ENCLOSURE 2

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Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Joseph M. Farley Units 1 and 2 Nuclear Power Plants for the License Renewal Program

Westinghouse

WCAP-12825-NP Addendum 1

Revision 0

Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Joseph M. Farley Units 1 and 2 Nuclear Power Plants for the License Renewal Program

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1.0 INTRODUCTION

1.1 BACKGROUND

A Leak-Before-Break (LBB) evaluation was performed to demonstrate that pipe breaks in the Reactor Coolant Systems (RCS) primary loop piping of the Farley Units 1 and 2 plants need not be considered in the structural design basis. The evaluation was documented in Westinghouse topical report WCAP-12825 (Reference 1-2) and approved by the NRC (Reference 1-3).

Westinghouse also performed a LBB analysis to support steam generator replacement and steam generator snubber elimination that demonstrated continued compliance with LBB acceptance criteria for the Farley Units 1 and 2 reactor coolant loop piping. The analysis results were documented in WCAP-15097 Revision 1 (Reference 1-4).

Westinghouse also performed LBB analyses to demonstrate that pipe breaks in the pressurizer surge line of the Farley Units 1 and 2 plants need not be considered in the structural design basis. The analyses were documented in Westinghouse topical reports WCAP-12835 (Reference 1-5) and WCAP-12835 Supplement 1(Reference 1-6). Since the surge line does not contain any Cast Austenitic Stainless Steel (CASS) and the transients and cycles for 60 year plant life remain the same as those of 40 year plant life, no revision of WCAP-12835 and WCAP-12835 Supplement 1 is required for license renewal.

1.2 OBJECTIVES

The objective of this evaluation is to demonstrate leak-before-break for the primary loops in Farley Units 1 and 2 on a plant specific basis for the 60 year plant life. The recommendations and criteria proposed in Reference 1-7 are used in this evaluation.

This is accomplished by demonstrating the following:

a. An ample margin exists between critical crack size and a postulate crack that yields a detectable leak rate.

b. Sufficient margin exists between the leakage through a postulated crack and the leak detection capability of the plant.

c. Ample margins on applied loads are present.

There is no change in loads in the primary loop piping for the plant life extension program and therefore the evaluation described in this report includes the loads due to the replacement of the Units 1 and 2 Steam Generators and the elimination of the Steam Generator snubbers for Units 1 and 2. The effects of thermal aging degradation of the cast stainless steel material for the 60 year plant life were included in this evaluation.

This report provides a fracture mechanics demonstration of primary loop integrity for the Farley Units 1 and 2 Plants based on the latest LBB methodology and consistent with the NRC position for exemption from consideration of dynamic effects.

1.3 REFERENCES

1-1 WCAP-7211, Revision 4, "Energy Systems Business Unit Policy and Procedures for Management, Classification, and Release of Information," January 2001.

1-2 WCAP-12825, "Technical Justification for Eliminating Large primary Loop Pipe Rupture as the Structural Design Basis for the Joseph M. Farley Units 1 and 2 Nuclear Power Plants," January 1991.

1-3 Nuclear Regulatory Commission Docket #'s 50-348 and 50-364 Letter from Stephen T. Hoffman, Project manager Project Directorate II-1 Division of Reactor projects I/II Office of Nuclear Reactor Regulation, to W. G. Hairston III, Senior Vice President Alabama Power Company, dated August 12, 1991.

1-4 WCAP-15097 Revision 1," Farley Nuclear Plant Units 1 and 2 Replacement Steam Generator Program NSSS Engineering Report, Book 1," March 2001.

1-5 WCAP-12835, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for Farley Units 1 and 2," April 1991.

1-6 WCAP-12835 Supplement 1, " Additional Information in Support of Eliminating Pressurizer Surge Line Rupture from the Structural Design Basis for Farley Units 1 and 2," September 1991.

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.

2.0 LOADS AND STRESSES

The normal operating loads and stresses, the faulted condition loads and stresses used in the original analysis are provided in Tables 3-1 through 3-4 of Reference 1-2. The corresponding loads resulting from the revised configuration (Replacement Steam Generators and the elimination of the Steam Generator snubbers) are provided in Tables 2-1 through 2-4 in this report and are also applicable for the license renewal program.

2.1 NATURE OF THE LOADS

Figure 2-1 shows schematic layout of the Farley Units 1 and 2 primary loop piping and identifies the weld locations. The stresses due to axial loads and moments were calculated by the following equation:

$$\sigma = \frac{F}{A} + \frac{M}{Z} \qquad (2\text{-}1)$$

where,

σ	=	Stress
F	=	Axial Load
M	=	Moment
A	=	Metal Cross-Sectional Area
Z	=	Section Modulus

The moment for the desired loading combination was calculated by the following equation:

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

where,

M	=	Moment for Required Loading
M_Y	=	Y Component of Bending Moment
M_Z	=	Z Component of Bending Moment

The axial load and moments for crack stability analysis and leak rate predictions are computed by the methods to be explained in Sections 2.2 and 2.3.

2.2 LOADS FOR CRACK STABILITY ANALYSIS

In accordance with the Standard Review Plan 3.6.3 (Reference 1-7), the absolute sum of loading components can be applied which results in higher magnitude of combined loads. If

crack stability is demonstrated using these loads, the LBB margin on loads can be reduced from √2 to 1. The faulted loads for the crack stability analysis were calculated by the absolute sum method as follows:

$$F = |F_{DW}| + |F_{TH}| + |F_p| + |F_{SSE}| \quad (2\text{-}3)$$

$$M_Y = |M_{Y\,DW}| + |M_{Y\,TH}| + |M_{Y\,SSE}| \quad (2\text{-}4)$$

$$M_Z = |M_{Z\,DW}| + |M_{Z\,TH}| + |M_{Z\,SSE}| \quad (2\text{-}5)$$

where

DW = Deadweight

TH = Normal Thermal expansion

P = Load Due To Internal Pressure

SSE = SSE Loading Including Seismic Anchor Motion

2.3 LOADS FOR LEAK RATE EVALUATION

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

$$F = F_{DW} + F_{TH} + F_p \quad (2\text{-}6)$$

$$M_Y = M_{Y\,DW} + M_{Y\,TH} \quad (2\text{-}7)$$

$$M_Z = M_{Z\,DW} + M_{Z\,TH} \quad (2\text{-}8)$$

The parameters and subscripts are the same as those explained in Section 2.2.

Loads shown in Tables 2-1 through Table 2-4 envelope the Replacement Steam Generators for Units 1 and 2 and the elimination of the Steam Generator snubbers for Units 1 and 2. All the weld locations are identified in Figure 2-1. The operating parameters shown in Figure 2-1 are obtained from References 2-1 and 2-2.

2.4 REFERENCES

2-1 PCWG-2742," Farley Unit 2 (APR): Category IIIP (for Limited Scope Contract) Approval of PCWG Parameters to Support Upflow Conversion Program," February 28, 2002 (Westinghouse Proprietary).

2-2 PCWG-2719," Farley Units 1 & 2 (ALA/APR): Approval of Category IV PCWG Parameters to Support Uprate Program," December 11, 2001 (Westinghouse Proprietary).

Table 2-1 Dimensions, Normal Loads and Normal Stresses for Farley Unit 1					
Location*	Outside Diameter (in)	Minimum Thickness (in)	Axial Load** (kips)	Bending Moment (in-kips)	Total Stress (ksi)
1	33.78	2.28	1,554	24,380	21.53
2	33.78	2.28	1,554	12,218	14.26
3	36.96	2.88	1,917	20,403	14.58
4	36.76	2.88	1,775	6,583	8.52
5	36.05	2.42	1,811	4,991	9.56
6	36.05	2.42	1,812	4,652	9.40
7	36.05	2.42	1,716	1,534	7.47
8	36.05	2.42	1,720	3,184	8.31
9	37.16	2.98	1,627	8,078	8.27
10	32.03	2.16	1,339	7,689	12.02
11	32.03	2.16	1,339	4,896	10.06
12	32.03	2.56	1,269	4,471	8.12

* See Figure 2-1
** Includes Pressure

Table 2-2 Faulted Loads and Stresses for Farley Unit 1			
Location*,**	Axial Load*** (kips)	Bending Moment (in-kips)	Total Stress (ksi)
1	1,804	31,586	26.96
2	1,812	21,218	20.77
3	2,056	30,824	19.30
4	1,823	13,582	11.58
5	1,825	10,135	12.17
6	1,851	6,226	10.33
7	1,776	4,293	9.08
8	1,774	6,211	10.02
9	1,847	12,230	10.60
10	1,456	11,171	15.05
11	1,440	7,370	12.30
12	1,358	6,573	9.79

* See Figure 2-1

** See Table 2-1 for dimensions

*** Includes Pressure

Table 2-3 Dimensions, Normal Loads and Normal Stresses for Farley Unit 2					
Location*	Outside Diameter (in)	Minimum Thickness (in)	Axial Load** (kips)	Bending Moment (in-kips)	Total Stress (ksi)
1	33.81	2.30	1,554	24,380	21.33
2	33.81	2.30	1,543	12,281	14.09
3	36.20	2.50	1,917	20,403	17.02
4	36.20	2.50	1,775	6,583	9.86
5	36.11	2.45	1,811	4,491	9.43
6	36.11	2.45	1,812	4,660	9.28
7	36.11	2.45	1,716	1,534	7.37
8	36.11	2.45	1,720	3,184	8.20
9	37.52	3.16	1,616	8,078	7.72
10	32.07	2.18	1,330	7,689	11.86
11	32.07	2.18	1,339	4,896	9.96
12	32.14	2.22	1,336	4,471	9.46

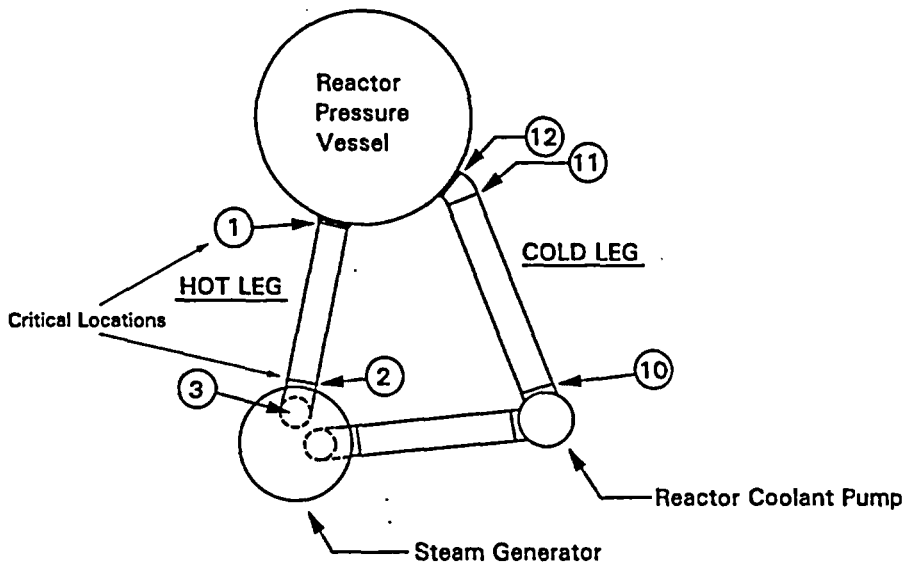
* See Figure 2-1
** Includes Pressure

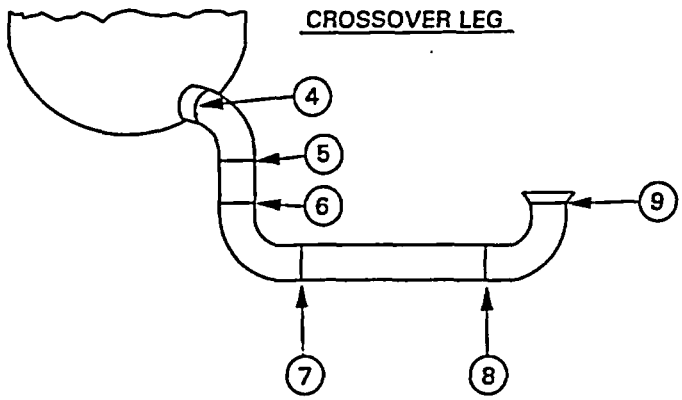
Table 2-4 Faulted Loads and Stresses for Farley Unit 2			
Location*,**	Axial Load*** (kips)	Bending Moment (in-kips)	Total Stress (ksi)
1	1,804	31,586	26.72
2	1,801	21,218	20.54
3	2,056	30,824	22.53
4	1,823	13,582	13.39
5	1,825	10,135	12.01
6	1,851	6,236	10.20
7	1,776	4,293	8.96
8	1,774	6,211	9.89
9	1,836	12,230	9.90
10	1,446	11,171	14.86
11	1,440	7,370	12.18
12	1,426	6,573	11.33

* See Figure 2-1

** See Table 2-3 for dimensions

*** Includes Pressure





HOT LEG

Temperature 613.3°F, Pressure: 2250 psia

CROSS-OVER LEG

Temperature 540.8°F, Pressure: 2250 psia

COLD LEG

Temperature 541.1°F, Pressure: 2250 psia

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

3.0 MATERIAL CHARACTERIZATION

3.1 PRIMARY LOOP PIPE, FITTINGS MATERIALS AND WELD PROCESS

The primary loop piping material for both Farley Unit 1 and Farley Unit 2 is SA351 CF8A. The elbow fittings for Farley Unit 1 are SA351 CF8M, while for Farley Unit 2, they are SA351 CF8A. The field welds are SMAW following GTAW root passes. The shop welds are SAW.

3.2 TENSILE PROPERTIES

The piping Certified Materials Test Reports (CMTRs) for Farley Units 1 and 2 were used to establish the tensile properties for the Leak-Before-Break analysis. 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 Tables 4-1, 4-2 of Reference 3-1 for Farley Units 1 and 2 respectively. Mechanical properties for Farley Unit 1 material at the operating temperatures are shown in Table 4-3 of Reference 3-1. Mechanical properties for Farley Unit 2 material at the operating temperatures are shown in Table 4-4 of Reference 3-1. Mechanical properties for Farley Units 1 and 2 material at the operating temperature for the critical locations for the current evaluation are shown in Table 3-1 and 3-2 and they are calculated using the information from Tables 4-3 and 4-4 of Reference 3-1 and Reference 3-2.

The average and lower bound yield strengths and lower bound ultimate strengths are given in Tables 3-1 and 3-2. The ASME code moduli of elasticity are also given in these Tables, and poisson's ratio was taken as 0.3.

3.3 FRACTURE TOUGHNESS PROPERTIES

The pre-service fracture toughnesses of cast stainless steels in terms of J_{Ic} (J at Crack Initiation) have been found to be very high at 600°F. 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 end of life fracture toughness values calculated by Westinghouse methodology and shown in WCAP-12825 (Reference 3-1) were conservative and were not used for this current evaluation, an alternate method as described below was used to calculate the end of life toughness properties for the cast material.

In 1994, the Argonne National Laboratory (ANL) completed an extensive research program in assessing the extent of thermal aging of cast stainless steel materials. The ANL research

program measured mechanical properties of cast stainless steel materials after they have 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). From this database, ANL developed correlations for estimating the extent of thermal aging of cast stainless steel (References 3-3 and 3-4).

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 ANL procedures produced conservative estimates that were about 30 to 50 percent less than actual measured values. The procedure developed by ANL in Reference 3-4 was used to calculate the end of life fracture toughness values for this analysis. ANL research program was sponsored and the procedure was accepted (Reference 3-5) by the NRC.

The chemical compositions are available from CMTRs and are provided in Appendix B of Reference 3-1. The following equations are taken from Reference 3-4.

$$Cr_{eq} = Cr + 1.21(Mo) + 0.48(Si) - 4.99 \quad (3\text{-}1)$$

$$Ni_{eq} = (Ni) + 0.11(Mn) - 0.0086(Mn)^2 + 18.4(N) + 24.5(C) + 2.77 \quad (3\text{-}2)$$

where Cr_{eq} and Ni_{eq} are in percent weight

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

δ_c, ferrite content is in percent volume.

For CF 8 steel the saturation value of RT impact energy C_{vsat} (J/cm^2) is the lower value determined from

$$\log_{10} C_{Vsat} = 1.15 + 1.36 \exp(-0.035\phi) \quad (3\text{-}4)$$

where the material parameter ϕ is expressed as

$$\phi = \delta_c (Cr + Si)(C + 0.4N) \quad (3\text{-}5)$$

and from

$$\log_{10} C_{Vsat} = 5.64 - 0.006\delta_c - 0.185Cr + 0.273Mo - 0.204Si + 0.044Ni - 2.12(C + 0.4N) \quad (3\text{-}6)$$

For CF 8M steel with <10% Ni the saturation value of RT impact energy C_{vsat} (J/cm^2) is the lower value determined from

$$\log_{10}C_{Vsat} = 1.10 + 2.12 \exp(-0.041\phi) \quad (3\text{-}7)$$

where the material parameter ϕ is expressed as

$$\phi = \delta_c (Ni + Si + Mn)^2 (C + 0.4N)/5; \quad (3\text{-}8)$$

and from

$$\log_{10}C_{Vsat} = 7.28 - 0.011\delta_c - 0.185Cr - 0.369Mo - 0.451Si - 0.007Ni - 4.71(C + 0.4N) \quad (3\text{-}9)$$

For CF 8M steel with >10% Ni, the saturation value of RT impact energy C_{vsat} (J/cm^2) is the lower value determined from

$$\log_{10}C_{Vsat} = 1.10 + 2.64 \exp(-0.064\phi) \quad (3\text{-}10)$$

where the material parameter ϕ is expressed as

$$\phi = \delta_c (Ni + Si + Mn)^2 (C + 0.4N)/5 \quad (3\text{-}11)$$

and from

$$\log_{10}C_{Vsat} = 7.28 - 0.011\delta_c - 0.185Cr - 0.369Mo - 0.451Si - 0.007Ni - 4.71(C + 0.4N) \quad (3\text{-}12)$$

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 Appendix B of Reference 3-1and also provided in Table 3-3 and 3-4 of this report.

The saturation J-R curve at 290°C (554°F), for static-cast CF 8 steel is given by

$$J_d = 102 (C_{Vsat})^{0.28} (\Delta a)^n \quad (3\text{-}13)$$

$$n = 0.21 + 0.09 \log_{10} (C_{Vsat}) \quad (3\text{-}14)$$

The saturation J-R curve at 290°C (554°F), for static-cast CF 8M steel is given by

$$J_d = 49 (C_{Vsat})^{0.41} (\Delta a)^n \quad (3\text{-}15)$$

$$n = 0.23 + 0.06 \log_{10} (C_{Vsat}) \quad (3\text{-}16)$$

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

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The correlation presented in Reference 3-4 is applicable to cast stainless steels used in the U. S. Nuclear Industry, the steels contain <25% ferrite in almost all cases. [

]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 strength 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 3-5 will be used as the criteria against which the applied fracture toughness values will be compared.

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

3.4 REFERENCES

3-1 WCAP-12825, "Technical Justification for Eliminating Large primary Loop Pipe Rupture as the structural Design basis for the Joseph M. Farley Units 1 and 2 Nuclear Power Plants," January 1991.

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

3-3 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.

3-4 O. K. Chopra, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," NUREG-CR-4513, Revision 1, U. S. Nuclear Regulatory Commission, Washington, DC, August 1994.

3-5 "Flaw Evaluation of Thermally aged Cast Stainless Steel in Light-Water Reactor Applications," Lee, S.; Kuo, P. T.; Wichman, K.; Chopra, O.; Published in International Journal of Pressure Vessel and Piping, June 1997.

Table 3-1 Mechanical Properties for Farley Unit 1 Materials at Operating Temperatures				
Material	Temperature* (°F)	Average Yield Strength (psi)	Lower Bound	
			Yield Stress (psi)	Ultimate Strength (psi)
A351 CF8A	614	24,642	21,436	66,470
A351 CF8M	614	26,025	21,649	52,200
Modulus of Elasticity				
E = 25.23x 10^6 psi, at 614°F				
Poisson's ratio: 0.3				

Table 3-2 Mechanical Properties for Farley Unit 2 Materials at Operating Temperatures				
Material	Temperature* (°F)	Average Yield Strength (psi)	Lower Bound	
			Yield Stress (psi)	Ultimate Strength (psi)
A351 CF8A	614	22,575	20,217	66,600
Modulus of Elasticity				
E = 25.23x 10^6 psi, at 614°F				
Poisson's ratio: 0.3				

Note: * Actual temperature is 613.3°F. For analysis used 614°F.

a,c,e

Notes:

*A351 CF8A material; **A351 CF8M material; [1]From Equations 3-4, or 3-7, 3-10; [2]From Equations 3-6 or 3-9, 3-12; [3]Minimum of C_{Vsat}^{1} and C_{Vsat}^{2} ; N is assumed as 0.05

a,c,e

Notes: All A351 CF8A material; [1]From Equation 3-4; [2]From Equation 3-6; [3] Minimum of C_{Vsat}^{1} and C_{Vsat}^{2} ; N is assumed as 0.05

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4.0 CRITICAL LOCATIONS AND EVALUATION CRITERIA

4.1 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 in Section 2.0 and the material properties established in Section 3.0. These locations are defined below for Farley Units 1 and 2. Table 2-2, Table 2-4 as well as Figure 2-1 are used for this evaluation.

Critical Locations

Unit 1:

The highest stressed location for the straight pipe with SA351 CF8A material is at Location 1 (in the Hot Leg) (See Figure 2-1) at the reactor vessel outlet nozzle to pipe weld. The highest stressed location for the elbows with SA351 CF8M material is at Location 2 (in the Hot Leg) (See Figure 2-1). Locations 1 and 2 are the critical locations for all the weld locations in the primary loop piping.

Unit 2:

The highest stressed location for the straight pipe and the elbows with SA351 CF8A material is at location 1 (in the hot leg) (see figure 2-1) at the reactor vessel outlet nozzle to pipe weld. Location 1 is the critical locations for all the weld locations in the primary loop piping.

4.2 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;

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

Where:

J_{app} = Applied J; J_{IC}=J at Crack Initiation; J_{max}=Maximum J value of the material

T_{app} = Applied Tearing Modulus; T_{mat}=Material Tearing Modulus

For critical locations, the limit load method discussed in Section 6.1 was also used.

5.0 LEAK RATE PREDICTIONS

5.1 LEAK RATE CALCULATIONS

Leak rate calculations were made as a function of crack length at the critical locations previously identified in Section 4.1. The normal operating loads of Table 2-1 and Table 2-3 were applied, in these calculations. The leak rates were calculated using the same methodology as described in section 6 of Reference 3-1. The average material properties of Section 3.0 (see Tables 3-1 and 3-2) were used for these calculations.

The flaw sizes that yield a leak rate of 10 gpm were calculated at the governing locations and are given in Table 5-1. The flaw sizes so determined are called leakage flaw sizes.

The Farley Units 1 and 2 RCS pressure boundary leak detection system meets the intent of Reg. Guide 1.45, which is 1 gpm in 1 hour or less. 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.

Additional leak rate calculations were performed for the Alloy 82/182 weld at location 1 and the results are shown at the bottom of Table 5-1.

Table 5-1 Flaw Sizes Yielding a Leak Rate of 10 gpm at the Governing Locations	
Location*	**Leakage Flaw Size (in)**
1 Unit 1	3.28
2 Unit 1	4.96
1 Unit 2	3.14

* [ ] a,c,e
[ ] a,c,e

6.0 FRACTURE MECHANICS EVALUATION

6.1 RESULTS OF CRACK STABILITY EVALUATION

J-integral calculation results

J-integral stability analyses were performed at the critical locations established in section 4.1. The elastic-plastic fracture mechanics (EPFM) J-integral analyses for through-wall circumferential flaws were performed using the same methodology of Section 7.1 of Reference 3-1.

The lower-bound material properties from Table 3-1 and Table 3-2 were applied. The fracture toughness properties established in Section 3.3 (shown in Table 3-5) and the normal plus SSE loads given in Table 2-2 and Table 2-4 were used for EPFM calculations. The postulated flaw sizes were twice those giving a leak rate of 10 gpm as established in section 5.0 (see Table 5-1). Evaluations were performed at the critical locations identified in section 4.1. The results of the EPFM J-Integral evaluations are provided in Table 6-1. It can be seen that the fracture criteria are satisfied at all the critical locations. Specifically a margin of 2 on flaw size is demonstrated. Since the faulted loads are combined by absolute summation method, the required margin on load of 1.0 is also accomplished as described in SRP 3.6.3(Reference 1-7).

Fracture criteria as described in section 4.2 are satisfied

Limit Load Results

At the critical locations Limit Load analysis was performed with the same methodology of Section 7.2 of Reference 3-1. The 'Z' factor correction was applied in the limit load calculations. The applied loads were increased by the 'Z' factor and a plot of Limit load versus crack length was generated as shown in Figures 6-1 through 6-3. Table 6-2 summarizes the results of the stability analyses based on limit load. The leakage size flaws are also presented on the Table 6-2.

Alloy 82/182 weld was used in the reactor vessel inlet (weld location 12) and outlet nozzle (weld location 1) locations. Location 1 governs with higher faulted stress than location 12. Alloy 82/182 weld toughness does not degrade due to the thermal aging. For the Alloy 82/182 welds the 'Z' factor is 1.0. The critical flaw size(s) and the leakage flaw size(s) for Alloy82/182 are shown at the bottom of Table 6-2.

Table 6-1 Stability Results for Farley Units 1 and 2 Based on Elastic J-Integral Evaluations a,c,e

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Table 6-2 Stability Results for Farley Units 1 and 2 Based on Limit Load

Location	Critical Flaw Size (in)	Leakage Flaw Size (in)
1 Unit 1*	16.36	3.28
2 Unit 1	16.11	4.96
1 Unit 2**	16.25	3.14

* [ ]a,c,e

** [ ]a,c,e

a,c,e

[ ]

Flaw Length (inches)

OD = 33.78 in	σ_y = 21.44 ksi	F = 1804 kips
t = 2.28 in	σ_u = 66.47 ksi	M = 31586 in-kips

SA351 CF8A with SMAW weld

Figure 6-1 Critical Flaw Size Prediction at Hot Leg at Location 1 (Unit 1)

a,c,e

[ ]

Flaw Length (inches)

OD = 33.78 in	**σ_y = 21.65 ksi**	**F = 1812 kips**
t = 2.28 in	**σ_u = 52.20 ksi**	**M = 21218 in-kips**

SA351 CF8M with SAW weld

Figure 6-2 Critical Flaw Size Prediction at Hot Leg Location 2 (Unit 1)

a,c,e

[ ]

Flaw Length (inches)

OD = 33.81 in	σ_y = 20.22 ksi	F = 1804 kips
t = 2.30 in	σ_u = 66.60 ksi	M = 31586 in-kips

SA351 CF8A with SMAW weld

Figure 6-3 Critical Flaw Size Prediction at Hot Leg Location 1 (Unit 2)

7.0 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, Location [ ]a,c,e of Figure 2-1. This region was selected because crack growth calculated here would be typical of that in the entire primary loop. Crack growth calculated at other locations would be expected to show less than a 10% variation.

A [ ]a,c,e of a 3 loop plant typical in geometry and operational characteristics to any Westinghouse PWR system. The dimensions (see Table 2-1) for the Farley Unit 1 inlet nozzle are 32.03 inches in diameter and a 2.56-inch wall thickness and for the Farley Unit 2 inlet nozzles the dimensions (see Table 2-3) are 32.14 inches in diameter and a 2.22-inch wall thickness. The nozzle dimensions are also shown in Figure 7-1. The fatigue crack growth analysis performed in this report was for the Farley Unit 1 and 2 plant specific geometry, transients and cycles. The fatigue crack growth analysis documented in WCAP-12825 (Reference 3-1) was for a generic analysis.

The normal, upset, and test conditions were considered. A summary of the applicable applied transients (Reference 7-1) is provided in Table 7-1. Circumferentially oriented surface flaws were postulated in the region, assuming the flaw was located in three different locations, as shown in Figure 7-1. Specifically, these were:

Cross Section A: [ ]a,c,e

Cross Section B: [ ]a,c,e

Cross Section C: [ ]a,c,e

Fatigue crack growth rate laws were used [

]a,c,e The law for stainless steel was derived from Reference 7-2, a compilation of data for austenitic stainless steel in a PWR water environment was presented in Reference 7-3, 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 7-2.

For stainless steel, the fatigue crack growth formula is:

[

]a,c,e

[

]a,c,e

[ ]a,c,e

where: [ ]a,c,e
The unit for crack growth rate da/dn is in equation is inches per cycle, and the unit for K_{eff} is ksi√in

where: ΔK is the stress intensity factor range.

The calculated fatigue crack growth for semi-elliptic surface flaws of circumferential orientation and various depths is summarized in Table 7-2 and Table 7-3, and shows that the crack growth for 60 year is very small, [ ]a,c,e and therefore the FCG is not a concern for the Farley Units 1 and 2 primary loop piping.

The transients and cycles (shown in Table 7-1) for the Farley plants for 40 years are the same as those for 60 years. It is therefore, concluded that the fatigue crack growth analysis shown in Table 7-2 and Table 7-3 are applicable for 40 years as well as 60 years plant life.

7.1 REFERENCES

7-1 WCAP-15097 Revision 1," Farley Nuclear Plant Units 1 and 2 Replacement Steam Generator Program NSSS Engineering Report, Book 2," March 2001.

7-2 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.

7-3 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.

7-4 James, L. A., "Fatigue Crack Propagation Behavior of Inconel 600," in Hanford Engineering Development Labs Report HEDL-TME-76-43, May 1976.

7-5 Hale, D. A., et al., "Fatigue Crack Growth in Piping and RPV Steels in Simulated BWR Water Environment," Report GEAP 24098/NUREG CR-0390, Jan. 1978.

Table 7-1 Summary of Farley Units 1 and 2 Transients		
Number	**Transient Identification**	**Number of Cycles**
	Normal conditions and Upset Conditions	
1	Heat up/Cool Down at 100° F/hr (pressurizer cool down 200° F/hr	200
2	Load Follow Cycles (Unit loading and unloading at 5% of full power/min.)	18300
3	Step load increase and decrease	2000
4	Large step load decrease, with steam dump	200
5	Steady state fluctuation	Infinite *
6	Loss of load, without immediate turbine or reactor trip	80
7	Loss of power (blackout with natural circulation in the Reactor Coolant System)	40
8	Loss of Flow (partial loss of flow, one pump only)	80
9	Reactor Trip from full power	400
10	Turbine roll test	10
11	Hydrostatic test conditions, Primary side	5
	Hydrostatic test conditions, Primary side leak test	50
12	Cold Hydrostatic test	10
13	Feedwater/ Cycling/Hot Standby operation	2000
14	Inadvertent Auxiliary Pressurizer Spray	10
15	Operating Basis Earthquake (OBE)	5

* $3x10^6$ cycles were used for the FCG analysis

Table 7-2 Fatigue Crack Growth at [ ]a,c,e (40 and 60 years) for Farley Unit 1			
FINAL FLAW (in.)			
[			
			]a,c,e

Table 7-3 Fatigue Crack Growth at [ ]a,c,e (40 and 60 years) for Farley Unit 2			
FINAL FLAW (in.)			
[			
			]a,c,e

a,c,e

[ ]

PLANT	Thickness* (inches)	Radius** (inches)
Farley Unit 1	2.56	13.46
Farley Unit 2	2.22	13.85

Figure 7-1. Cross-Section of [ ] a,c,e

a,c,e

[ ]

Figure 7-2 Reference Fatigue Crack Growth Curves for [ ]a,c,e

[ ] a,c, e

Figure 7-3 Reference Fatigue Crack Growth Law for [ ]a,c,e in a Water Environment at 600^{0}F

8.0 ASSESSMENT OF MARGINS

The results of the leak rates of Section 5.1 and the corresponding stability and fracture toughness evaluations of Section 6.1 are used in performing the assessment of margins. Margins are shown in Table 8-1.

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 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 on loads of 1(see Section 2.2 for explanation) using the absolute summation of faulted load combinations is satisfied. This satisfied the requirement of action item 10 of the NRC FSER (Final Safety Evaluation Report) for WCAP-14575-A (Reference 8-1) for margin on loads.

   Note: No CASS (Cast Austenitic Stainless Steel) material was replaced for Farley Units 1 and 2 primary loop piping and therefore second component of action item 10 of the NRC FSER is not applicable for Farley Units 1 and 2 primary loop piping.

8.1 REFERENCE

8-1 WCAP-14575-A,"Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components," December 2000.

Table 8-1 Leakage Flaw Sizes, Critical Flaw Sizes and Margins for Farley Units 1 and 2			
Location	**Leakage Flaw Size**	**Critical Flaw Size**	**Margin**
1 Unit 1*	3.28 in.	16.36$^{\alpha}$ in.	5$^{\alpha}$
1 Unit 1	3.28 in.	6.56 $^{\beta}$ in.	>2$^{\beta}$
2 Unit 1	4.96 in.	16.11$^{\alpha}$ in.	3$^{\alpha}$
2 Unit 1	4.96 in.	9.92$^{\beta}$ in.	>2$^{\beta}$
1 Unit 2 **	3.14 in.	16.25$^{\alpha}$ in.	5$^{\alpha}$
1 Unit 2	3.14 in.	6.28$^{\beta}$ in.	>2$^{\beta}$

* [ ]^{a,c,e}

** [ ]^{a,c,e}

$^{\alpha}$based on limit load

$^{\beta}$ based on J integral evaluation

9.0 CONCLUSIONS

This report justifies the elimination of RCS primary loop pipe breaks from the structural design basis for the 60 year plant life of Joseph M. Farley Units 1 and 2 as follows:

1) 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 (for discussions see Section 2.1 of Reference 1-2).

   Note: Currently an ERPI MRP program is underway to address the Alloy 82/182 PWSCC (Primary Water Stress Corrosion Cracking) issue for the industry due to the V. C. Summer cracking incident.

2) Water hammer should not occur in the RCS piping because of system design, testing, and operational considerations (for discussions see Section 2.2 of Reference 1-2).

3) The effects of low and high cycle fatigue on the integrity of the primary piping are negligible (for discussions see Section 2.3 of Reference 1-2).

4) Ample margin exists between the leak rate of small stable leakage flaws and the capability of the Joseph M. Farley Units 1 and 2 reactor coolant system pressure boundary Leakage Detection System.

5) Ample margin exists between the small stable leakage flaw sizes of item d and the larger critical stable flaws.

6) Ample margin exists in stability using the end of life (60 year) thermal aging material properties.

For the critical locations, flaws are identified that will be stable because of the ample margins described in items 4, 5 and 6 above.

Based on the above, the Leak-Before-Break conditions are satisfied for the Joseph M. Farley 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 of the Joseph M. Farley Units 1 and 2 Nuclear Power Plants for 60 year plant life as part of the license renewal program.