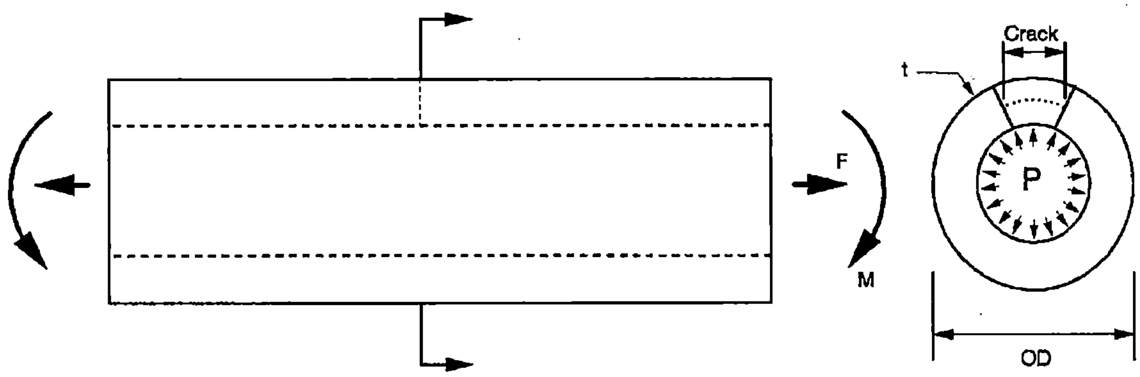
Location[a,b]	Axial Load[c] (kips)	Moment (in-kips)	Total Stress (ksi)
1	1640	24646	20.93
2	1640	2652	8.41
3	1639	12918	14.25
4	1941	20101	12.62
5	1927	22956	13.89
6	1870	15673	14.23
7	1864	9928	11.52
8	1839	8475	10.75
9	1839	11375	9.43
10	1885	14032	10.53
11	1413	7850	11.84
12	1413	7390	11.54
13	1420	6032	10.66
14	1422	7923	8.99
15	1418	10829	10.42

Notes:

a. See Figure 3-2

b. See Table 3-1 for dimensions

c. Included Pressure



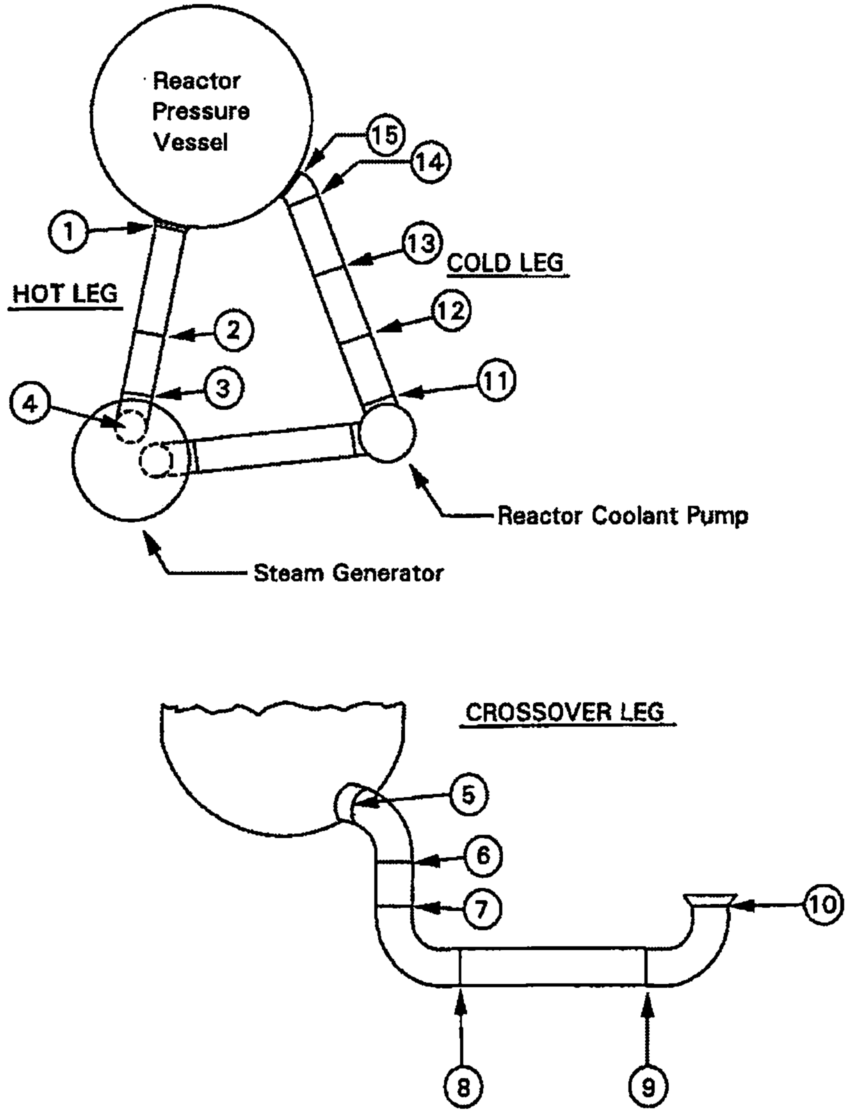
OD[a] = 34.00 in
t[a] = 2.395 in

Location 1

Normal Loads[a]		Faulted Loads[b]	
Force[c]:	1482 kips	Force[c]:	1640 kips
Bending Moment:	19815 in-kips	Bending Moment:	24646 in-kips

[a] See Table 3-1
[b] See Table 3-2
[c] Includes the force due to a pressure of 2250 psia

Figure 3-1 Hot Leg Coolant Pipe



<u>HOT LEG</u>	Temperature 609.1°F	Pressure: 2250 psia
<u>CROSS-OVER LEG</u>	Temperature 542.6°F	Pressure: 2250 psia
<u>COLD LEG</u>	Temperature 542.9°F	Pressure: 2250 psia

Figure 3-2 Schematic Diagram of Surry Units 1 and 2 Primary Loop Showing Weld Locations

4.0 MATERIAL CHARACTERIZATION

4.1 PRIMARY LOOP PIPE AND FITTINGS MATERIALS

The primary loop pipe is A376-TP316 and the elbow fittings are A351-CF8M for Surry Units 1 and 2.

4.2 TENSILE PROPERTIES

The Pipe Certified Materials Test Reports (CMTRs) for Surry Units 1 and 2 were used to establish the tensile properties for the leak-before-break analyses. The CMTRs include tensile properties at room temperature and/or at 650°F for each of the heats of material. These properties are given in Table 4-1 for the Surry Unit 1 pipe, Table 4-2 for the Surry Unit 2 pipe, Table 4-3 for Unit 1 elbows and in Table 4-4 for Unit 2 elbows.

The representative properties at 609°F (represents actual 609.1°F for Hot Leg) for the pipe were established from the tensile properties at 650°F given in Tables 4-1 and 4-2 by utilizing Section II of the 1999 ASME Boiler and Pressure Vessel Code (Reference 4-1). Code tensile properties at 609°F were obtained by interpolating between the 600°F and 650°F tensile properties. Ratios of the code tensile properties at 609°F to the corresponding tensile properties at 650°F were then applied to the 650°F tensile properties given in Tables 4-1 and 4-2 to obtain the plant specific properties for A376-TP316 at 609°F. It should be noted that there is no significant impact by using the 1999 ASME Code Section II edition for material properties for the LBB analysis, as compared to the Surry ASME code of record.

The representative properties at 609°F (represents actual 609.1°F for Hot Leg) and 543°F (represents actual 542.9°F for Cold Leg and 542.6°F for Crossover Leg) for the elbows were established from the tensile properties at room temperature properties given in Tables 4-3 and 4-4 by utilizing Section II of the 1999 ASME Boiler and Pressure Vessel Code (Reference 4-1). Code tensile properties at 609°F and 543°F were obtained by interpolating between the 500°F, 600°F and 650°F tensile properties. Ratios of the code tensile properties at 609°F and 543°F to the corresponding tensile properties at room temperature were then applied to the room temperature tensile properties given in Tables 4-3 and 4-4 to obtain the plant specific properties for A351-CF8M at 609°F and 543°F.

The average and lower bound yield strengths and ultimate strengths are given in Table 4-5. The ASME Code moduli of elasticity values are also given, and Poisson's ratio was taken as 0.3.

Updated CMTRs from Replacement Steam Generator (RSG) replacement elbows on Surry Unit 1 are also considered. The added tensile properties are shown in Table 4-3 (marked by grey-shaded color). Of these added properties, the minimum Yield Strength is 38350 psi and the minimum Ultimate Strength is 81200 psi. These values are bounded by lower bound values in Table 4-5. It has also been reviewed that the updated tensile data has negligible impact to the average Yield Strength in Table 4-5.

4.3 FRACTURE TOUGHNESS PROPERTIES

The pre-service fracture toughness (J) of cast stainless steels that are of interest are in terms of J_{Ic} (J at Crack Initiation) and have been found to be very high at 600°F. [

]^{a,c,e} However, cast

stainless steel is susceptible to thermal aging at the reactor operating temperature, that is, about 290°C (550°F). Thermal aging of cast stainless steel results in embrittlement, that is, a decrease in the ductility, impact strength, and fracture toughness of the material. Depending on the material composition, the Charpy impact energy of a cast stainless steel component could decrease to a small fraction of its original value after exposure to reactor temperatures during service.

The susceptibility of the material to thermal aging increases with increasing ferrite contents. The molybdenum bearing CF8M shows increased susceptibility to thermal aging.

The method described below was used to calculate the end of life toughness properties for the cast material of the Surry Units 1 and 2 primary coolant loop piping and elbows.

In 1994, the Argonne National Laboratory (ANL) completed an extensive research program in assessing the extent of thermal aging of cast stainless steel materials (Reference 4-2). The ANL research program measured mechanical properties of cast stainless steel materials after they had been heated in controlled ovens for long periods of time. ANL compiled a data base, both from data within ANL and from international sources, of about 85 compositions of cast stainless steel exposed to a temperature range of 290-400°C (550-750°F) for up to 58,000 hours (6.5 years). In 2015 the work done by ANL was augmented, and the fracture toughness database for CASS materials was aged to 100,000 hours at 290-350°C (554-633°F). The methodology for estimating fracture properties has been extended to cover CASS materials with a ferrite content of up to 40%. From this database (NUREG/CR-4513, Revision 2), ANL developed correlations for estimating the extent of thermal aging of cast stainless steel (Reference 4-3).

ANL developed the fracture toughness estimation procedures by correlating data in the database conservatively. After developing the correlations, ANL validated the estimation procedures by comparing the estimated fracture toughness with the measured value for several cast stainless steel plant components removed from actual plant service. The procedure developed by ANL in Reference 4-3 was used to calculate the end of life fracture toughness values for this analysis. The ANL research program was sponsored and the procedure was accepted by the NRC.

Based on NUREG/CR-4513, Revision 2, the fracture toughness correlations used for the full aged condition is applicable for plants operating at and beyond 15 EFPY (Effective Full Power Years) for the CF8M materials (elbows for Surry Units 1 and 2). As of January 2017, Surry Units 1 and 2 are operating at 33.78 and 33.69 EFPY, respectively. Therefore, the use of the fracture toughness correlations described below is applicable for the fully aged or saturated condition of the Surry Units 1 and 2 elbow materials made of CF8M.

The chemical compositions of the Surry Units 1 and 2 primary loop elbow fitting material are available from CMTRs and are provided in Table 4-6 and Table 4-7 of this report. The following equations are taken from Reference 4-3 and applicable for CF8M type material:

$$Cr_{eq} = Cr + 1.21(Mo) + 0.48(Si) - 4.99 = \text{(Chromium equivalent)} \tag{4-1}$$

$$Ni_{eq} = (Ni) + 0.11(Mn) - 0.0086(Mn)^2 + 18.4(N) + 24.5(C) + 2.77 = \text{(Nickel equivalent)} \tag{4-2}$$

$$\delta_c = 100.3(Cr_{eq} / Ni_{eq})^2 - 170.72(Cr_{eq} / Ni_{eq}) + 74.22 = \text{(Ferrite Content)} \tag{4-3}$$

where the elements are in percent weight and δ_c is ferrite in percent volume.

The saturation room temperature (RT) impact energies of the cast stainless steel materials were determined from the chemical compositions available from CMTRs and provided in Tables 4-6 and 4-7.

For CF8M steel with < 10% Ni, the saturation value of RT impact energy Cv_{sat} (J/cm^2) is the lower value determined from

$$\log_{10}Cv_{sat} = 0.27 + 2.81 \exp(-0.022\phi) \quad (4\text{-}4)$$

where the material parameter ϕ is expressed as

$$\phi = \delta_c (Ni + Si + Mn)^2(C + 0.4N)/5.0 \quad (4\text{-}5)$$

and from

$$\log_{10}Cv_{sat} = 7.28 - 0.011\delta_c - 0.185Cr - 0.369Mo - 0.451Si - 0.007Ni - 4.71(C + 0.4N) \quad (4\text{-}6)$$

For CF8M steel with $\geq$ 10% Ni, the saturation value of RT impact energy Cv_{sat} (J/cm^2) is the lower value determined from

$$\log_{10}Cv_{sat} = 0.84 + 2.54 \exp(-0.047\phi) \quad (4\text{-}7)$$

where the material parameter ϕ is expressed as

$$\phi = \delta_c (Ni + Si + Mn)^2(C + 0.4N)/5.0 \quad (4\text{-}8)$$

and from

$$\log_{10}Cv_{sat} = 7.28 - 0.011\delta_c - 0.185Cr - 0.369Mo - 0.451Si - 0.007Ni - 4.71(C + 0.4N) \quad (4\text{-}9)$$

The saturation J-R curve at RT, for static-cast CF8M steel is given by

$$J_d = 1.44\,(Cv_{sat})^{1.35}(\Delta a)^n \quad \text{for } Cv_{sat} < 35 \text{ J/cm}^2 \quad (4\text{-}10)$$

$$J_d = 16\,(Cv_{sat})^{0.67}(\Delta a)^n \quad \text{for } Cv_{sat} \geq 35 \text{ J/cm}^2 \quad (4\text{-}11)$$

$$n = 0.20 + 0.08 \log_{10}(Cv_{sat}) \quad (4\text{-}12)$$

where J_d is the "deformation J" in kJ/m^2 and Δa is the crack extension in mm.

The saturation J-R curve at 290-320°C (554-608°F), for static-cast CF8M steel is given by

$$J_d = 5.5\,(Cv_{sat})^{0.98}(\Delta a)^n \quad \text{for } Cv_{sat} < 46 \text{ J/cm}^2 \quad (4\text{-}13)$$

$$J_d = 49\,(Cv_{sat})^{0.41}(\Delta a)^n \quad \text{for } Cv_{sat} \geq 46 \text{ J/cm}^2 \quad (4\text{-}14)$$

$$n = 0.19 + 0.07 \log_{10}(Cv_{sat}) \quad (4\text{-}15)$$

where J_d is the "deformation J" in kJ/m^2 and Δa is the crack extension in mm.

[

]a,c,e

[

]a,c,e

The results from the ANL Research Program indicate that the lower-bound fracture toughness of thermally aged cast stainless steel is similar to that of submerged arc welds (SAWs). The applied value of the J-integral for a flaw in the weld regions will be lower than that in the base metal because the yield stress for the weld materials is much higher at the temperature.[1]

Therefore, weld regions are less limiting than the cast material.

In the fracture mechanics analyses that follow, the fracture toughness properties given in Table 4-8 will be used as the criteria against which the applied fracture toughness values will be compared.

As indicated in the record of revisions table, this stress report is revised to address CMTR errors and to address updated CMTRs from RSG replacement elbows on Unit 1. As shown in the revised Tables 4-6 and 4-7, the corrections of CMTR errors (shown in bold font) and updates due to additional CMTR's (marked by grey-shaded color) have been included in this stress report. It has been reviewed that the revisions do not impact the evaluation results in this Section 4 material characterization. The summary of limiting fracture toughness properties provided in Table 4-8 remains valid and applicable. The revisions also do not affect the conclusions of this stress report.

4.4 REFERENCES

4-1 ASME Boiler and Pressure Vessel Code, An International Code, Section II, Materials, Part D-Properties, 1999 Addenda, July 1,1999.

4-2 O. K. Chopra and W. J. Shack, "Assessment of Thermal Embrittlement of Cast Stainless Steels," NUREG/CR-6177, U.S. Nuclear Regulatory Commission, Washington, DC, May 1994.

4-3 O. K. Chopra, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," NUREG/CR-4513, Revision 2, U.S. Nuclear Regulatory Commission, Washington, DC, May 2016.

[1] In the report all the applied J values were conservatively determined by using base metal strength properties.

Table 4-1 Measured Tensile Properties (psi) for Surry Unit 1 Primary Loop Pipes					
		At Room Temp.		At 650°F	
Heat No./Serial No.	Location	Yield Strength	Ultimate Strength	Yield Strength	Ultimate Strength
F0212/2867Z	X-over Leg	43500	84200	26400	67000
F0212/2867Z	X-over Leg	43900	82500	N/A	N/A
F0215/2894Y	X-over Leg	40000	84800	21700	66800
F0215/2894Y	X-over Leg	41900	87700	N/A	N/A
F0222/2900Y	X-over Leg	44000	86500	21500	66200
F0222/2900Y	X-over Leg	41500	83200	N/A	N/A
8777/2883X	X-over Leg	36100	78200	21000	62000
8777/2883X	X-over Leg	38500	77800	N/A	N/A
8785/2881X	X-over Leg	36100	74200	20400	57200
8785/2881X	X-over Leg	39700	79800	N/A	N/A
8915/2885X	X-over Leg	38500	77400	24200	62300
8915/2885X	X-over Leg	38600	77200	N/A	N/A
8777/2874Y	Cold Leg	35300	79200	24100	65600
8777/2874Y	Cold Leg	34900	78200	N/A	N/A
8913/2875Y	Cold Leg	35100	78400	24300	68500
8913/2875Y	Cold Leg	41100	84800	N/A	N/A
F0228/2996	Cold Leg	45100	88600	24700	69800
F0228/2996	Cold Leg	43900	87400	N/A	N/A
F0229/2952	Cold Leg	42200	85300	24500	68200
F0229/2952	Cold Leg	39700	83400	N/A	N/A
F0162/2859	Cold Leg	46000	83500	27600	69400
F0162/2859	Cold Leg	52900	87900	N/A	N/A
F0162/2860	Cold Leg	43700	83900	N/A	N/A
F0162/2860	Cold Leg	47400	90400	N/A	N/A
F0216/2861	Cold Leg	40600	83400	21300	66600
F0216/2861	Cold Leg	40000	80500	N/A	N/A
F0227/2947	Cold Leg	42500	83700	20900	66600
F0227/2947	Cold Leg	43200	87100	N/A	N/A
F0229/2951	Cold Leg	42000	84400	24500	68200
F0229/2951	Cold Leg	44500	86800	N/A	N/A
8785/2963X	Hot Leg	33400	75700	22000	61000
8785/2963X	Hot Leg	33600	75300	N/A	N/A
F0190/2847Y	Hot Leg	42000	88800	21300	58200
F0190/2847Y	Hot Leg	43000	86000	N/A	N/A
F0058/2629	Hot Leg	41200	85200	27000	74000
F0058/2629	Hot Leg	44300	89500	N/A	N/A
F0188/2845	Hot Leg	40900	83000	26100	67000
F0188/2845	Hot Leg	43000	84000	N/A	N/A
E1482/3353	Hot Leg	41800	84400	24500	66200
E1482/3353	Hot Leg	41700	84600	N/A	N/A

Note: N/A = Not Applicable

Table 4-2 Measured Tensile Properties (psi) for Surry Unit 2 Primary Loop Pipes					
		At Room Temperature		At 650°F	
Heat No./Serial No.	Location	Yield Strength	Ultimate Strength	Yield Strength	Ultimate Strength
F0213/2889	Hot Leg	43200	84800	23700	69800
F0213/2889	Hot Leg	44000	88400	N/A	N/A
F0213/2890X	Hot Leg	41900	82500	N/A	N/A
F0213/2890X	Hot Leg	46500	84500	N/A	N/A
F0213/2891X	Hot Leg	44000	86000	N/A	N/A
F0213/2891X	Hot Leg	41000	83000	N/A	N/A
F0227/2897X	Hot Leg	41800	86800	N/A	N/A
F0227/2897X	Hot Leg	42500	85500	N/A	N/A
F0373/3169	Cold Leg	41400	85800	25600	71900
F0373/3169	Cold Leg	44800	89700	N/A	N/A
F0221/2866X	X-over Leg	44000	83800	N/A	N/A
F0221/2866Y	X-over Leg	42700	86000	21600	65200
F0222/2900X	X-over Leg	44000	86500	21500	66200
F0222/2900X	X-over Leg	41500	83200	N/A	N/A
52154/2843	Cold Leg	43500	86500	23100	57400
52154/2843	Cold Leg	33300	75600	N/A	N/A
F0228/2948	Cold Leg	42500	87400	24700	69800
F0228/2948	Cold Leg	45000	87500	N/A	N/A
F0229/2994	Cold Leg	41000	82800	24500	68200
F0229/2994	Cold Leg	45000	87900	N/A	N/A
E1490/3347Y	Hot Leg	43100	86000	23700	68000
E1490/3347Y	Hot Leg	43000	82700	N/A	N/A
F0189/2869Y	X-over Leg	37700	80600	N/A	N/A
F0189/2869Z	X-over Leg	44100	91000	N/A	N/A
F0189/2868X	X-over Leg	37700	80600	25200	70000
F0189/2868X	X-over Leg	44100	91000	N/A	N/A
F0226/2946	Cold Leg	43000	85000	21500	66400
F0226/2946	Cold Leg	42100	86000	N/A	N/A
K2011/3683X	Cold Leg	32100	75900	20600	56200
K2011/3683X	Cold Leg	38400	81900	N/A	N/A
E1478/3257	Cold Leg	38000	82900	25300	72400
E1478/3257	Cold Leg	42400	84900	N/A	N/A
V0629/3262	Cold Leg	48400	88800	21100	67400
V0629/3262	Cold Leg	41300	82200	N/A	N/A
F0229/2953	Cold Leg	41000	84900	24500	68200
F0229/2953	Cold Leg	45000	86200	N/A	N/A
F0215/2892Y	Hot Leg	43000	83000	21700	66800
F0215/2892Y	Hot Leg	42000	84600	N/A	N/A

Note: N/A = Not Applicable

Table 4-3 Measured Tensile Properties (psi) for Surry Unit 1 Primary Loop Elbows			
Heat No.	Location	At Room Temperature	
		Yield Strength	Ultimate Strength
10360-1	X-over Leg	46500	88000
10844-1	X-over Leg	39000	78000
11046-1	X-over Leg	43500	87000
11246-1	X-over Leg	42000	82500
11441-1	X-over Leg	48000	88500
11937-1	X-over Leg	45000	88500
10442-1	X-over Leg	48000	88500
11168-1	X-over Leg	46500	88000
12198-1	X-over Leg	45000	85000
12623-1	X-over Leg	42000	80000
10482-1	X-over Leg	45750	80250
10723-1	X-over Leg	46200	88600
29943-2	Cold Leg	39300	77300
30690-1	Cold Leg	39700	80200
29943-1	Hot Leg	38400	74900
31011-5	Hot Leg	43000	84800
28387-2	Hot Leg	42300	85800
28387-1	Cold Leg	42300	85800
30597-2	X-over Leg	42300	84800
10128-2	X-over Leg	46500	89000
10243-2	X-over Leg	48000	90000
44632-1*	X-over Leg	44700	86400
44756-1*	X-over Leg	41300	87100
44004-1*	X-over Leg	41600	84750
45395-2*	X-over Leg	46900	84950
44662-2*	Hot Leg	42450	86400
45396-1*	X-over Leg	46900	84950
44756-2*	Hot Leg	41300	87100
44286-2*	Hot Leg	38350	81450
44504-1*	X-over Leg	38700	81200

Note:

* tensile properties from the added heats are not included in the average tensile property calculations in Table 4-5, but the impact of the added tensile data are negligible.

Table 4-4 Measured Tensile Properties (psi) for Surry Unit 2 Primary Loop Elbows			
Heat No.	Location	At Room Temperature	
		Yield Strength	Ultimate Strength
30080-2	Hot Leg	42400	82400
32535-2	Cold Leg	44100	87500
31571-6	Cold Leg	42800	85600
14008-1	X-over Leg	48000	87000
14085-1	X-over Leg	45000	86000
14165-1	X-over Leg	48000	89000
12709-1	X-over Leg	48000	85500
14507-1	X-over Leg	40500	82000
13579-1	X-over Leg	48000	85500
13781-5	X-over Leg	4500	89500
13826-2	X-over Leg	48000	87500
30690-2	Hot Leg	38100	82200
31427-2	Hot Leg	41250	86250
14786-3	X-over Leg	42000	84500
14990-1	X-over Leg	43500	84500
15384-1	X-over Leg	45000	88500
15769-1	X-over Leg	46500	85500
30080-1	Cold Leg	45100	84200
12087-3	X-over Leg	42000	80500
12547-2	X-over Leg	40500	83500
13051-4	X-over Leg	40500	83500

Table 4-5 Mechanical Properties for Surry Units 1 and 2 Materials at Operating Temperatures

Material	Temperature* (°F)	Average Yield Strength (psi)	Modulus of Elasticity (psi)	Lower Bound	
				Yield Strength (psi)	Ultimate Strength (psi)
A376 TP316	609	23835	25.255×10^6	20,762	56,200
A351 CF8M	609	27468	25.255×10^6	23,785	71,904
	543	28493	25.585×10^6	24,672	71,904
Poisson's ratio: 0.3					

Note: Representative temperature. The actual temperatures are provided in Figure 3-2.

Table 4-6 Chemistry and Fracture Toughness Elbow Properties of the Material Heats of Surry Unit 1

a,c,e

Table 4-6 Chemistry and Fracture Toughness Elbow Properties of the Material Heats of Surry Unit 1

a,c,e

Table 4-7 Chemistry and Fracture Toughness Elbow Properties of the Material Heats of Surry Unit 2

a,c,e

Table 4-8 Fracture Toughness Properties for Surry Units 1 and 2 Primary Loops for Leak-Before-Break Evaluation at Critical Locations

a,c,e

a,c,e

Figure 4-1 Pre-Service J vs. Δa for SA351-CF8M Cast Stainless Steel at 600°F

5.0 CRITICAL LOCATION AND EVALUATION CRITERIA

5.1 CRITICAL LOCATIONS

The leak-before-break (LBB) evaluation margins are to be demonstrated for the limiting locations (governing locations). Such locations are established based on the loads (Section 3.0) and the material properties established in Section 4.0. These locations are defined below for Surry Units 1 and 2. Table 3-2 as well as Figure 3-2 are used for this evaluation.

Critical Locations

The highest stressed location for the entire primary loop is at Location 1 (in the Hot Leg) (See Figure 3-2) at the reactor vessel outlet nozzle to pipe weld. Location 1 is critical for all the weld locations of pipe.

Since the elbows are made of cast materials, the critical locations for the elbows are: for the hot leg, the highest stressed location is at weld location 3; for the cross-over leg, the highest stressed location is at weld location 6; and for the cold leg, the highest stressed location is at weld location 15. It is thus concluded that the enveloping locations in Surry Units 1 and 2 for which LBB methodology is to be applied are locations 1, 3, 6 and 15. The tensile properties and the allowable toughness for the critical locations are shown in Tables 4-5 and 4-8.

5.2 FRACTURE CRITERIA

As will be discussed later, fracture mechanics analyses are made based on loads and postulated flaw sizes related to leakage. The stability criteria against which the calculated J and tearing modulus are compared are:

(1) If $J_{app} < J_{Ic,}$ then the crack will not initiate and the crack is stable;

(2) If $J_{app} \geq J_{Ic}$; and $T_{app} < T_{mat}$ and $J_{app} < J_{max}$, then the crack is stable.

Where:

J_{app} = Applied J
J_{Ic} = J at Crack Initiation
T_{app} = Applied Tearing Modulus
T_{mat} = Material Tearing Modulus
J_{max} = Maximum J value of the material

For critical locations, the limit load method discussed in Section 7.0 was also used.

6.0 LEAK RATE PREDICTIONS

6.1 INTRODUCTION

The purpose of this section is to discuss the method which is used to predict the flow through postulated through-wall cracks and present the leak rate calculation results for through-wall circumferential cracks.

6.2 GENERAL CONSIDERATIONS

The flow of hot pressurized water through an opening to a lower back pressure causes flashing which can result in choking. 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

6.3 CALCULATION METHOD

The basic method used in the leak rate calculations is the method developed by [

]a,c,e

The flow rate through a crack was calculated in the following manner. Figure 6-1 from Reference 6-2 was used to estimate the critical pressure, P_c, for the primary loop enthalpy condition and an assumed flow. Once P_c was found for a given mass flow, the [ ]a,c,e was found from Figure 6-2 (taken from Reference 6-2). 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 6-3, where P_o is the operating pressure. Now using the assumed flow rate, G, the frictional pressure drop can be calculated using

$$\Delta P_f = [\quad]^{a,c,e} \tag{6-1}$$

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

The frictional pressure drop using equation 6-1 is then calculated for the assumed flow rate and added to the [ ]a,c,e to obtain the total pressure drop from the primary system to the atmosphere. That is, for the primary loop:

$$\text{Absolute Pressure} - 14.7 = [\quad]^{a,c,e} \tag{6-2}$$

for a given assumed flow rate G. If the right-hand side of equation 6-2 does not agree with the pressure difference between the primary loop and the atmosphere, then the procedure is repeated until equation 6-2 is satisfied to within an acceptable tolerance which in turn leads to flow rate value for a given crack size.

6.4 LEAK RATE CALCULATIONS

Leak rate calculations were made as a function of crack length at the governing locations previously identified in Section 5.1. The normal operating loads of Table 3-1 were applied in these calculations. The crack opening areas were estimated using the method of Reference 6-3, and the leak rates were calculated using the two-phase flow formulation described above. The average material properties of Section 4.0 (see Table 4-5) were used for these calculations.

The flaw sizes to yield a leak rate of 10 gpm were calculated at the governing locations and are given in Table 6-1 for Surry Units 1 and 2. The flaw sizes so determined are called leakage flaw sizes.

The Surry Units 1 and 2 RCS pressure boundary leak detection system meets the intent of Regulatory Guide 1.45, and the plant leak detection capability is 1 gpm. Thus, to satisfy the margin of 10 on the leak rate, the flaw sizes (leakage flaw sizes) are determined which yield a leak rate of 10 gpm.

6.5 REFERENCES

6-1 [

]a,c,e

6-2 M. M, El-Wakil, "Nuclear Heat Transport, International Textbook Company," New York, N.Y, 1971.

6-3 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.

Table 6-1 Flaw Sizes Yielding a Leak Rate of 10 gpm at the Governing Locations

Location	Leakage Flaw Size (in)
1	4.02
3	5.85
6	6.84
15	7.96

Note: The flaw size in the above table refers to the flaw length of through-wall circumferential crack.

a,c,e

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

STAGNATION ENTHALPY (10^2 Btu/lb)

Figure 6-1 Analytical Predictions of Critical Flow Rates of Steam-Water Mixtures

a,c,e

CRITICAL PRESSURE RATIO (p_c/p_o)

LENGTH/DIAMETER RATIO (L/D)

Figure 6-2 [ ]a,c,e Pressure Ratio as a Function of L/D

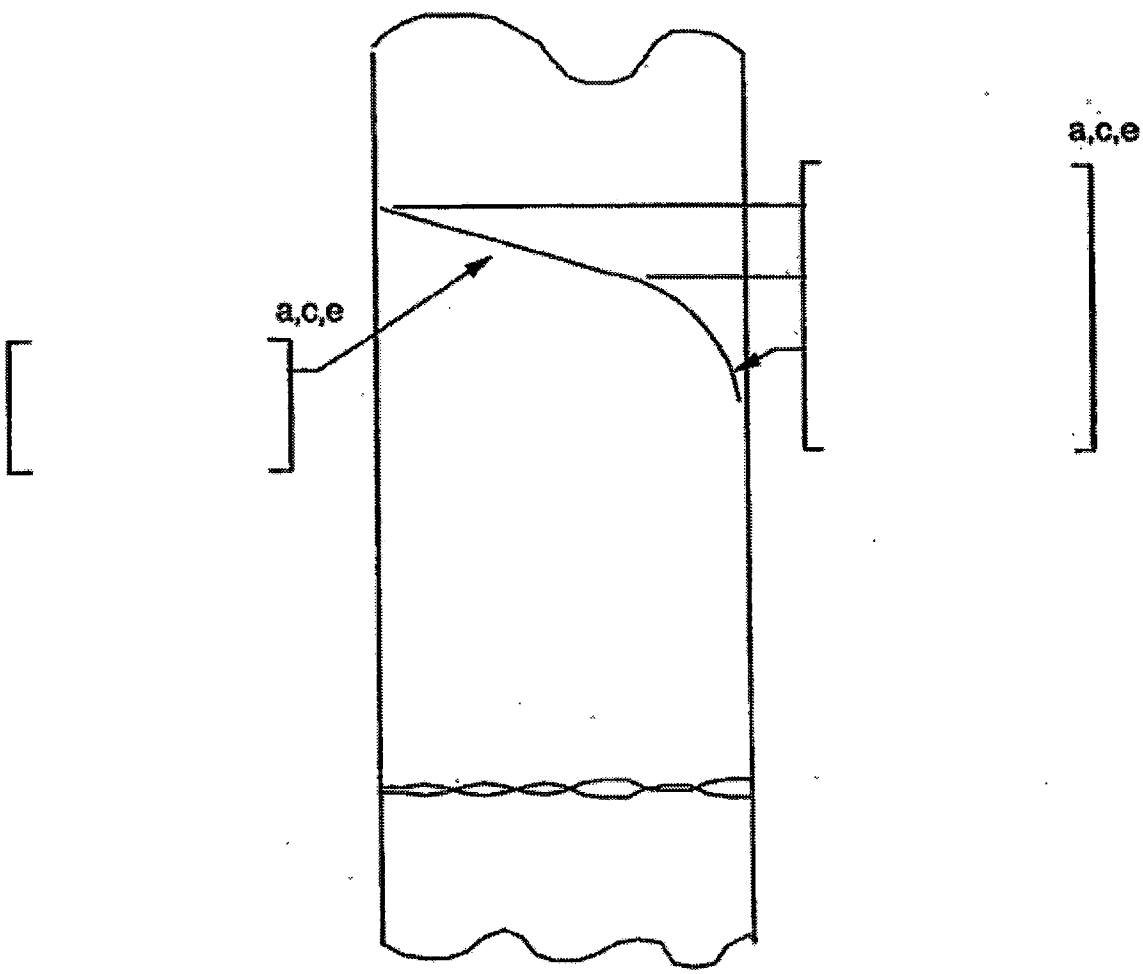


Figure 6-3 Idealized Pressure Drop Profile Through a Postulated Crack

7.0 FRACTURE MECHANICS EVALUATION

7.1 LOCAL FAILURE MECHANISM

The local mechanism of failure is primarily dominated by the crack tip behavior in terms of crack-tip blunting, initiation, extension and final crack instability. The local stability will be assumed if the crack does not initiate at all. It has been accepted that the initiation toughness measured in terms of J_{Ic} from a J-integral resistance curve is a material parameter defining the crack initiation. If, for a given load, the calculated J-integral value is shown to be less than the J_{Ic} of the material, then the crack will not initiate. If the initiation criterion is not met, one can calculate the tearing modulus as defined by the following relation:

$$T_{app} = \frac{dJ}{da} x \frac{E}{\sigma_f^2}$$

where:

T_{app}	=	applied tearing modulus
E	=	modulus of elasticity
σ_f	=	$0.5\ (\sigma_y + \sigma_u)$ = flow stress
a	=	crack length
σ_y, σ_u	=	yield and ultimate strength of the material, respectively

Stability is said to exist when ductile tearing does not occur if T_{app} is less than T_{mat}, the experimentally determined tearing modulus. Since a constant T_{mat} is assumed a further restriction is placed in J_{app}. J_{app} must be less than J_{max} where J_{max} is the maximum value of J for which the experimental T_{mat} is greater than or equal to the T_{app} used.

As discussed in Section 5.2 the local crack stability criteria is a two-step process:

(1) If $J_{app} < J_{Ic,}$ then the crack will not initiate and the crack is stable;

(2) If $J_{app} \geq J_{Ic}$; and $T_{app} < T_{mat}$ and $J_{app} < J_{max}$, then the crack is stable.

7.2 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 plastic instability method, based on traditional plastic limit load concepts, but accounting for strain hardening 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 termed as the flow stress. The flow stress is generally taken as the average of the yield and ultimate tensile strength of the material at the temperature of interest. This methodology has been shown to be applicable to ductile piping through a large number of experiments and will be used here to predict the critical flaw size in the primary coolant piping. The failure criterion has been obtained by requiring equilibrium of the section containing the flaw (Figure 7-1) 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

σ_f = $0.5\,(\sigma_y + \sigma_u)$ = flow stress, psi

[

]a,c,e

The analytical model described above accurately accounts for the piping 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 7-1). For application of the limit load methodology, the material, including consideration of the configuration, must have a sufficient ductility and ductile tearing resistance to sustain the limit load.

7.3 CRACK STABILITY EVALUATIONS

Local Failure Mechanism:

[

]a,c,e

Global failure mechanism:

A stability analysis based on limit load was performed for all the critical locations (locations 1, 3, 6 and 15) as described in Section 7.2. The field welds are made of GTAW and SMAW combination weld. The shop welds are made of GTAW, SMAW or SAW combination weld. Field welds are at the Critical locations 1, 3 and 15. Shop weld is at critical location 6. The "Z" factor correction for SMAW was applied (References 7-5 and 7-6) at the field weld critical locations (locations 1, 3 and 15) and the "Z" factor correction for SAW was applied (References 7-5 and 7-6) at the shop weld location (location 6) and the equations as follows:

$$Z = 1.15 [1.0 + 0.013 (OD-4)] \qquad \text{for SMAW}$$

$$Z = 1.30 [1.0 + 0.01 (OD-4)] \qquad \text{for SAW}$$

where OD is the outer diameter of the pipe in inches.

The Z-factors were calculated for the critical locations, using the dimensions given in Table 3-1. The Z factor was 1.599 for locations 1 and 3, 1.72 for location 6 and 1.592 for location 15. The applied loads were increased by the Z factors and plots of limit load versus crack length were generated as shown in Figures 7-2, 7-3, 7-4 and 7-5. Table 7-2 summarizes the results of the stability analyses based on limit load. The leakage flaw sizes are also presented on the same table.

7.4 REFERENCES

7-1 Kanninen, M. F., et. al., "Mechanical Fracture Predictions for Sensitized Stainless Steel Piping with Circumferential Cracks," EPRI NP-192, September 1976.

7-2 Johnson, W. and Mellor, P. B., Engineering Plasticity, Van Nostrand Relmhold Company, New York, (1973), pp. 83-86.

7-3 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.

7-4 Irwin, G. R., "Plastic Zone near a Crack and Fracture Toughness," Proc.7th Sagamore Conference, P. IV-63 (1960).

7-5 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.

7-6 NUREG-0800 Revision 1, March 2007, Standard Review Plan: 3.6.3 Leak-Before-Break Evaluation Procedures.

Table 7-1 Stability Results for Surry Units 1 and 2 Based on Elastic-Plastic J-Integral Evaluations

a,c,e

Table 7-2 Stability Results for Surry Units 1 and 2 Based on Limit Load

Critical Location	Critical Flaw Size (in)	Leakage Flaw Size (in)
1	19.87	4.02
3	34.08	5.85
6	34.81	6.84
15	40.07	7.96

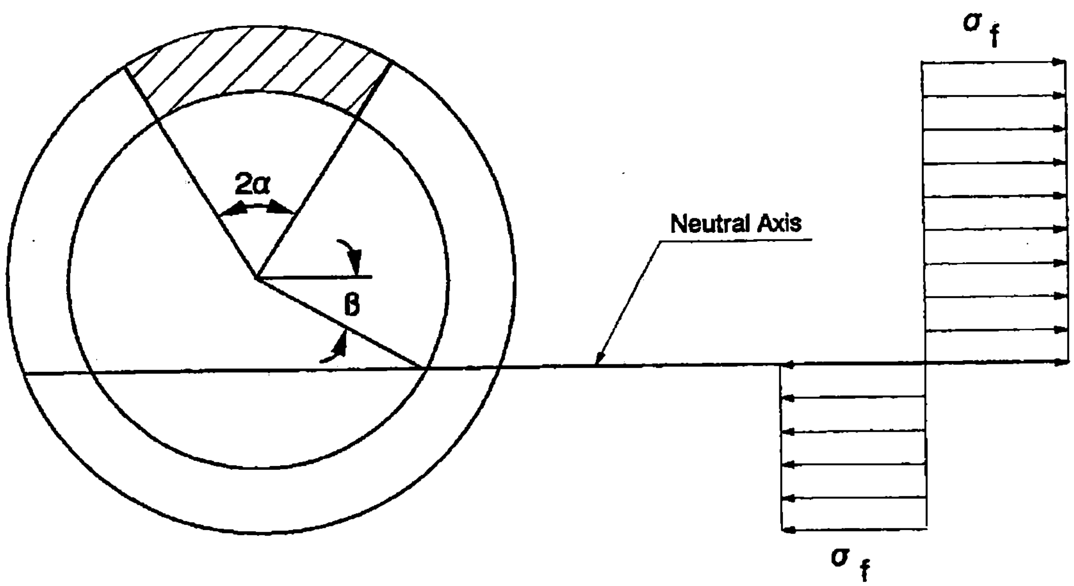
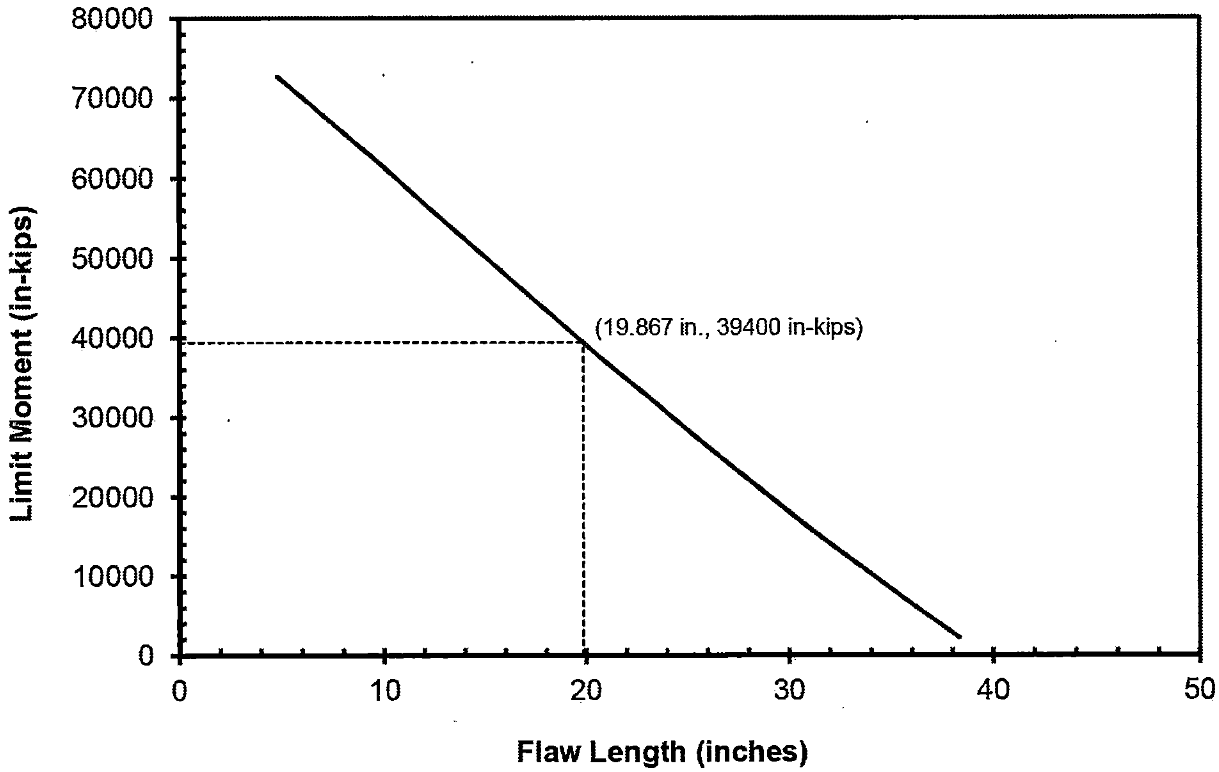
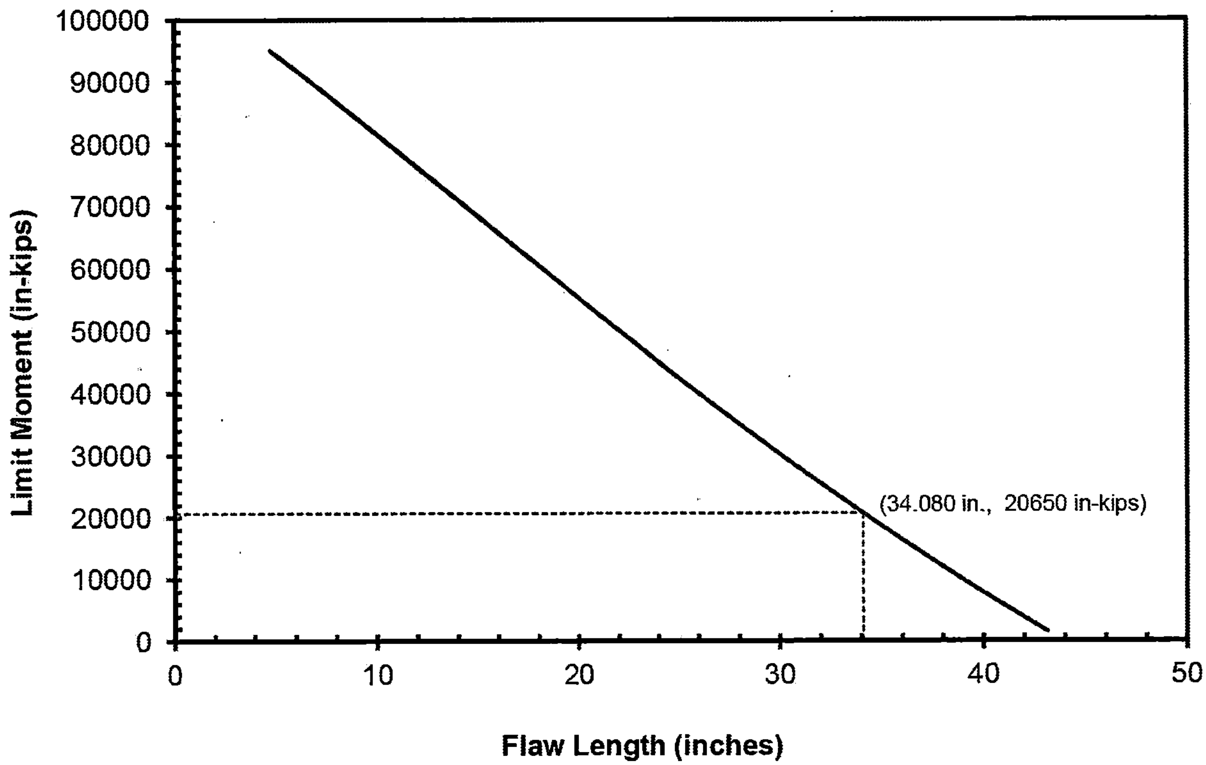


Figure 7-1 [ ]a,c,e Stress Distribution



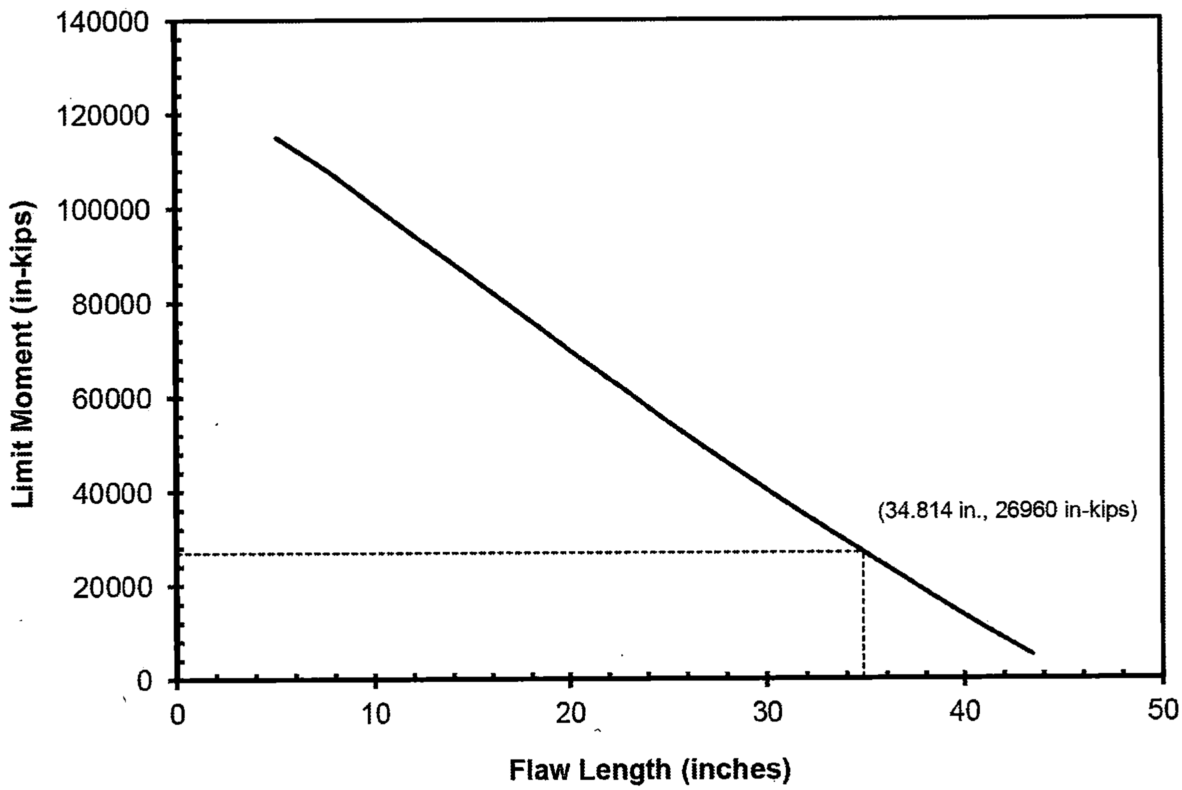
OD = 34.00 in.	σ_y = 20.76 ksi	F = 1640 kips
t = 2.395 in.	σ_u = 56.20 ksi	M = 24646 in-kips
A376-TP316 Material with SMAW Weld		

Figure 7-2 Critical Flaw Size Prediction – Hot Leg at Location 1



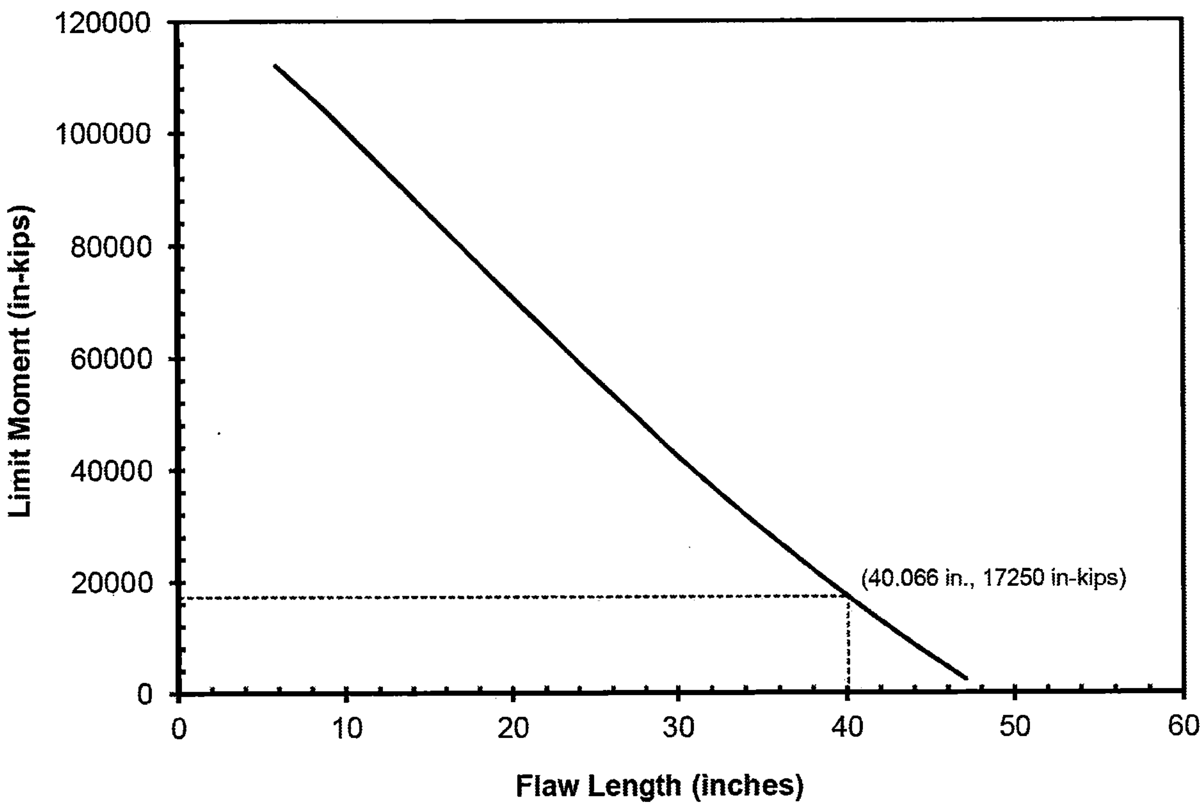
OD = 34.00 in.	σ_y = 23.78 ksi	F = 1639 kips
t = 2.395 in.	σ_u = 71.90 ksi	M = 12918 in-kips
A351-CF8M Material with SMAW Weld		

Figure 7-3 Critical Flaw Size Prediction – Hot Leg at Location 3



OD = 36.32 in.	σ_y = 24.67 ksi	F = 1870 kips
t = 2.555 in.	σ_u = 71.90 ksi	M = 15673 in-kips
A351-CF8M material with SAW Weld		

Figure 7-4 Critical Flaw Size Prediction – Crossover Leg at Location 6



OD = 33.60 in.	σ_y = 24.67 ksi	F = 1418 kips
t = 2.940 in.	σ_u = 71.90 ksi	M = 10829 in-kips
A351-CF8M material with SMAW Weld		

Figure 7-5 Critical Flaw Size Prediction – Cold Leg at Location 15

8.0 FATIGUE CRACK GROWTH ANALYSIS

To determine the sensitivity of the primary coolant system to the presence of small cracks, a fatigue crack growth analysis was carried out for the [ ]a,c,e region of a typical system (see Location [ ]a,c,e of Figure 3-2). This region was selected because crack growth calculated here will be typical of that in the entire primary loop. Crack growths calculated at other locations can be expected to show less than 10% variation.

A [ ]a,c,e of a plant typical in geometry and operational characteristics to any Westinghouse PWR System. [

]a,c,e The normal, upset, and test conditions were considered. A summary of generic applied transients is provided in Table 8-1. Circumferentially oriented surface flaws were postulated in the region, assuming the flaw was located in two different locations, as shown in Figure 8-1. Specifically, these were:

Cross Section A: Stainless Steel

Cross Section B: SA 508 Cl. 2 or 3 Low Alloy Steel

Fatigue crack growth rate laws were used [

]a,c,e The law for stainless steel was derived from Reference 8-1, a compilation of data for austenitic stainless steel in a PWR water environment was presented in Reference 8-2, and it was found that the effect of the environment on the crack growth rate was very small. From this information it was estimated that the environmental factor should be conservatively set at [ ]a,c,e in the crack growth rate equation from Reference 8-1.

For stainless steel, the fatigue crack growth formula is:

[

]a,c,e

The calculated fatigue crack growth for semi-elliptic surface flaws of circumferential orientation and various depths is summarized in Table 8-2, and shows that the crack growth is very small, regardless of which material is assumed.

The reactor vessel transients and cycles for Surry Units 1 and 2 are shown in Table 8-3. By comparing the transients and cycles for the generic analysis shown in Table 8-1 and the Surry plant specific transients and cycles shown in Table 8-3, it is concluded that the generic transients and cycles used for the fatigue crack growth analysis enveloped the Surry transients and cycles. The transients and cycles (shown in Table 8-3) for the Surry plants for 60 years are the same as those of 40 years, and remain applicable for 80 years of operation as well. Also any changes in the cycles for the 80 year design transients will not have a significant impact on the fatigue crack growth conclusions, since there is insignificant growth of small surface flaws as shown in Table 8-2.

It is therefore, concluded that the generic fatigue crack growth analysis shown in Table 8-2 is representative of the Surry plants fatigue crack growth and also applicable for 80 years.

8.1 REFERENCES

8-1 James, L.A. and Jones, D.P., "Fatigue Crack Growth Correlations for Austenitic Stainless Steel in Air, Predictive Capabilities in Environmentally Assisted Cracking," ASME Publication PVP-99, December 1985.

8-2 Bamford, W. H., "Fatigue Crack Growth of Stainless Steel Piping in a Pressurized Water Reactor Environment," Trans. ASME Journal of Pressure Vessel Technology, Vol. 101, Feb. 1979.

Table 8-1 Summary of Reactor Vessel Transients		
Number	**Typical Transient Identification**	**Number of Cycles**
1	Turbine Roll	10
2	Cold Hydro	10
3	Heatup/Cooldown	200
4	Loading and Unloading	18300
5	10% Step Load Decrease/Increase	2000
6	Large Step Decrease	200
7	Steady State Fluctuation	1000000
8	Loss of Load from Full Power	80
9	Loss of Power	40
10	Partial Loss of Flow	80
11	Reactor Trip from Full Power	400
12	Hot Hydro Test	50

Table 8-2 Typical Fatigue Crack Growth at [ ]a,c,e (40, 60, and 80 years)

FINAL FLAW (in.)

a,c,e

Table 8-3 Summary of Reactor Vessel Transients For Surry Units 1 and 2 (40, 60, 80 years)		
Number	**Typical Transient Identification**	**Number of Cycles**
1	Heatup/Cooldown	200
2	Loading and Unloading	18300
3	10% Step Load Decrease/Increase	2000
4	Large Step Decrease	200
5	Reactor Trip from Full Power	400
6	Hot Hydro Test	45

a,c,e

Figure 8-1 Typical Cross-Section of [ ]a,c,e

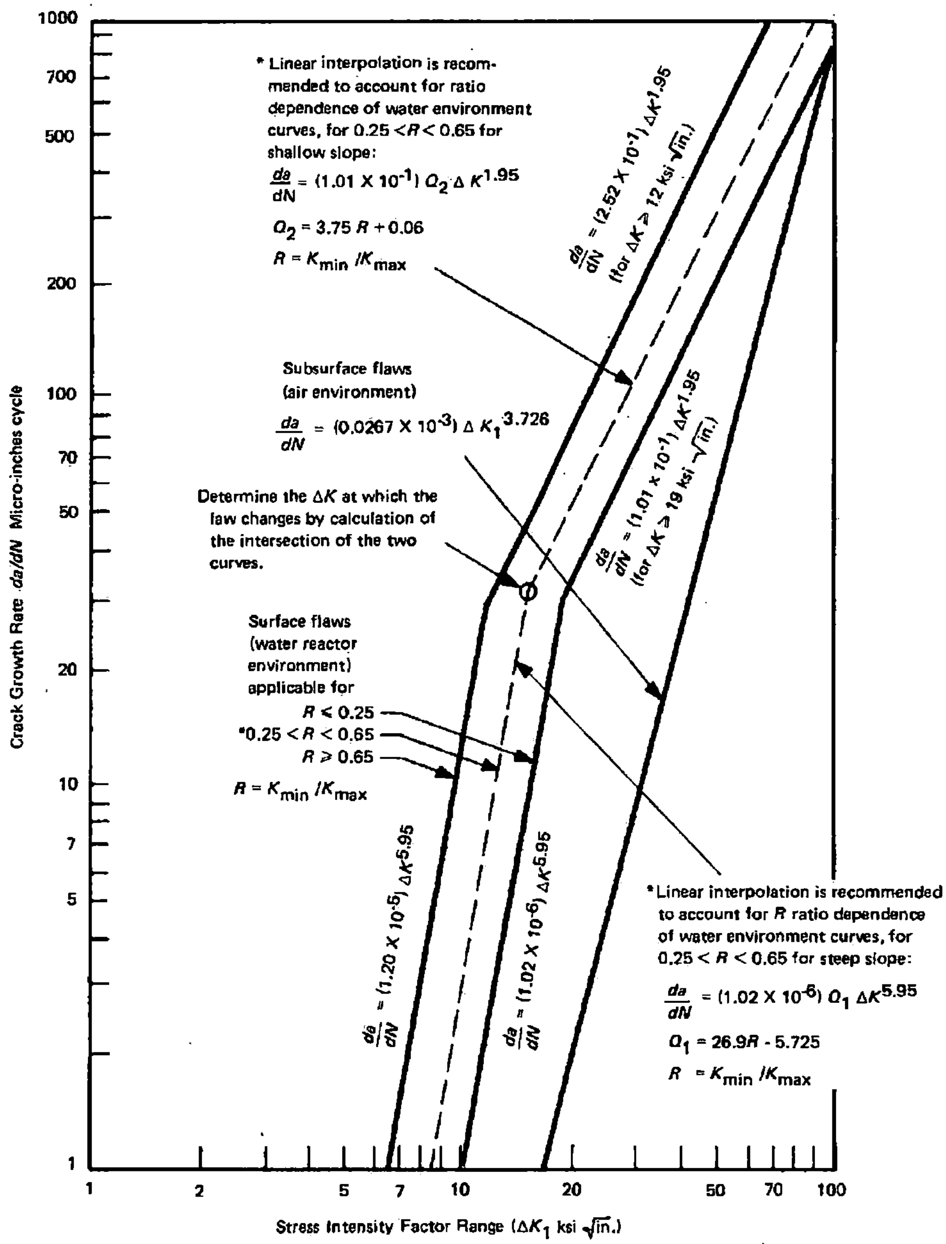


Figure 8-2 Reference Fatigue Crack Growth Curves for Carbon & Low Alloy Ferritic Steels

9.0 ASSESSMENT OF MARGINS

The results of the leak rates of Section 6.4 and the corresponding stability and fracture toughness evaluations of Sections 7.1, 7.2 and 7.3 are used in performing the assessment of margins. Margins are shown in Table 9-1. All of the LBB recommended margins are satisfied.

In summary, at all the critical locations relative to:

1. Flaw Size - Using faulted loads obtained by the absolute sum method, a margin of 2 or more exists between the critical flaw and the flaw having a leak rate of 10 gpm (the leakage flaw).

2. Leak Rate - A margin of 10 exists between the calculated leak rate from the leakage flaw and the plant leak detection capability of 1 gpm.

3. Loads - At the critical locations the leakage flaw was shown to be stable using the faulted loads obtained by the absolute sum method (i.e., a flaw twice the leakage flaw size is shown to be stable; hence the leakage flaw size is stable). A margin of 1 on loads using the absolute summation of faulted load combinations is satisfied.

Table 9-1 Leakage Flaw Sizes, Critical Flaw Sizes and Margins for Surry Units 1 and 2

Location	Leakage Flaw Size	Critical Flaw Size	Margin
1	4.02 in.	19.87[a] in.	4.9[a]
3	5.85 in.	34.08[a] in.	5.8[a]
3	5.85 in.	11.70[b] in.	>2.0[b]
6	6.84 in.	34.81[a] in.	5.1[a]
6	6.84 in.	13.68[b] in.	>2.0[b]
15	7.96 in.	40.07[a] in.	5.0[a]
15	7.96 in.	15.92[b] in.	>2.0[b]

[a]based on limit load
[b]based on J integral evaluation

10.0 CONCLUSIONS

This report justifies the elimination of RCS primary loop pipe breaks from the structural design basis for the 80 year plant life of Surry Units 1 and 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. There is no Alloy 82/182 material present in the welds for the Surry Units 1 and 2 Reactor Coolant System (RCS) primary loop piping.

b. Water hammer should not occur in the RCS piping because of system design, testing, and operational considerations.

c. The effects of low and high cycle fatigue on the integrity of the primary piping are negligible.

d. Ample margin exists between the leak rate of small stable flaws and the capability of the Surry Units 1 and 2 reactor coolant system pressure boundary Leakage Detection System.

e. Ample margin exists between the small stable flaw sizes of item (d) and larger stable flaws.

f. Ample margin exists in the material properties used to demonstrate end-of-service life (fully aged) stability of the critical flaws.

For the critical locations, flaws are identified that will be stable because of the ample margins described in d, e, and f above.

Based on the above, the Leak-Before-Break conditions and margins are satisfied for the Surry Units 1 and 2 primary loop piping. All the recommended margins are satisfied. It is therefore concluded that dynamic effects of RCS primary loop pipe breaks need not be considered in the structural design basis for Surry Units 1 and 2 Nuclear Power Plants for the 80 year plant life (subsequent license renewal program).

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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Reviewer Approval Johnson Eric D Mar-14-2019 12:16:52
Manager Approval Leber Benjamin A Mar-14-2019 12:27:28

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