STRUCTURAL INTEGRITY ASSOCIATES, INC.

CALCULATION PACKAGE

FILE NO: CPL-34Q-301

PROJECT NO: CPL-34Q

PROJECT NAME: WELD OVERLAY REPAIR OF BRUNSWICK FEEDWATER NOZZLES.

CLIENT: CAROLINA POWER & LIGHT

CALCULATION TITLE: WELD OVERLAY REPAIR DESIGN FOR BSEP UT. 1 FW SAFE-END WELDS

PROBLEM STATEMENT, OR OBJECTIVE OF THE CALCULATION: See p. 1

DOCUMENT REVISION	AFFECTED PAGES	REVISION DESCRIPTION	PROJECT MANAGER APPROVAL/DATE	SIGNATURE/INITIALS OF PREPARERS AND CHECKERS – WITH DATES –
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				Miessi GAM 11/15/94
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				Miessi GAM 12/14/94

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(Subsequent Page)

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Document Revision	Affected Pages	Revision Description	Proj. Mgr. Approval /Date	Signature, Initials and Date of Preparers and Checkers
2	2, 4	REVISED LOAD	[signature] for HLG 12/15/94	CCM [signature] 12/15/94 [signature] GAM 12/15/94

WELD OVERLAY REPAIR DESIGN FOR BSEP UNIT 1 FEEDWATER SAFE END WELDS

I. Objective

During the 1992 inspection, a flaw believed to be indicative of IGSCC was observed in weld 1B21N4D-5-SW1-2 of BSEP Unit 1, Nozzle D. The flaw appeared to be located in the Inconel 182/82 weld metal on the carbon steel pup piece. Although the flaw was accepted for continued operation at that time, CP&L wants to be prepared to perform repairs should this weld show further degradation, or should other welds be determined to be flawed.

The purpose of the present calculation is to develop a set of weld overlay designs which could be applied to these welds on any of the four feedwater nozzles. Figure 3 shows the susceptible welds. Although the base metal is either carbon steel or Inconel, depending on the specific weld, Inconel 182/82 weld metal are present. The weld overlays will be designed using E-52 (Inconel) weld material as the repair material. The weld overlays will be prepared assuming that a "standard" weld overlay repair will be applied, in accordance with NUREG-0313 Revision 2 [1]. Since this design basis takes no credit for the underlying base material in demonstrating structural adequacy, use of NUREG-0313 Revision 2 and ASME Section XI, IWB-3641 is appropriate for these repairs, even with the ferritic base materials [1,2].

II. Geometry

The feedwater inlet piping is 12 in. NPS [3,4]. Figure 1 shows the geometry for the nozzle/safe end. (Note: this figure was used to obtain the thickness at the safe-end piece only.) The welds have been numbered for convenience in Figure 2; specific identification for these welds are given in Table 1.

Material properties:

The weld overlay repair will be applied using E-52 material. This material is the weld metal equivalent of Alloy 690 Inconel (ASME SB-163). For this material, Sm = 23.3 ksi for temperatures from 0 to 800°F [ASME Section IID, Table 28 (1992)] [2]. The mechanical properties for the base metal do not need to be considered in the design, since the standard design approach does not take credit for the strength of the base metal.

III. Assumptions

The following assumptions are made in this calculation. These need to be confirmed or shown to be bounding prior to finalizing the design.

a. The outer diameter (OD) of the nozzle safe end is 13.75 [5]. The OD of the extension piece (12 in. Schedule 100 Piping) is 12.75 in. [3,4]. (All four nozzles have identical

b. The wall thickness (t) at the nozzle/safe end is 0.875 in. [5]. "t" of the extension piece is 0.843 in. [3,4]. (All four nozzles have identical dimensions.)

c. The distance between the affected welds is shown in Figure 2 [6,7]. (All four nozzles have identical dimensions.)

d. All welds were made using a standard 37.5° weld prep.

e. The flaw location, if any, is in the Inconel weld metal.

f. Due to its high chromium content, E-52 weld material can be considered to be highly resistant to IGSCC, so propagation into this material can be neglected.

g. Residual stress has no affect on the design, since arrest of any flaw prior to encountering the E-52 material is not assumed. However, residual stress resulting from the repair will tend to inhibit any new flaw initiation.

IV. Stress Determination

The applied deadweight, pressure, thermal, OBE (operating basis earthquake), and DBE (design basis earthquake) seismic loads for the seven welds located at the extension piece and nozzle safe-end (Nodes 107 and 114) for both loops have been obtained from [8 and 9] and shown in Tables 2 through 9.

For purposes of this analysis, only the axial stress was calculated. Since the loads given by References 8 and 9* are in the global coordinate system, transformation in the axial direction is required. For Nodes 107 and 114 in Loops A and B, the following equations were used; based on the geometry provided by References 3 and 4.

The membrane (pressure) stress can be calculated with the following equation:

$$\text{Pressure (axial)} = P\left(\frac{D_i^2}{D_o^2 - D_i^2}\right)$$

$$F_{axial} = |F_x \bullet \cos 45°| + |F_z \bullet \cos 45°| \text{ (Node 107)}$$
$$F_{axial} = |-F_x \bullet \cos 45°| + |F_z \bullet \cos 45°| \text{ (Node 114)}$$

* These loads have been compared to those given in References 11 and 12 and will not have an effect on the weld overlay thickness.

b. The wall thickness (t) at the nozzle/safe end is 0.875 in. [5]. "t" of the extension piece is 0.843 in. [3,4]. (All four nozzles have identical dimensions.)

c. The distance between the affected welds is shown in Figure 2 [6,7]. (All four nozzles have identical dimensions.)

d. All welds were made using a standard 37.5° weld prep.

e. The flaw location, if any, is in the Inconel weld metal.

f. Due to its high chromium content, E-52 weld material can be considered to be highly resistant to IGSCC, so propagation into this material can be neglected.

g. Residual stress has no affect on the design, since arrest of any flaw prior to encountering the E-52 material is not assumed. However, residual stress resulting from the repair will tend to inhibit any new flaw initiation.

IV. Stress Determination

The applied deadweight, pressure, thermal, OBE (operating basis earthquake), and DBE (design basis earthquake) seismic loads for the seven welds located at the extension piece and nozzle safe-end (Nodes 107 and 114) for both loops have been obtained from [8 and 9] and shown in Tables 2 through 9.

For purposes of this analysis, only the axial stress was calculated. Since the loads given by References 8 and 9* are in the global coordinate system, transformation in the axial direction is required. For Nodes 107 and 114 in Loops A and B, the following equations were used; based on the geometry provided by References 3 and 4.

The membrane (pressure) stress can be calculated with the following equation:

$$\text{Pressure (axial)} = P\left(\frac{D_i^2}{D_o^2 - D_i^2}\right)$$

$$F_{axial} = |F_x \bullet \cos 45°| + |F_z \bullet \cos 45°| \text{ (Node 107)}$$
$$F_{axial} = |-F_x \bullet \cos 45°| + |F_z \bullet \cos 45°| \text{ (Node 114)}$$

* These loads have been compared to those given in References 11 and 12 and will not have an effect on the weld overlay thickness.

For Loop B:

$$f_{axial} = |F_x \bullet \cos45°| + |-F_z \bullet \cos45°| \text{ (Node 107)}$$
$$f_{axial} = |-F_x \bullet \cos45°| + |-F_z \bullet \cos45°| \text{ (Node 114)}$$

The axial stress for all other loads can be calculated with the following equation:

$$\sigma_{axial} = |F_{axial} \ / A| + \frac{\sqrt{M_x^2 + M_y^2 + M_z^2}}{z}$$

where:

$$A = \frac{\pi}{4}(OD^2 - ID^2)$$
$$Z = \frac{\pi}{32}\frac{(OD^4 - ID^4)}{OD}$$

The primary bending stress for normal/upset conditions is equal to the combination of axial stresses due to deadweight and OBE loads. (For completeness, bending for emergency/faulted conditions and thermal stresses have been reported in Tables 2 through 9.) Calculated pressure and primary bending stresses for normal/upset conditions at the extension piece and safe-end in Loops A and B are shown in Table 10. It should be noted that using the moment in all three directions to calculate the resultant moment is conservative.

V. Weld Overlay Design

a. Weld Overlay Thickness

Based upon the above set of assumptions and calculated stresses, a pc-CRACK calculation of required weld overlay thickness was made using the Codes and Standards/Structural Reinforcement module and the IWB-Tables option [10]. The resulting thicknesses for the extension piece to safe-end weld and safe-end to nozzle weld are 0.2810 and 0.2917 in., respectively. The pc-CRACK output files are included in Appendix A.

b. Weld Overlay Length

Historically, it has been assumed that a reinforcement must extend approximately SQRT (Rt) on either side of the repair location for effective load transfer. For the extension piece (with R = 6.375 in. and t = 0.843 in.) the half length is then 2.32 in. For the safe-end (with R = 6.875 in. and t = 0.875 in.), the half length is then 2.45 in. If Weld 5 is being repaired, this means that the repair would run up to the Weld 6. If either Weld 3 or Weld 4 is being repaired, the overlay repair would be extended to include both welds.

Figures 4 through 15 show the weld overlay design for the susceptible welds in Loops A and B.

VI. References

1. NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping", Final Report, Revision 2. January 1988.
2. ASME Boiler and Pressure Vessel Code, Sections II, 1992 Edition and XI, 1989 Edition, IWB-3640.
3. Carolina Power & Light Co., Drawing No. F-28046 Sheet No. 516 Rev. 2, "Stress Analysis Diagram Feedwater System Loop B", SI File CPL-34Q-206.
4. Carolina Power & Light Co., Drawing No. F-28046 Sheet No. 660 Rev. 2, "Stress Analysis Diagram Feedwater System Loop A", SI File CPL-34Q-207.
5. General Electric, Drawing No. 767E723, Rev. 1, "Safe End Feedwater Nozzle", SI File: CPL-34Q-203.
6. Carolina Power & Light Co., Drawing No. C-24004 Sheet 13-1, Rev. 2, "Unit No. 1 Inservice Inspection Isometric For Feedwater System Loop "A" Weld Location", SI File: CPL-34Q-224.
7. Carolina Power & Light Co., Drawing No. C-24004 Sheet 14-1, Rev. 2, "Unit No. 1 Inservice Inspection Isometric For Feedwater System Loop "B" Weld Location", SI File: CPL-34Q-225.
8. Carolina Power & Light Co., Calculation SA-B21-660, Rev. 1, "Piping Design Turnover Phase II for Reactor Feedwater Inlet Piping (Loop A)", November 1992, SI File: CPL-34Q-216.
9. Carolina Power & Light Co., Calculation SA-B21-516, Rev. 1, "Piping Design Turnover Phase II for Reactor Feedwater Inlet Piping (Loop B)", November 1992, SI File: CPL-34Q-217.
10. Structural Integrity, Computer Program pc-CRACK, Version 2.1, 1992.
11. Exerpt from Calculation SA-B21-660-0001, Rev. 1, "Equipment Load Summary", pp. 60-61.
12. Exerpt from Calculation SA-B21-516-0002, Rev. 1, "Equipment Load Summary", pp. 60-61.

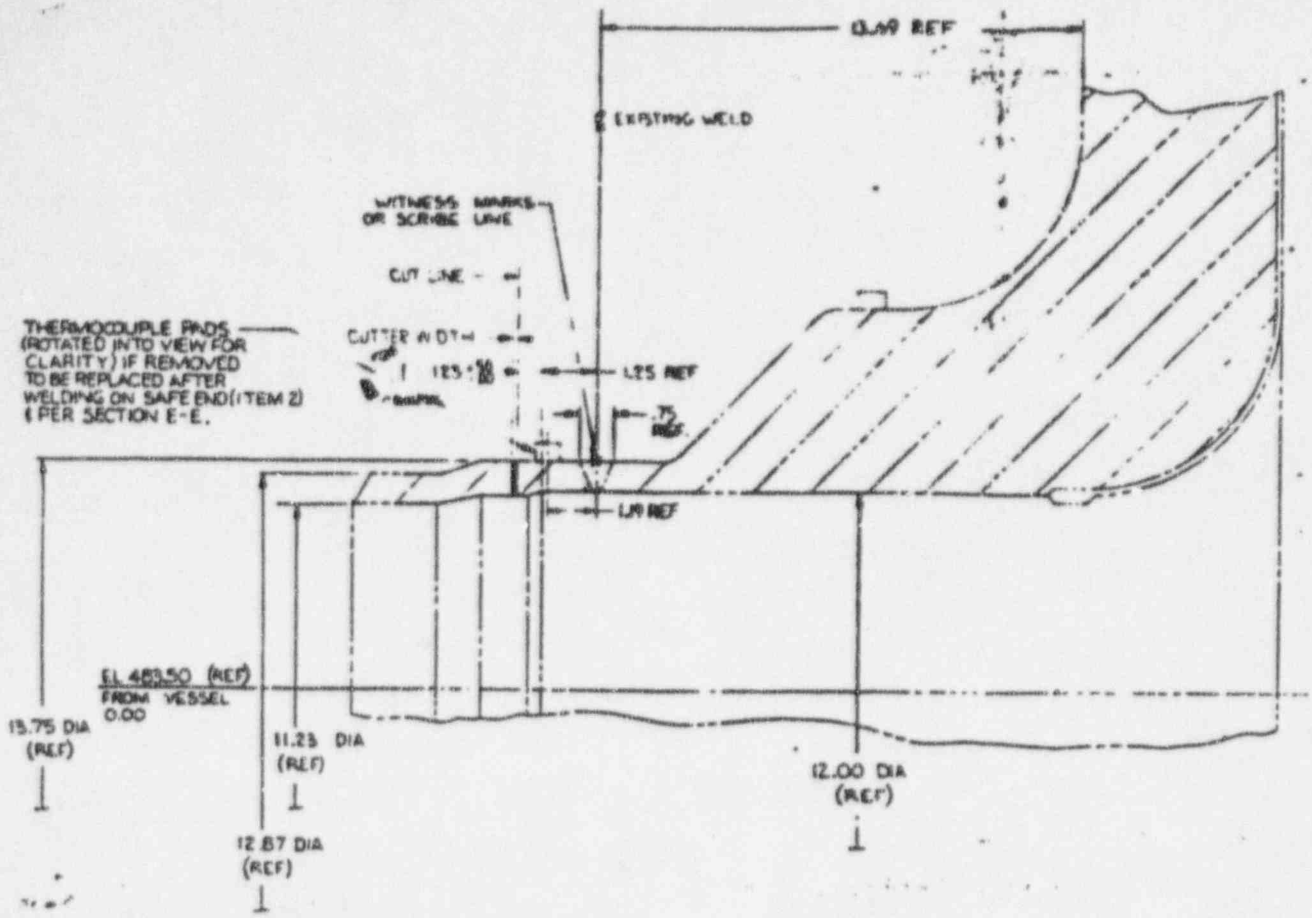
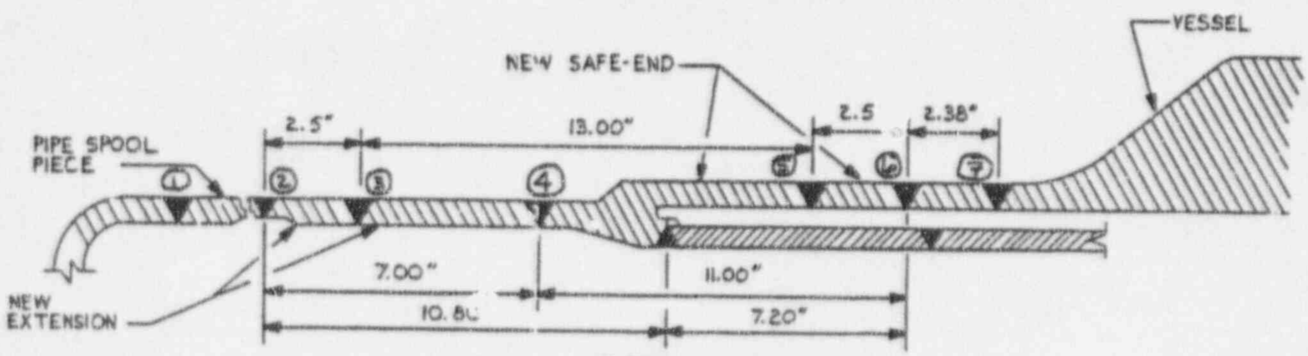


FIGURE 1

NOTE: THIS FIGURE WAS USED TO OBTAIN THE THICKNESS AND O.D. AT THE SAFE-END PIECE ONLY.



DETAIL-B

N4C/D NOZZLE, SAFE END, AND EXTENSION

FIGURE 2

	Revision	0				
	Preparer/Date	CDM 11/15/94				
	Checker/Date	GAM 11/15/94				
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Revision: 0
Preparer/Date: CBM 11/15/94
Checker/Date: GAM 11/15/94
File No. CECG-01Q-301

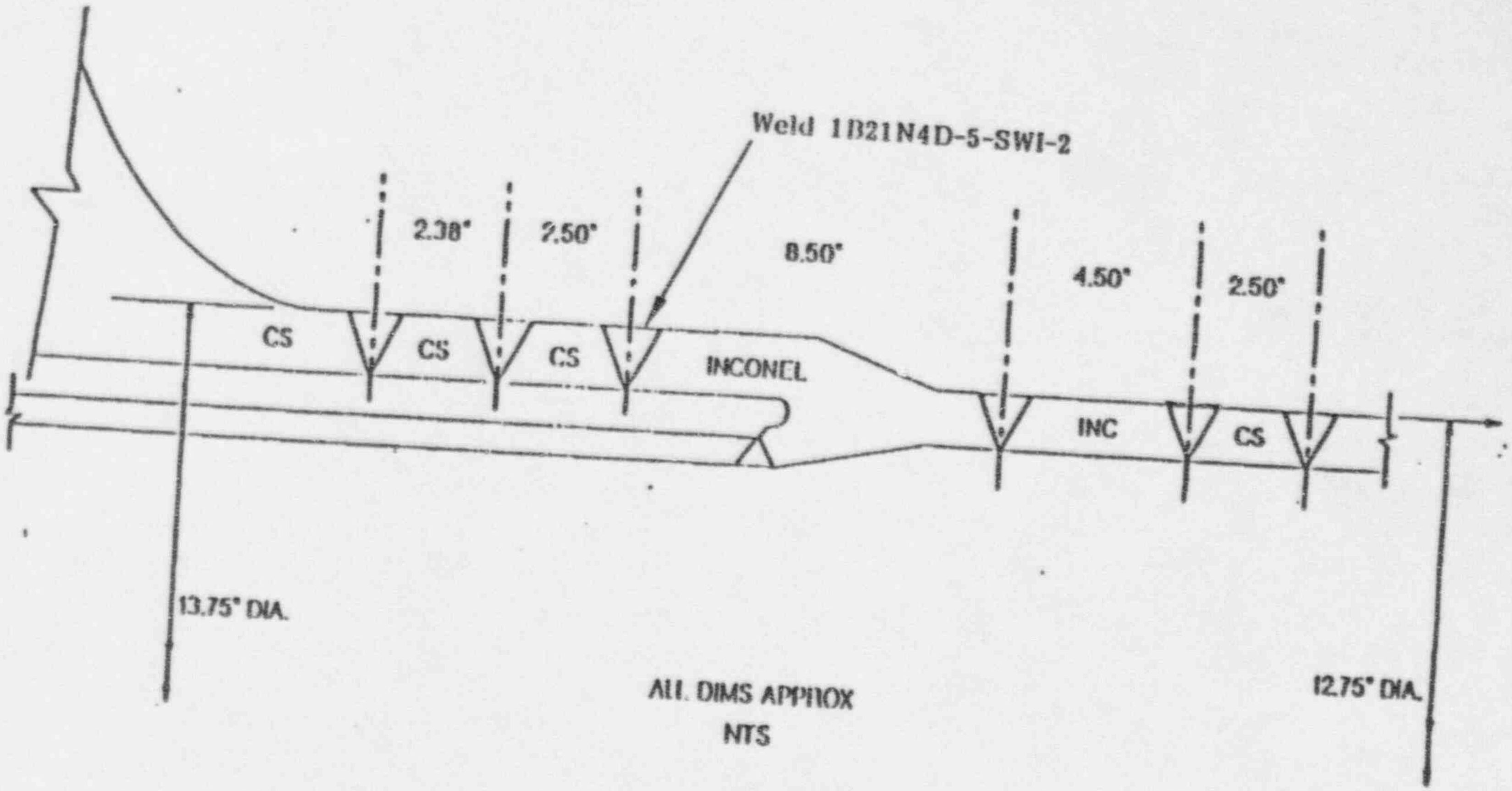

FIGURE 3

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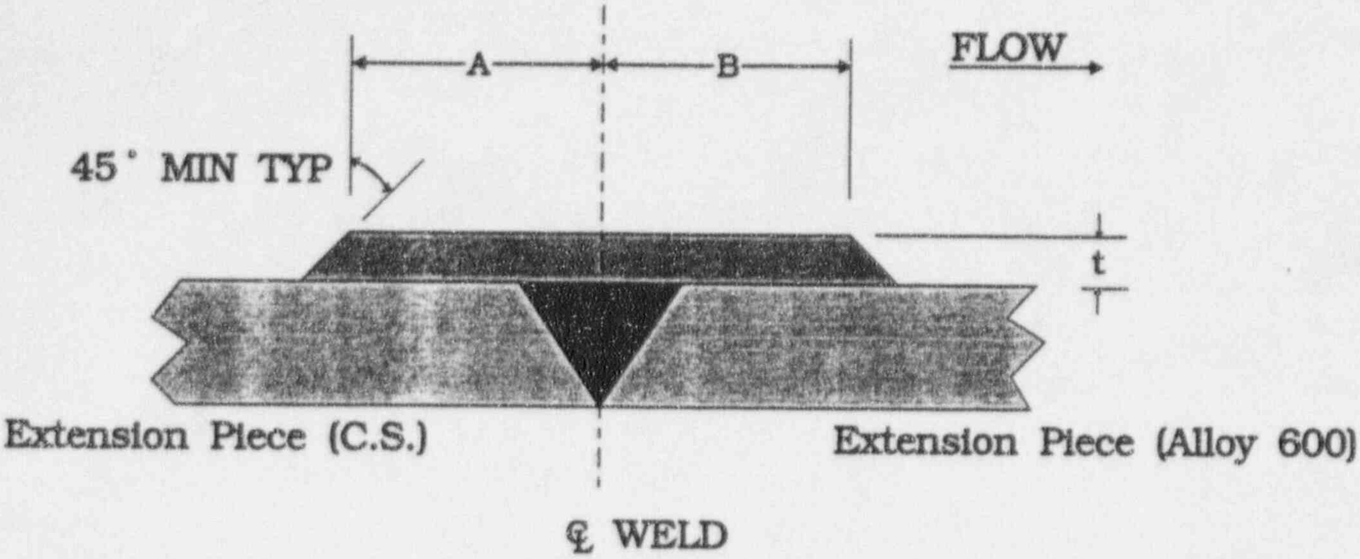
FIGURE 4



Not to Scale

WELD NUMBER	FLAW CHARACTERIZATION	DESIGN DIMENSIONS			COMMENTS
		t	A	B	
1B21N4A-2-SW2-3	Assumed 360° Circ. 100% throughwall flaw	0.28"	2.3"	6.8" (1),(2)	

94775r0

NOTE: (1) This length is the minimum required length. If feasible, end of overlay should be blended with safe-end OD taper.
(2) Length is sufficient to cover adjacent susceptible weld.

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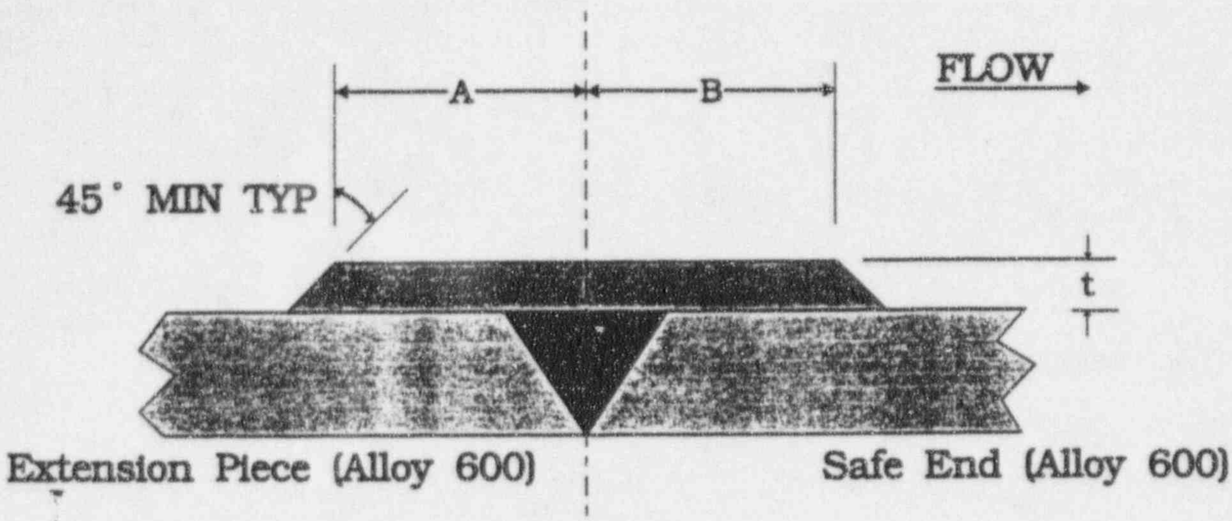
FIGURE 5



Not to Scale

WELD NUMBER	FLAW CHARACTERIZATION	DESIGN DIMENSIONS			COMMENTS
		t	A	B	
1B21N4A-2-FWN4A45-3	Assumed 360° Circ. 100% throughwall flaw	0.28"	6.8"(2)	2.3"(1)	

94775r0

NOTE: (1) This length is the minimum required length. If feasible, end of overlay should be blended with safe-end OD taper.
(2) Length is sufficient to cover adjacent susceptible weld.

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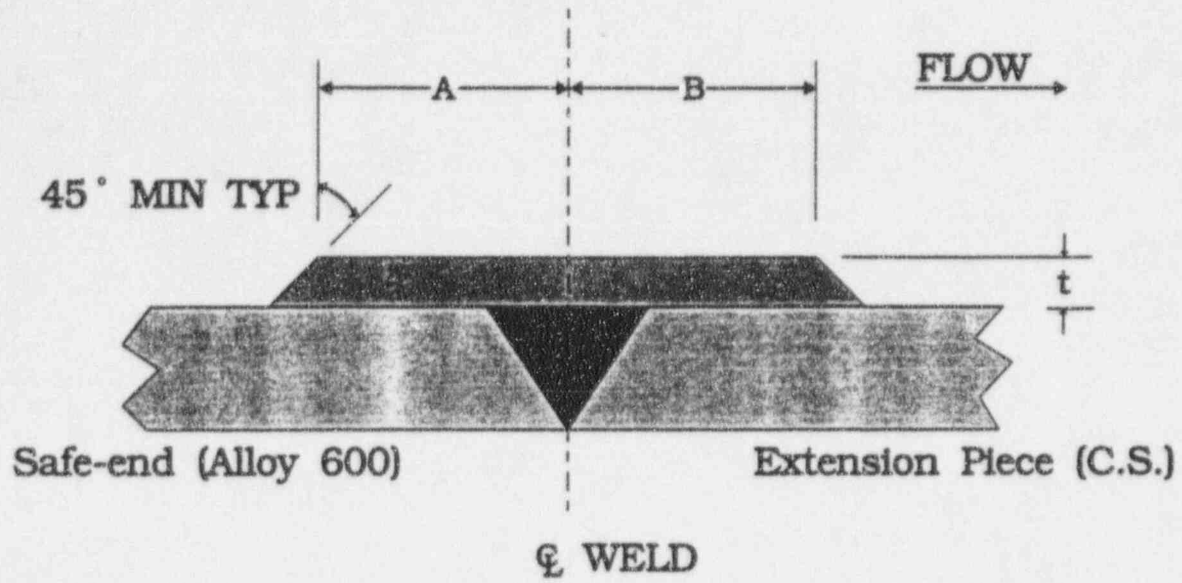
FIGURE 6



WELD NUMBER	FLAW CHARACTERIZATION	DESIGN DIMENSIONS			COMMENTS
		t	A	B	
1B21N4A-2-SW1-2	Assumed 360° Circ. 100% throughwall flaw	0.29"	2.4"	2.4"	

94775r0

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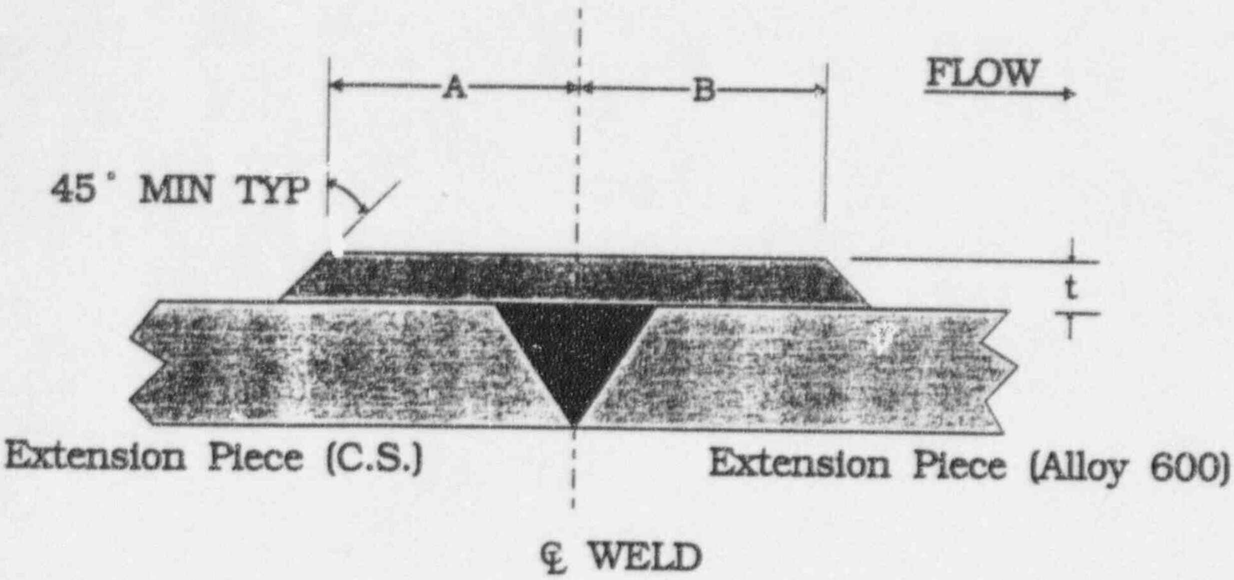
FIGURE 7



Not to Scale

WELD NUMBER	FLAW CHARACTERIZATION	DESIGN DIMENSIONS			COMMENTS
		t	A	B	
1B21N4B-3-SW2-3	Assumed 360° Circ. 100% throughwall flaw	0.28"	2.3"	6.8" (1),(2)	

94775r0

NOTE: (1) This length is the minimum required length. If feasible, end of overlay should be blended with safe-end OD taper.
(2) Length is sufficient to cover adjacent susceptible weld.

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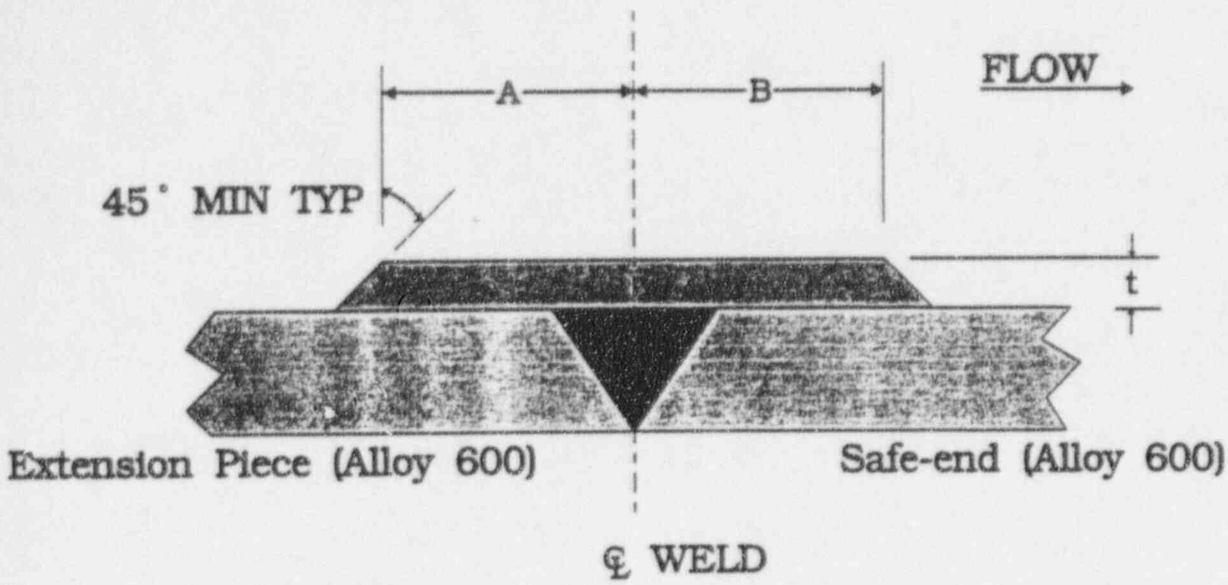
FIGURE 8



Not to Scale

WELD NUMBER	FLAW CHARACTERIZATION	DESIGN DIMENSIONS			COMMENTS
		t	A	B	
1B21N4B-3-FWN4B135-3	Assumed 360° Circ. 100% throughwall flaw	0.28"	6.8"(2)	2.3"(1)	

94775r0

NOTE: (1) This length is the minimum required length. If feasible, end of overlay should be blended with safe-end OD taper.
(2) Length is sufficient to cover adjacent susceptible weld.

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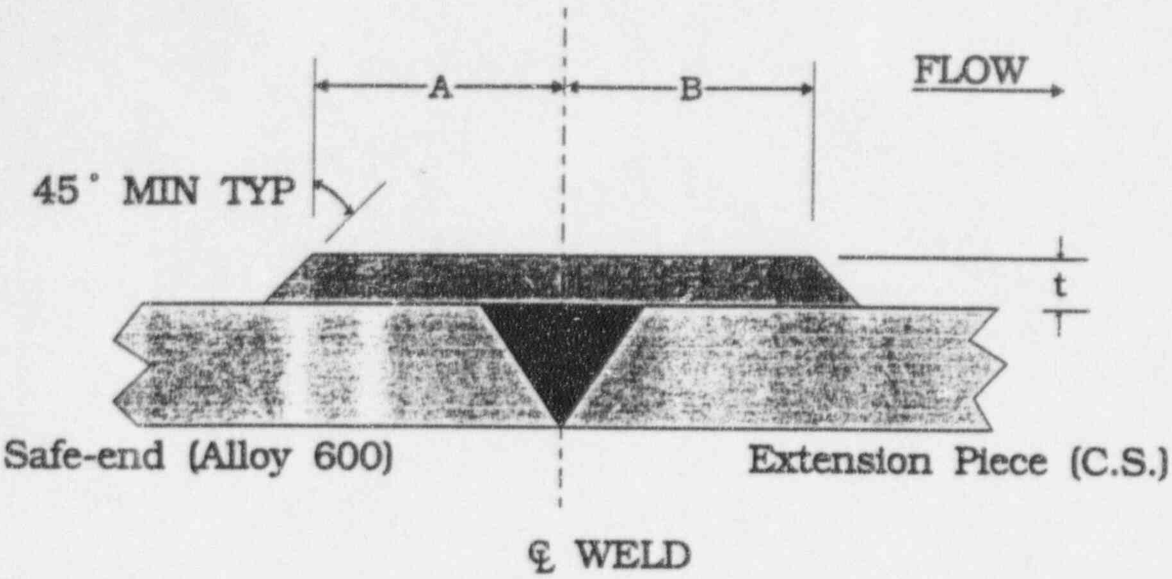
FIGURE 9



Not to Scale

WELD NUMBER	FLAW CHARACTERIZATION	DESIGN DIMENSIONS			COMMENTS
		t	A	B	
1B21N4B-3-SW1-2	Assumed 360° Circ. 100% throughwall flaw	0.29"	2.4"	2.4"	

94775r0

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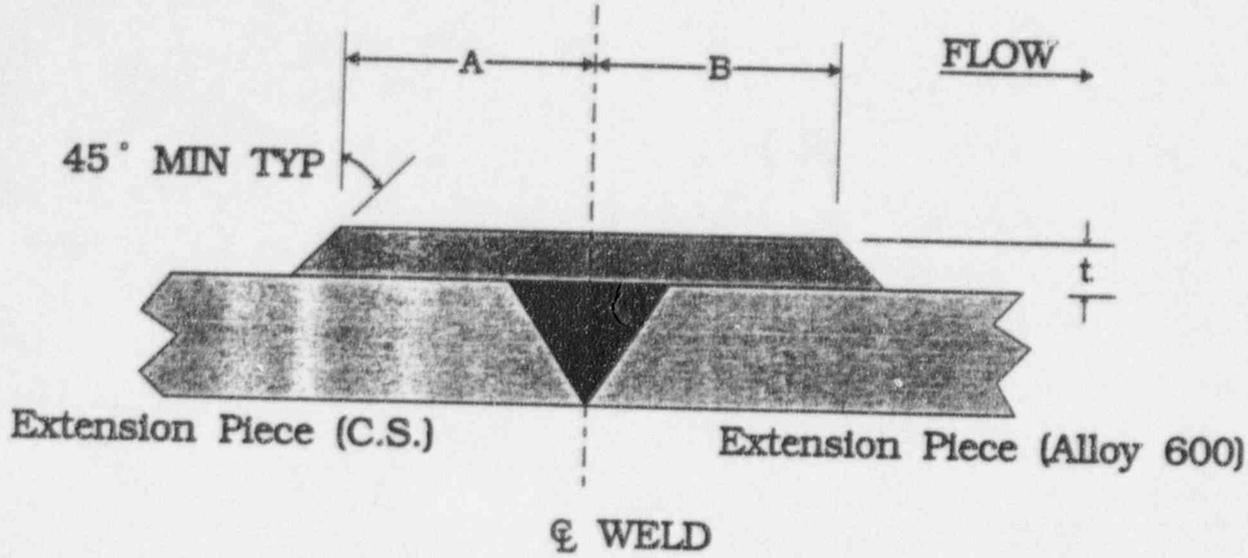
FIGURE 10



Not to Scale

WELD NUMBER	FLAW CHARACTERIZATION	DESIGN DIMENSIONS			COMMENTS
		t	A	B	
1B21N4C-6-SW2-3	Assumed 360° Circ. 100% throughwall flaw	0.28"	2.3"	6.8" (1),(2)	

94775r0

NOTE: (1) This length is the minimum required length. If feasible, end of overlay should be blended with safe-end OD taper.
(2) Length is sufficient to cover adjacent susceptible weld.

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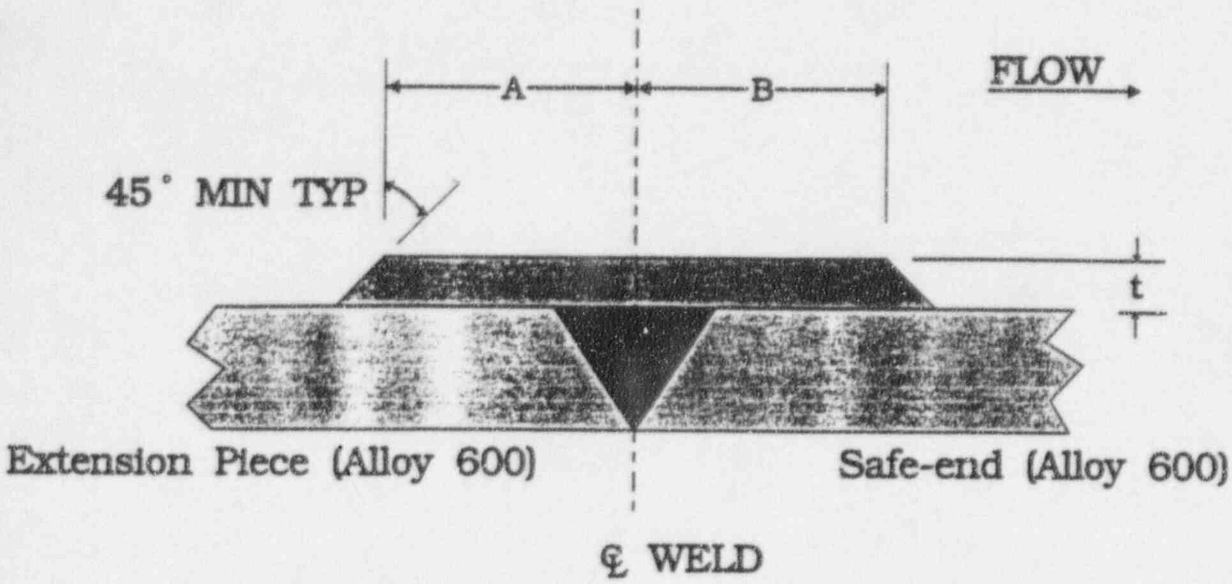
FIGURE 11



Not to Scale

WELD NUMBER	FLAW CHARACTERIZATION	DESIGN DIMENSIONS			COMMENTS
		t	A	B	
1B21N4C-6-FWN4C225-3	Assumed 360° Circ. 100% throughwall flaw	0.28"	6.8"(2)	2.3"(1)	

94775-0

NOTE: (1) This length is the minimum required length. If feasible, end of overlay should be blended with safe-end OD taper.
(2) Length is sufficient to cover adjacent susceptible weld.

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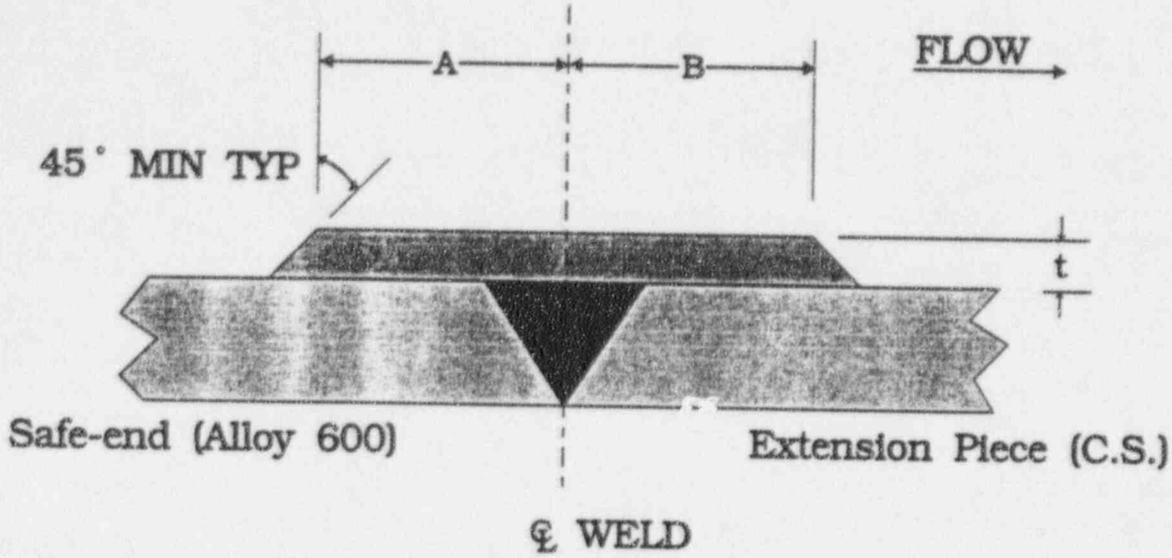
FIGURE 12



Not to Scale

WELD NUMBER	FLAW CHARACTERIZATION	DESIGN DIMENSIONS			COMMENTS
		t	A	B	
1B21N4C-6-SW1-2	Assumed 360° Circ. 100% throughwall flaw	0.29"	2.4"	2.4"	

94775r0

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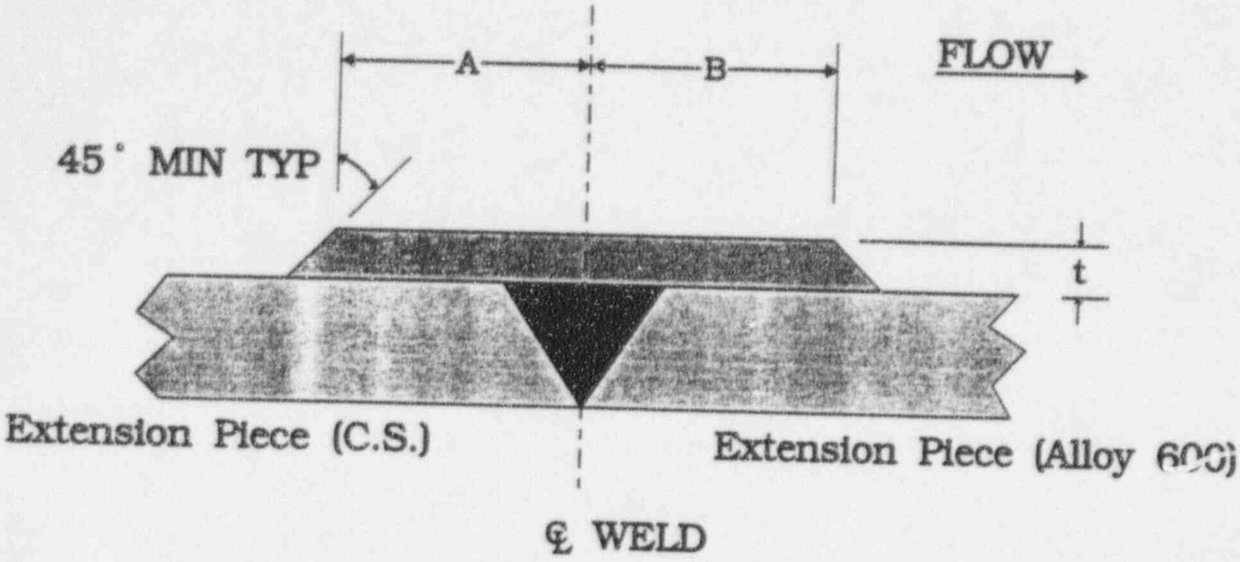
FIGURE 13



Not to Scale

WELD NUMBER	FLAW CHARACTERIZATION	DESIGN DIMENSIONS			COMMENTS
		t	A	B	
1B21N4D-5-SW2-3	Assumed 360° Circ. 100% throughwall flaw	0.28"	2.3"	6.8" (1),(2)	

94775r0

NOTE: (1) This length is the minimum required length. If feasible, end of overlay should be blended with safe-end OD taper.
(2) Length is sufficient to cover adjacent susceptible weld.

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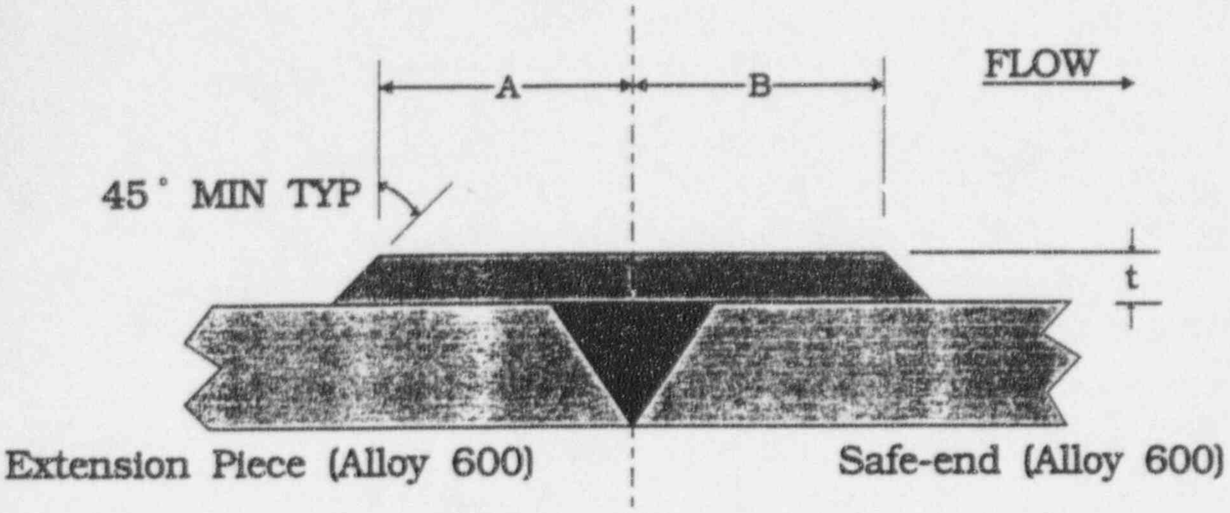
FIGURE 14



Not to Scale

WELD NUMBER	FLAW CHARACTERIZATION	DESIGN DIMENSIONS			COMMENTS
		t	A	B	
1B21N4D-5-FWN4D315-3	Assumed 360° Circ. 100% throughwall flaw	0.28"	6.8"(2)	2.3"(1)	

94775r0

NOTE: (1) This length is the minimum required length. If feasible, end of overlay should be blended with safe-end OD taper.
(2) Length is sufficient to cover adjacent susceptible weld.

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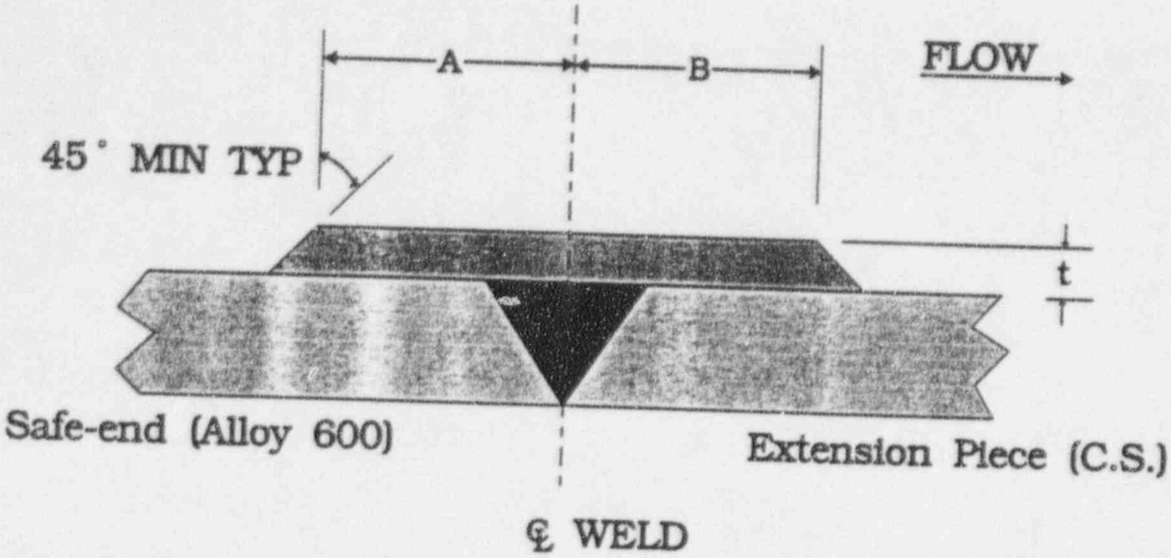
FIGURE 15



Not to Scale

WELD NUMBER	FLAW CHARACTERIZATION	DESIGN DIMENSIONS			COMMENTS
		t	A	B	
1B21N4D-5-SW1-2	Assumed 360° Circ. 100% throughwall flaw	0.29"	2.4"	2.4"	

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TABLE 1

Loop	Nozzle	Weld No.	Weld ID
A	N4-A	1	1B212-2-2-SWB
		2	1B212-2-2-FWRN1A
		3	1B21N4A-2-SW2-3
		4	1B21N4A-2-FWN4A45-3
		5	1B21N4A-2-SW1-2
		6	1B21N4A-2-FWN4A45-1
		7	1B11N4A-RPV-SWAB
	N4-B	1	1B213-3-3-SWB
		2	1B213-3-3-FWRN2A
		3	1B21N4B-3-SW2-3
		4	1B21N4B-3-FWN4B135-3
		5	1B21N4B-3-SW1-2
		6	1B21N4B-3-FWN4B135-1
		7	1B11N4B-RPV-SWAB
B	N4-C	1	1B216-3-6-SWB
		2	1B216-3-6-FWRN3A
		3	1B21N4C-6-SW2-3
		4	1B21N4C-6-FWN4C225-3
		5	1B21N4C-6-SW1-2
		6	1B21N4C-6-FWN4C225-1
		7	1B11N4C-RPV-SWAB
	N4-D	1	1B215-2-5-SWB
		2	1B215-2-5-FWRN4A
		3	1B21N4D-5-SW2-3
		4	1B21N4D-5-FWN4D315-3
		5	1B21N4D-5-SW1-2
		6	1B21N4D-5-FWN4D315-1
		7	1B11N4D-RPV-SWAB

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CPL-34Q, BRUNSWICK NUCLEAR PLANT Unit 1									
FEEDWATER SYSTEM - RV NOZZLE N4A									
Extension to Safe-End Weld - LOOP A									
NODE NO. 107									
				OD (IN):	12.75				
				T (IN):	0.843				
				ID (IN):	11.064				
				P (PSI):	1005				
				A (IN^2):	31.534				
				Z (IN^3):	88.102				
	FORCES AND MOMENTS								
LOAD	Fx (LB)	Fy (LB)	Fz (LB)	Faxial (LB)	Mx (FT-LB)	My (FT-LB)	Mz (FT-LB)	TOTAL MOMENT (IN-LB)	AXIAL STRESS (PSI)
P	-----	-----	-----		-----	-----	-----	-----	3064.1
DW	-86	10	-183	190.21	656	922	300	14048	165.5
OBE	814	1437	884	1200.67	2935	3588	2822	65123	777.3
DBE	1314	1926	1432	1941.72	4025	5794	3979	97195	1164.8
THERMAL425	-1608	2695	50	1172.38	-10604	-4877	-15278	230715	2655.9
THERMAL40	2017	-6895	-18	1438.96	25460	8133	-2080	321699	3697.1

Pm	Pb, Norm. DW+OBE	Pb, Emerg. DW+DBE	Thermal
(PSI)	(PSI)	(PSI)	(PSI)
3064.1	942.7	1330.3	3697.1

Prepared by CCM 12/14/94
Checked by: GAM 12/14/94
File No. CPL-34Q-301, rev. 1
Page 20 of 28

CPL-34Q, BRUNSWICK NUCLEAR PLANT Unit 1									
FEEDWATER SYSTEM - RV NOZZLE N4A									
Safe-End to Nozzle Weld - LOOP A									
NODE NO. 107									
				OD (IN):	13.75				
				T (IN):	0.875				
				ID (IN):	12				
				P (PSI):	1005				
				A (IN^2):	35.392				
				Z (IN^3):	107.161				
	FORCES AND MOMENTS								
LOAD	Fx (LB)	Fy (LB)	Fz (LB)	Faxial (LB)	Mx (FT-LB)	My (FT-LB)	Mz (FT-LB)	TOTAL MOMENT (IN-LB)	AXIAL STRESS (PSI)
P	-----	-----	-----		-----	-----	-----	-----	3211.5
DW	-86	10	-183	190.21	656	922	300	14048	136.5
OBE	814	1437	884	1200.67	2935	3588	2822	65123	641.6
DBE	1314	1926	1432	1941.72	4025	5794	3979	97195	961.9
THERMAL425	-1608	2695	50	1172.38	-10604	-4877	-15278	230715	2186.1
THERMAL40	2017	-6895	-18	1438.96	25460	8133	-2080	321699	3042.7

	Pm		Pb, Norm. DW+OBE		Pb, Emerg. DW+DBE		Thermal		
	(PSI)		(PSI)		(PSI)		(PSI)		
	3211.5		778.1		1098.3		3042.7		

Prepared by: CCM 12/14/94
Checked by: GAM 12/14/94
File No. CPL-34Q-301, rev 1
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TABLE 4

CPL-34Q, BRUNSWICK NUCLEAR PLANT Unit 1									
FEEDWATER SYSTEM - RV NOZZLE N4B									
EXTENSION to Safe-End Weld- LOOP A									
NODE NO. 114									
				OD (IN):	12.75				
				T (IN):	0.843				
				ID (IN):	11.064				
				P (PSI):	1005				
				A (IN^2):	31.534				
				Z (IN^3):	88.102				
		FORCES AND MOMENTS							
LOAD	Fx (LB)	Fy (LB)	Fz (LB)	Faxial (LB)	Mx (FT-LB)	My (FT-LB)	Mz (FT-LB)	TOTAL MOMENT (IN-LB)	AXIAL STRESS (PSI)
P	-----	-----	-----		-----	-----	-----	-----	3064.1
DW	-61	-417	8	48.79	82	-183	460	6022	69.9
OBE	780	1615	647	1009.04	5765	3108	8592	129643	1503.5
DBE	1128	2146	988	1496.24	7422	4703	10443	163773	1906.3
THERMAL425	-463	43	140	426.39	-2404	-3216	-11299	143895	1646.8
THERMAL40	326	517	-450	548.71	3053	3184	4847	78645	910.1
	Pm		Pb, Norm. DW+OBE		Pb, Emerg. DW+DBE		Thermal		
	(PSI)		(PSI)		(PSI)		(PSI)		
	3064.1		1573.4		1976.2		1646.8		

Prepared by: CCM 12/14/94
Checked by: GAM 12/14/94
File No. CPL-34Q-301, rev. 1
Page 22 of 28

TABLE 5

CPL-34Q, BRUNSWICK NUCLEAR PLANT Unit 1

FEEDWATER SYSTEM - RV NOZZLE N4B

Safe-End to Nozzle Weld- LOOP A

NODE NO. 114

OD (IN):	13.75
T (IN):	0.875
ID (IN):	12
P (PSI):	1005
A (IN^2):	35.392
Z (IN^3):	107.161

FORCES AND MOMENTS

LOAD	Fx (LB)	Fy (LB)	Fz (LB)	Faxial (LB)	Mx (FT-LB)	My (FT-LB)	Mz (FT-LB)	TOTAL MOMENT (IN-LB)	AXIAL STRESS (PSI)
P	-----	-----	-----		-----	-----	-----	-----	3211.5
DW	-61	-417	8	48.79	82	-183	460	6022	57.6
OBE	780	1615	647	1009.04	5765	3108	8592	129643	1238.3
DBE	1128	2146	988	1496.24	7422	4703	10443	163773	1570.6
THERMAL425	-463	43	140	426.39	-2404	-3216	-11299	143895	1354.8
THERMAL40	326	517	-450	548.71	3053	3184	4847	78645	749.4

Pm (PSI)	Pb, Norm. DW+OBE (PSI)	Pb, Emerg. DW+DBE (PSI)	Thermal (PSI)
3211.5	1295.9	1628.1	1354.8

Prepared by: CCM 12/14/94
Checked by: GAM 12/14/94
File No. CPL-34Q-301, rev.1
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CPL-34Q, BRUNSWICK NUCLEAR PLANT Unit 1									
FEEDWATER SYSTEM - RV NOZZLE N4D									
Extension to Safe-End Weld- LOOP B									
NODE NO. 107									
				OD (IN):	12.75				
				T (IN):	0.843				
				ID (IN):	11.064				
				P (PSI):	1005				
				A (IN^2):	31.534				
				Z (IN^3):	88.102				
	FORCES AND MOMENTS							TOTAL	AXIAL
	Fx	Fy	Fz	Faxial	Mx	My	Mz	MOMENT	STRESS
LOAD	(LB)	(LB)	(LB)	(LB)	(FT-LB)	(FT-LB)	(FT-LB)	(IN-LB)	(PSI)
P	-----	-----	-----		-----	-----	-----	-----	3064
DW	-100	109	172	192.33	-186	-935	499	12912	152.7
OBE	1082	5151	651	1225.42	10700	2869	9485	175005	2025.3
DBE	1418	6148	1041	1738.78	12883	4494	11709	215756	2504.1
THERMAL42	-1609	2718	-42	1167.43	10580	4887	-15203	229871	2646.2
THERMAL40	1971	-6744	24	1410.68	-25147	-8021	-2101	317745	3651.3
	Pm		Pb, Norm.		Pb, Emerg.		Thermal		
			DW+OBE		DW+DBE				
	(PSI)		(PSI)		(PSI)		(PSI)		
	3064.1		2177.9		2656.7		3651.3		

Prepared by: CCM 12/14/94
Checked by: GAM 12/14/94
File No. CPL-34Q-301, rev. 1
Page 24 of 28

CPL-34Q, BRUNSWICK NUCLEAR PLANT Unit 1									
FEEDWATER SYSTEM - RV NOZZLE N4D									
Safe-End to Nozzle Weld - LOOP B									
NODE NO. 107									
				OD (IN):	13.75				
				T (IN):	0.875				
				ID (IN):	12				
				P (PSI):	1005				
				A (IN^2):	35.392				
				Z (IN^3):	107.161				
		FORCES AND MOMENTS						TOTAL	AXIAL
	Fx	Fy	Fz	Faxial	Mx	My	Mz	MOMENT	STRESS
LOAD	(LB)	(LB)	(LB)	(LB)	(FT-LB)	(FT-LB)	(FT-LB)	(IN-LB)	(PSI)
P	-----	-----	-----		-----	-----	-----	-----	3212
DW	-100	109	172	192.33	-186	-935	499	12912	125.9
OBE	1082	5151	651	1225.42	10700	2869	9485	175005	1667.7
DBE	1418	6148	1041	1738.78	12883	4494	11709	215756	2062.5
THERMAL425	-1609	2718	-42	1167.43	10580	4887	-15203	229871	2178.1
THERMAL40	1971	-6744	24	1410.68	-25147	-8021	-2101	317745	3005.0
	Pm		Pb, Norm.		Pb, Emerg.		Thermal		
			DW+OBE		DW+DBE				
	(PSI)		(PSI)		(PSI)		(PSI)		
	3211.5		1793.7		2188.4		3005.0		

Prepared by: CAM 12/14/94
Checked by: GAM 12/14/94
File No. CPL-34Q-301, rev.1
Page 25 of 28

CPL-34Q, BRUNSWICK NUCLEAR PLANT Unit 1									
FEEDWATER SYSTEM - RV NOZZLE N4C									
EXTENSION to Safe-End Weld- LOOP B									
NODE NO. 114									
				OD (IN):	12.75				
				T (IN):	0.843				
				ID (IN):	11.064				
				P (PSI):	1005				
				A (IN^2):	31.534				
				Z (IN^3):	88.102				
	FORCES AND MOMENTS							TOTAL	AXIAL
	Fx	Fy	Fz	Faxial	Mx	My	Mz	MOMENT	STRESS
LOAD	(LB)	(LB)	(LB)	(LB)	(FT-LB)	(FT-LB)	(FT-LB)	(IN-LB)	(PSI)
P	-----	-----	-----		-----	-----	-----	-----	3064
DW	-17	-153	29	32.53	996	-236	-88	12328	141.0
OBE	582	1352	569	813.88	6820	3660	9324	145416	1676.3
DBE	823	1811	805	1151.17	8421	4841	11219	178076	2057.7
THERMAL42	-456	33	-138	420.02	2345	3177	-11248	143052	1637.0
THERMAL40	326	497	430	534.57	-3075	-3134	4898	78934	912.9
	Pm		Pb, Norm.		Pb, Emerg.		Thermal		
			DW+OBE		DW+DBE				
	(PSI)		(PSI)		(PSI)		(PSI)		
	3064.1		1817.3		2198.7		1637.0		

Prepared by: CCM 12/14/94
Checked by: GAM 12/14/94
File No. CPL-34Q-301, rev.1
Page 26 of 28

TABLE 9

CPL-34Q, BRUNSWICK NUCLEAR PLANT Unit 1

FEEDWATER SYSTEM - RV NOZZLE N4C

Safe-End to Nozzle Weld- LOOP B

NODE NO. 114

OD (IN):	13.75
T (IN):	0.875
ID (IN):	12
P (PSI):	1005
A (IN^2):	35.392
Z (IN^3):	107.161

FORCES AND MOMENTS

LOAD	Fx (LB)	Fy (LB)	Fz (LB)	Faxial (LB)	Mx (FT-LB)	My (FT-LB)	Mz (FT-LB)	TOTAL MOMENT (IN-LB)	AXIAL STRESS (PSI)
P	-----	-----	-----		-----	-----	-----	-----	3212
DW	-17	-153	29	32.53	996	-236	-88	12328	116.0
OBE	582	1352	569	813.88	6820	3660	9324	145416	1380.0
DBE	823	1811	805	1151.17	8421	4841	11219	178076	1694.3
THERMAL42	-456	33	-138	420.02	2345	3177	-11248	143052	1346.8
THERMAL40	326	497	430	534.57	-3075	-3134	4898	78934	751.7

Pm (PSI)	Pb, Norm. DW+OBE (PSI)	Pb, Emerg. DW+DBE (PSI)	Thermal (PSI)
3211.5	1495.9	1810.2	1346.8

Prepared by: CCM 12/14/94

Checked by: GAM 12/14/94

File No. CPL-34Q-301, rev.1

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TABLE 10

Membrane & Primary Bending Stresses				
LOOP	NODE	LOCATION	Pm (ksi)	Pb (ksi)
A	107	Extension	3.064	0.943
	107	Safe-End	3.212	0.778
	114	Extension	3.064	1.544
	114	Safe-End	3.212	1.270
B	107	Extension	3.064	2.149
	107	Safe-End	3.212	1.768
	114	Extension	3.064	0.801
	114	Safe-End	3.212	1.495

Prepared by: CEM 11/16/94
Checked by: GAM 11/16/94
File No. CPL-34Q-301, rev.0
Page 28 of 28

APPENDIX A

pc-CRACK Output Files

	Revision	0				
	Preparer/Date	CJM 11/11/94				
	Checker/Date	GAM 11/15/94				

Date: 15-Nov-1994
Time: 9: 5: 3.13

STRUCTURAL REINFORCEMENT SIZING EVALUATION

STRUCTURAL REINFORCEMENT SIZING FOR CIRCUMF. CRACK,WROUGHT/CAST STAINLESS

EXTENSION TO SAFE-END WELD, LOOP A, NODE 107

WALL THICKNESS= 0.8430
MEMBRANE STRESS= 3.0640
BENDING STRESS= 0.9430
STRESS RATIO= 0.1720
ALLOWABLE STRESS= 23.3000
FLOW STRESS= 69.9000

	L/CIRCUM					
	0.00	0.10	0.20	0.30	0.40	0.50
FINAL A/T	0.7500	0.7500	0.7500	0.7500	0.7500	0.7500
REINFORCEMENT THICK.	0.2810	0.2810	0.2810	0.2810	0.2810	0.2810

Prepared by: CCM 11/15/94
Checked by: GAM 11/15/94
File No. CPL-34Q-30, rev.0
Page A2 of A9

Date: 15-Nov-1994
Time: 9: 6:18.71

STRUCTURAL REINFORCEMENT SIZING EVALUATION

STRUCTURAL REINFORCEMENT SIZING FOR CIRCUMF. CRACK,WROUGHT/CAST STAINLESS

SAFE-END TO NOZZLE WELD, LOOP A (NODE 107)

WALL THICKNESS= 0.8750
MEMBRANE STRESS= 3.2120
BENDING STRESS= 0.7780
STRESS RATIO= 0.1712
ALLOWABLE STRESS= 23.3000
FLOW STRESS= 69.9000

L/CIRCUM	0.00	0.10	0.20	0.30	0.40	0.50
FINAL A/T	0.7500	0.7500	0.7500	0.7500	0.7500	0.7500
REINFORCEMENT THICK.	0.2917	0.2917	0.2917	0.2917	0.2917	0.2917

Prepared by: CCM 11/15/94
Checked by: GAM 11/15/94
File No. CPL-34Q-301, rev.0
Page A3 of A9

Date: 15-Dec-1994
Time: 2:52:41.25

STRUCTURAL REINFORCEMENT SIZING EVALUATION

STRUCTURAL REINFORCEMENT SIZING FOR CIRCUMF. CRACK,WROUGHT/CAST STAINLESS

EXTENSION TO SAFE-END WELD, LOOP A (NODE 114)

WALL THICKNESS= 0.8430
MEMBRANE STRESS= 3.0640
BENDING STRESS= 1.5734
STRESS RATIO= 0.1990
ALLOWABLE STRESS= 23.3000
FLOW STRESS= 69.9000

	L/CIRCUM					
	0.00	0.10	0.20	0.30	0.40	0.50
FINAL A/T	0.7500	0.7500	0.7500	0.7500	0.7500	0.7500
REINFORCEMENT THICK.	0.2810	0.2810	0.2810	0.2810	0.2810	0.2810

Prepared by: CCM 12/14/94
Checked by: GAM 12/14/94
File No. CPL-34D-301, rev. 1
Page A4 of A9

Date: 15-Dec-1994
Time: 2:59:54.12

STRUCTURAL REINFORCEMENT SIZING EVALUATION

STRUCTURAL REINFORCEMENT SIZING FOR CIRCUMF. CRACK,WROUGHT/CAST STAINLESS

SAFE-END TO NOZZLE WELD, LOOP A (NODE 114)

WALL THICKNESS= 0.8750
MEMBRANE STRESS= 3.2120
BENDING STRESS= 1.2959
STRESS RATIO= 0.1935
ALLOWABLE STRESS= 23.3000
FLOW STRESS= 69.9000

	L/CIRCUM					
	0.00	0.10	0.20	0.30	0.40	0.50
FINAL A/T	0.7500	0.7500	0.7500	0.7500	0.7500	0.7500
REINFORCEMENT THICK.	0.2917	0.2917	0.2917	0.2917	0.2917	0.2917

Prepared by: CCM 12/14/94
Checked by: GAM 12/14/94
File No. CPL-34Q-301, rev.1
Page A5 of A9

ate: 15-Dec-1994
ime: 3: 2: 6.93

STRUCTURAL REINFORCEMENT SIZING EVALUATION

STRUCTURAL REINFORCEMENT SIZING FOR CIRCUMF. CRACK,WROUGHT/CAST STAINLESS

EXTENSION TO SAFE-END WELD, LOOP B (NODE 107)

WALL THICKNESS= 0.8430
MEMBRANE STRESS= 3.0640
BENDING STRESS= 2.1779
STRESS RATIO= 0.2250
ALLOWABLE STRESS= 23.3000
FLOW STRESS= 69.9000

	L/CIRCUM					
	0.00	0.10	0.20	0.30	0.40	0.50
FINAL A/T	0.7500	0.7500	0.7500	0.7500	0.7500	0.7500
REINFORCEMENT THICK.	0.2810	0.2810	0.2810	0.2810	0.2810	0.2810

Prepared by: CCM 12/14/94
Checked by: GAM 12/14/94
File No. CR-34Q-301, rev.1
Page A6 of A9

Date: 15-Dec-1994
Time: 3: 6:11.46

STRUCTURAL REINFORCEMENT SIZING EVALUATION

STRUCTURAL REINFORCEMENT SIZING FOR CIRCUMF. CRACK,WROUGHT/CAST STAINLESS

SAFE-END TO NOZZLE WELD, LOOP B (NODE 107)

WALL THICKNESS= 0.8750
MEMBRANE STRESS= 3.2120
BENDING STRESS= 1.7937
STRESS RATIO= 0.2148
ALLOWABLE STRESS= 23.3000
FLOW STRESS= 69.9000

	L/CIRCUM					
	0.00	0.10	0.20	0.30	0.40	0.50
FINAL A/T	0.7500	0.7500	0.7500	0.7500	0.7500	0.7500
REINFORCEMENT THICK.	0.2917	0.2917	0.2917	0.2917	0.2917	0.2917

Prepared by: CDM 12/14/94
Checked by: GAM 12/14/94
File No. CPL-34Q-301 rev.1
Page A7 of A9

tm
pc-CRACK

STRUCTURAL INTEGRITY ASSOCIATES, INC.
SAN JOSE, CA (408)978-8200
VERSION 2.1

Date: 15-Dec-1994
Time: 3: 9:32. 4

STRUCTURAL REINFORCEMENT SIZING EVALUATION

STRUCTURAL REINFORCEMENT SIZING FOR CIRCUMF. CRACK,WROUGHT/CAST STAINLESS

EXTENSION TO SAFE-END WELD, LOOP B (NODE 114)

WALL THICKNESS= 0.8430
MEMBRANE STRESS= 3.0640
BENDING STRESS= 1.8173
STRESS RATIO= 0.2095
ALLOWABLE STRESS= 23.3000
FLOW STRESS= 69.9000

L/CIRCUM	0.00	0.10	0.20	0.30	0.40	0.50
FINAL A/T	0.7500	0.7500	0.7500	0.7500	0.7500	0.7500
REINFORCEMENT THICK.	0.2810	0.2810	0.2810	0.2810	0.2810	0.2810

Prepared by: CCM 12/14/94
Checked by: GAM 12/14/94
File No. CPL-340-301 Rev. 1
Page A8 of A9

15-Dec-1994
3:15:21.70

STRUCTURAL REINFORCEMENT SIZING EVALUATION

STRUCTURAL REINFORCEMENT SIZING FOR CIRCUMF. CRACK,WROUGHT/CAST STAINLESS

SAFE-END TO NOZZLE WELD, LOOP B (NODE 114)

WALL THICKNESS= 0.8750
MEMBRANE STRESS= 3.2120
BENDING STRESS= 1.4959
STRESS RATIO= 0.2021
ALLOWABLE STRESS= 23.3000
FLOW STRESS= 69.9000

	L/CIRCUM					
	0.00	0.10	0.20	0.30	0.40	0.50
FINAL A/T	0.7500	0.7500	0.7500	0.7500	0.7500	0.7500
REINFORCEMENT THICK.	0.2917	0.2917	0.2917	0.2917	0.2917	0.2917

Prepared by: CCM 12/14/94
Checked by: GAM 12/14/94
File No. CPL-34Q-301, rev.1
Page A9 of A9

ENCLOSURE 3

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1
NRC DOCKET NO. 50-325
OPERATING LICENSE NO. DPR-71
WELD OVERLAY REPAIR OF FEEDWATER PIPING AND SAFE-ENDS

Engineering Service Request 94-00458,
"Unit 1 FW Piping and Safe-End Weld Overlay"
Brunswick Steam Electric Plant, Unit 1

Form 1	ENGINEERING SERVICE REQUEST		
ESR # 9400458	Rev # 0	WR/JO #	Other Documents (ACR, FACTS, etc.)
Plant/Unit BNP 1	Primary System # 1005	Primary System Name B21,B11-NUCLEAR BOILER (INC.RX VESSEL &	☐ Multiple Systems Affected
Title Unit 1 FW piping and safe-end weld overlay		Originator/Phone (print) BERTZ, STEVEN L /8503182	

Problem/Proposed Solution/Justification

PCN G0029B, Replacement of the Unit 1 Feedwater Spargers. the project scope has been revised such that replacement is no longer scheduled. the spargers will be repaired if necessary.

The feedwater piping and safe-end contain 3 Alloy 600 Alloy 82/182 welds in each of the 4 FW lines which require frequent inspection in acccordance with NUREG-0313 Rev.2. These welds are susceptible to IGSCC and rapid crack growth such that repairs would be necessary in the RFO that identified them.

A weld overlay will be designed for the 12 welds and a contingency PM developed

☒ Continued

MANAGEMENT REVIEW

Assigned Manager	PRIORITY
W. BLANE WILTON	3 ROUTINE
Responsible Engineer	DUE DATE
STEVEN L BERTZ	12-15-94

SCREENING

Quality Class	Is a 10CFR 50.59 Safety Review required per (plant specific procedure)?	Response Type
A Safety-Related	☒ Yes (See attached safety evaluation for signatures) ☐ No (Concurrence of two QSRs required below) ☐ N/A (Engineering Reply ESR) 1st QSR: ________ Date: ____ 2nd QSR: ________ Date: ____	MOD-MAJOR

Engineering Disciplines *(Print Name, Sign, Date)*

Mechanical STEVEN L. BERTZ 12/21/94

Civil/Seismic J.McINTYRE 12/22/94

Materials Michael W. Guthrie 12/22/94

Welding Michael W. Guthrie 12/22/94 MICHAEL W. GUTHRIE

Engineering/Plant Programs *(Print Name, Sign, Date)*

ALARA

ISI 1/12/95

Fire Prot 1-6-95

RECEIVED BY BNP

JAN 13 1994

NUCLEAR DOCUMENT CONTROL

Stop Think Act Review

WORKING COPY VERIFIED

Lancaster 1/13/95 1/20/95

SIGNATURE DATE EXP DATE

SIGNATURE DATE EXP DATE

SIGNATURE DATE EXP DATE

Form 1 **ENGINEERING SERVICE REQUEST**

ESR #	Rev #	Title
9400458	0	Unit 1 FW piping and safe-end weld overlay

Plant Customers *(Print Name, Sign, Date)*

System Eng Phillip Gore 1-13-95

Installer Chuck Raines 1.12.95

Specialty Reviews

Design Verification

☐ NAS Review Before Implementation Reference: ______

☒ NAS Review Before Closeout

☐ PNSC Review Before Implementation Reference: ______

☒ NRC Review Before Implementation Reference: OPLP-08, GL-88-01

Problem Resolution:

MODIFICATION SUMMARY

The feedwater inlet piping to the reactor vessel is carbon steel up to the nozzle safe-end. A plant modification removed the original carbon steel safe-end, except for a short stub, and installed an Alloy 600 safe-end that accommodated a new feedwater sparger thermal sleeve design prior to initial plant start-up. The modification included three welds at each safe-end that are of a material which is susceptible to Intergranular Stress Corrosion Cracking (IGSCC) as defined in NUREG-0313 Rev. 2. Generic Letter 88-01/NUREG-0313 Rev. 2 provides options for the repair or replacement of piping susceptible to IGSCC. This modification implements a weld overlay in accordance with the NRC Staff positions of Generic Letter 88-01/NUREG-0313 Rev. 2, ASME Section XI, 1989 ed., no addenda, and ASME Code Committee guidance of Code Case N-504 using material that is highly resistant to IGSCC.

NRC review and approval is required prior to implementing this repair on a flawed weldment since this type of repair is not detailed in ASME Section XI IWA-4000 and Code Case N-504 specifically addresses austenitic stainless steel

☒ Continued

APPROVAL

Is this a modification which constitutes a reduction in design margin?

☐ Yes (PGM approval is required)

☒ No (Engineering Mgr signs for PGM)

Interim Approval Required? ☐ Yes ☒ No ☐ N/A

Engineer *(Print Name, Sign, Date)* Steven L. Bertz 1/13/95

Engineering Manager *(Print Name, Sign, Date)* Paul T. Cafarella 1/13/95

Plant General Manager *(Print Name, Sign, Date)* William Levis 1/13/95

Form 1 **ENGINEERING SERVICE REQUEST**

ESR #	Rev #	Title
9400458	0	Unit 1 FW piping and safe-end weld overlay

Problem/Proposed Solution/Justification

for implementation. An RFQ will be develped and evaluated to select the best vendor to apply the overlay. A contingency contract will be awarded for the vendor to support the 1995 U1 RFO. The contract will be activated if defects are identified which require repair.

BESS Mechanical will develop the design and modification. The NRC will be notified by letter of the design and asked for comment.

Projects will develop the RFQ and select the vendor for implementation.

Form 1		ENGINEERING SERVICE REQUEST
ESR # 9400458	Rev # 0	Title Unit 1 FW piping and safe-end weld overlay

Problem Resolution:

but not carbon steel.

This project provides a weld overlay design for the three Inconel 82/182 welds connecting the RPV nozzle, safe-end, and feedwater piping in each feedwater line. These welds are included in Generic Letter 88-01/NUREG-0313 Rev.2, as Category "D". The weld overlay will provide full structural reinforcement assuming a 360 through-wall crack. The overlay design uses a weld filler metal, Alloy 52, which is highly resistant to IGSCC and will mitigate crack growth into the weld overlay.

The overlay geometry is designed to permit future inspections by visual, ultrasonic, and radiographic techniques to satisfy Augmented (Generic Letter 88-01/NUREG-0313 Rev. 2) and In-service Inspection requirements.

The weld overlays may be applied at adjacent welds to minimize the dose associated with the repairs even if the adjacent weld does not have an identified flaw.

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1.0 BASIC FUNCTIONS

1.1 The basic function of the feedwater supply system is to return the condensed steam from the turbine (via the condensate system) back to the reactor pressure vessel (RPV). The feedwater piping affected by this project is classified as part of the feedwater system. The feedwater piping is connected to the RPV nozzle via a safe-end and four transition pieces, which are considered part of the Nuclear Boiler System.

1.2 This project provides a weld overlay design for the three Inconel 82/182 welds connecting the RPV nozzle, safe-end, and feedwater piping in each feedwater line that are in NUREG-0313 Rev. 2, as Category "D". NUREG-0313 requires inspections of these welds and provides repair options if unacceptable flaws, per ASME Section XI (ref. 3.2.1.3), are identified. A weld overlay is a repair option. The weld overlay will provide full structural reinforcement assuming a 360° through-wall crack. The overlay design uses a weld filler metal which is highly resistant to IGSCC and will mitigate crack growth into the weld overlay. The weld overlay design is in accordance with the requirements of ASME Section XI, IWA-4300, 1989 ed., with guidance taken from Code Case N-504, and is contained in reference 3.3.6.13. This will assure that the overlay weld stresses imposed on the base material will remain within acceptable limits.

1.3 Unflawed Inconel welds on the feedwater lines at these safe-end locations may be overlaid to mitigate IGSCC to minimize future schedule impacts, resource costs and minimize accumulated dose.

1.4 The overlay geometry is designed to permit future inspections by visual, ultrasonic, and radiographic techniques to satisfy Augmented (NUREG-0313 Rev. 2) and In-service Inspection requirements.

1.5 This weld overlay repair is termed a non-ASME Code repair per 0PLP-08 and NRC approval is required prior to implementation per Generic Letter 88-01 and 0PLP-08.

2.0 PERFORMANCE REQUIREMENTS

2.1 The structural integrity of the reactor coolant pressure boundary shall not be degraded by this project. Performance requirements for structural integrity of the Nuclear Boiler system and RPV are unchanged by thir project. The overlay design was verified by analysis to be acceptable for the existing design loads and materials (ref. 3.3.6.13).

2.2 The hydraulic performance requirements, i.e., flow rate of the feedwater lines are unchanged by this project.

3.0 CODES, STANDARDS, REGULATORY REQUIREMENTS, & DBD REFERENCES

This section identifies docketed regulatory commitments that establish the requirements for the Nuclear Boiler System (B21) design basis. These are original codes and standards, Regulatory Guides, Regulations, Generic Letters, etc., to which BNP is committed with respect to the portion of the B21 system affected by this modification.

3.1 *QUALITY ASSURANCE PROGRAM REQUIREMENTS*

The specific provisions of the Corporate QA Program are contained in the Corporate Quality Assurance Manual (Ref: 3.3.1.7) as described in Section 2.1.1 for GDC 1 requirements.

3.1.1 *10 CFR 50, Appendix B, Quality Assurance Program Requirements* (Ref: 3.3.8.9)

BNP's Quality Assurance Program shall meet the requirements of 10 CFR 50 App. B (ref: 3.3.1.1, p. 13.4-2)

3.1.2 *R.G. 1.33, Revision 0, Quality Assurance Program Requirements* (Ref: 3.3.9.3)

BNP complies with the provisions of RG-1.33, Rev. 0, including the requirements and recommendations for administrative controls described in ANSI N18.7-1976 with the exception listed in UFSAR, Section 1.8. (Refs: 3.3.1.1, pp. 13.4-2 and 13.4-9a; 3.3.1.2, Section 1.8; and 3.3.1.7).

3.1.3 *R.G. 1.64, Revision 0, Quality Requirements for the Design of Nuclear Power Plants* (Ref: 3.3.9.4)

Compliance with the provisions of R.G. 1.64, Rev. 0, for those areas of the QA Program applicable to the design and modification of the plant shall be met by complying with the applicable guidance of ANSI N45.2.11-1974, with the exception listed in UFSAR Section 1.8 (Refs. 3.3.1.2, Section 1.8; and 3.3.1.7)

3.1.4 *R.G. 1.74, Revision 0, Quality Assurance Terms and Definitions* (Ref: 3.3.9.5)

BNP shall comply with the provisions of R.G. 1.74, Rev. 0, and ANSI N45.2.10-1973. (Refs: 3.3.1.2, Sect 1.8; and 3.3.1.7)

3.2 *CODES/STANDARDS OF RECORD*

The following is a listing of codes and standards which are applicable, in part or entirely, to the portion of the Nuclear Boiler System (B21) affected by this project and which are tied to actual commitments. In the cases where only a certain section of the standard is design basis, that section is specified. A reference to one section of a standard does not indicate commitment with the entire standard. Codes and standards which are utilized for good engineering practice, but are not required to meet the design basis may be found in the applicable specifications. The current code and standards of record are listed with references to original design and fabrication standards where applicable.

3.2.1 *ANSI/ASME STANDARDS*

3.2.1.1 *(USAS) B31.1, Code For Pressure Piping, Power Piping, 1967* for the piping design unless otherwise reconciled.

3.2.1.1 (cont.)

The original feedwater supply piping affected by this project was designed and fabricated in accordance with USAS B31.1, 1967 (Ref 3.3.1.1., pp. A-2, 3, Table A-1).

3.2.1.2 *American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 1989 Edition, no addenda, "Rules for Inservice Inspection of Nuclear Power Plant Components".*

The changes to the existing FW supply piping as a result of this project, are classified as a permanent repair and designed in accordance with ASME Section XI, 1989 ed., no addenda. Guidance from Code Case N-504 was used to develop the weld overlay design. (Reference 3.3.6.13).

3.2.1.3 *ASME Boiler and Pressure Vessel Code, Section XI, 1980 Edition through Winter 1981 Addenda, "Rules for Inservice Inspection of Nuclear Power Plant Components".*

The Updated Inservice Inspection Testing Program is in accordance with Section XI of the ASME Code, 1980 Edition through the Winter 1981 Addenda, (Ref. 3.3.1.2, para 3.9.3.1, Amendment 7 and 5.2.4, Amendment 5). This project is within the ISI boundary.

3.2.1.4 *ASME Boiler and Pressure Vessel Code, Section III, Division 1, Nuclear Power Plant Components, 1965 edition with Addenda up to and including Summer 1967.*

The reactor pressure vessel, nozzle, safe-ends and extensions were originally designed in accordance with this document and this is the "code of record" for the reactor pressure vessel design. The original quality classification was designated as Class A. Additional requirements from the Winter 1967 Addenda were added for fracture toughness testing (Ref. 3.3.1.2 para. 5.3.1.5).

3.2.1.5 *ASME Boiler and Pressure Vessel Code, Section IX, Welding and Brazing Qualifications, latest Edition and Addenda.*

3.3 *DESIGN BASIS DOCUMENT REFERENCES*

This is a list of documents used as references in this DBD/ESR and other major documents important to system design. This listing is not intended to be all encompassing in order to minimize the revisions to this ESR. Supplemental sources such as EDBS, BRMS, and NRCS can be utilized for a detailed listing of all documents related to the feedwater piping and nozzles (N4 A/D).

3.3.1 *LICENSE BASIS DOCUMENTS*

3.3.1.1 Final Safety Analysis Report (FSAR) Brunswick Steam Electric Plant, through Amendment 31

3.3.1.2 Updated Final Safety Analysis Report (UFSAR), Brunswick Steam Electric Plant

3.3.1.3 NG-75-300, 02/27/75, Quality Assurance Program

3.3.1.4 NG-75-1278, 08/26/75, Quality Assurance Program

3.3.1.5 Appendix A to License Nos. DPR-62 & 71, BSEP Technical Specifications, Unit 2 and Unit 1

3.3.1.6 Appendix B to License Nos. DPR-62 & 71, BSEP Environmental Technical Specifications, Unit 2 and Unit 1

3.3.1.7 OQA-81-026, 03/18/81, Quality Assurance Program

3.3.2 *CORRESPONDENCE*

3.3.2.1 MISC-00489, 08/02/74, Commitments Made Concerning Brunswick Steam Electric Plant

3.3.2.2 B-3513, 12/03/74, Telephone Conversation Memorandum on Outstanding AEC Concerns, between R.G. Black (CP&L) and Ray Powell (AEC)

3.3.2.3 MISC-04912, 01/24/75, Trip Report for Brunswick Site Visit January 21 through January 22, 1975

3.3.3 *PROCEDURES AND NON-TECHNICAL MANUALS*

3.3.3.1 PAM, current revision, Carolina Power & Light Company, Nuclear Generation, Procedure Administration Manual

3.3.3.2 CQAM, current revision, Carolina Power & Light Company, Corporate Quality Assurance Manual

3.3.3.3 ENP-16, current revision, Procedure for Administrative Control of Inservice Inspection Activities

3.3.3.4 Corporate Radiation Control and Protection Manual, current revision

3.3.3.5 SD-01, System Description for Nuclear Boiler System.

3.3.3.6 OP-01, Operating Procedure for Nuclear Boiler System.

3.3.3.7 CP&L Engineering Procedure 0-ENP-16.2, Administrative Control of ASME Section XI Non-Destructive Examination Program

3.3.3.8 Inservice Inspection Program for BSEP Units 1 and 2 Second Inspection Interval, CP&L 02

3.3.3.9 E&RC Procedure-4150, current revision, Temporary Shielding

3.3.3.10 DBD-104, Generic Issue Design Basis Document for Shielding

3.3.4 *DRAWINGS*

3.3.4.1 REACTOR BUILDING PIPING DIAGRAM, NUCLEAR STEAM SYSTEM, UNIT #1, D-25021, SHT. 1C

3.3.4.2 UNIT 1 STRESS ANALYSIS DIAGRAM, FEEDWATER SYSTEM LOOP "A" 18" & 12" LINE INSIDE DRYWELL, F-28046 SHEET NO. 660

3.3.4.3 UNIT 1 STRESS ANALYSIS DIAGRAM, FEEDWATER SYSTEM LOOP "B" 18" & 12" LINE INSIDE DRYWELL, F-28046 SHEET NO. 516

3.3.4.4 FEEDWATER INLET NOZZLES N4 A/D (Forgings), FP-9527-5005

3.3.4.5 12" DIA. FEEDWATER NOZZLES MARK N4 A/D, FP-9527-5060

3.3.4.6 INSERVICE INSPECTION ISO FOR FEEDWATER SYST. LOOP "A" WELD LOCATION, C-24004, SHT. 13-1

3.3.4.7 INSERVICE INSPECTION ISO FOR FEEDWATER SYST. LOOP "B" WELD LOCATION, C-24004, SHT. 14-1

3.3.4.8 Installation Kit Safe-End Feedwater Nozzle 767E723, FP-55126

3.3.4.9 Outline Safe-End 131C9063, FP-55116

3.3.4.10 Outline Extension 131C9061, FP-55114

3.3.4.11 Unit 1 Stress Analysis Diagram FW System Loop "B" 18" and 12" Line Inside Drywell, F-28046 Sht 516

3.3.4.12 Unit 1 Stress Analysis Diagram FW System Loop "A" 18" and 12" Line Inside Drywell, F-28046 Sht 660

3.3.4.13 Safe-End QA Requirements Specification, 22A4447

3.3.4.14 SA-B21-660, Reactor Feedwater Inlet Piping-Loop A, Pipe Stress Analysis

3.3.4.15 SA-B21-516, Reactor Feedwater Inlet Piping-Loop B, Pipe Stress Analysis

3.3.5 *SPECIFICATIONS*

Specifications listed in this Section are listed without revision levels. For installation or purchases, the latest revision of the specification shall be used (See NRCS for latest Specification Revisions). For information on existing equipment, the revision of the Specification in effect at the time of purchase and installation of the particular component of interest must be used.

3.3.5.1 CPL-XXXX-W-01, current revision, Welding Filler Metals and Materials Procurement for Nuclear Power Plants, ASME Section III Applications

3.3.5.2 ASME Material specifications as specified in the referenced specifications.

3.3.5.3 GE 22A6286 - Specification for Feedwater Nozzle

3.3.5.4 248-145, current revision, Specification for Fluid System Cleanliness

3.3.5.5 CPL-XXXX-W-003, current revision, Specification for Control of Contamination Limits for Consumables Used in Contact with Stainless Steel and Nickel Base Alloys

3.3.5.6 248-117, current revision, Specification for Installation of Piping Systems for BNP Units 1 and 2

3.3.5.7 248-003, current revision, Specification for Reactor Coolant Pressure Boundary Piping

3.3.6 *MISC TECHNICAL REFERENCES/REPORTS/EERS/CALCULATIONS*

3.3.6.1 CP&L Corporate Welding Manual, current revision

3.3.6.2 UE&C Study Report No. 7992.017-S-M-045, Piping Specification(s)/Code Reconciliation for CP&L, Brunswick Steam Electric Plant, Units 1 & 2, April 9, 1987.

3.3.6.3 NED-DG-001, current revision, Design Guide Incorporating ALARA

3.3.6.4 DG-VIII.53, current revision, BNP Human Factors Engineering

3.3.6.5 DG-VIII.58, current revision, Human Factors Evaluation of Plant Modifications

3.3.6.6 Feedwater Dynamic Analysis Data, General Electric Document No. 158B8898J, Sheets 2, 3, 3A, 16A, 16B,& 16C, (0-FP-50447, recd 12/27/93)

3.3.6.7 Nuclear Boiler System (P&ID), General Electric Doc. No. 729E616 BB, Sht 1 & 2, (0-FP-05547, BSEP recd 12/27/75)

3.3.6.8 Nuclear Boiler System Process Diagram, General Electric Company Document No. 117C3230 B, Sheet No. 1, (0-FP-5094, BSEP recd 1/14/77)

3.3.6.9 Calculation No. 9527-2-FW-52-F, Pressures and Temperatures in BOP Feedwater, 1/29/87

3.3.6.10 Calculation No. 9527-2-FW-10-F, Feedwater Design Criteria, 9/3/81

3.3.6.11 Calculation No. 9527-2-FW-6-F, Condensate & Feedwater System Pressure Drop, 9/3/81

3.3.6.12 CB&I Stress Report, FP-50585

3.3.6.13 Calculation CPL-34Q-301, Weld Overlay Repair Design for BSEP UT-1 FW Safe-End Welds, 11/94

3.3.6.14 NEDC-30634 Rev.1, Brnswick Unit 1 Feedwater Nozzle Fracture Mechanics Analysis, May 1991

3.3.7 *ADDITIONAL CODES AND STANDARDS*

3.3.7.1 AISC 1963, Specification for Design Fabrication, and Erection of Structural Steel for Buildings. This is the original design and fabrication standard for the structural steel (Ref. 3.3.1.1 and FSAR comment MC.22-2 and Ref. 3.3.1.2, Para. 3.8.4.5).

3.3.7.2 AISC, 8th Ed., Specification for Design Fabrication, and Erection of Structural Steel for Buildings. This is the current edition applicable to BSEP (Ref. 3.3.1.2, Section 3.8.1).

3.3.7.3 ANS-3.3-1982, Security For Nuclear Power Plants

3.3.7.4 ANSI N18.7-1976, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants

3.3.7.5 ANSI N45.2.8-1975, Supplementary Quality Assurance Requirements for Installation, Inspections, and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants

3.3.7.6 ANSI N45.2.10-1973, Quality Assurance Terms and Definitions

3.3.7.7 ANSI N45.2.11-1974, Quality Assurance Requirements for the Design of Nuclear Power Plants

3.3.7.8 ANSI N45.2.2-1972, Packaging, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants

3.3.7.9 ANSI N52.1, Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants

3.3.7.10 ANSI N45.2.13-1976, Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants

3.3.7.11 ANSI N45.2.1-1973, Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants

3.3.7.12 Cases of ASME Boiler and Pressure Vessel Code - Case 2142: F-Number Grouping for Ni-Cr-Fe, Classification UNS N06052 Filler Metal, Section XI

3.3.7.13 ASME Boiler and Pressure Vessel Code, Section II, Materials, Part C, Specification for Welding Rods, Electrodes, and Filler Metals, latest Edition and Addenda.

3.3.8 *FEDERAL REGULATIONS*

3.3.8.1 Code of Federal Regulations, Title 10. Part 20, (10 CFR 20), current issuance, Standard for Protection Against Radiation

3.3.8.2 Code of Federal Regulations, Title 10, Part 50.44 (10 CFR 50.44), current issuance, Standards For Combustible Gas Control Systems in Light-Water-Cooled Power Reactors

3.3.8.3 Code of Federal Regulations, Title 10, Part 50.48, (10 CFR 50.48), current issuance, Fire Protection

3.3.8.4 Code of Federal Regulations, Title 10, Part 50.49 (10 CFR 50.49), current issuance, February 22, 1983, Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants

3.3.8.5 Code of Federal Regulations, Title 10, Part 50.54 (10 CFR 50.54), current issuance, Conditions of Licenses

3.3.8.6 Code of Federal Regulations, Title 10, Part 50.55a (10 CFR 50.55a), current issuance, Codes and Standards

3.3.8.7 Code of Federal Regulations, Title 10, Part 50.63, (10 CFR 50.63), current issuance, Loss of All Alternating Current Power

3.3.8.8 Code of Federal Regulations, Title 10, Part 50, Appendix A, (10 CFR 50, App. A), 1971, General Design Criteria

3.3.8.9 Code of Federal Regulations, Title 10, Part 50, Appendix B, (10 CFR 50, App. B), current issuance, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants

3.3.8.10 Code of Federal Regulations, Title 10, Part 50, Appendix J, (10 CFR 50, App. J), current issuance, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors

3.3.8.11 Code of Federal Regulations, Title 10, Part 50, Appendix R, (10 CFR 50, App. R), current issuance, Fire Protection Program for Nuclear Facilities Operating Prior to January 1, 1979

3.3.8.12 Code of Federal Regulations, Title 10, Part 73, (10 CFR 73), current issuance, Physical Protection of Plants and Materials

3.3.8.13 Code of Federal Regulations, Title 10, Part 100, (10 CFR 100), current issuance, Reactor Site Criteria

3.3.8.14 Code of Federal Regulations, Title 29, Part 1910, (29 CFR 1910), current issuance, Occupational Safety and Health Act

3.3.9 *REGULATORY GUIDES*

3.3.9.1 NRC Regulatory Guides per commitments in the CP&L Corporate QA Manual, Appendix II

3.3.9.2 Reg Guide 1.26, draft revision 3, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants.

3.3.9.3 Reg Guide 1.33, Revision 0, 1972, Quality Assurance Program Requirements

3.3.9.4 Reg Guide 1.64, Revision 0, 1973, Quality Assurance Requirements for the Design of Nuclear Power Plants

3.3.9.5 Reg Guide 1.74, Revision 0, Quality Assurance Terms and Definitions

3.3.10 *USNRC SAFETY EVALUATIONS AND SERS*

3.3.10.1 MISC-00678, Safety Evaluation Report of BSEP 1&2-License Application with Appendices Through 7/26/76 - Pages 7-10, 9-12 & 9-13, 1/01/73

3.3.10.2 NLU-79-219, 06/07/79, Review of Piping Reanalysis per IE Bulletin 79-07 Safety Evaluation Report

3.3.10.3 NLU-79-484, 10/26/79, NRC Evaluation of CP&L Responses to IE Bulletin 79-08

3.3.10.4 NLS-85-141, Safety Evaluation for Final Resolution of Environmental Qualification of Electric Equipment Important to Safety, March 5, 1985.

3.3.10.5 NLS-85-297, 05/14/85, Emergency Response Capability - Conformance to Regulatory Guide 1.97, Rev. 2 - BSEP, Units 1 & 2

3.3.10.6 NRC-89-812, 12/06/89, Appendix R Safety Evaluation Clarification and Revision - BSEP

3.3.11 *MISC USNRC DOCUMENTS*

3.3.11.1 Generic Letter 84-01, 01/05/84, NRC Use of Terms Important to Safety and Safety Related. CP&L reference NLU-84-31

3.3.11.2 Generic Letter 88-01, 1/25/88, NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping

3.3.11.3 Generic Letter 88-01, Supplement 1, 2/26/92, NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping

3.3.11.4 NUREG-0313, Rev. 2, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping, 1/1988

3.3.11.5 IE Bulletin 79-07, 04/14/79, Seismic Stress Analysis of Safety-Related Piping

3.3.11.6 IE Bulletin No. 79-08, 04/14/79, Events Relevant to Boiling Water Power Reactors Identified During Three Mile Island Accident, CP&L Letter No. NLU-79-127

3.3.12 *IE INSPECTION REPORTS*

3.3.12.1 IE Inspection Report No. 50-324/75-2, 02/13/75

4.0 DESIGN CONDITIONS

4.1 The design pressure and temperature for the reactor pressure vessel nozzle "N4" and safe-end shall be as follows: (Ref: 3.3.1.2, Table 5.3.3-1)

4.1.1 Pressure: 1250 psig

4.1.2 Temperature: 575° F

4.2 The design pressure and temperature of the feedwater supply piping is per Specification 248-117 (Ref: 3.3.5.6).

5.0 DESIGN LOADS

5.1 The design loads to be considered in the design of the weld overlay shall be based upon the requirements of the Codes, Standards, References and Regulatory requirements listed in Section 3.0 of this DBD.

5.2 The affected feedwater supply lines are seismically supported to maintain the integrity of the reactor coolant pressure boundary during design basis accidents. Design loading is given in the UFSAR Sections 3.7 and 3.9 (Ref. 3.3.1.2).

5.3 A thermal analysis of the feedwater nozzle safe-ends was performed as part of the reactor vessel in reference 3.3.6.12. The nozzle blend radius is the most fatigue sensitive. The weld overlay is upstream of the blend radius and is in a lower fatigue usage area. The fatigue analysis for the feedwater nozzle is not adversely affected.

5.4 Pipe stress analyses have been performed on these feedwater lines (ref. 3.3.4.14 and 3.3.4.15). These analyses accounted for pressure, dead weight, temperature, seismic, and displacement up to the outboard extension piece. The weld overlay is designed to provide full structural design margins for these loads. The axial shrinkage of the feedwater pipe due to the weld overlay is expected to be less than 1/2 inch and may result in adjustment to supports, but will not have a significant effect on the analysis.

6.0 ENVIRONMENTAL CONDITIONS

6.1 The normal and accident conditions which would be observed inside of primary containment are given in Table 3.11.1-1 of the UFSAR (Ref. 3.3.1.2).

6.2 The interior of the feedwater nozzle, safe-ends and the piping are exposed to saturated steam and demineralized water during normal operating conditions.

6.3 Neutron fluence is negligible at the location of the feedwater nozzles; flux is $<1 \times 10^{17}$ n/cm^2/yr, >1 Mev. Similar materials do not indicate significant property changes until the fluence approaches 10^{20} n/cm^2, >1 Mev.

6.4 During installation of the weld overlay the Drywell will be de-inerted. The feedwater lines shall either be filled with reactor water or drained. The overlay shall not be applied with the feedwater line partially filled with water.

6.5 The feedwater line shall be isolated and feedwater flow is not permited during the application of the weld overlay.

6.6 If water is in the feedwater line during the weld overlay process, the water is expected to range between 80-125°F.

7.0 CLASSIFICATION, BOUNDARY, AND INTERFACE REQUIREMENTS

7.1 Line nos. 1-B21-2-12-900, 1-B21-3-12-900, 1-B21-5-12-900, 1-B21-6-12-900 are part of the Nuclear Boiler System (System 1005) and are classified as Quality Classification A, per EDBS (screen function 404). The reactor pressure vessel is classified as Quality Classification A.

7.2 The affected feedwater supply (Nuclear Boiler) lines and reactor vessel are ASME Code, Class 1 components.

7.3 The boundary between the reactor pressure vessel design and the piping design is the outboard safe-end weld, i.e., the safe-end to pipe weld. (The safe-end is considered part of the reactor pressure vessel Ref. 3.2.1.2.)

7.4 Seismic classification is given in UFSAR Section 3.2 (Ref. 3.3.1.2). The affected portion of the feedwater supply (Nuclear Boiler) system is classified as seismic Class 1.

7.5 A thermocouple pad is welded to the old safe-end stub of the N4D and N4B nozzles. These pads may have to be removed and reinstalled in accordance with FP-55126, if the 1B21N4D-5-SW1-2 OR 1B21N4B-3-SW1-2 welds require overlay. Performance requirements will not change since the pads will be returned to their as-found condition.

8.0 MATERIAL REQUIREMENTS

8.1 No IGSCC susceptible material, as defined by NUREG-0313, Rev. 2 (Ref: 3.3.11.4), shall be used in the design of the weld overlay.

8.2 UNS N06052 Ni-Cr-Fe (Alloy 52) filler material shall be purchased in accordance with Specification CPL-XXXX-W-01 (ref 3.3.5.1), Code Case 2142 (ref. 3.3.7.12), and applicable portions of ASME Section II, Part C, SFA 5.14 (ref. 3.3.7.13).

8.3 Materials, e.g., tapes, markers, wrapping materials, cleaning solvents including demineralized water, etc., that come in contact with the reactor coolant pressure boundary components shall be in accordance with Specification CPL-XXXX-W-003, Attachment 1 (Ref: 3.3.5.5).

8.4 E-52 weld metal is equivalent to Alloy 690, and contains approximately twice the chromium (28-31%) of the more conventional Alloy 82 material (15%). This material has been shown in studies by EPRI and others to be highly resistant to IGSCC initiation and propagation, which is the postulated flaw mechanism at the Brunswick feedwater welds. The E-52 weld metal composition has been modified from Alloy 690 composition to enhance weldability, by increasing the content of such elements as aluminum.

8.5 Due to the high chromium content in the Alloy E-52, the material will be effective in mitigating crack propagation even when the effects of dilution with the base material are considered. Therefore, the first layer of the overlay deposited shall constitute the first layer of the weld reinforcement design thickness, without field verification of the alloy content.

9.0 MECHANICAL REQUIREMENTS

9.1 The mechanical design loads to be considered in the design of the weld overlay shall be those specified in the Pipe Stress Analyses (Ref. 3.3.4.14 and 3.3.4.15). These loads were developed based upon the requirements of the Codes, Standards, References and Regulatory requirements listed in Section 3.0.

9.2 The weld overlay shall have thickness and length as required in reference 3.3.6.13. The taper at the ends of the overlay should be approximately 30° from horizontal and shall not exceed 45°. The end of the overlay shall not be closer than 1/2" from the HAZ of adjacent welds. The weld overlay may terminate in the weld crown. Welds 1B21N4D-5-SW1-2 OR 1B21N4B-3-SW1-2 will require the overlay to terminate in the adjacent weld's crown, see para. 7.5.

10.0 STRUCTURAL REQUIREMENTS

10.1 The piping system design after installation of the weld overlay, will vary slightly from the existing configuration. The supports upstream of the weld overlay shall be inspected in accordance with the ISI Program (ref. 3.2.1.3). Adjustments to the supports will be made in accordance with plant procedures, if necessary. The existing supports shall be verified to be adequate for the new design and the existing stress analyses shall be reconciled with the new configuration and materials (ref. 3.3.4.14 and 3.3.4.15).

11.0 TEST REQUIREMENTS

11.1 The installation of the weld overlay shall be pressure tested to verify leak tightness in accordance with the ISI Program (ref. 3.2.1.3) and the requirements of OPT-080.1.

12.0 ACCESSIBILITY, MAINTENANCE, REPAIR, AND IN-SERVICE INSPECTION REQUIREMENTS

12.1 The overlay welds are in the reactor coolant pressure boundary and are subject to the inspection requirements of ASME Code, Section XI, Subsection IWB, *Rules for Inservice Inspection of Nuclear Power Plant Components* (Ref. 3.2.1.3).

12.2 The new configuration adds weld overlays for up to twelve (12) inconel welds in the reactor coolant pressure boundary portion of the feedwater supply (Nuclear Boiler) system. These welds are classified as Category D by NUREG-0313. Rev 2 (Ref. 3.3.11.4). These overlay welds will become part of the augmented inservice inspection program as administered by ENP-016 (Ref. 3.3.3.3). This document will have to be revised as a result of this project to incorporate these overlay welds in the NUREG-0313 inspection program.

13.0 CLEANLINESS, HANDLING AND STORAGE

13.1 The cleanliness of existing portions of the Nuclear Boiler system affected by this project shall not be degraded . The cleanliness of the existing portions of the Nuclear Boiler system shall be equal to or better than the condition found.

14.0 MATERIALS, PROCESSES, PARTS AND EQUIPMENT SUITABLE FOR APPLICATION

14.1 Welding Procedure Specifications shall be prepared and qualified in accordance with ASME Section III (ref. 3.2.1.4), ASME Section IX (ref. 3.2.1.5), Code Case 2142 (ref. 3.3.7.12), ANSI B31.1 (ref. 3.2.11), and Specification 248-003 (ref. 3.3.5.7).

14.2 The weld parameters will be established for dry conditions, additional testing using the same parameters will be performed on a sample filled with water to assure that no brittle fracture problems are introduced. The sample will be Charpy impact tested to the orginal design requirements.

14.3 A test coupon shall be prepared and tested to assure that the Charpy impact values for the base material Heat Affected Zone (HAZ) meet original design requirements after installation of the overlay welds. The test shall be on an ASME P No. 1 Group 2 material 0.843" thick ± 0.125" backed with water. The water temperature shall be controlled from 70° to 150° F. Weld a full thickness overlay 1" wide minimum tapered at 30° on the edge.

14.4 Obtain three Charpy specimens from the HAZ and three from the unaffected base material. The location for the specimens is at the toe of the weld, 1/4" from the toe, and 1/2" from the toe. The coupons shall be taken transverse to the weld and etched to define the HAZ. The notch will be aligned parallel to the fusion line.

14.4.1 The unaffected base material specimens shall be removed at approximately the same depth as the HAZ specimens. The axis of the specimens shall be the same as the HAZ specimens with the notch normal to the surface of the base material.

14.4.2 The Charpy V notch test will be performed in accordance with SA-370. Specimens shall be in accordance with SA-370, Figure 11, Type A. The test shall consist of full size 10mm x 10mm, specimens.

14.4.3 Acceptance criteria for the HAZ specimens is per ASME Boiler and Pressure Vessel Code, Section III, Division 1, Nuclear Power Plant Components, 1965 Edition Winter 1967 Addenda, Table N-421 for A 508 Class 1 (ref. 3.2.1.4). [Average of three specimens: 20 ft-lbs minimum; one individual: 15 ft-lbs minimum at a maximum temperature of 40°F].

15.0 INSTALLATION METHODS

15.1 Installation methods shall take into account that the work will be performed in a high radiation work environment.

15.2 The weld overlay will be applied using automatic welding machines and remote cameras.

15.3 The pipe is expected to be filled with water during the weld overlay. The pipe shall not be partially filled with water.

15.4 The profile of the overlay will support ISI and Augmented NUREG-0313 Rev. 2 inspection requirements. The examination area should be free of irregularities, loose material, and coatings which interfere with ultrasonic wave transmission. The short range surface finish should be 250 RMS or better and the long range waviness or out of flatness tolerance of 1/32" or less over a 2.0" flat area.

16.0 ALARA

16.1 The placing of temporary shielding on any safety related or potentially safety related component shall be evaluated in accordance with E&RC Procedure-4150 (Ref. 3.3.3.9). This is a requirement of the Generic Issue Design Basis Document for Shielding, DBD-104 (Ref. 3.3.3.10).

16.2 The feedwater nozzles are in an area of high radiation and lessons learned from previous weld overlays should be taken into account in the mock-up testing.

17.0 APPLICABILITY OF SECTION XI

17.1 This project is a repair under the auspices of ASME Code, Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components* (Ref. 3.2.1.3).

17.2 The design of the weld overlay shall be in accordance with ASME Code, Section XI 1989 ed., no addenda, with guidance from Code Case N-504. (Ref. 3.2.1.2).

17.3 Additionally, a liquid penetrant (LP) examination of the area(s) to be overlaid shall be performed prior to the weld overlay. The completed overlay shall be examined by LP and ultrasonic techniques. The acceptance criteria shall be to ASME Section XI (ref. 3.2.1.3)

17.4 This weld overlay is a repair to a piping system classified as ASME Section XI Class 1, and shall be documented on a NIS-2 form.

MECHANICAL IMPACT EVALUATION

Twelve (12) Inconel welds joining the feedwater lines to the safe-ends are to be examined during the next refueling outage in accordance with NUREG-0313 Rev. 2 per the In-Service-Inspection (ISI) Program. If a flaw is identified, it will be evaluated in accordance with the ISI Program requirements. The evaluation will determine if the flawed weld needs to be repaired by a weld overlay. If an overlay is needed, this modification will be activated.

During this modification, Inconel Alloy 52 weld metal will be applied to the outside of the existing feedwater inlet piping/safe-ends to a thickness and configuration capable of assuming all loads on the pipe, even assuming the original material has a 360° crack all the way through the original material. These overlays may be applied to restore full structural integrity to the three (3) welds on each safe-end susceptible to intergranular stress corrosion cracking (IGSCC), as well as to two carbon steel welds located nearby.

Once a setup is made to apply an overlay to one flawed Inconel weld, additional unflawed Inconel welds may also be overlaid at the same time both to mitigate IGSCC and to minimize future schedule impacts, resource costs and dose associated with future inspections which could be avoided by application of the overlay.

Weld overlays will be added to the ISI Inspection Program as an Augmented Inspection in accordance with the requirements of NUREG-0313 Rev.2. For unflawed welds that are weld overlaid, NUREG-0313 Rev. 2 has provisions for reduced future inspection frequencies.

These weld overlays are in a radiation area. Neutron fluence is negligible at the location of the feedwater nozzles; flux is $<1 \times 10^{17}$ n/cm^2/yr, >1 Mev. Similar materials do not indicate significant property changes until the fluence approaches 10^{20} n/cm^2 >1 Mev (ref. EER 93-0536).

The weld overlay will result in a slight shrinkage of the pipe inside diameter but will not affect flow. The safe-end and thermal sleeve have a smaller inside diameter than the resultant inside diameter after shrinkage of the overlays. The inside diameter will be reduced less than 1/2 inch due to shrinkage.

The Alloy 52 weld material has been shown in studies by EPRI and others to be highly resistant to IGSCC initiation and propagation. A crack in the pipe base metal due to IGSCC will terminate at the weld overlay to pipe interface and prevent leakage from occurring. The weld overlay thickness is also sized such that a through-wall flaw of the base pipe will not exceed the ASME Section XI, IWB-3500 allowable flaw sizes.

The profile of the overlay will support ISI and Augmented NUREG-0313 Rev. 2 inspection requirements. The examination area should be free of irregularities, loose material, and coatings which interfere with ultrasonic wave transmission.

The overlays will be inspected per NUREG-0313 Rev. 2 requirements to assure that no unacceptable flaws were introduced into the overlay. Pressure testing of the piping will also be performed in accordance with the ISI Program per ASME Section XI (ref. 3.2.1.3).

CIVIL/STRUCTURAL DESIGN IMPACT EVALUATION

The weld overlay will increase the outside diameter of the feedwater inlet piping to the reactor vessel. Pipe line shrinkage, axial and radial, from the weld overlay may affect the pin to pin settings of existing supports. Therefore, the supports must be evaluated and adjusted if necessary after the overlays are completed.

The safe-ends, extension pieces, and reactor nozzles are analyzed as part of the reactor, reference 3.3.6.12. The weld overlay does not affect these nozzle loads and also provides full design structural margin for the feedwater piping. Therefore reanalysis is not required.

MATERIALS DESIGN IMPACT EVALUATION

The weld overlay uses Alloy 52 filler material which has been shown in studies by EPRI and others to be highly resistant to IGSCC initiation and propagation. It will stop propagation of an existing through-wall crack at the weld metal interface. Alloy 52 is an acceptable overlay material for the base materials at each feedwater piping/safe-end to RPV location.

The weld overlay design uses successive weld beads to provide a cross sectional thickness which can meet all imposed loads and stresses on the piping even if the original material has a 360° crack through the full section of the original material. The overlay design requires sufficient overlap beyond the flaw to assure that stress loading of the original piping underneath the overlay is maintained within design requirements. The overlay design also requires tapered ends of approximately 30° from horizontal to avoid stress concentrations. The weld profile also supports inspection by UT.

The design also specifies a minimum distance of 1/2 inch to be maintained between the overlay and any other weld to assure that no embrittlement of the heat affected zone could result.

A mock up test is also planned to demonstrate that even if the weld is made with the pipe filled with water, the welding parameters used will not result in a reduction of base metal toughness below design requirements.

WELDING DESIGN IMPACT EVALUATION

The weld overlay uses Alloy 52 filler material which has been shown in studies by EPRI and others to be highly resistant to IGSCC initiation and propagation. It will stop propagation of an existing through-wall crack at the weld metal interface. Alloy 52 is an acceptable overlay material for the base materials at each feedwater piping/safe-end to RPV location.

Charpy impact tests will be performed on a test sample welded while filled with water to demonstrate compliance with design minimums.

The original Construction Code for the safe end and transition piece welds was ASME Section III (ref. 3.2.1.4). The original Construction Code for the feedwater piping was USAS B31.1.0 (ref. 3.2.1.1), supplemented by UE&C Specification 9527-248-003 (ref. 3.3.5.7). The UE&C Specification imposed ASME Section III Charpy impact test requirements on the qualification of the piping welds. Therefore, all impact test requirements are covered using ASME Section III (ref. 3.2.1.4), which will be used for qualification of the WPS and evaluation of the demonstration test samples.

SYSTEM IMPACT EVALUATION

The feedwater inlet piping to the reactor vessel is carbon steel up to the nozzle safe-end. A plant modification removed the original carbon steel safe-end, except for a short stub, and installed an Alloy 600 safe-end that accommodated a new feedwater sparger thermal sleeve design prior to initial plant start-up. The modification included three welds at each safe-end that are of a material which is susceptible to Intergranular Stress Corrosion Cracking (IGSCC) as defined in NUREG-0313 Rev. 2. NUREG-0313 Rev. 2 provides options for the repair or replacement of piping susceptible to IGSCC. This modification implements a weld overlay in accordance with ASME Section XI, 1989 ed. using material that is not susceptible to IGSCC.

The basic function of the feedwater supply system is to return the condensed steam from the turbine (via the condensate system) back to the reactor pressure vessel (RPV). The portion of the feedwater supply system affected by this project is inside primary containment and is classified as part of the reactor coolant pressure boundary Nuclear Boiler (B21) System. The feedwater lines are connected to the RPV nozzle via a safe-end.

- This project provides a weld overlay design for the three Inconel 82/182 welds connecting the RPV nozzle, safe-end, and feedwater piping in each feedwater line. These welds are in NUREG-0313 Rev. 2, as Category "D". The weld overlay will provide full structural reinforcement assuming a 360° through-wall crack. The overlay design uses a weld filler metal, Alloy 52, which is highly resistant to IGSCC and will mitigate crack growth into the weld overlay. The weld overlay design is in accordance with the requirements of ASME Section XI, 1989 ed., with guidance taken from Code Case N-504.

- The overlay geometry is designed to permit future inspections by visual, ultrasonic, and radiographic techniques to satisfy Augmented (NUREG-0313 Rev. 2) and In-service Inspection requirements.

- The weld overlays may be applied over adjacent welds to minimize the dose associated with future repairs even if the adjacent weld does not have an identified flaw.

The structural integrity of the reactor coolant pressure boundary is not degraded by this project. Performance requirements for structural integrity of the Nuclear Boiler system and RPV are unchanged by this project. The weld overlay design meets the repair requirements of ASME Section XI, 1989 ed. The overlay weld material, Alloy 52, is highly resistant to IGSCC and will mitigate crack growth into the overlay. The overlay design has been verified by analysis to be acceptable for the existing design loads and materials. The weld overlay will not prevent the pipe supports from performing their intended function and does not add a significant load on the supports.

The overlay is designed to provide full structural design margin with no credit taken for the base material at the crack location. The weld overlay is a series of weld beads that builds up the outside of the pipe to meet the minimum wall thickness required by ASME Section XI. Alloy 52 has a high chromium content and is considered highly resistant to IGSCC, so crack propagation into this material is unlikely.

Applying the weld overlay onto the carbon steel portion of the feedwater piping may be necessary. The feedwater piping may be filled with water during the weld overlay process. Parameters will be tested to assure that the original design requirements for Charpy impact tests are met.

The profile of the overlay accommodates future inspections required by ASME Section XI and NUREG-0313 Rev. 2 to verify any identified flaw remains within design allowables. This ensures that structural integrity and corresponding safety margins of the piping will be maintained.

The hydraulic performance requirements, i.e., flow rate of the feedwater lines are unchanged by this project. While the weld overlay may result in a slight constriction of the piping resulting from weld metal shrinkage; the safe-end and thermal sleeve have a smaller inside diameter than at the locations where the weld overlays will be applied, approximately 1/2". Since shrinkage from the weld overlays will be less than 1/2", the thermal sleeve remains the most restrictive diameter.

NUREG-0313
Rev. 2

Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping

Final Report

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ABSTRACT

This report updates and supersedes the technical recommendations of NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," published in July 1977, and its subsequent revision published in July 1980.

This report provides the technical bases for the NRC staff's revised recommended methods to control the intergranular stress corrosion cracking susceptibility of BWR piping. For piping that does not fully comply with the material selection, testing, and processing guideline combinations of this document, varying degrees of augmented inservice inspection are recommended. This revision also includes guidance and NRC staff recommendations (not requirements) regarding crack evaluation and weld overlay repair methods for long-term operation or for continuing interim operation of plants until a more permanent solution is implemented.

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ABBREVIATIONS

ANL	Argonne National Laboratory
ASME	American Society of Mechanical Engineers
ASTM	American Society of Testing Materials
BWR	Boiling Water Reactors
BWROG	Boiling Water Reactor Owners Group
CFR	Code of Federal Regulations
CRC	Corrosion Resistant Cladding
EPRI	Electric Power Research Institute
GE	General Electric
GMAW	Gas Metal Arc Welding
GTAW	Gas Tungsten Arc Welding
HAZ	Heat Affected Zone
HSW	Heat Sink Welding
HWC	Hydrogen Water Chemistry
IE	Office of Inspection and Enforcement
IGSCC	Intergranular Stress Corrosion Cracking
IHSI	Induction Heating Stress Improvement
ISI	Inservice Inspection
LPHSW	Last Past Heat Sink Welding Process
MSIP	Mechanical Stress Improvement Process
NDE	Nondestructive Examinations
NRC	Nuclear Regulatory Commission
PNL	Pacific Northwest Laboratory
PPB	Part per Billion
PT	Penetrant Inspection
RHR	Residual Heat Removal
SECY	Office of the Secretary of the Commission
SHT	Solution Heat Treatment
SI	Stress Improvement
UT	Ultrasonic Testing

EXECUTIVE SUMMARY

This revision to NUREG-0313, Rev. 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping" provides the technical bases for the staff's recommendations regarding actions that can be taken to ensure that the integrity and reliability of BWR piping will be maintained.

The staff long-range plan regarding BWR pipe cracks was presented to the Commission in SECY 84-301. A major task in this plan was to revise NUREG-0313 to include the recommendations of the Piping Review Committee Task Group on Pipe Cracking, issued as NUREG-1061, Vol. 1.

The subjects covered by this revision include recommendations regarding piping and weld material, special processing to minimize crack susceptibility, improvements in BWR primary coolant chemistry and control, inspection requirements, repair methods, and leak detection. These recommendations and conclusions are consistent with those made in NUREG-1061, Vol. 1, and are summarized as follows:

BWR piping weldments made of austenitic stainless steel are susceptible to intergranular stress corrosion cracking (IGSCC). The three elements that, in combination, cause IGSCC are, a susceptible (sensitized) material, a significant tensile stress, and an aggressive environment.

The staff technical recommendation is that improvements in all three of these elements should be pursued. Nevertheless, significant reduction in the probability of IGSCC can be accomplished even by improving one or two of these three elements. From a practical standpoint, this is more readily accomplished in the near term, and can provide acceptable assurance of continued integrity and reliability.

There is no practical way to reduce the sensitization of weldments already installed, so the only way to reduce the susceptibility of the material is to replace the piping with material that is resistant to sensitization by welding. Solution heat treatment of individual spool pieces in the pipe fabrication shop before field erection is practicable, and is recommended. Austenitic materials considered by the staff to be adequately resistant to sensitization by welding are the following:

(1) Low carbon wrought austenitic stainless steel. These include 304L, 304NG, 316L, 316NG, 347NG, and similar types.

(2) Low carbon weld metal of type 308L and similar grades with a minimum of 7.5% ferrite as deposited. This may also be used as a cladding on the inside of the pipe.

(3) Cast austenitic stainless steel with less than 0.035% carbon and a minimum of 7.5% ferrite.

(4) Other materials such as nickel base alloys, etc., may be sufficiently resistant, and may be evaluated in special cases. Inconel 82 is the only nickel base weld metal considered to be resistant.

Service-induced stresses on most BWR piping are relatively low. The source of the high stress primarily responsible for IGSCC is the high tensile stress on the inside of the pipe caused by normal welding practice. Stress Improvement (SI) can be accomplished on weldments already installed by the Induction Heating Stress Improvement (IHSI) process, or by the Mechanical Stress Improvement Process (MSIP).

SI can be applied to new or replaced piping, or can be applied at any time during plant life. The staff strongly recommends that SI be applied on all new or replacement piping, and preferably within two years for piping already installed. For piping with more than 2 years of operation, SI is considered to be less effective, because cracking may already be present.

BWR primary coolant normally contains oxygen from radiolytic dissociation of water, and also contains other impurities such as chlorides, carbonates, and sulfur species. If the oxygen levels are reduced by using hydrogen injection, and other impurities are kept to very low levels, IGSCC of even sensitized material will be drastically reduced. This combination of water chemistry improvement is referred to as Hydrogen Water Chemistry (HWC). The staff recommends that HWC be implemented as soon as the practical and safety aspects have been worked out.

Some utilities have decided not to replace piping at this time. The staff has developed guidelines for interim actions that should be taken in these instances. Augmented inspection schedules for susceptible and repaired weldments are based on judgment regarding the probability that significant cracks or leaks will develop, considering the effectiveness of any repair or mitigative actions applied.

The staff believes that replacing degraded, susceptible piping with IGSCC resistant materials will provide the highest degree of assurance against future cracking problems. Nevertheless, the staff concludes that if the recommendations provided herein are implemented, adequate levels of piping integrity and reliability can be achieved.

The approved Staff Positions derived from the recommendations in this Report are implemented by Generic Letter 88-01.

TECHNICAL REPORT ON MATERIAL SELECTION AND PROCESSING GUIDELINES FOR BWR COOLANT PRESSURE BOUNDARY PIPING

1.0 INTRODUCTION

1.1 History

The subject of intergranular stress corrosion cracking (IGSCC) at welds in boiling water reactor (BWR) piping has been of continuous concern for almost 20 years. An ever-increasing amount of research and developmental activity related to understanding the causes of the cracking and ways to prevent it has been going on during this time period. Under the auspices of NRC, two Pipe Crack Study Groups have reviewed the problem in BWRs--one in 1975 and the other in 1979. Reports of the findings of these groups were published (NUREG-75/067 and NUREG-0531), and staff guidelines prepared to implement their recommendations were published as NUREG-0313 entitled "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," and NUREG-0313, Revision 1.

Until recently, significant cracking of large-diameter piping (12-in. diameter and larger) was considered to be relatively unlikely, and even if it occurred it was expected that cracks would remain shallow. In Japan some cracks had been detected in the 12-in-diameter recirculation riser pipes. Because of this, NUREG-0313, Rev. 1 recommended that augmented inservice inspection (ISI) on a sampling basis be performed for these pipes. Shallow cracking was discovered in pipes larger than 12-in-diameter in Germany but it was not clear that either the Japanese or German experience was relevant to plants built in the United States.

During a hydrostatic test in March 1982, slight leakage was detected at two of the furnace-sensitized recirculation safe ends at Nine Mile Point. When these safe ends had been examined ultrasonically 9 months earlier, no cracking was reported. Additional ultrasonic testing (UT) using more sensitive procedures disclosed cracks at many of the 28-in-diameter recirculation piping welds.

This finding was important for two reasons:

(1) It could no longer be believed that large pipes were relatively immune to significant cracking.

(2) It cast doubt on the adequacy of the UT procedures used at that time to detect cracks in large pipes.

IE Bulletin 82-03 was issued to specify augmented inspections of large piping in the recirculation systems of plants (9 units) with outages scheduled in late 1982 and spring 1983. It also specified that

inspection teams demonstrate that they could detect and properly identify cracks in large-diameter pipe welds. IE Bulletin 83-02 was later issued to require inspections at all other operating BWRs (14 units) with more than 2 years of operating service, and to upgrade the UT performance capability demonstrations required of the inspection teams. Reinspections at the next refueling outage were required by Generic Letter 84-11, which also provided specific guidance regarding flaw evaluation and repair for interim operation.

The results of these inspections varied greatly from plant to plant. Some found very little, if any, cracking. Others found very significant cracking in a large percentage of the recirculation, residual heat removal (RHR) system, and reactor water cleanup system piping welds.

The discovery of significant cracking in the large-diameter piping, the development of ASME Code procedures for evaluating flaws in such piping, and results of further development of materials and processes to mitigate or prevent IGSCC led to the decision to revise NUREG-0313.

1.2 Revision 1 of NUREG-0313

NUREG-0313 was revised in 1980 to provide guidance and recommendations regarding materials and processes that could be used to minimize IGSCC and to provide recommendations about augmentation of the extent and frequency of ISI on welds considered to be susceptible to IGSCC.

Revision 1 also provided recommendations about upgrading leak detection systems and leakage limits for plants with susceptible welds.

1.3 Revision 2 of NUREG-0313

This present (second) revision updates these recommendations and adds several subjects:

(1) It provides guidance for performing ASME Code, Section XI, IWB 3600, calculations for flaw evaluation.

(2) It provides recommendations regarding repair of cracked piping.

(3) It recommends formal performance demonstration tests for UT examiners, such as those prescribed by IE Bulletins 82-03 and 83-02 and currently being conducted under the NDE Coordination Plan, agreed upon by NRC, EPRI, and the BWROG. This will provide additional assurance that inspections for IGSCC in BWR piping will be performed in an effective manner.

The approach used in previous editions of NUREG-0313 to identify welds that require augmented inspection is simplified, but is expanded to include consideration of reinspections of welds found to be cracked, with or without repair or mitigation actions. The current approach is based on the following:

(1) All stainless steel welds in high-temperature BWR systems are considered to be subject to IGSCC unless measures have been taken to make them resistant.

(2) The frequency and sample size used to inspect all safety related piping welds in BWR plants will depend on the material and processing used. Simple bases are provided for such classification.

(3) Some utilities may choose not to replace, or to operate for some interim period of time before making major modifications or replacing piping. This would mean that operation with cracked or repaired welds will be desired. Guidance is provided to cover these situations.

1.4 Bases for Recommendations

Extensive work sponsored by industry through the Electric Power Research Institute (EPRI), General Electric (GE), and the U.S. Nuclear Regulatory Commission (NRC) has been carried out since the second Pipe Crack Study Group reported in 1978-1979 (NUREG-0531). It is not the objective of this report to cover this work in detail. NUREG-1061, Vol. 1 was prepared by the Pipe Crack Task Group of the Piping Review Committee. It represents an in-depth discussion of the technical aspects of IGSCC in BWR piping, and provides recommendations regarding materials and processes available to mitigate or eliminate the problem. It also includes a discussion of the technical basis for the guidelines for interim operation used by the staff.

This revision is based primarily on the information presented in NUREG-1061, as modified by more recent advances in ultrasonic testing and fracture mechanics evaluation methods. It also takes cognizance of work in progress related to serviceability of cracked pipes reinforced by weld overlay or mitigated by IHSI being performed at General Electric and PNL under EPRI and the BWROG sponsorship, and related work at ANL funded by the NRC, as well as public comments received on NUREG-1061, Vol. 1.

1.5 Piping Replacement

As stated in the staff paper to the Commission (SECY-84-301), it is the staff's long range goal to bring all affected plants in line with regulations without undue reliance on augmented inspections. Although not required, utilities with degraded and repaired piping systems should consider replacing such piping in their future plans, taking into account relevant aspects of their situations.

Procedural guidance regarding pipe replacement licensing activities is provided in Generic Letter 84-07, dated March 14, 1984.

2.0 METHODS TO REDUCE OR ELIMINATE IGSCC

There are three primary ways to minimize the occurrence of IGSCC in BWR piping:

(1) Use material that is not subject to sensitization by welding, or solution heat treat after welding.

(2) Use processes that reduce the tensile stress level at the inner surface of the pipe near the weld.

(3) Modify the BWR water chemistry to control the levels of oxygen and other aggressive contaminants to very low levels.

Each of these three basic approaches are discussed below, and recommendations regarding each are presented.

2.1 Materials for New or Replacement Piping

Sensitization involves carbon diffusion out of solution forming carbides at grain boundaries upon moderate heating; therefore, reducing the carbon content of the material will result in reducing the degree of sensitization resulting from a given thermal exposure, assuming that other factors remain equal. However, because the susceptibility of an austenitic stainless steel is also affected by other variables, such as grain size, previous heat treatment, amount of cold work, trace impurities, and overall compositional balance, complete dependence on reduced carbon content may not be effective unless the carbon level is very low. Nevertheless, a high degree of protection against IGSCC will result if the carbon content is kept below 0.035%, as specified for type 304L grade material. Freedom from sensitization will be much more certain if the carbon levels are controlled to even lower levels.

If carbon is limited to very low levels (such as below 0.02%), the strengthening effect of the carbon is lost, and the material has lower strength, which results in lower Code-allowable stresses. Some heats of type 304L material will also have strength levels too low to meet the minimum specified strength level for standard type 304. Therefore, the replacement of piping with low carbon grades may require redesigning or using thicker wall pipes.

Industry has overcome these problems by developing special grades of austenitic stainless steel. Carbon content is kept very low, and the reduction in strength is compensated for by adding controlled amounts of nitrogen. Molybdenum is often added; it enhances strength and resistance to sensitization.

The grades of austenitic stainless steels developed for increased resistance to sensitization are listed below.

Series 300 stainless steels developed for increased resistance to sensitization

Steel	C %, max.	Cr %	Ni %	Mo %	N %
304L	0.035	18.0-20.0	8-10.5	--	--
304NG*	0.02	18.0-20.0	8-12.0	--	0.06-0.10
316L	0.035	16.0-18.0	10.0-14.0	2.0-3.0	--
316NG*	0.02	16.0-18.0	10.0-14.0	2.0-3.0	0.06-0.10
347NG	0.03	17.0-19.0	9.0-13.0	**	

*304NG and 316NG were formerly called 304K and 316K, respectively.
**Minimum Nb + Ta = 10 x %C.

Weld metal with low carbon and controlled ferrite (such as 308L with 7.5% minimum ferrite) is resistant to sensitization and IGSCC. This resistance is also somewhat dependent on the microstructure produced by the specific welding process used. Weld passes diluted with high carbon base material will not have suitable resistance

Cast austenitic stainless steel with low carbon and high ferrite content is also resistant to sensitization and IGSCC.

Other common materials such as carbon steels are suitable for many BWR piping systems and are immune to the problem of sensitization and resultant IGSCC. Higher strength alloy steels are less desirable; they may be subject to other types of cracking.

2.1.1 Staff Recommendations on Materials

The materials considered resistant to sensitization and IGSCC in BWR piping systems are:

(1) Low carbon wrought austenitic stainless steel, which includes types 304L, 304NG, 316NG and similar low carbon grades with a maximum carbon content of 0.035%. Type 347, as modified for nuclear use, will be resistant with somewhat higher carbon content, the usual maximum of 0.04% is adequate. These materials are generally tested for resistance to sensitization in accordance with ASME A262-A, -E1, or equivalent standard.

(2) Low carbon weld metal, including types 308L, 316L, 309L and similar grades, with a maximum carbon content of 0.035% and a minimum of 7.5 percent (or FN) ferrite as deposited. Low carbon weld filler material especially developed for joining modified type 347 is also resistant as deposited.

Welds joining resistant material that meet the ASME Boiler and Pressure Vessel Code requirement of 5 percent (or FN) ferrite, but are below 7.5% may be sufficiently resistant, depending on carbon content and other factors. These will be evaluated on an individual case basis.

(3) Piping weldments are considered resistant to IGSCC if the weld heat affected zone on the inside of the pipe is protected by a cladding of resistant weld metal. This is often referred to as corrosion resistant cladding (CRC).

(4) Cast austenitic stainless steel with a maximum of 0.035% carbon and a minimum of 7.5 percent (or FN) ferrite. Weld joints between resistant piping and cast valve or pump bodies that do not meet these requirements are considered to be special cases, and are covered in the Staff Position on Inspection Schedules below.

(5) Austenitic stainless steel piping that does not meet the requirements of (1) above is considered to be resistant if it is given a solution heat treatment after welding.

(6) Other austentic materials, including nickel base alloys such as Inconel 600, will be evaluated on an individual case basis. Inconel 82 is the only commonly used nickel base weld considered to be resistant.

The staff recommends that no austenitic material be considered to be resistant to cracking in the presence of a crevice, such as formed by a partial penetration weld, where the crevice is exposed to reactor coolant.

2.2 Processes for New, Replacement, or Older Piping

Special or controlled processing during or after fabrication can provide protection from IGSCC in three ways:

(1) removing sensitization,
(2) preventing sensitization, and
(3) providing favorable state of residual stress.

There are several special processes that have proved effective in one or more of these ways; they are discussed below:

Solution Heat Treatment

The normal metallurgical treatment used to ensure freedom from sensitization is to perform a complete solution heat treatment (SHT) to the piece after welding or other processing. It consists of heating the material to a high enough temperature to dissolve all carbides, then cooling fast enough to retain the carbon in solution. Standard specifications are used to control the process; the chief concern is providing fast cooling.

Note that the solution heat treatment must be performed after welding, and complex piping sections may be difficult to cool fast enough from the solution temperature. Interiors of long or complex piping runs may pose a particular problem.

To be effective, solution heat treatments must be performed in accordance with written procedures that have been proven to be effective for the size and geometry of the piece, and must be in accordance with applicable specifications.

Heat Sink Welding

Heat sink welding (HSW) is a term applied to a method of butt welding pipes or fittings in which the major portion of the weld is produced with cooling water inside the pipe. The cooling effect of the water minimizes the sensitization caused by the welding process, and in addition, produces a steep temperature gradient through the pipe wall during welding. This steep temperature gradient causes tensile thermal stresses on the inside of the pipe to exceed the yield strength of the material. After the welding is completed and the weldment is cooled, the inner portion of the weld is under high compressive residual stress. This is the opposite of what is caused by normal welding. The high compressive stresses are maintained through about half the wall thickness. The combination of reduced sensitization and high beneficial residual stresses provides significant resistance to IGSCC.

Stress Improvement Processes

One of the major sources of stress causing IGSCC is the residual tensile stress that remains on the inside of the weld joint after the normal butt welding process. Processes have been developed that effectively reverse this residual stress distribution, and actual pipe tests have shown that this is very effective in inhibiting IGSCC in sensitized welds that have been treated by a Stress Improvement Process (SIP). There are two such processes that are considered fully qualified to provide this mitigation.

Induction Heating Stress Improvement (IHSI)

Induction heating stress improvement (IHSI) is a process originally developed in Japan for treating piping weldments already fabricated or installed in a plant. It consists of heating the outside of the pipe by induction coils to controlled temperatures (≈800°F) while cooling water is circulated inside the pipe. The high gradients produce the same effect as HSW. The inside of the pipe is plastically strained in tension during the process, causing residual compressive stresses after the process is completed.

Mechanical Stress Improvement Process

The Mechanical Stress Improvement Process (MSIP) is a later development that uses a hydraulic system to uniformly compress the entire pipe at a location near the weld joint. It also causes slight plastic strain, and the residual stresses remaining after the treatment are compressive in the location susceptible to IGSCC because of weld sensitization.

Last Pass Heat Sink Welding

The last pass heat sink welding (LPHSW) process is similar to HSW, except that only the last welding passes are performed when there is cooling water inside the pipe. Although some preliminary tests appear promising, it cannot be considered to be fully effective at this time.

2.2.1 Staff Recommendations on Processes

The processes considered to be qualified for providing resistance to IGSCC in BWR piping welds are:

(1) Solution Heat Treatment (SHT)
(2) Heat Sink Welding (HSW)
(3) Induction Heating Stress Improvement (IHSI)
(4) Mechanical Stress Improvement Process (MSIP)

Although last pass heat sink welding (LPHSW) is not considered to be fully qualified, specific cases may be evaluated individually.

2.3 Water Chemistry Modifications

Intergranular stress corrosion cracking of sensitized and stressed stainless steel requires a corrosive environment. Although BWR reactor coolant is comparatively pure water, the small amounts of impurities usually present are enough to cause IGSCC. These impurities fall into two general classes; those that increase the oxidizing potential, and those that increase the electrical conductivity of the water. Both must be reduced to very low levels to achieve an electrochemical potential below which IGSCC cannot be initiated or propagated.

Oxygen is formed in the core of light water reactors by the disassociation of water by radiolysis. This reaction can be inhibited by the addition of hydrogen to the water, as is done in pressurized water reactors. Until recently, this was not considered to be feasible in boiling water reactors, therefore, the normal oxygen content of BWR reactor water is about 200 parts per billion (PPB), providing an oxidizing environment conducive to IGSCC in the entire BWR primary system.

Efforts to find ways to reduce the oxygen levels in BWRs led to the development of a hydrogen addition methodology that appears to be effective and practicable. Tests conducted in the Dresden 2 plant over the past several years indicate that oxygen levels can be reduced to levels of 10 to 20 PPB, although occasional excursions to higher levels may occur. Tests indicate that IGSCC will not occur at an oxygen level of 20 PPB or less, if other contaminants are controlled to keep conductivity low.

Contaminants that increase the conductivity of the reactor water can come from several sources, such as condenser leakage, resin beds, etc. They include chlorides, carbonates, and sulfur species. Because the electrochemical potential causing IGSCC depends on both the oxidizing state and the conductivity of the water, the conductivity must be held to very low levels. Laboratory tests have indicated that conductivity levels should be kept to a maximum of 0.3 micro-Siemens (µS) per centimeter with oxygen at 20 PPB or less to prevent IGSCC. Although the tests in Dresden 2 indicated that such conductivity levels could be attained, occasional excursions must be anticipated, and plant to plant variations are likely to be significant in this regard.

This combination of oxygen and conductivity control is commonly referred to as Hydrogen Water Chemistry, or HWC. Although tests have shown that HWC can inhibit IGSCC, some questions regarding radiation effects, fuel

performance, etc. are still being resolved. Field implementation and engineering are being actively pursued by the industry, and it is expected that within the next few years, HWC will be considered a practical method of control.

2.3.1 Staff Recommendation on Water Chemistry

The use of hydrogen water chemistry, together with stringent controls on conductivity, will inhibit the initiation and growth of IGSCC. However, the responses of BWRs to hydrogen injection differs from plant to plant, and the development and verification of a generic HWC specification is not yet complete. For these reasons, reduction in piping inspection frequency based on the use of HWC will be considered on an individual case basis at the present time. Staff criteria for evaluating the effectiveness of HWC are under development. If fully effective HWC is maintained, a factor of two in reduction of inspection frequency may be justified for susceptible weldments.

3.0 EVALUATION AND REPAIR OF CRACKED WELDMENTS

When cracks are found in BWR piping, several alternatives (and combinations) are available to provide assurance of further safe operation of affected welds.

If the cracking is not too severe, the rules of ASME Code Section XI, IWB 3600 (as modified and expanded in Section 8) may be used for short-term interim operation. Further, SI may be applied to reduce the probability of further crack growth.

If the cracking is too severe to meet these rules, the affected piping must be repaired or replaced before the plant can be returned to service.

3.1 Repair Procedures

IGSCC in BWR piping initiates at the inner surface of the pipe and grows progressively through the wall toward the outside. It commonly initiates near the weld root and progresses up the heat-affected zone (HAZ) close to the weld, and sometimes in the weld. Therefore, cracking can affect a region of the pipe longer in axial extent than the maximum width of the weld if cracks occur on both sides of the weld. The usual repair process during construction is to grind out the defective area and fill the area with weld metal. This is not practical for repair of IGSCC, because IGSCC starts from the inside surface, requiring removal of essentially the entire weld and HAZ area.

There are several repair methods available for at least short-term operation:

(1) Weld overlay reinforcement
(2) Partial replacement
(3) SI (for minor cracks)
(4) Approved clamping devices

These are discussed below.

3.1.1 Weld Overlay Reinforcement

Weld overlay reinforcement consists of applying weld metal over the weld and for a specified minimum distance beyond the weld on both sides. This is done completely around the outside surface of the pipe overlapping each pass. IGSCC-resistant low-carbon, high-ferrite type 308L weld metal is used, and the process is usually performed with an automatic welding machine using the Gas Tungsten arc (GTAW) or Gas Metal arc (GMAW) processes. Weld overlay is performed with cooling water in the pipe during welding, and there is no need to drain the pipe during repair. More specific design details and quality control recommendations are covered in Section 4.0.

3.1.2 Partial Replacement

A very effective repair method is to cut out a section of the pipe containing the defective weldments and to weld in another piece of pipe. The major drawback to this approach is that the affected run of pipe must be drained and dried. Either all fuel must be removed from the reactor vessel or special plugs must be installed when this type of repair is used in portions of piping that cannot be isolated.

If this method can be used, a fully effective repair can be made with resistant material, using welding processes such as heat sink welding for the new installation welds and high-ferrite type 308L weld material. SI can also be applied.

Another disadvantage of this process (assuming that draining is feasible) is that high radiation exposures to workers may be encountered at older plants from the inner surfaces of the pipes. Prior decontamination can alleviate this problem.

Both weld overlay and partial replacement cause the pipe to shrink in the axial direction. If several such repairs are made in one length of pipe, additional stresses will be introduced by this shrinkage which must be taken into account in the stress analysis required for the repair, and in the fracture mechanics analyses of crack growth in other welds of the pipe system. Measurements of shrinkage on weld procedure qualification test pieces can provide guidance regarding how much shrinkage can be expected. Actual measurements made during the repair should be used in the final stress analysis.

3.1.3 Stress Improvement

As discussed above, SI alters the residual stress pattern, putting the inner part of the pipe wall in compression, thus inhibiting crack initiation. If cracks are present, the situation is more complex. If cracks are shallow the process will probably prevent further growth, as long as the residual stress pattern remains favorable. The process may stretch cracks open but tests have shown that they are not extended in depth by the process. Such stretching may even be beneficial for shallow cracks because it enhances the resulting compressive stress around the crack tip.

The tips of deeper cracks, particularly those penetrating deeper than half way through the pipe wall, are likely to be in a general tensile stress field after SI processes. This could cause such cracks to propagate through the wall, faster than they would without the SI treatment. Cracks will not be expected to grow longer because of the beneficial residual stress on the inside portion of the pipe. Therefore, neither short cracks of medium depth or longer shallow cracks are expected to grow to a significant size after an SI treatment.

3.1.4 Mechanical Clamping Devices

Another approach to reinforcing a cracked weldment is to use a mechanical clamp. One advantage of this approach is that the clamp may be periodically removed for weld examination. Such clamping devices will be reviewed for adequacy of mechanical design, materials of construction, and installation methods on a case basis.

3.2 Staff Recommendations on Repairs

3.2.1 Staff Recommendations on Weld Overlay Reinforcement

Weld overlay reinforcement made in accordance with recommendations described in this report are considered to be acceptable at least for short-term operation. Weld overlay may be considered for longer term operation provided:

(1) The overlays are in conformance with the criteria of Section 4.0 of this report; and

(2) they are inspected in accordance with the criteria of Section 5.0 by UT examiners and procedures qualified to inspect overlayed welds.

Weld overlays not meeting (1) above may be reinforced to the extent necessary to meet the staff position, if desired.

3.2.2 Staff Recommendations on Partial Replacement

Repair of cracked weldments by partial replacement can be considered to be fully effective if appropriate materials and weld processes are used, and therefore are considered to be resistant to IGSCC.

3.2.3 Staff Recommendations on SI of Cracked Weldments

SI may be considered as a partial mitigation process when applied to weldments with short or shallow cracks. Details of allowable crack sizes in this regard are covered in the next section. Note that SI is only considered effective if it is followed by a qualified UT examination, and if cracks are found they must be sized, both in depth and length, by procedures and personnel qualified to perform sizing examinations according to recommendations given in Section 5.1 of this report.

3.2.4 Staff Recommendations on Clamping Devices

Clamping devices may be used for temporary reinforcement of cracked weldments. Each case must be reviewed and approved on an individual basis.

4.0 CRACK CHARACTERIZATION AND REPAIR CRITERIA

4.1 Flaw and Repair Evaluation Criteria

This section provides guidance and staff positions regarding methods to evaluate IGSCC cracks for limited further operation. It also covers evaluation methods and acceptance criteria for repairs if immediate pipe replacement is not practicable.

The methods and criteria described in this section are generally in accordance with IWB 3640 of Section XI of the ASME Boiler and Pressure Vessel Code. In particular, IWB 3642 provides for flaw evaluation using fracture mechanics or other applicable methods. The Code requires that crack growth be calculated, and the flawed joint is acceptable for further operation only for the time period that the flaw remains small enough that the Code-intended safety or design margins are maintained.

In IWB 3641, the Code (Winter 83 Addenda) provided simple tables of allowable crack depth as a function of the primary stress level and crack length. These tables are based on limit load calculations, and assume that the material is tough. An overall margin of about 2.77 against net section collapse (limit load) failure mode is factored into the tables.

It was recognized that these tables did not provide an acceptable level of margin against failure for low toughness materials such as fluxed welds (SAW, SMAW). This is because low toughness material may fail at load levels below limit load, and secondary stresses (not considered in the original IWB 3641 tables) may also contribute to failure of low toughness materials.

This problem has now been addressed by the Code, and the 1986 Edition provides appropriate criteria for all types of welds.

4.2 Crack Growth Calculations

The rate of growth of cracks by IGSCC has been the subject of discussion and controversy for many years. Part of the problem is that the rate of growth as a function of stress is affected by the degree of sensitization of the material and the severity of the environment. A further complication has been that ways to measure the degree of sensitization have proved to be inaccurate or n relevant to the particular problem of BWR piping. For these reasons, many crack growth tests have been performed that were either too severe or not severe enough. The staff recommends a crack growth rate curve that is believed to be near the upper bound for weld-sensitized material in actual BWR environments. (See Appendix A)

Crack growth by IGSCC appears to follow a classical trend. If the logarithm of the growth rate is plotted against the logarithm of severity of loading, measured by the stress intensity factor (a fracture mechanics parameter) K_I, a linear relationship is found. As the K_I changes with crack growth, iterative calculations will track the growth of the crack with time. The calculational procedures recommended by the staff to predict crack growth are detailed in Appendix A.

Actual circumferential cracks in welds are usually very long in relation to their depth; therefore, crack growth in a congruent manner (maintaining the same shape) cannot be assumed, particularly for large-diameter pipes. The growth in the length direction, therefore, may be more than in the depth direction. Specifically, the growth along the length should be assumed to increase the aspect ratio (length to depth) by the same factor that the depth is increased. For example, if a crack with an aspect ratio of 3 to 1 grows to twice the original depth, the new length will be assumed to give an aspect ratio of 6 to 1. Cracks with aspect ratios over 20 to 1 are assumed not to change shape with crack growth.

Although axially oriented cracks are not likely to grow significantly beyond the sensitized zones on each side of the weld, they will grow through the weld if the weld metal is marginal in resistance to sensitization, and therefore was sensitized during welding. Axial cracks will therefore be assumed to grow through the wall but the length is limited to 1.5 times the thickness of the pipe.

4.3 Multiple and Complex Crack Characterization

Case 1

If Multiple cracks are present that will remain less than 20% of the circumference in total length after crack growth, they may be treated as one crack with the length equal to the sum of the lengths.

Case 2

If multiple cracks are present that will remain less than 30% of the circumference in total length after crack growth, they may be treated as one crack with the length equal to the sum of the lengths, provided that after crack growth each crack is separated by at least 20% of the circumference from all other cracks.

Case 3

All other situations regarding multiple cracks will be considered as a single 360° crack.

Case 4

Cracks on both sides of the weld will be treated as if they were all on the side of the weld with the thinnest wall; overlapping cracks or overlapping areas are considered as one crack.

4.4 Weld Overlay Design Criteria

4.4.1 Standard Overlay Design

The standard overlay should be designed to provide a nominal margin of 2.77 against limit load failure, assuming that the original crack was completely through the wall for 360°. The calculation method described in Section 4.1 is recommended. Because none of the original weld or heat affected zone is considered in the analysis, the stresses to be used in the analysis depend only on the kind of weld metal used for the overlay. Specifically, if the overlay is made using GTAW or GMAW processes, secondary stress need not be considered. Calculations are made using the as-overlayed joint dimensions and stress levels.

4.4.2 Design Overlays

In cases where cracks are perpendicular to the weld (axial) or short in the circumferential direction, even a small amount of overlay will prevent further growth in the length direction, because high compressive stresses are induced at the inner surface of the pipe. In such cases the overlay will also act to prevent leakage.

Weldments with a total length of circumferential cracking less than approximately 10% of the circumference, with no more than four axial cracks, are considered apropriate for repair by a designed overlay. A standard overlay should be used for more severe cracking.

The thickness of the designed overlay should be at least two layers of weld metal after the surface has passed surface examination by penetrant inspection (PT). If credit is taken for the thickness of the first layer, it should be shown by actual test to contain a minimum of 7.5% ferrite, and the original surface must have passed PT.

Because designed overlays take credit for part of the original pipe in their design, there are several ways that the lower toughness of the original fluxed weld may be taken into account. An acceptable design approach is to assume that the crack or cracks requiring the overlay are completely through the original pipe wall for the total length of crack involved. The overlay thickness is calculated so that the as-overlayed cracked weldment meets the IWB 3641 tables in Section XI of the ASME Boiler and Pressure Vessel Code.

Other approaches to overlay design may be evaluated on a case basis. In general, it is recommended that highly stressed welds should be reinforced with standard overlays.

4.4.3 Limited Service Overlays

Overlay designs not meeting the above criteria for either Standard or Designed overlays are only recommended for limited service, such as one fuel cycle of operation. (See 5.3.2.6)

4.5 SI Crack Mitigation Criteria

In general, SI is only recommended for use on weldments with minor cracking. This is because the tips of deep cracks can be in an area of high tensile stress caused by the process, and further crack growth may even be accelerated by the SI treatment. Because the effectiveness of the SI treatment is also related to the applied stress on the weldment, mitigation by SI is not recommended for weldments with service stresses over 1.0 S_m, dual; and 1.5 cracks deeper than 30% of the wall, circumferential cracking longer than 10% of the circumference, or axial cracks of any extent. (See 5.3.2.6)

5.0 INSPECTION OF PIPING FOR IGSCC

5.1 Weldments Subject to Inspection

The discussion and recommendations in this section apply to BWR piping made of austenitic stainless steel that is four inches or larger in nominal diameter and contains reactor coolant at a temperature above 200°F during power operation regardless of code classification. It also applies to reactor vessel attachments and appurtenances such as jet pump instrumentation penetration assemblies and head spray and vent components.

This section does not apply to piping made of carbon steel classified as P-1 by the ASME Boiler and Pressure Vessel Code.

5.2 Inspection Methods

One positive result of the extensive investigations performed on BWR piping is that no significant mode of degradation other than IGSCC has been noted. This means that inspections can focus on those approaches that are best suited for detecting and evaluating IGSCC. A less favorable finding is that special methods and specific operator training are required to reliably detect and characterize IGSCC in the presence of the variable geometric configurations of the weldments.

It is not the intent of this report to provide specific guidance to operators regarding details of equipment and procedures. This function is best handled by Code activities in which industry and regulatory participants reach a consensus. It is not a simple problem; finding and recognizing IGSCC by UT is still as much an art as it is a science. The intent of the recommendations in this report is to ensure that the UT operators inspecting BWR piping for IGSCC can detect and characterize IGSCC in the welds they inspect, and that they will accomplish these two functions reliably in the field.

5.2.1 Staff Recommendations on Inspection Methods and Personnel

Although examinations should be performed in general accordance with the ultrasonic examination requirements of the applicable edition of the ASME Code, details of the examination method, acceptance criteria, and personnel qualification should be upgraded to ensure that the examinations will be effective.

All examination procedures and the specific equipment used in the field inspections, and all level 2 and 3 NDE examiners or operators for flaw detection and sizing should demonstrate their field performance capability on cracked, preferably service-induced, samples in a manner acceptable to the NRC. No NDE examiner or operator should perform examinations of BWR piping without proving his competence even if he must take special training to gain specific skills and knowledge required to perform these inspections. The program being conducted at EPRI NDE center in Charlotte, North Carolina, in accordance with the NDE Coordination Plan agreed upon by NRC, EPRI, and BWROG, as upgraded in September 1985 is considered to be acceptable. Any future changes in this program should be in conformance with the Coordination Plan and approved by the Executive Director for Operations, NRC

Specialized radiographic techniques developed for detection of IGSCC may be used in cases where ultrasonic examination is not practical, or to augment the UT method.

5.2.2 Flaw Size Uncertainty

Inspections performed under IE Bulletins 82-03 and 83-02 were often performed by examiners with limited knowledge and experience in sizing IGSCC. Although the length of the cracks could usually be defined satisfactorily, must UT operators could not determine their throughwall depth accurately and reliably. After this was shown to be true in industry-wide evaluation projects, the industry developed more effective and diverse techniques, and the NDE Center initiated a

training and qualification program specifically for crack depth sizing. The NRC staff participated in this effort by defining acceptable levels of performance, based on the level of accuracy required to ensure safe operation. The staff now believes that flaw sizes determined by examiners and procedures qualified by test will not be grossly underestimated or overestimated provided that an inspectable weld joint configuration and weld surface exist.

The depth of cracks not sized by fully qualified personnel or with limitations to examination (such as wide weld crowns, obstructions, or other adverse geometrical configurations) should be assumed to be at least 75% of the wall in depth, and the flaw so evaluated.

5.3 Inspection Frequency

5.3.1 Weldment IGSCC Condition Category Definitions

The purpose of inservice inspection of piping is to provide continued assurance that the structural integrity and reliability (e.g., see 10 CFR 50.55a(g)(6)(ii)) of the piping is maintained and that there continues to be an extremely low probability of abnormal leakage (10 CFR 50 Appendix A, Criterion 14). Piping with weldments that are susceptible to degradation mechanisms such as IGSCC require more frequent inspections to provide such continued assurance. Weldments in BWRs will have different degrees of susceptibility to IGSCC depending on the materials and processing involved. Therefore, the inspection frequencies recommended by the staff are based on the condition of each weldment

The extent of augmented inspection recommended depends on the number of cracked welds in the plant as well as the condition of each individual weldment. In addition, welds that have already been found to be cracked will have varying degrees of susceptibility to further cracking, depending on the remedial actions taken.

Some may be considered repaired, at least on a conditional basis; whereas others with marginal or no repair are considered fit for only very limited service without additional action. These seven categories of weldment conditions are listed in Table 1 and defined in detail below.

5.3.1.1 Definition of IGSCC Category A Weldments

IGSCC Category A Weldments are those with no known cracks, that have a low probability of incurring IGSCC problems, because they are made entirely of IGSCC resistant materials or have been solution heat treated after welding. CRC is considered to be IGSCC resistant, and welds joining cast pump and valve bodies to resistant piping are considered to be resistant weldments.

5.3.1.2 Definition of IGSCC Category B Weldments

IGSCC Category B Weldments are those not made of resistant materials but have had an SI performed either before service or within two years of operation. If the SI is performed after plant operation, a UT examination after SI to ensure that they are not cracked is required.

5.3.1.3 Definition of IGSCC Category C Weldments

IGSCC Category C Weldments are those not made of resistant materials (see 2.1.1), and have been given an SI process after more than two years of operation. An ultrasonic examination to ensure that they are not cracked should be performed after the SI treatment as part of the process.

5.3.1.4 Definition of IGSCC Category D Weldments

IGSCC Category D Weldments are those not made with resistant materials, and have not been given an SI treatment, but have been inspected by examiners and procedures in conformance with section 5.2.1, and found to be free of cracks.

5.3.1.5 Definition of IGSCC Category E Weldments

IGSCC Category E Weldments are those with known cracks but have been reinforced by an acceptable weld overlay or have been mitigated by an SI treatment with subsequent examination by qualified examiners and procedures to verify the extent of cracking. Guidelines for acceptable weld overlay reinforcement and extent of cracking considered amenable to SI treatment are covered in Sections 3.2 and 4.5 of this document.

5.3.1.6 Definition of IGSCC Category F Weldments

IGSCC Category F Weldments are those with known cracks that have been approved by analysis for limited additional service without repair. Weldments found to have significant cracking or a questionable extent of cracking that have been minimally overlay reinforced (not in conformance with Section 4.1) are considered acceptable only for interim operation. Weldments with significant cracking that have been SI treated may also be considered to be in this category. Detailed guidelines used to evaluate specific cases are provided in Sections 3.0 and 4.0 of this document.

5.3.1.7 Defini.ion of IGSCC Category G Weldments

IGSCC Category G Weldments are those not made of resistant materials, have not been given an SI treatment and have not been inspected in accordance with Section 5.2.1. Stress improved welds that were not inspected after the SI treatment are considered to be Category G weldments until the post-SI inspection has been performed.

Cat. G is not refueling outage

5.3.2 Staff Recommendations on Inspection Schedules

The staff recommendations in the extent and frequency of inspection for various weldments categorized in accordance with 5.3.1 are discussed in detail below and summarized in Table 1.

5.3.2.1 Inspection Schedule for IGSCC Category A Weldments

IGSCC Category A welds should be inspected according to a schedule similar to that called for in Section XI of the Code. A representative sample of 25% of the welds should be examined every 10 year interval. The sample selection should reflect the best technical judgment of the plant owner.

5 3.2.2 Inspection Schedule for IGSCC Category B Weldments

IGSCC Category B welds are more likely to develop cracking than Category A welds, so a larger sample size is needed. Specifically, a representative sample of 50% of IGSCC Category B welds should be examined every 10 year interval.

5 3.2.3 Inspection Schedule for IGSCC Category C Weldments

IGSCC Category C welds have longer service life prior to SI than IGSCC Category B welds, so are more likely to contain undetected cracking. All IGSCC Category C welds should be inspected within two refueling cycles after the post-SI inspection, and every 10 years thereafter.

5.3.2.4 Inspection Schedule for IGSCC Category D Weldments

Category D Weldments should be inspected at least once every two refueling cycles. Approximately half of the IGSCC Category D weldments in the plant should be inspected each refueling outage.

5.3.2.5 Inspection Schedule for IGSCC Category E Weldments

Repaired and stress improved cracked weldments, IGSCC Category E should be inspected at least once every two refueling cycles after repair. Approximately half of them should be inspected during the first refueling outage after repair.

If it is desired to operate for more than two fuel cycles with overlay reinforcement repairs, the overlayed weldments should be inspected to ensure that the overlays will continue to provide the necessary safety margin. For standard and designed overlays meeting the requirements of Section 4.0, the inspection method should provide positive assurance that cracks have not progressed into the overlay. It is also desirable that the inspection procedure be capable of detecting cracks that originally were deeper than 75% of the original wall thickness, or that have grown to be deeper than 75% of the original wall thickness. Ultrasonic inspections should be performed using a procedure that has been demonstrated to be reliable and effective, and should be performed by personnel that have been trained and qualified in the specific methods for inspections of overlays.

5.3.2.6 <u>Inspection Schedule for IGSCC Category F Weldments</u>

IGSCC Category F Weldments are approved for limited service only, and should be inspected every refueling outage, unless a shorter service period has been specified. Weldments that are classified as IGSCC Category F because overlay repairs or SI treatment mitigation is not according to recommendations in Sections 3.2 and 4.5 may be upgraded to IGSSC Category E after 4 successive examinations indicate no adverse change in cracking condition.

5.3.2.7 <u>Inspection Schedule for IGSCC Category G Weldments</u>

IGSCC Category G Weldments should be inspected at the next refueling outage.

5.3.3 <u>Inspection Schedules with HWC</u>

If improved water chemistry control, including hydrogen additions is implemented, the time schedule for inspections may be extended. Although specific details of such extensions will be evaluated on a case basis, it is anticipated that periods between inspections could be lengthened by about a factor of two for category B, C, D and E weldments.

5.3.4 <u>Staff Recommendations on Sample Expansion</u>

If one or more cracked welds in IGSCC Categories A, B, or C, are found by a sample inspection during the 10 year interval, an additional sample of the welds in that category should be inspected, approximately equal in number to the original sample. This additional sample should be similar in distribution (according to pipe size, system, and location) to the original sample, unless it is determined that there is a technical reason to select a different distribution. If any cracked welds are found in this second sample, all of the welds in that IGSCC Category should be inspected.

If significant crack growth, or additional cracks are found during the inspection of one or more IGSCC Category E welds, all other Category E welds should be examined.

a) Significant crack growth for overlayed welds is defined as crack extension to deeper than 75% of the original wall thickness, or for cracks originally deepter than 75% of the pipe wall, evidence of crack growth into the effective weld overlay.

b) Significant crack growth for SI mitigated Category E welds is defined as growth to a length or depth exceeding the criteria for SI mitigation. (10% of circumference or 30% in depth).

TABLE 1

SUMMARY OF RECOMMENDED INSPECTION SCHEDULES FOR BWR PIPING WELDMENTS

DESCRIPTION OF WELDMENTS	NOTES	IGSCC CATEGORY	INSPECTION EXTENT & SCHEDULE
Resistant Materials		A	25% every 10 years (at least 12% in 6 years)
Nonresistant Matls SI within 2 yrs of operation (1)	(1)	B	50% every 10 years (at least 25% in 6 years)
Nonresistant Matls SI after 2 yrs of operation	(1)	C	All within the next 2 refueling cycles, then all every 10 years (at least 50% in 6 years)
Non Resistant Matl No SI	(1)	D	All every 2 refueling cycles
Cracked Reinforced by weld overlay or mitigated by SI	(1)(2)	E	50% next refueling outage, then all every 2 refueling cycles
Cracked Inadequate or no repair	(2)	F	All every refueling outage
Non Resistant Not Inspected	(3)	G	All next refueling outage

Notes:

(1) All welds in non-resistant material should be inspected after a stress improvement process as part of the process. Schedules shown should be followed after this initial inspection.

(2) See recommendations for acceptance weld overlay reinforcements and stress improvement mitigation.

(3) Welds that are not UT inspectable should be replaced, "sleeved", or local leak detection applied. RT examination or visual inspection for leakage may also be considered.

6.0 LEAK DETECTION

The staff reviewed the leak detection and leakage limits that have been applied to BWRs by past revisions of NUREG-0313, Bulletins, and Generic Letter 84-11. In NUREG 1061 Vol. 1, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee," the report of the Pipe Crack Task Group, it was recommended that leakage detection equipment should be improved, and that the upper limit on unidentified leakage should be decreased from 5 gpm to 3 gpm.

As a result of this review, the staff concluded that if the other recommendations of this report are followed, present leak detection systems will be adequate. Further, the staff concluded that the decrease in the limit on unidentified leakage recommended in NUREG-1061 Vol. 1, would constitute a backfit that could not be justified by a supporting Regulatory Analysis, in accordance with the new backfit rule, 10 CFR50.109.c.

Accordingly, the staff recommendations on leak detection and leakage limits are in accordance with past staff positions on the subject. Relaxation of the operability requirements for those plants with resistant or mitigated noncracked piping is also in accordance with past staff positions.

6.1 Staff Recommendations on Leak Detection

Leakage detection systems should be in conformance with Position C of Regulatory Guide 1.45 "Reactor Coolant Pressure Boundary Leakage Detection Systems, or as otherwise approved by the NRC.

1. Plant shutdown should be initiated for inspection and corrective action when, within any period of 24 hours or less, any leakage detection system indicates an increase in rate of unidentified leakage in excess of 2 gpm or its equivalent, or when the total unidentified leakage attains a rate of 5 gpm or equivalent, whichever occurs first. For sump level monitoring systems with fixed-measurement-interval methods, the level should be monitored at approximately 4-hour intervals or less.

2. Unidentified leakage should include all leakage other than

   (a) leakage into closed systems, such as pump seal or valve packing leaks that are captured, flow metered, and conducted to a sump or collecting tank, or

   (b) leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operations of unidentified leakage monitoring systems or not to be from a throughwall crack in the piping within the reactor coolant pressure boundary.

3. For plants operating with any IGSCC Category D, E, F, or G welds, at least one of the leakage measurement instruments associated with each sump shall be operable, and the outage time for inoperable instruments shall be limited to 24 hours, or immediately initiate an orderly shutdown.

APPENDIX A - CRACK GROWTH CALCULATIONS

Introduction

Crack growth calculations are required to evaluate the continued structural integrity of a weld with known cracks, if it is desired to continue operation without repair or reinforcement. The rate of growth of IGSCC is not easy to predict, because the several important factors are usually imperfectly known. Research work in this area has been helpful in defining the general effect of these factors but a large uncertainty in crack growth predictions still remain.

Nevertheless crack growth calculations can be performed within certain limits with enough confidence to ensure plant safety without excessive conservatism.

Crack growth calculations are based on the fundamental concept that the crack growth rate of a specific material in a specific environment will be a function of the applied stress intensity factor, K_I. Laboratory crack growth data are usually presented in this manner, details of the calculational methods used to calculate K_I are provided later in this Appendix but an important point to note here is that K_I depends on the crack depth therefore it changes continuously during crack growth.

Crack growth analysis methods are, therefore, iterative in nature. Given an initial crack depth, the K_I is calculated for the particular stress distribution of interest. Knowing the K_I, the amount of growth for a specific time is calculated, the growth is added to the initial crack depth, a new K_I is calculated, and the process is repeated Time intervals selected can vary from 1 hour to 1000 hours, depending on the rate of growth and rate of change in K_I with crack depth.

Selection of Crack Growth Rate Parameters

Although only two parameters, crack growth rate and K_I, are used, they are both highly dependent on several factors.

Crack growth rate is affected by the degree of sensitization of the material and by the severity of the environment. Our interest as it relates to BWR piping is primarily in a degree of sensitization normally caused by welding, and in an environment similar to normal BWR water conditions.

Most formal crack growth studies are carried out with standard fracture mechanics specimens, which makes K_I determination easy. These specimens are not readily machined from pipe walls, so the material is given an artificial sensitization treatment, intended either to simulate the effect of welding or, in some cases, the more severe effect of furnace sensitization. Tests to ascertain whether the intended degree of sensitization has been obtained are still inexact, causing significant scatter in laboratory test results intended to apply to a similar metallurgical state.

Tests to simulate the BWR environment are usually run at operating temperature in high purity water containing 0.2 gpm oxygen. This is generally accepted to be a representative condition, although higher oxygen levels could occur locally for short periods of time. Tests are also often run in water containing up to 8 gpm oxygen, usually to achieve accelerated comparisons of materials or conditions.

In addition to these standardized tests for crack growth rate, results of actual pipe tests are available. Many hundreds of welds have been tested in General Electric's pipe test facility. These tests, although generally more relevant in terms of material condition and environment, are more difficult to evaluate. K_I is more difficult to calculate, and accurate crack growth rates are also more difficult to measure. Nevertheless this body of data has been used to augment those data from the more standard laboratory tests to select appropriate crack growth rates.

Figure 1 (from NUREG/CR-3292) * shows much of the relevant laboratory data in the conventional form, where measured rates are plotted against K_I. This plot clearly shows the large scatter resulting from a wide variation in material condition and environment. This information, together with additional information from actual pipe tests, was used to select a crack growth curve that is appropriate for use in safety evaluations. Note that if the fastest crack growth rate shown in Figure 1 is used, cracks would be predicted to grow completely through pipe walls in a matter of days. Clearly this would not reasonably represent reality.

The curve selected for use by the NRC staff is shown on Figure 2. Note that it is a curved line on the semilogarithmic chart used in Figure 1. On log-log coordinates, as used in Figure 2, it plots as a straight line. In calculations, it is expressed as:

$$da/dt = 3.590 \times 10^{-8} \times K_I^{2.161} \text{ inches per hour}$$

As can be seen, the crack growth rate is a very strong function of K_I. In laboratory tests, K_I is easily determined with good accuracy. This is not the case for real pipes and real pipe cracks. There are two major sources of uncertainty: knowledge of the actual crack size and shape, and the actual stress distribution in the area of the crack to be evaluated. The service distribution at a pipe weld is made up of the stress caused by the service loading and the residual stresses caused by the welding process. Of these, knowledge of the residual stress is the more uncertain. Nevertheless, a residual stress distribution through the pipe wall must be defined, if realistic crack growths are to be calculated. Although this is covered later in more detail, several comments are in order here.

The residual stress distribution caused by welding is the major stress component causing IGSCC. Welding causes a high tensile residual stress on the inside surface of the pipe near the root of the weld where the material is sensitized.

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*Shack, W.J., et al., "Environmentally Assisted Cracking in Light Water Reactors: Annual Report, October 1981 - September 1982" NUREG/CR-3292, Washington DC. U.S. Nuclear Regulatory Commission, June 1983.

This residual stress level has been calculated and measured to be up to or above the yield strength of the material. It typically is four or five times as high as the service-induced stress. In fact, without this very high residual stress at the sensitized area, IGSCC would not be a problem in BWR piping. This fundamental observation is helpful; wherever this combination of stress and sensitization occurs, cracking occurs. In actual cases, if there are significant cracks, there must be significant tensile residual stresses, and this should be accounted for in the crack growth analysis. The method used by the staff is described below.

Stress Intensity Factor Calculations

There are several relatively standard analytical solutions available for calculating the stress intensity factor (K_I) caused by stress distributions of the type found at BWR pipe welds. The method using influence functions is the one used by the staff and will be summarized here. Other methods, such as those described in the ASME Boiler and Pressure Vessel Code, Section XI, Appendix A, may also be used where appropriate.

Stress Analysis

The total stress state, including residual stress, pressure stress, and other stresses caused by normal operation must be known or assumed. Note that factors such as stress indices used for other purposes should not be used when calculating stress levels that apply to K_I calculations.

Residual Stress

The laboratory-measured throughwall axial residual stresses on pipe wall thickness $\geq$ 1 inch are presented in Figure 3 (from NUREG/CR-3292). The solid line in Figure 3 is the axial residual stress distribution used for the calculation of stress intensity factors for pipe sizes of 12" diameter and larger. The residual stress distribution is the most complex analytical problem involved. This is handled by fitting the curve of residual stress distribution through the wall by an analytical expression. For this particular residual stress distribution, the nondimensional expression given below is used.

$$\sigma/\sigma_i = \sum_{j=o}^{4} \sigma_j \ \xi^j$$

where

σ_o = 1.0
σ_1 = -6.910
σ_2 = 8.687
σ_3 = -0.480
σ_4 = -2.027
ξ = x/t
σ_i = stress magnitude at ξ = 0 (inner surface)

The above formula permits calculation of the residual stress value at any point (x) through the vessel wall thickness (t) as a function of the peak residual stress value at the inside diameter (ID), σ_i.

The stress intensity factor caused by the residual stress from welding (K_{IR}), is calculated using influence functions taken from NUREG CR-3384,* page A.19, Table (7). The influence functions i_j, given in this Appendix are for a 360° circumferential crack in a cylinder with a R/t ratio of 10. In view of other analytical conservatisms and uncertainties (i.e., assumed crack geometry and initial depths), it is believed that they may be used for cylinders with R/t ratios of from 9 to 11 to obtain reasonable and conservative estimates of crack growth versus time. For R/t ratios significantly different from 10, other influence functions or other analytical methods should be used.

The specific formula used by the staff is:

$$K_{IR}/(\sigma_i\sqrt{t}) = \sqrt{\pi\alpha}\sum_{j=o}^{4} \sigma_j \quad a^j \quad i_j$$

where:

$\sigma_o, \ldots \sigma_4$ and σ_i are as above
$i_o = 1.1220 + 0.3989\,\alpha + 1.5778\,\alpha^2 + 0.6049\,\alpha^3$
$i_1 = 0.6830 + 0.1150\,\alpha + 0.7556\,\alpha^2 + 0.1667\,\alpha^3$
$i_2 = 0.5260 + 0.1911\,\alpha - 0.1000\,\alpha^2 + 0.5802\,\alpha^3$
$i_3 = 0.4450 + 0.0783\,\alpha + 0.0556\,\alpha^2 + 0.3148\,\alpha^3$
$i_4 = 0.3880 + 0.1150\,\alpha - 0.1333\,\alpha^2 + 0.3519\,\alpha^3$
α = a/t
a = crack depth
t = wall thickness

<u>Membrane Stress</u>

The membrane stresses are assumed constant through the wall thickness, so

$$\sigma_m = \sigma_p$$

where

σ_p = membrane stress (σ_m) from pressure

*Stevens, D.L., et al., "VISA-A Computer Code for Predicting the Probability of Reactor Pressure Vessel Failure" NUREG/CR-3384, PNL-4774, Washington, D.C. U.S. Nuclear Regulatory Commission, September 1983.

The stress intensity factor for a 360° circumferential crack from pressure K_{IP}, is calculated from

$$K_{IP} = (PR/2t) \sqrt{t} \sqrt{\pi a} \quad (1.122 + 0.3989\, a + 1.5778\, a^2 + 0.6049\, a^3)$$

where

a, t are as above
P = pressure
R = radius to center of pipe wall

The total stress intensity factor, K_{IT}, is given by

$$K_{IT} = K_{IP} + K_{IR}$$

where

K_{IP} and K_{IR} are defined as above.

Correlation with Service Experience

Although the residual stress is assumed to be the same for all welds, the applied stresses, primary and secondary, vary from weld to weld; therefore, calculations must be performed for each weld evaluated. Figure 4 shows the results of K_I calculations for several pipe sizes using a nominal applied stress of 7500 psi. Note that at relatively shallow depths the K_I is high; therefore, the crack growth rate will be relatively fast. However, the K_I actually diminishes as the crack grows to about half way through the wall. This prediction is consistent with service experience; very few, if any, actual cracks of significant circumferential extent have been found deeper than about 50% of the wall thickness.

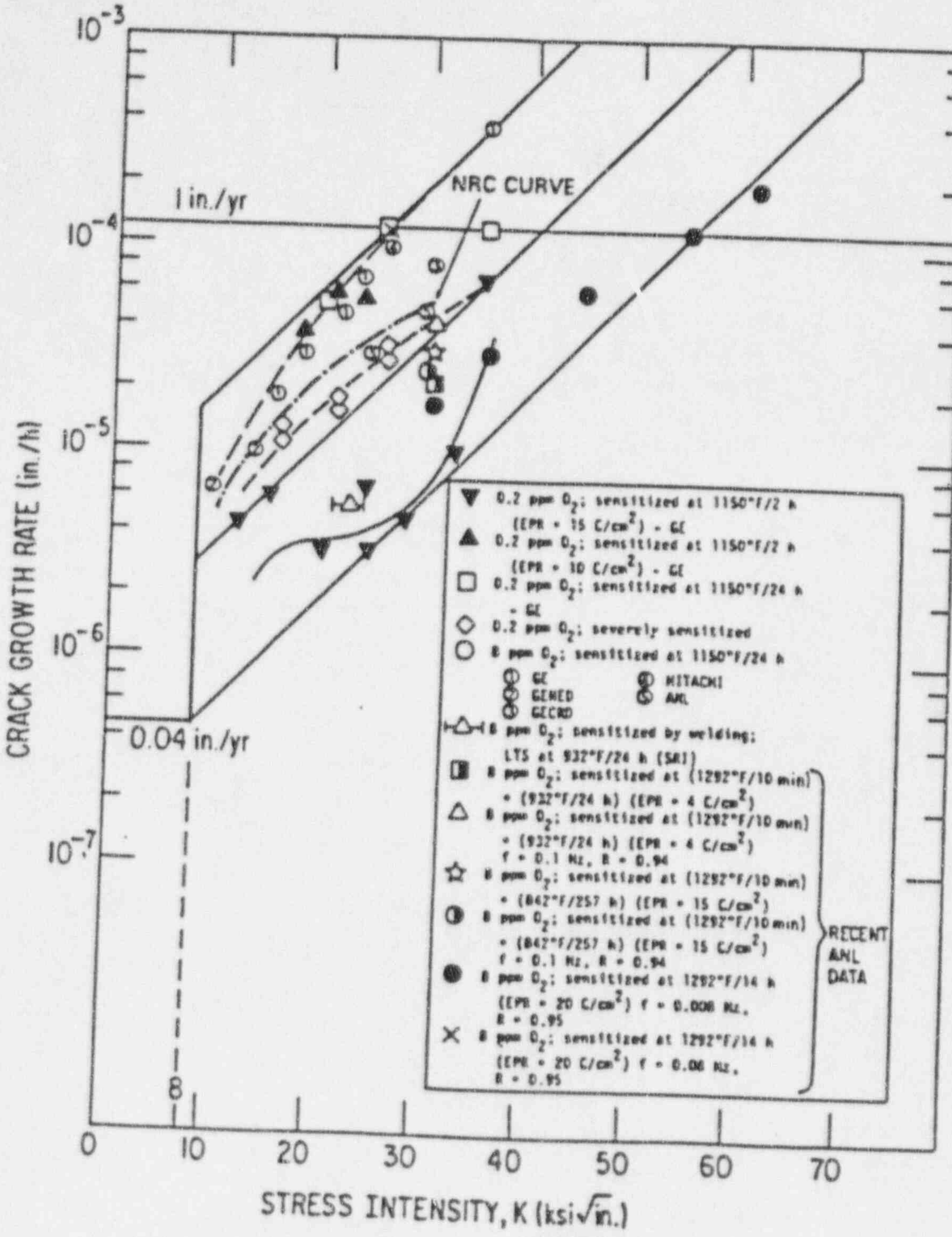

Figure 1
CRACK GROWTH RATE DATA

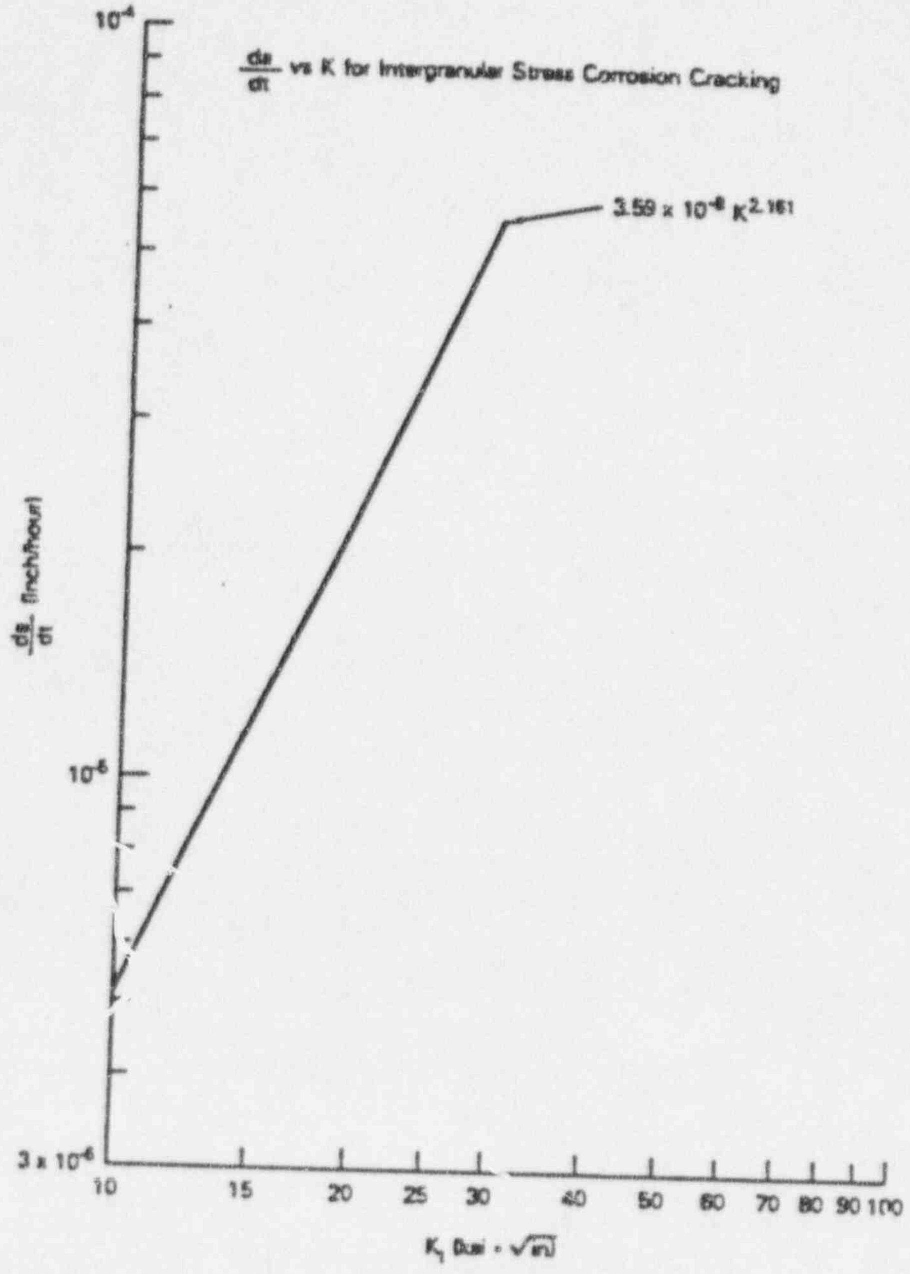

Figure 2
$\frac{da}{dt}$ vs K for Intergranular Stress Corrosion Cracking

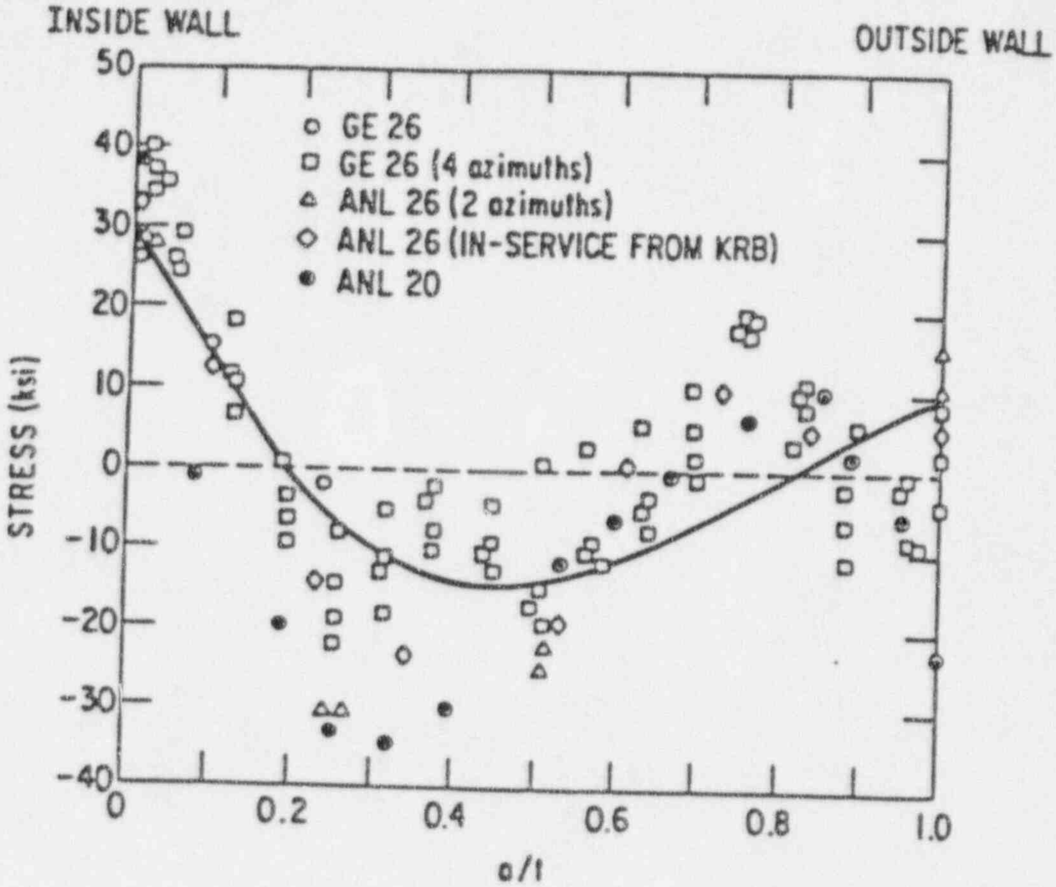

Figure 3
Through-wall Distribution of Axial Residual Stress in Large-Diameter Pipes (t $\geq$ 1 in.)

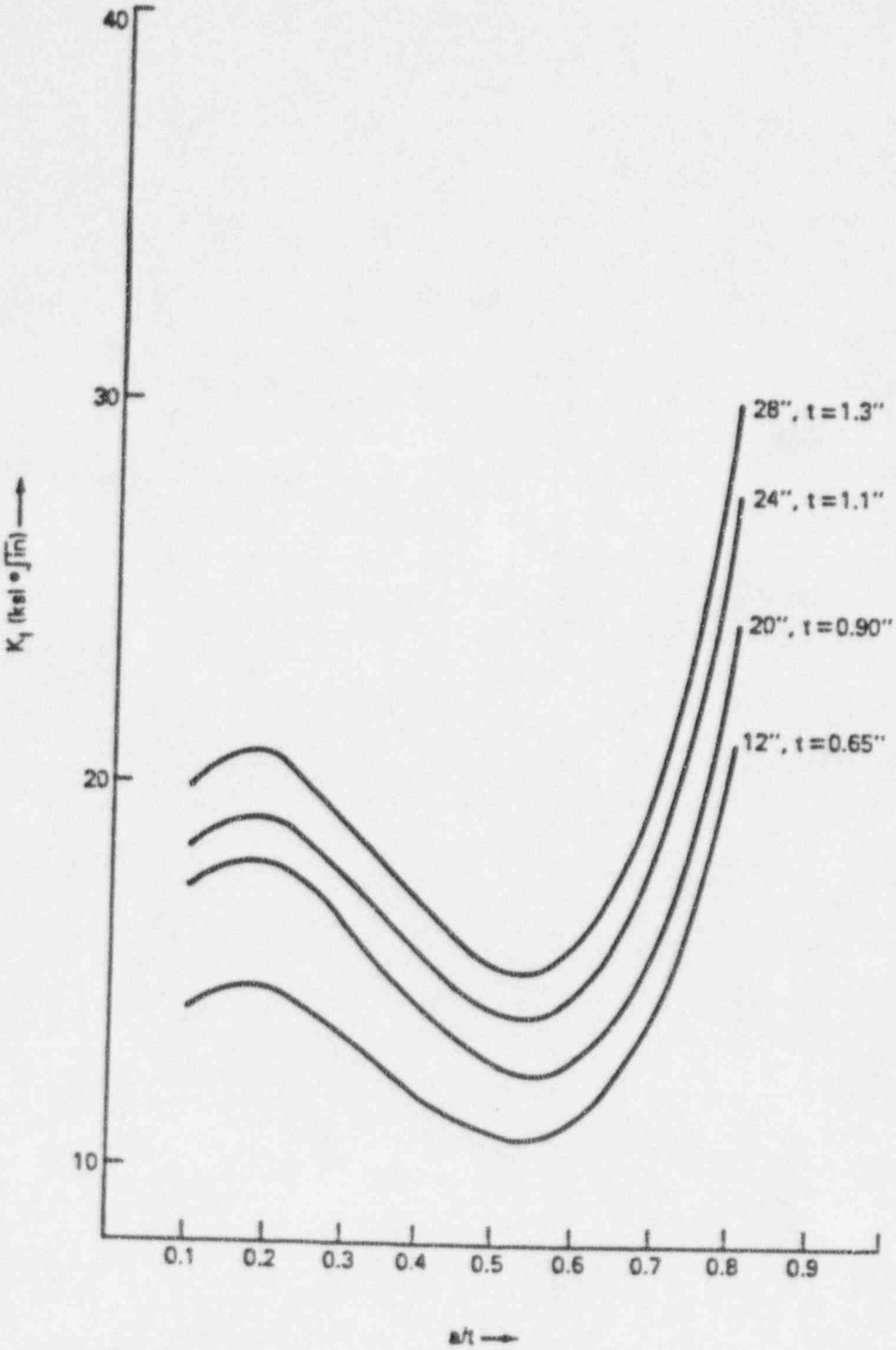

Figure 4
THROUGH-WALL DISTRIBUTION K_I WITH APPLIED STRESS OF 7500 PSI

FORM 2

ALARA Pre-approval Walkdown RECORD

Plant BNP ESR No. 94-00458 Revision 0

Title Unit 1 Feedwater Piping and Safe-End Weld Overlay

Scope of Work (by RE)

The feedwater piping and safe-end welds will be inspected by ISI as a normal program requirement. If a flaw is identified in one of the IGSCC susceptible welds, a weld overlay may be applied. The weld overlay will require a contractor to install an automatic welding machine with cameras on the piping and monitor progress from a remote location. Upon completion, the contractor will remove the welding equipment, prepare the as welded surface for a UT examination by grinding, machining, sanding, or equivalent. The QC inspector will perform the as-left examination. Repairs by welding may be necessary if the examination results are not acceptable.

This impact evaluation and estimate of the exposure is based on overlaying one feedwater safe-end at two adjacent weld locations. The adjacent weld locations are less than 24" apart and within 48" of the reactor vessel outside diameter, at the feedwater nozzle.

Special ALARA Considerations (by ALARA)

1) Piping should be filled with water during drywell work.
2) N9 and N11 nozzles should be flushed or shielded, as appropriate.
3) N4 nozzles should be flushed prior to ISI inspections.
4) Shielding for the feedwater piping should be available for use during manual operations, as needed.
5) Mock-up training and testing of equipment should be performed prior to entering the RCA.

Estimated Installation Dose 4.0 Person-Rem per nozzle. (An additional 0.5 Person-Rem is required if the thermocouples on the N4B and N4D nozzles must be removed and reinstalled).

1/11/95
Responsible Engineer Date

1/12/95
ALARA Specialist Date

Installation Time and Manpower Chuck Raines 1.12.95
Installation Representative Date

Estimate is attached.

DESIGN VERIFICATION
of the
COMPLETED DESIGN PACKAGE

Plant BNP
Project G0029B Q
File No. 1005 Level
Document No. ESR 94-00458 Rev. 0

[✓] Q (Class A)
[ ] Seismic (Class B)
[ ] FP-Q (Class D)
[ ] Other

Mark each Discipline or Area of expertise addressed by Discipline Technical Reviews.

Discipline		Discipline	
Mechanical	[✓]	Civil Structural	[✓]
HVAC	[ ]	Seismic Equip. Qual.	[ ]
Electrical	[ ]	Civil Stress	[✓]
I&C	[ ]	Fire Protection	[ ]
		Environmental Qualification	[ ]
		Human Factors	[ ]
		Materials	[✓]
Other Welding			[✓]

Mark each item yes, no, or not applicable and initial each item checked by you.

1. Is the design adequately interfaced between design disciplines providing input? YES
2. Are sufficient design documents and procedures included or referenced to allow implementation to be carried out in a planned and controlled manner? YES
3. Have adequate provisions for in-process or post-installation examinations, holdpoints, inspections, and testing been specified to assure quality of work and verification that the design performs as intended? (Including dimensions for inspection?) YES
4. Have adequate provisions been provided to document installation and results of examinations, inspections, and testing within the package or documents referenced? YES
5. Has consideration been given to design change operability, reliability, maintainability, safety (Industrial and Nuclear) and adherence to appropriate codes, standards, and regulatory requirements? YES
6. Have appropriate Discipline Technical Reviews been performed for the Completed Design Package? YES
7. Are specified materials and processes suitable for the intended application? YES
8. Is the design technically adequate with respect to the design basis? YES
9. Does the package comply with design related procedures (i.e., POM and NED procedures)? YES

For each question on the check list not answered yes, explain on page 2. If "Not Applicable," give reason.

Complete Design Package Acceptable: Yes [✓] No [ ] - comments attached.

Lead Verifier [signature] Date 12/16/94

Acknowledgement of Verification:
(DPE) [signature] Date 12/16/94

Resolution of Comments:
Comments Resolved (See Attached):
(LE) ________ Date ________

Action taken makes Design Documents Acceptable:
Lead Verifier ________ Date ________

(DPE) ________ Date ________

DESIGN VERIFICATION of COMPLETED DESIGN PACKAGE
COMMENT SHEET

Plant BNP

Project G0029B

File No. 1005

Document No. ESR 94-00458 Rev. 0

This sheet is only required when comments are being made.

Comment No.	Comment	Resolution	Resolved Initial/Date
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DISCIPLINE DESIGN VERIFICATION RECORD

I. Instructions to Verification Personnel

Plant BNP
Project G0029B
File No.
Document No. ESR 94-00458 Rev 0

Q Level

[X] Q (Class A)
[ ] Seismic (Class B)
[ ] FP-Q (Class D)
[ ] Other

Design verification should be done in accordance with ANSI N45.2.11, Section 6, as amended by Regulatory Guide 1.64, Rev. 2.

Special Instructions:

Discipline Project Engineer Mark Duth

II. Verification Documentation

Applicability

Discipline		Discipline	
Mechanical	[ ]	Civil Structural	[ ]
HVAC	[ ]	Seismic Equip. Qual.	[ ]
Electrical	[ ]	Civil Stress	[ ]
I&C	[ ]	Fire Protection	[ ]
		Environmental Qualification	[ ]
		Human Factors	[ ]
		Materials	[✓]
			[ ]

Other Welding [✓]

Verification Methods Used:
[✓] Design Review [ ] Alternate Calculations [ ] Qualification Testing

Design Document Acceptable: Yes [ ] No [✓] - comments attached.
Design Verifier Michael W. Guthrie Date 12/15/94

Acknowledgement of Verification:
(DPE) Mark Duth Date 12/15/94

III. Resolution of Comments:

Comments Resolved (See Attached):
(RE) Date 12/16/94
Action taken makes Design Documents Acceptable:
Design Verifier Michael W. Guthrie Date 12/16/94

(DPE) Mark Duth Date 12/16/94

DISCIPLINE DESIGN VERIFICATION RECORD
COMMENT SHEET

Plant BNP

Project G0029B

File No. ____________

Document No. ESR 94-00458 Rev. 0

This sheet is only required when comments are being made.

Comment No.	Comment	Resolution	Resolved Initial/Date
1	Various typographical corrections.	INCORPORATED	PLTS 12/16/94
2	Add specific material requirements.	INCORPORATED	PLTS 12/16/94
3	Add welding parameters and testing requirements	INCORPORATED	PLTS 12/16/94
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DISCIPLINE DESIGN VERIFICATION RECORD

I. Instructions to Verification Personnel

Plant	BNP		[X] Q (Class A)
Project	G0029B	Q	[ ] Seismic (Class B)
File No.		Level	[ ] FP-Q (Class D)
Document No.	ESR 94-00458 Rev 0		[ ] Other

Design verification should be done in accordance with ANSI N45.2.11, Section 6, as amended by Regulatory Guide 1.64, Rev. 2.

Special Instructions:

Discipline Project Engineer Mark [signature]

II. Verification Documentation

Applicability

Discipline		Discipline	
Mechanical	[ ]	Civil Structural	[✓]
HVAC	[ ]	Seismic Equip. Qual.	[ ]
Electrical	[ ]	Civil Stress	[✓]
I&C	[ ]	Fire Protection	[ ]
		Environmental Qualification	[ ]
		Human Factors	[ ]
		Materials	[ ]
			[ ]
Other			[ ]

Verification Methods Used:
[✓] Design Review [ ] Alternate Calculations [ ] Qualification Testing

Design Document Acceptable: Yes [✓] No [ ] - comments attached.
Design Verifier JC DAY Date 12-19-94

Acknowledgement of Verification
(DPE) M.R. [signature] Date 12-15-94

III. Resolution of Comments:

Comments Resolved (See Attached):
(RE) ______ Date ______
Action taken makes Design Documents Acceptable:
Design Verifier ______ Date ______

(DPE) ______ Date ______

DISCIPLINE DESIGN VERIFICATION RECORD
COMMENT SHEET

Plant BNP

Project G0029B

File No.

Document No. ESR 94-00458 Rev. 0

This sheet is only required when comments are being made.

Comment No.	Comment	Resolution	Resolved Initial/Date
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Comment Resolution Impact Statement

Form 6

Plant BNP ESR No. 94-00458 Rev. No. 0

Title Unit 1 Feedwater Piping and Safe-End Weld Overlay

COMPLETION

This document integrates the work of disciplines after resolution of comments. It also considers and integrates the impact of the latest revisions of the plant Tech Specs, FSAR, drawings, and procedures. Resolution of the comments does not increase the approved scope of the project.

1/13/95
Responsible Engineer Date

DESIGN VERIFICATION/CHECK

[✓] Resolution of comments does not affect the original design verification/check.

1/13/95
[✓] Design Verifier Date
[ ] Checker

[ ] Original Design Verification/Check has been revised to accommodate comments.

SAFETY REVIEW

[X] Resolution of comments does not affect the original safety evaluation.

1/13/95
1st Safety Reviewer Date

Kayne C Nel 1/13/95
2nd Safety Reviewer Date

[ ] Original safety evaluation has been revised.

APPROVAL

1/13/95
Responsible Manager Date

MODIFICATION SUMMARY

The feedwater inlet piping to the reactor vessel is carbon steel up to the nozzle safe-end. A plant modification removed the original carbon steel safe-end, except for a short stub, and installed an Alloy 600 safe-end that accommodated a new feedwater sparger thermal sleeve design prior to initial plant start-up. The modification included three welds at each safe-end that are of a material which is susceptible to Intergranular Stress Corrosion Cracking (IGSCC) as defined in NUREG-0313 Rev. 2. Generic Letter 88-01/NUREG-0313 Rev. 2 provides options for the repair or replacement of piping susceptible to IGSCC. This modification implements a weld overlay in accordance with the NRC Staff positions of Generic Letter 88-01/NUREG-0313 Rev. 2, ASME Section XI, 1989 ed., and ASME Code Committee guidance of Code Case N504 using material that is highly resistant to IGSCC.

- This project provides a weld overlay design for the three Inconel 82/182 welds connecting the RPV nozzle, safe-end, and feedwater piping in each feedwater line. These welds are in Generic Letter 88-01/NUREG-0313 Rev. 2, as Category "D". The weld overlay will provide full structural reinforcement assuming a 360° through-wall crack. The overlay design uses a weld filler metal, Alloy 52, which is highly resistant to IGSCC and will mitigate crack growth into the weld overlay.

- The overlay geometry is designed to permit future inspections by visual, ultrasonic, and radiographic techniques to satisfy Augmented (Generic Letter 88-01/NUREG-0313 Rev. 2) and In-service Inspection requirements.

- The weld overlays may be applied at adjacent welds to minimize the dose associated with future repairs even if the adjacent weld does not have an identified flaw.

10CFR50.59 PROGRAM MANUAL
ATTACHMENT A
CP&L SAFETY REVIEW PACKAGE
SAFETY REVIEW COVER SHEET

DOCUMENT NO. ESR 94-00458 REV. NO. 0
DESCRIPTION OF TITLE: Unit 1 Feedwater Piping Weld Overlay

1. Assigned Responsibilities:
   Safety Analysis Preparer: Steven L. Bertz
   Lead 1st Safety Reviewer: Steven L. Bertz
   2nd Safety Reviewer: Wayne Nelson

2. Safety Analysis Preparer: Complete PART I, SAFETY ANALYSIS

   Safety Analysis Preparer [signature] Date 12/21/94

3. Lead 1st Safety Reviewer: Complete Part II, Item Classification.

4. Lead 1st Safety Reviewer: III may be completed. If either question 1 or 2 is "yes," then Part IV is not required.

5. Lead 1st Safety Reviewer: Determine which DISCIPLINES are required for review of this item (including own) and mark the appropriate blocks below.

DISCIPLINES Required:	(Print Name)	Signature/Date (Step 7)
[ ] Nuclear Plant Operations		
[ ] Nuclear Engineering		
[X] Mechanical	Steve Bertz	[signature] 12/21/94
[ ] Electrical		
[ ] Instrumentation & Control		
[X] Structural	J.C. Dail	JC Dail 12-21-94
[X] Metallurgy	M. Guthrie	M. Guthrie 12-22-94
[ ] Chemistry/Radiochemistry		
[ ] Health Physics		
[ ] Administrative Controls		

6. A QUALIFIED SAFETY REVIEWER will be assigned for each DISCIPLINE marked in step 5 and his/her name printed in the space provided. Each person shall perform a SAFETY REVIEW and provide input into the Safety Review Package.

7. The Lead 1st Safety Reviewer will assure that a Part III or Part IV is completed (see step 4 above) and a Part VI if required (see 9.d of Part II) Each person listed in step 5 shall sign and date next to his/her name in step 5, indicating completion of a SAFETY REVIEW.

8. 2nd Safety Reviewer: Perform a SAFETY REVIEW in accordance with Section 8.0

   2nd Safety Reviewer [signature] Date 12/22/94
   DISCIPLINE Mechanical

9. PNSC review required? If "yes" attach Part V and mark reason below: Yes [ ] No [x]
   [ ] Potential UNREVIEWED SAFETY QUESTION
   [ ] Question 9 of Part IV answered "Yes"
   [ ] Other (specify): ____________

PART I: SAFETY ANALYSIS
(See instructions in Section 8.4.1)
(Attach additional sheets as necessary)

DOCUMENT NO. ESR 94-00458 REV. NO. 0

DESCRIPTION OF CHANGE:

The feedwater inlet piping to the reactor vessel is ASME Code Class 1 carbon steel up to the nozzle safe-end. A plant modification removed the original carbon steel safe-end, except for a short stub, and installed an Alloy 600 safe-end that accommodated a new feedwater sparger thermal sleeve design prior to initial plant start-up. The modification included three welds at each safe-end that are of a material which is susceptible to Intergranular Stress Corrosion Cracking (IGSCC) as defined in Generic Letter 88-01/NUREG-0313 Rev. 2. Generic Letter 88-01/NUREG-0313 Rev. 2 provides options for the repair or replacement of piping susceptible to IGSCC. This modification implements a weld overlay in accordance with Generic Letter 88-01/NUREG-0313 Rev. 2, ASME Section XI, 1989 ed., and guidance from Code Case N-504, and Code Case 2142, using material that is highly resistant to IGSCC.

The basic function of the feedwater supply system is to return the condensed steam from the turbine (via the condensate system) back to the reactor pressure vessel (RPV). The portion of feedwater supply system affected by this project is inside primary containment and is classified as part of the reactor coolant pressure boundary Nuclear Boiler (B21) System. The feedwater lines are connected to the RPV nozzle via a safe-end.

- The weld overlay design, reference 3.3.6.13 in ESR 94-00458, analyzed this repair for structural reinforcement, residual stress, base metal stresses, ASME allowable flaw geometry, and material that is highly resistant to IGSCC. The weld overlay design specifies weld thickness and length that satisfies the design requirements and has been reviewed/accepted by the required Owner's Review.

- This project provides a weld overlay design for the three Inconel 82/182 welds connecting the RPV nozzle, safe-end, and feedwater piping in each feedwater line. These welds are in NUREG-0313 Rev. 2, and defined as Category "D", susceptible material without stress improvement treatment. The weld overlay will provide full structural reinforcement assuming a 360° through-wall crack. The weld overlay design is in accordance with the requirements of ASME Section XI, 1989 ed., with guidance taken from Code Case N-504.

- Alloy E-52 is a nickel based alloy with a high chromium content and is considered highly resistant to IGSCC, so crack propagation into this material is unlikely. This material is not listed in ASME Section II but will be used as permitted in Code Case 2142.

- Residual stress has no effect on the design, since arrest of any flaw prior to encountering the Alloy E-52 material is not assumed. However, residual stress resulting from the repair will tend to inhibit any new flaw initiation.

- The overlay geometry is designed to permit future inspections by visual, ultrasonic, and radiographic techniques to satisfy Augmented (NUREG-0313 Rev. 2) and In-service Inspection requirements.

PART I: SAFETY ANALYSIS
(See instructions in Section 8.4.1)
(Attach additional sheets as necessary)

DOCUMENT NO. ESR 94-00458 REV. NO. 0

- The weld overlays may be applied over adjacent inconel welds to minimize the dose associated with future repairs even if the adjacent weld does not have an identified flaw. Carbon steel welds adjacent to inconel welds will be included in the overlay due to their close proximity, thereby, avoiding a stress concentration.

ANALYSIS:

Summary
The application of the weld overlays will not adversely affect the safe operation of the plant.

The structural integrity of the reactor coolant pressure boundary is not degraded by this project. Performance requirements for structural integrity of the Nuclear Boiler system and RPV are unchanged by this project. The weld overlay design meets the repair requirements of ASME Section XI, 1989 ed. The overlay weld material, Alloy 52, is highly resistant to IGSCC and will mitigate crack growth. The overlay design has been verified by analysis to be acceptable for the existing design loads and materials. The leakage detection system meets NUREG-0313 Rev. 2 requirements and the plant is in an active LCO if the limits are exceeded.

This weld overlay repair is a non-ASME Code repair under the ASME Section XI 1980 edition, through Winter 1981 Addenda, and NRC approval is required prior to implementation. Additionally, Generic Letter 88-01 requires NRC review and approval of the weld overlay design and material. The weld material is not listed in ASME Section II, so NRC approval of Code Case 2142 is also required.

Details
A loss of coolant accident resulting from failure of a feedwater line at a nozzle safe-end will depressurize the reactor vessel. A pipe break at the feedwater nozzle is bounded by a pipe break of the recirculation pipe, since the feedwater nozzles are above the top of the active core versus below, and the feedwater pipe diameter is smaller than the recirculation pipe diameter postulated in the accident scenario. The low pressure coolant injection system is sized to provide adequate core cooling in the event of a recirculation line break. All the effects of a feedwater line break at the safe-end are bounded by a recirculation line break.

If excessive unidentified leakage occurs inside containment, i.e. a pipe leak in the primary coolant system, BNP has a leak detection system and corresponding Technical Specification LCO to provided the unit with safe shutdown capability. This system meets or exceeds the requirements of Generic Letter 88-01, ref. NLS 88-160.

Alloy E-52 (Inconel) weld metal is equivalent to Alloy 690, and contains approximately twice the chromium (28-31%) of the more conventional Alloy 82 material (15%). These materials have been shown in studies by EPRI and others to be highly resistant to IGSCC initiation and propagation, which is the postulated flaw mechanism at the Brunswick feedwater welds. Due to the high chromium content in the Alloy E-52, the material will be effective in mitigating crack propagation even when the effects of dilution with the base material are considered. Therefore, the first layer of the overlay deposited shall constitute the first layer of the weld reinforcement design thickness, without field verification of the alloy content.

PART I: SAFETY ANALYSIS
(See instructions in Section 8.4.1)
(Attach additional sheets as necessary)

DOCUMENT NO. ESR 94-00458 REV. NO. 0

ANALYSIS (continued):

The overlay is designed to provide full structural design margin with no credit taken for the base material at the crack location. The weld overlay is a series of weld beads that builds up a minimum of two layers on the outside of the pipe to meet the minimum wall thickness required by ASME Section XI. The overlay length and taper are designed to transfer the loads to the base metal within design allowables. Code Case 2142 permits use of the Alloy E-52 material for this application.

The profile of the overlay accommodates future inspections required by ASME Section XI and NUREG-0313 Rev. 2 to verify any identified flaw does not propagate into the weld overlay, and the flaw size remains within allowables. These required inspections ensure that structural integrity and corresponding safety margins of the piping will be maintained.

The weld overlay will not prevent the pipe supports from performing their intended function and does not add a significant load to the supports. Axial shrinkage of the pipe line, less than 1/2", may result in slight adjustments in existing supports. The modification has a requirement to inspect the supports and make necessary adjustments.

The weld overlay will result is a slight radial shrinkage of the pipe inside diameter but will not affect flow to the feedwater spargers nor significantly affect thermal sleeve performance. The safe-end and thermal sleeve have an inside diameter which is approximately 1/2" smaller than at the locations where the upstream weld overlays will be applied.

The weld overlay is integrally attached to the feedwater piping/safe-end and will not become a loose part or projectile during an accident, therefore, equipment important to safety will not be affected by this modification.

Applying the weld overlay onto the carbon steel portion of the feedwater piping may be necessary. The feedwater piping may be filled with water during the weld overlay process. Probability of brittle fracture is considered, and suitable limits are established that avoid conditions where brittle fracture is possible. Parameters used in the welding procedure will be tested to confirm properties of the carbon steel are maintained within design requirements. Therefore, the probability of a pipe rupture is not increased.

REFERENCES:

Documents: NUREG-0313 Rev.2, Generic Letter 88-01, NLS-88-160
FSAR Sections:3.1, 3.2, 3.6, 4.1, 5.1, 5.2, 5.3.1, 5.3.3, 5.4.3, 5.4.9, 6.3.1, 6.3.2, 7.4, 3.9.1, 3.9.3, 5.4, 15.0.1, 15.0.2, 15.0.3, 15.6.3, 15.6.4
Tech Specs:3/4.3.7, 3/4.4.3, 3/4.4.8, 3/4.5.3, 4.0.5, 5.0, and associated Bases

PART II: ITEM CLASSIFICATION

DOCUMENT NO. ESR 94-00458 REV. NO. 0

	Yes	No
1. Does this item represent:		
a. A change to the facility as described in the SAFETY ANALYSIS REPORT?	[x]	[ ]
b. A change to the procedures as described in the SAFETY ANALYSIS REPORT?	[ ]	[x]
c. A test or experiment not described in the SAFETY ANALYSIS REPORT?	[ ]	[x]
2. Does this item involve a change to the individual plant Operating License or to its Technical Specifications?	[ ]	[x]
3. Does this item require a revision to the FSAR?	[x]	[ ]
4. Does this item involve a change to the Offsite Dose Calculation Manual?	[ ]	[x]
5. Does this item constitute a change to the Process Control Program?	[ ]	[x]
6. Does this item involve a major change to a Radwaste Treatment System?	[ ]	[x]
7. Does this item involve a change to the Technical Specification Equipment List?	[ ]	[x]
8. Does this item impact the NPDES Permit (all 3 sites) or constitute an "unreviewed environmental question" (SHNPP Environmental Plan Section 3.1) or a "significant environmental impact" (BSEP)?	[ ]	[x]
9. Does this item involve a change to a previously accepted:		
a. Quality Assurance Program	[ ]	[x]
b. Security Plan (including Training, Qualification, and Contingency Plans)?	[ ]	[x]
c. Emergency Plan?	[ ]	[x]
d. Independent Spent Fuel Storage Installation license? (If yes, refer to Section 8.4.2, "Question 9," for special considerations. Complete Part VI in accordance with Section 8.4.6)	[ ]	[x]

SEE SECTION 8.4.2 FOR INSTRUCTIONS FOR EACH "YES" ANSWER.

REFERENCES. List FSAR and Technical Specification references used to answer questions 1-9 above. Identify specific reference sections used for any "Yes" answer.

FSAR Sections:3.1, 3.2, 3.6, 4.1, 5.1, 5.2, 5.3.1, 5.3.3, 5.4.3, 5.4.9, 6.3.1, 6.3.2, 7.4, 3.9.1, 3.9.3, 5.4, 15.0.1, 15.0.2, 15.0.3, 15.6.3, 15.6.4
Tech Specs:3/4.3.7, 3/4.4.3, 3/4.4.8, 3/4.5.3, 4.0.5, 5.0, and associated Bases
For questions answered yes, see UFSAR 5.2.3.3.

PART III: UNREVIEWED SAFETY QUESTION DETERMINATION SCREEN

DOCUMENT NO. ESR 94-00458 REV. NO. 0

	YES	NO
1. Is this change fully addressed by another completed UNREVIEWED SAFETY QUESTION determination? (See Section 7.2.1, 7.2.2.5, and 7.9.1.1)	[ ]	[x]

REFERENCE DOCUMENT: ________________ REV. ________

	YES	NO
2. For procedures, is the change a non-intent change which only (check all that apply): (See Section 7.2.2.3)	[ ]	[ ]

[ ] Correct typographical errors which do not alter the meaning or intent of the procedure; or,

[ ] Add or revise steps for clarification (provided they are consistent with the original purpose or applicability of the procedure); or,

[ ] Change the title of an organizational position; or,

[ ] Change names, addresses, or telephone numbers of persons; or,

[ ] Change the designation of an item of equipment where the equipment is the same as the original equipment or is an authorized replacement; or,

[ ] Change a specified tool or instrument to an equivalent substitute; or,

[ ] Change the format of a procedure without altering the meaning, intent, or content; or

[ ] Deletes a part or all of a procedure, the deleted portions of which are wholly covered by approved plant procedures?

If the answer to either Question 1 or Question 2 in PART III is "Yes," then PART IV need not be completed.

PART IV: UNREVIEWED SAFETY QUESTION DETERMINATION

DOCUMENT NO. ESR 94-00458 REV. NO. 0

Using the SAFETY ANALYSIS developed for the change, test or experiment, as well as other required references (LICENSING BASIS DOCUMENTATION, Design Drawings, Design Basis Documents, codes, etc.), the preparer of the SAFETY EVALUATION must directly answer each of the following seven questions and make a determination of whether an UNREVIEWED SAFETY QUESTION exists.

A WRITTEN BASIS IS REQUIRED FOR EACH ANSWER — Yes / No

1. May the proposed activity increase the probability of occurrence of an accident evaluated previously in the SAFETY ANALYSIS REPORT? [ ] [x]

The probability of loss of feedwater supply by a pipe rupture is not increased by the application of the overlay. The application of a weld overlay to an existing weld provides full structural margin should a crack develop in the existing weld through-wall for 360°. The weld process will be developed so as to avoid a brittle transition zone in the carbon steel materials as the overlay is applied. The overlay also adds compressive stress to the pipe inside diameter and minimizes the initiation of any new cracks.

2. May the proposed activity increase the consequences of an accident evaluated previously in the SAFETY ANALYSIS REPORT? [ ] [x]

The application of the weld overlay will not increase the consequences of an accident previously analyzed in the UFSAR. The weld overlay will result is a slight shrinkage of the pipe inside diameter but will not affect flow. The safe-end and thermal sleeve have an inside diameter which is approximately 1/2" smaller than at the locations where the weld overlays will be applied. Radial shrinkage from the weld overlays will be less than 1/2". The feedwater nozzles are also located above the top of active fuel. The ECC systems are designed to keep the core covered if a recirculation pipe were to break. The feedwater line rupture inside containment is bounded by a loss of coolant from a rupture of the recirculation pipe.

3. May the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAFETY ANALYSIS REPORT? [ ] [x]

The application of a weld overlay returns the full structural design margin of a flawed weld assuming the weld was to develop a crack through-wall for 360°. The overlay length and taper are designed to transfer the loads to the base metal within design allowables. Parameters used in the welding procedure will be tested to confirm properties of the carbon steel are maintained within design requirements. Therefore, the probability of a pipe rupture is not increased

4. May the proposed activity increase the consequence of a malfunction of equipment important to safety evaluated previously in the SAFETY ANALYSIS REPORT? [ ] [x]

Failure of the piping at an overlay location is the same as failure of the piping without the weld overlay; water would be sprayed on equipment important to safety and the ECCS would be placed in service to ensure that the core remains covered. The material used in the design of the weld overlay, Alloy 52, is compatible with the primary coolant and will remain essentially intact if the feedwater pipe were to rupture at an overlay location.

5. May the proposed activity create the possibility of an accident of a different type than any evaluated previously in the SAFETY ANALYSIS REPORT? [ ] [x]

No new accident types are created. The overlay only affects the feedwater piping to which it is applied. The weld overlay is designed to improve the performance of the existing weld for the remainder of the plant design life. If the feedwater piping were to rupture at a weld overlay location, the consequences of such an accident have been analyzed and bounded by a recirculation pipe break.

PART IV (Continued)

DOCUMENT NO. ESR 94-00458 REV. NO. 0

	Yes	No
6. May the proposed activity create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAFETY ANALYSIS REPORT?	[ ]	[x]

Failure of the piping at an overlay location is the same as failure of the piping without the weld overlay; water would be sprayed on equipment important to safety and the ECCS would be placed in service to ensure that the core remains covered. Equipment is qualified for service in the containment based on accidents which bound a feedwater line break and, therefore, the possibility of equipment important to safety malfunctioning is not changed

	Yes	No
7. Does the proposed activity reduce the margin of safety as defined in the basis of any Technical Specification?	[ ]	[x]

The application of a weld overlay returns the full structural design margin of a flawed weld assuming the weld was to develop a crack through-wall for 360°. The weld overlay material has a high chromium content which will prevent the crack from propagating into the weld overlay and bound the crack's growth. Safety margins are not reduced by the application of the weld overlay.

	Yes	No
8. Based on the answers to questions 1 - 7, does this item result in an UNREVIEWED SAFETY QUESTION? If the answer to any of the questions 1-7 is "Yes", then the item is considered to constitute an UNREVIEWED SAFETY QUESTION.	[ ]	[x]
9. Is PNSC review required for any of the following reasons?	[ ]	[x]

If, in answering questions 1 or 3 "No", it was determined that the probability increase was small relative to the uncertainties; or, in answering question 2 or 4 "No", it was determined that the doses increased, but that the dose was still less than the NRC ACCEPTANCE LIMIT; or in answering question 7 "No", a parameter would be closer to the NRC ACCEPTANCE LIMIT, but the end result was still within the NRC ACCEPTANCE LIMIT; then PNSC review is required.

REFERENCES:

FSAR Sections:3.1, 3.2, 3.6, 4.1, 5.1, 5.2, 5.3.1, 5.3.3, 5.4.3, 5.4.9, 6.3.1, 6.3.2, 7.4, 3.9.1, 3.9.3, 5.4, 15.0.1, 15.0.2, 15.0.3, 15.6.3, 15.6.4
Tech Specs:3/4.3.7, 3/4.4.3, 3/4.4.8, 3/4.5.3, 4.0.5, 5.0, and associated Bases

This Unreviewed Safety Question Determination is for the following DISCIPLINE(s): (Additional Part IV forms may be included as appropriate.)

[ ] Nuclear Plant Operations
[ ] Nuclear Engineering
[X] Mechanical
[ ] Electrical
[ ] Instrumentation & Control
[X] Structural
[X] Metallurgy
[ ] Chemistry/Radiochemistry
[ ] Health Physics
[ ] Administrative Controls

Will the evaluation, on either a temporary or permanent basis:

1. Justify the deletion of equipment/common components from the BSEP EQ program?

   [ ] Yes [x] No

2. Justify the addition of (already existing) equipment/common components to the BSEP EQ program?

   [ ] Yes [x] No

3. Authorize the repair of EQ equipment/common components with other than qualified like-in-kind equipment/components parts?

   [ ] Yes [x] No

4. Affect the existing installation or interface (of EQ equipment/common component applications) as may be designated in EDBS and/or in the qualification data package (including changing the type of interface/installation)?

   [ ] Yes [x] No

5. Justify the (quality class) upgrade of equipment/common components or component parts which could be utilized in EQ applications?

   [ ] Yes [x] No

6. (Re)Define qualification parameters (e.g., normal or LOCA/HELB environmental conditions, postaccident operating time requirements, essential passive/active postaccident operating requirements, qualified life assumptions/results, etc.) for specific EQ equipment?

   [ ] Yes [x] No

7. Provide an EQ-related justification for continued operation (as required per PLP-02, Section 4.4.3.3 or 4.4.4)?

   [ ] Yes [x] No

8. Provide the resolution of a qualification problem (as required per PLP-02, Section 4.4.4)?

   [ ] Yes [x] No

Notes:
1. If all no, then no further EQ consideration is required. Mark the EER Traveler accordingly as required by ENP-12 and include this completed EER-EQIF within the EER package. An EQ Technical Review is not required.

2. If any yes, an EQ impact assessment (per Section 5.3) must be performed during the evaluation process. Mark the EER Traveler accordingly and include this completed EER-EQIF within the EER package. An EQ technical review is required.

BSEP FSAR CHANGE REVIEW AND APPROVAL FORM

Log No. ____________

Subject: U1 FW NOZZLE SAFE-END WELD OVERLAY NPP = ____________
(Brief Description of Change)

Initiating Document: (Include Rev. #) ESR 94-00458 REV. 0

Basis for Change: (SEE ATTACHED)

Affected Pages/Figures: (attach mark-ups) 5.2.3-4

Effective Date: 6/1/95 TARGET ACTUAL (circle one)

Safety Review Package Attached: Yes No

(If "No", Provide Justification Below)

Initiator: STEVEN L. BERTZ (Print) / NED Organization / 12/2/94 Date

Supervisor Approval: [signature] (Sign) NED Organization, 12/22/94 Date

REVIEW AND APPROVALS:

Licensing:

(Name)	(Date)	Comments

NOTES:

1) ATTACH ADDITIONAL SHEETS AS NECESSARY TO ASSURE COMPLETENESS
2) APPLICABLE REVIEW/APPROVALS TO BE DETERMINED BY LICENSING
3) USE CURRENT UPDATED FSAR PAGES FOR MARK-UP

Basis For Change:

The basic function of the feedwater supply system is to return the condensed steam from the turbine (via the condensate system) back to the reactor pressure vessel (RPV). The feedwater piping affected by this project is classified as part of the feedwater system. The feedwater piping is connected to the RPV nozzle via a safe-end and four transition pieces, which are considered part of the Nuclear Boiler System.

This project, ESR 94-00458, provides a weld overlay design for the three Inconel 82/182 welds connecting the RPV nozzle, safe-end, and feedwater piping in each feedwater line that are in NUREG-0313 Rev. 2, as Category "D". NUREG-0313 requires inspections of these welds and provides repair options if unacceptable flaws, per ASME Section XI, are identified. A weld overlay is a repair option. The weld overlay will provide full structural reinforcement assuming a 360° through-wall crack. The overlay design uses a weld filler metal which is highly resistant to IGSCC and will mitigate crack growth into the weld overlay. The weld overlay design is in accordance with the requirements of ASME Section XI, IWA-4300, 1989 ed., with guidance taken from Code Case N-504. This will assure that the overlay weld stresses imposed on the base material will remain within acceptable limits.

Unflawed Inconel welds on the feedwater lines at these safe-end locations may be overlaid to mitigate IGSCC to minimize future schedule impacts, resource costs and minimize accumulated dose.

The overlay geometry is designed to permit future inspections by visual, ultrasonic, and radiographic techniques to satisfy Augmented (NUREG-0313 Rev. 2) and In-service Inspection requirements.

This weld overlay repair is termed a non-ASME Code repair per OPLP-08 and NRC approval is required prior to implementation per Generic Letter 88-01 and OPLP-08.

The structural integrity of the reactor coolant pressure boundary shall not be degraded by this project. Performance requirements for structural integrity of the Nuclear Boiler system and RPV are unchanged by this project. The overlay design was verified by analysis to be acceptable for the existing design loads and materials.

The following criteria were applied to the reactor pressure vessel:

a) ASME B and PV Code, Section III, Paragraph N-331

b) Paragraphs 1.9.29 and 1.9.30 of Appendix J, Reactor Pressure Vessel - Data Sheet, Document No. 21A1100AR, and

c) 1) Impact properties of all "as-fabricated" carbon and low alloy steel used in the main closure flanges and the shell and head materials connecting to these flanges meet the requirements of the ASME Code, Section III, Paragraph N-330 at a temperature no higher than 10°F. In addition, this material has an NDT temperature no higher than 10°F as determined per ASTM E208.

2) Impact properties of all other "as-fabricated" carbon and low alloy steel pressure containing material and the vessel support skirt material meet the requirements of the ASME Code, Section III, N-330 at a temperature no higher than 40°F. In addition, this material has an NDT temperature no higher than 40°F as determined per ASTM E208.

3) Hardness tests were made on all main vessel closure bolting to demonstrate that heat treatment was performed. Studs, nuts, and bushings were hardness-treated individually. One sample from each lot of washers was hardness tested. Impact tests required by ASME Code, Section III, paragraph N-330 meet the Code requirements at a temperature no higher than 10°F.

ADD: 5.2.3.3.2 (ATTACHED)

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steels

5.2.3.4.1 Avoidance of Stress-Corrosion Cracking

No material inspection program was used for the Brunswick plant to verify the non-susceptibility of unstabilized austenitic stainless steel to intergranular attacks. At the time there was no valid screening test available to judge the service performance of austenitic stainless steel and, consequently, stainless steel materials were not tested in this manner. ASTM A262-Practice E measures susceptibility to intergranular attack in a strong acid solution and is not relatable to stress corrosion cracking in high purity reactor water.

The following controls were, therefore, specified for core support components and reactor coolant pressure containing members of components manufactured for the Brunswick plant in order to provide austenitic stainless steel with optimum mechanical and corrosion resistant properties. These controls for the most part supplemented the requirements specified in the referenced codes and material specifications (ASTM or ASME).

4 5.2.3.4.1.1 Purchase of Material

Unstabilized austenitic stainless steel materials for core support components and reactor coolant pressure containing members were purchased in the solution heat treated condition in accordance with the requirements of the applicable ASME/ASTM specification. These specifications require the material to be heated to solution heat treat temperatures, held for a sufficient time to put grain boundary carbides into solution, and thereafter quenched in water or cooled rapidly by other means.

5.2 INTEGRITY OF THE REACTOR COOLANT PRESSURE BOUNDARY

5.2.3.3.2 Reactor Coolant Pressure Boundary Piping Repair

BSEP has adopted a plan to rectify the non-conformities as they are discovered, by repair welding the affected areas as discussed below. Significant indications may be repaired by weld overlays or pipe replacement with non-service sensitive materials.

5.2.3.3.2.1 Weld Overlay Repair Description

The weld overlay repair method consists of weld-depositing a circumferential bank of nickel based Alloy E-52 material over the affected areas. The design philosophy for the weld overlay process is as follows: The weld overlay is applied utilizing a remotely controlled automatic welding machine. The automatic welder deposits a circumferential band of uniform, high integrity weld metal over the affected area. Postweld shrinkage of the band places the damaged area into a state of permanent compressive stress. This stress removes available energy that initiates and propagates stress corrosion cracking (SCC). In addition, the Alloy E-52 overlay is highly resistant to SCC and effectively halts the progression of through-wall cracking into the overlay thereby preventing leakage.

The repair method, as described above, has been used on the areas of the feedwater piping and safe-end(s) that were inspected and found to have unacceptable flaws or indications.

5.2.3.3.2.2 Design Basis.

The construction code for the original feedwater piping system is USAS Standard Code for pressure piping, power piping ANSI B31.1, 1967 edition. The safe-end is part of the reactor vessel and meets the original construction code, ASME Section III 1965 Edition, through Summer 1967 Addendum, and material requirements through Winter 1967.

The analysis and design of weld overlays were performed in accordance with Generic Letter 88-01/NUREG-0313 Rev. 2, ASME Section XI, 1989 ed., and guidance from Code Case N-504, and Code Case 2142, using material that is highly resistant to IGSCC.

The weld overlay as a repair is not addressed in the construction code and specifications or the ASME Boiler and Pressure Vessel Code Section XI. Also, the geometric configuration of the overlay is not directly covered by Section III. However, materials, fabrication procedures, and quality assurance requirements are in accordance with applicable portions of the above code sections. Non-destructive examination requirements are also consistent with these documents and are as follows: The pipe and weld surface will be liquid penetrant examined before the overlay is applied to insure that adequate fusion can be obtained. The overlay surface will be liquid penetrant tested and the overlay itself will be ultrasonically tested.

Applying the weld overlay onto the carbon steel portion of the feedwater piping may be necessary. The feedwater piping may be filled with water during the weld overlay process. Probability of brittle fracture is considered, and suitable limits are established that avoid conditions where brittle fracture is possible. Parameters used in the welding procedure will be tested to confirm properties of the carbon steel are maintained within design requirements. Therefore, the probability of a pipe rupture is not increased.

5.2.3.3.2.3 Safety Evaluation.

If repairs are required, the overlay will be ground back, the crack will be repaired manually (if necessary) and the surface will be liquid penetrant tested. The overlay will then be restored by automatic welding. All non-destructive examination will be done to insure that there is sound overlay adequately fused to the original pipe, flaws in the original joint will be noted and evaluated for consistency with the overlay design assumptions. Ultrasonic testing of the overlay will establish a new preservice record and an inservice leak test will insure piping integrity.

The repair method does not constitute an unreviewed safety question. The repairs have been made in accordance with applicable ASME Codes and NRC approved industry practices. Safety margins have not been reduced because the design of the weld repairs complies with ASME Section III stress limitation requirements, as applicable.

Recognized methods of ASME Code Section XI were used in the development of the weld overlay concept. In sizing the overlays, the conservative assumption was made that cracks will continue to propagate in the original weld heat-affected zone. However, this is highly unlikely due to favorable residual stresses produced by the overlay. Also, the nickel based high chromium weld material is not susceptible to intergranular stress corrosion cracking in a BWR environment.

The feedwater safe-end(s) and piping functions do not deviate in any manner due to implementation of this plant modification. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated does not increase.

The weld overlays installed will restore the feedwater piping and safe-end(s) to original design requirements so that the reactor coolant pressure boundary integrity is maintained.

The material for applying the weld overlay is of equivalent quality as the existing pipe material. All welding has been done in accordance with ASME qualified welding procedures. The weld overlay process will not cause intergranular stress corrosion cracking to propagate significantly during repairs. The weld overlay repair program does not affect any feedwater system process parameters or setpoints.

5.2.3.3.2.4 Augmented Inspection Program

An augmented inspection program monitors the development of indications in susceptible pipe with specialized ultrasonic examination and other techniques. Any indications are evaluated using fracture mechanics techniques. The results of this evaluation are considered in developing the plan and schedule for mitigation and repair.

Installation Package

ESR Bill of Material

ESR. No. 94-00458 **Rev. No.** 0

Installation					Design			
Item	Quan. / Units	PO Number	NIRF/Req. by	Part No.	Description	Spec./CGID	Q Status Buy	Q Status Use
1								
2								
3								
4								
5								
6								

Form 5

Drawing List

ENGINEERING SERVICE REQUEST DRAWING/DOCUMENT UPDATE FORM (DUF)	ESR NO. 94-00458 REV.NO. 0 PAGE NO. IP-C1

DWG./DOC#(SKETCH)	REV	PERMANENT PLANT DRAWING	UPDATE PRIOR TO [1]	PURPOSE I/R/D	SPECIFIC INFO (i.e, tag #,sheet #)
Installation will be accomplished in accordance with the design provided in Structural Integrity Document, CPL-34Q-301. This design document and the sketches contained therein, shall be reviewed/approved by NED. An engineering hold point in the installation prequisites is provided to specify which welds will be overlayed. If a weld requires an overlay the scope of the overlay(s) will be recorded in the installation portion of the modification.		FP-55126	C	R	GE drawing 767E723, Safe End, Feedwater Nozzle. (Detail drawing of the feedwater safe-end modification implemented prior to initial plant service.)

[1] Mods: A - Testing
B - Operability/Turnover
C - Closeout

EEs: A - Installation
B - Returning equipment to Service
C - Closeout

I - Installation
R - Reference
D - ESR for Document Update Only

** Emergency Operating Calculation Impacts should be determined by contacting STA Staff (Ref ENP-4.143

Form 7

Carbon	Inconel	Inconel	Inconel	Carbon	Carbon
1B212-2-2-FWRN1A	1B21N4A-2-SW2-3	1B21N4A-2-FWN4A45-3	1B21N4A-2-SW1-2	1B21N4A-2-FWN4A45-1	1B11N4A-RPV-SWAB
1B213-3-3-FWRN2A	1B21N4B-3-SW2-3	1B21N4B-3-FWN4B135-3	* 1B21N4B-3-SW1-2	1B21N4B-3-FWN4B135-1	1B11N4B-RPV-SWAB
1B216-3-6-FWRN3A	1B21N4C-6-SW2-3	1B21N4C-6-FWN4C225-3	1B21N4C-6-SW1-2	1B21N4C-6-FWN4C225-1	1B11N4C-RPV-SWAB
1B215-2-5-FWRN4A	1B21N4D-5-SW2-3	1B21N4D-5-FWN4D315-3	* 1B21N4D-5-SW1-2	1B21N4D-5-FWN4D315-1	1B11N4D-RPV-SWAB

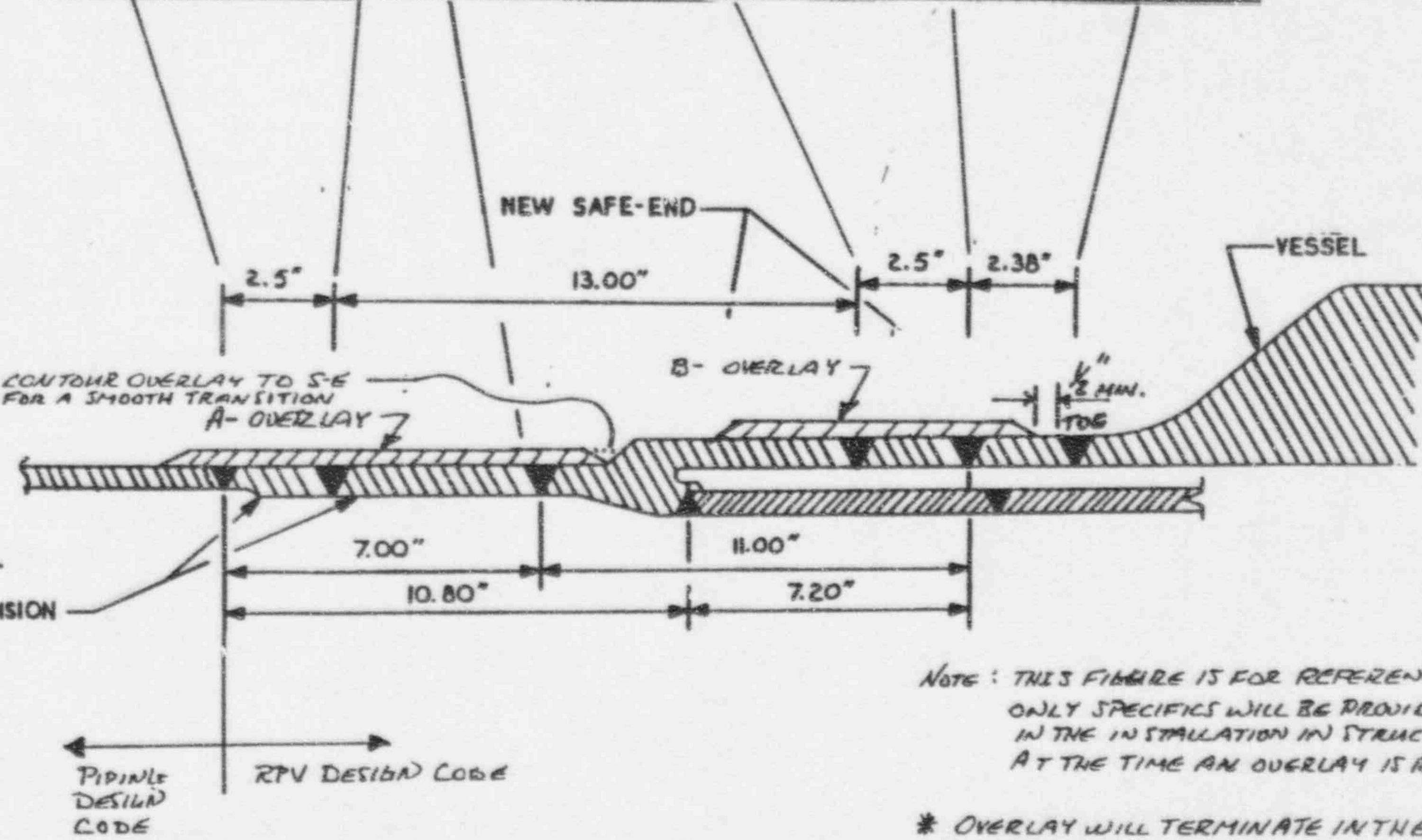


FIGURE 1

NOTE : THIS FIGURE IS FOR REFERENCE ONLY SPECIFICS WILL BE PROVIDED IN THE INSTALLATION INSTRUCTIONS AT THE TIME AN OVERLAY IS REQUIRED.

* OVERLAY WILL TERMINATE IN THE WELD CROWN OF THE ADJACENT WELD DUE TO A THERMOCOUPLE PAD

INSTALLATION CONSIDERATIONS

1.0 Purpose

The purpose of this procedure is to provide controls for the application of a weld overlay of the four feedwater safe-ends and piping. There are three (3) inconel 82/182 welds at each safe-end which are susceptible to IGSCC, and three carbon steel welds which are not susceptible. Two of the carbon steel welds are so close to the inconel weld that they would be covered up by the overlay also. Examinations are planned each refueling outage to determine if a flaw is present in the inconel weld. This modification is a contingency repair if a flaw is identified and a repair is necessary. Since the welds are in close proximity, a determination will be made as to the scope of the overlay when the flaw is identified and this may include adjacent unflawed welds. An engineering hold point in the installation prerequisites is provided to specify which welds will be overlayed. If a weld requires an overlay the scope of the overlay(s) will be recorded in the installation portion of the modification.

The feedwater lines are expected to be flooded and isolated from flow during the weld overlay process. The weld overlays may be applied with the pipe drained but not partially filled with water.

The Contractor's modification activities will be performed in accordance with the Contractor's QA/QC Program.

2.0 References

2.1 Contractor's procedures and drawing as accepted by CP&L.

2.2 FP-50445, Thermocouple

2.3 FP-55126, Safe End, Feedwater Nozzle

2.4 F-35046 Sht. 1, Reactor Building Drywell - Unit 1, Trays and Conduit

2.5 F-94005, Unit No.1, Reactor Vessel Temp. Monitoring Sys., Cable Diagram

2.6 F-95055 Sht. 9, Unit No. 1 Drywell Equipment Thermocouples, Interconnection Wiring Diagram

3.0 Responsibilities

3.1 NED - Design Group

NED shall coordinate design engineering for the modification.

NED shall develop field revisions as necessary to support design changes.

NED shall also provide engineering support during the modification installation to resolve installation problems affecting the design. Also, an engineering hold point in the installation prerequisites is provided to specify which welds will be overlayed. If a weld requires an overlay the scope of the overlay(s) will be recorded in the installation portion of the modification.

3.2 Projects Unit

Ensure work is accomplished in accordance with the plant modification procedures and WR/JO's. Also, coordinate the work sequence with other groups (i.e., Contractor, Operations, Plant Services).

Obtain and release necessary clearances.

Project turnover and plant modification operability.

Review and approval of documentation associated with the plant modification.

Provide temporary power, service air, work platforms, and requirements to Craft Resources for installation.

3.3 MSS/NDE Unit

Perform NDE examinations in accordance with site procedures.

Perform examinations required by the ASME Section XI, this modification, NUREG-0313 Rev.2

Review and acceptance of NDE records.

3.4 E&RC

Perform Health Physics functions as required to support work performed in accordance with this plant modification.

3.5 Operations

Be cognizant of the operational support required for the implementation of this plant modification.

Administer LCO's, Clearances, and manipulation of plant valves and equipment as required.

3.6 Installation Contractor

Responsible for all work associated with the installation of the weld overlay, including set-up of the automatic welding equipment, welding, and repairs in accordance with the Contractor's procedures, QA Manual, and the terms and conditions of the contract.

4.0 General Requirements

4.1 Work shall be performed in accordance with the installation instructions as set forth in this plant modification package WR/JO's, and the Contractor's work packages and procedures, as applicable and as accepted by CP&L. The Contractor's procedures, travelers, and instructions used to accomplish the work will be incorporated into this plant modification via the turnover package.

4.2 NED review and acceptance of Design Documents as required by the installation portion of the plant modification, shall be documented by sign-off prior to starting work.

4.3 Work performed per this plant modification in radiation environments shall be coordinated with other work in the same area to minimize delays and reduce exposure.

4.4 Personnel who will be performing work in high radiation areas or in the Drywell shall be trained on mock-ups or other suitable means.

4.5 An NIS-2 form is required by OPLP-08 to document the weld overlay repair and any changes to supports.

4.6 The system window required for this work is 14 days. Work will not be performed during fuel movement per ALARA requirements. This work may be performed on critical path in future outages.

4.7 The radial clearances for the safe-end and piping at the overlay locations can be found in drawing FP-18002.

4.8 Feedwater lines N4B and N4D have thermocouples pads welded between the two welds closest to the reactor vessel. These pads should be removed before the weld overlay is applied and reinstalled in accordance with FP-55126 after the weld overlay is completed.

5.0 General Prerequisites

5.1 NRC has reviewed and approved the weld overlay design.

______________________________	______________
Responsible Engineer or Designee	Date

5.2 PMS Project Manager shall notify the Contractor that the prerequisites are being met and work may proceed to mobilize for installation of the weld overlay(s).

______________________________	______________
PMS Project Manager or Designee	Date

5.3 Required Clearances have been obtained.

______________________________	______________
PMS Project Manager or Designee	Date

5.4 Temporary services have been installed.

______________________________	______________
Craft Resources Manager or Designee	Date

5.5 Temporary shielding has been installed as required by the HP/ALARA groups in order to reduce exposure, and personnel have been instructed to minimize stay time in the radiation area, and to observe all ALARA rules.

_______________________________ ______________

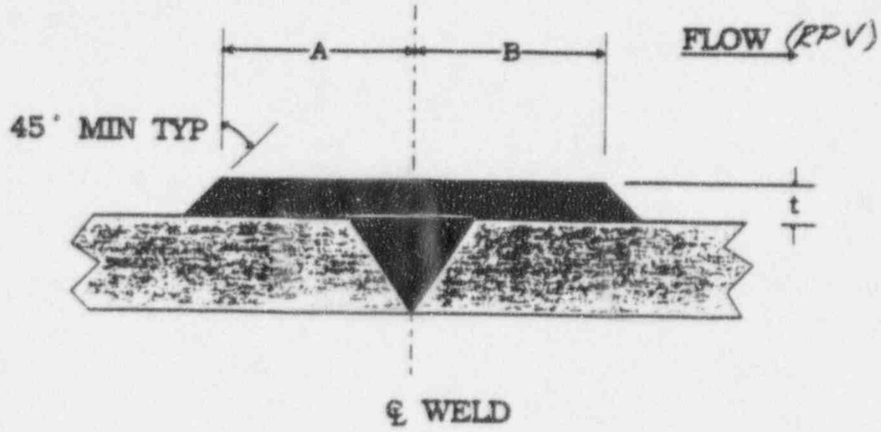
PMS Project Manager or Designee Date

5.6 Verify that the feedwater line which will receive the weld overlay has been isolated from flow and flooded or is isolated and drained of water at the safe-end location,and required LCO's and clearances are in place.

_______________________________ ______________

PMS Project Manager or Designee Date

5.7 The required examinations of the feedwater safe-end/pipe inconel welds have been evaluated in accordance with the requirements of the ISI Program and NUREG-0313 Rev. 2 and the following inconel welds require an overlay in accordance with the design requirements of this modification (adjacent welds are included):



Not to Scale

WELD NUMBER	FLAW CHARACTERIZATION	DESIGN DIMENSIONS			COMMENTS
		t	A	B	
	Assumed 360° Circ. 100% throughwall flaw				

Additional instructions: ______________________________

____

_______________________________ ______________

Responsible Engineer or Designee Date

5.8 Verify that the contractor has read and is familiar with the requirements of this plant modification. Verify that the Contractor's procedures have been reviewed and approved by CP&L prior to work. Also, verify that the contractor has located the correct weld(s) to be overlaid and the length/position to which the weld overlay must conform, based on the design requirements of this modification.

_______________________________ _______________

PMS Project Manager or Designee Date

6.0 Overlay Installation

6.1 If an overlay of 1B21N4D-5-SW1-2 OR 1B21N4B-3-SW1-2 is required, the Project Manager shall verify that the location of the thermocouple pads does not interfere with the weld overlay or the equipment. If an interference exists, issue a WR/JO to remove the thermocouple pads and perform a surface examination of the area in accordance with the ISI Program. Retain the thermocouple pad for installation after the weld overlay is complete.

□ An interference exists with the thermocouple pad, remove the thermocouple pad and perform inspections in accordance with WR/JO___________.

□ Do not remove the thermocouple pad. The thermocouple pad does not interfere with the weld overlay installation.

_______________________________ _______________

PMS Project Manager or Designee Date

6.2 The pipe/safe-end area to receive the weld overlay(s) has been examined by liquid penetrant and meets the acceptance criteria of the ISI Program and this modification.

_______________________________ _______________

MSS/NDE Project Manager or Designee Date

6.3 PMS Project Manager shall notify the Contractor that the prerequisites have been met and work may proceed to install the weld overlay(s) in accordance with CP&L approved Contractor's procedures.

_______________________________ _______________

PMS Project Manager or Designee Date

6.4 Weld overlay is complete and meets the profile and finish requirements for NDE as described in this modification.

_______________________________ _______________

PMS Project Manager or Designee Date

6.5 The weld overlay has been examined by liquid penetrant and ultrasonic techniques and meets the acceptance criteria of ASME Section XI, 1980 Edition through Winter 1981 Addenda, and this modification. (If unacceptable, repairs shall be performed by the contractor in accordance with CP&L approved Contractor procedures. Repeat steps 6.3 and 6.4, as necessary).

______________________________ ______________

MSS/NDE Project Manager or Designee Date

6.6 Thermocouple pad(s) have been reinstalled and examined in accordance with WR/JO __________ (ref. FP-55126). Applicable only if an overlay of 1B21N4D-5-SW1-2 OR 1B21N4B-3-SW1-2 is required.

______________________________ ______________

PMS Project Manager or Designee Date

6.7 The thermocouple(s) have been tested in accordance with OPIC-T/C001A and are acceptable (ref. FP-50445, F-35046, F-94005, F-95055, F-95059, and F-95085).

______________________________ ______________

PMS Project Manager or Designee Date

6.8 Installation of the weld overlay(s) has been completed.

______________________________ ______________

PMS Project Manager or Designee Date

6.9 Supports in the vertical run of feedwater piping upstream of the weld overlay have been examined and necessary adjustments have been made in accordance with site procedures.

N4A	N4B	N4C	N4D
1-B21-2CH5	1-B21-3CH14	1-B21-6CH28	1-B21-5CH19
1-B21-2SS230	1-B21-3SS13	1-B21-6SS244	1-B21-5SS238
1-B21-2SS4	1-B21-2SS235	1-B21-6SS27	1-B21-5SS18
1-B21-2SS229			1-B21-5SS239
WHIP RESTRAINTS ABOVE 50' El.	WHIP RESTRAINTS ABOVE 50' El.	WHIP RESTRAINTS ABOVE 50' El.	WHIP RESTRAINTS ABOVE 50' El.
Ref. F-28046 sht 660, F-19013, F-19007	Ref. F-28046 sht 660, F-19013, F-19007	Ref. F-28046 sht 516, F-19013, F-19007	Ref. F-28046 sht 516, F-19013, F-19007

______________________________ ______________

PMS Project Manager or Designee Date

6.10 Request Clearances be released.

PMS Project Manager or Designee — Date

6.11 Temporary services have been removed and the work area(s) are restored.

Craft Resources Manager or Designee — Date

6.12 Notify the Work Control Center that all work activities per this plant modification have been completed.

PMS Project Manager or Designee — Date

6.13 The NIS-2 form has been completed for the weld overlay(s) and support(s), as required, in accordance with 0PLP-08.

PMS Project Manager or Designee — Date

6.14 Contractor's final design documents have been received by CP&L and have been reviewed and accepted as documented by the completion of an Owners Review per 0ENP-310.

Responsible Engineer or Designee — Date

NOTE

Vendor documentation is to be included in the turnover package.

6.15 Contractor's completed procedures, in-process changes and supporting records have been reviewed by CP&L and found acceptable.

PMS Project Manager or Designee — Date

Responsible Engineer or Designee — Date

TESTING CONSIDERATIONS

This modification implements a structural repair, weld overlay, in accordance with the NRC Staff positions of Generic Letter 88-01/NUREG-0313 Rev. 2, ASME Section XI, 1989 ed., no addenda, and ASME Code Committee guidance of Code Case N504 using material that is highly resistant to IGSCC.

The pressure boundary will not be penetrated, therefore, a system leakage, inservice, or functional test may be performed rather than a hydrostatic test (per Code Case N-504). The test shall be in accordance with ASME Section XI, 1980 ed., through Winter 1981 Addenda.

ENCLOSURE 4

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1
NRC DOCKET NO. 50-325
OPERATING LICENSE NO. DPR-71
WELD OVERLAY REPAIR OF FEEDWATER PIPING AND SAFE-ENDS

Inco Alloys International, Inc.
Publication No. IAI-106
"INCONEL Filler Metal 52 and
INCONEL Welding Electrode 152"

INCONEL Filler Metal 52 and INCONEL Welding Electrode 152

INCONEL Filler Metal 52 is used for gas-metal-arc and gas-tungsten-arc welding of INCONEL alloy 690. INCONEL Welding Electrode 152 is used for shielded-metal-arc welding of INCONEL alloy 690. These nickel-chromium welding products were developed to meet the needs of the changing nuclear power industry. Current nuclear steam generators, and retrofits of existing units, require components with the higher chromium content found in INCONEL alloy 690. The higher chromium is needed for improved resistance to stress-corrosion cracking in the nuclear, pure-water environment.

Although INCONEL alloy 690 has good weldability as a base metal, it does not have good welding characteristics operated as a welding product. This limitation has been overcome, while maintaining excellent corrosion resistance, by optimizing the chemical composition. INCONEL Filler Metal 52 and INCONEL Welding Electrode 152 were designed to produce high-quality welds in all positions using a variety of processes in similar and dissimilar joints involving INCONEL alloy 690. The welding products also produce corrosion-resistant overlays on most low-alloy and stainless steels. Other uses include applications requiring resistance to oxidizing acids and welding of INCONEL alloy 690 "glass melters" used for disposal of nuclear waste.

Extensive information on stress-corrosion cracking is available in Electric Power Research Institute Report NP-5882S. In that report, the welding products are referred to by their developmental designations: R-127 for INCONEL Filler Metal 52 and R-135 for INCONEL Welding Electrode 152.

Publication No. IAI-106

INCONEL Filler Metal 52 and INCONEL Welding Electrode 152

INCONEL Filler Metal 52 is used for gas-metal-arc and gas-tungsten-arc welding of INCONEL alloy 690. INCONEL Welding Electrode 152 is used for shielded-metal-arc welding of INCONEL alloy 690. These nickel-chromium welding products were developed to meet the needs of the changing nuclear power industry. Current nuclear steam generators, and retrofits of existing units, require components with the higher chromium content found in INCONEL alloy 690. The higher chromium is needed for improved resistance to stress-corrosion cracking in the nuclear, pure-water environment.

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Publication No. IAI-106

INCONEL Filler Metal 52

Table 1—Limiting Chemical Composition, % (Filler Metal)

Nickel (plus Cobalt)	Remainder
Carbon	0.04 max.
Manganese	1.00 max.
Iron	7.0-11.0
Sulfur	0.015 max.
Silicon	0.50 max.
Molybdenum	0.50 max.
Copper	0.30 max.
Chromium	28.0-31.5
Titanium	1.00 max.
Aluminum	1.10 max.
Phosphorus	0.020 max.
Niobium (plus Tantalum)	0.10 max.
Aluminum + Titanium	1.5 max.
Others	0.50 max.

Table 2—Typical Mechanical Properties (As Welded)

Tensile Strength, psi	94,000
MPa	650
Elongation, %	40

Table 3—Conditions for Gas-Metal-Arc Welding

Metal Transfer	Wire Diameter		Voltage, V	Current,[a] A	Wire Feed		Shielding Gas[b]
	in	mm			in/min	m/min	
Short Circuiting	0.035	0.9	21-24	90-160	175-400	4.4-10.2	75% Ar, 25% He
Pulsing Arc	0.045	1.1	21-25[c]	100-160	130-150	3.3-3.8	75% Ar, 25% He
Spray	0.062	1.6	28-33	240-300	160-200	4.1-5.1	100% Ar

[a] Direct current, electrode positive.
[b] Flow rate of 60 ft³/h (1.7 m³/h).
[c] Peak voltage of 44-46 V.

Table 4—Conditions for Gas-Tungsten-Arc Welding

Wire Diameter		Current,[a] A	Shielding Gas	Gas Flow Rate	
in	mm			ft³/h	m³/h
0.093	2.4	80-140	100% Ar	15-20	0.42-0.57
0.045[b]	1.1	150-300	100% Ar	15-30	0.42-0.85

[a] Direct current, electrode negative.
[b] Automatic welding.

Sizes

36-in (0.91-m) straight lengths in diameters of 1/16, 3/32, and 1/8 in (1.6, 2.4, and 3.2 mm).

30-lb (13.6 kg) spools in diameters of 0.035, 0.045, and 0.062 in (0.9, 1.1, and 1.6 mm).

INCONEL Welding Electrode 152

Table 5—Limiting Chemical Composition, % (Deposited Weld Metal)

Nickel (plus Cobalt)	Remainder
Carbon	0.05 max.
Manganese	5.0 max.
Iron	7.0-12.0
Sulfur	0.015 max.
Silicon	0.75 max.
Molybdenum	0.50 max.
Copper	0.50 max.
Chromium	28.0-31.5
Titanium	0.50 max.
Aluminum	0.50 max.
Phosphorus	0.030 max.
Niobium (plus Tantalum)	1.0-2.5
Others	0.50 max.

Table 6—Typical Mechanical Properties (As Welded)

Tensile Strength, psi	97,000
MPa	670
Elongation, %	42

Table 7—Sizes and Welding Current

Diameter		Length		Current,[a] A
in	mm	in	mm	
3/32	2.4	9	229	45-70
1/8	3.2	14	356	75-100
5/32	4.0	14	356	95-130

[a] Direct current, electrode positive.

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Document Update Form

ENGINEERING SERVICE REQUEST DRAWING/DOCU`.`ENT UPDATE FORM (DUF)					ESR NO. 94-00458 REV.NO. 0 PAGE NO. IP-F1
DWG./DOC#(SKETCH)	REV	PERMANENT PLANT DRAWING	UPDATE PRIOR TO (1)	PURPOSE I/R/D	SPECIFIC INFO (i.e, tag #,sheet #)
DBD-01			N/A		
UFSAR			N/A		RCI 4.1 FORM SUBMITTED TO REG. COMPLIANCE
FP-55126			C		
C-24004 SH. 13-1			B		John Yadusky #2620
C-24004 SH 14-1			B		John Yadusky #2620
FATIGUE PRO SOFTWARE for fatigue usage evaluations			N/A		Incorporate a note that the weld overlay does not have a significant effect on the analysis and no changes were required.
0ENF-16.0			C		
0ENP-16.2			C		

(1) Mods: A - Testing
B - Operability/Turnover
C - Closeout

EEs: A - Installation
B - Returning equipment to Service
C - Closeout

I - Installation
R - Reference
D - ESR for Document Update Only

** Emergency Operating Calculation Impacts should be determined by contacting STA Staff (Ref ENP-4.143

Form 7