LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6 PRIM' . SYSTEM BOUNDARY	4.6. PRIMARY SYSTEM BOUNDARY
Applicability:	Applicability:
Applies to the operating status of the reactor coolant system.	Applies to the periodic examination and testing requirements for the reactor cooling system.
Objective:	Objective:
To assure the integrity and safe operation of the reactor coolant system	To determine the condition of the reactor coolant system and the operation of the safety devices related to it.
Specification:	Specification:
A. Thermal and Pressurization Limitations	A. Thermal and Pressurization Limitations
1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a one-hour period except when the vessel temperatures are above 450°F. The reactor vessel flange to adjacent reactor vessel shell temperature differential shall not exceed 145°F.	1. During heatups and cooldowns, with the reactor vessel temperature less than or equal to 450°F, the temperatures at the following locations shall be permanently logged at least every 15 minutes until the difference between any two readings at individual locations taken over a 45 minute period is less than 5°F: a. Reactor vessel shell adjacent to reactor vessel flange b. Reactor vessel shell flange c. Recirculation loops A and B
2. The reactor vessel shall not be pressurized for hydrostatic and/or leakage tests, and critical core operation shall not be conducted unless the reactor vessel temperatures are above those defined by the appropriate curve on Figures 3.6.1, 3.6.2, and 3.6.3. (Linear interpolation between curves is permitted). At stated pressure, the reactor vessel bottom head may be maintained at temperatures below those temperatures corresponding to the adjacent reactor vessel shell as shown in Figures 3.6.1 and 3.6.2.	2. Reactor vessel shell temperatures, including reactor vessel bottom head, and reactor coolant pressure shall be permanently logged at least every 15 minutes whenever the shell temperature is below 220°F and the reactor vessel is not vented. Test specimens of the reactor vessel base, weld and heat affected zone metal subjected to the highest fluence of greater than 1 Mev neutrons shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The specimens and sample program shall conform to the

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.A Thermal and Pressurization Limitations (Cont'd)	4.6.A Thermal and Pressurization Limitations (Cont'd)
In the event this requirement is not met, achieve stable reactor conditions with reactor vessel temperature above that defined by the appropriate curve and obtain an engineering evaluation to determine the appropriate course of action to take.	requirements of ASTM E 185-66. Selected neutron flux specimens shall be removed at the frequency required by Table 4.6.3 and tested to experimentally verify adjustments to Figures 3.6.1, 3.6.2, and 3.6.3 for predicted NDT temperature irradiation shifts.
3. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head is greater than 55°F.	3. When the reactor vessel head bolting studs are tensioned and the reactor is in a Cold Condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.
4. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.	4. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.
5. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.	5. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.
6. Thermal-Hydraulic Stability Core thermal power shall not exceed 25% of rated thermal power without forced recirculation.	
B. Coolant Chemistry	B. Coolant Chemistry
1. The reactor coolant system radioactivity concentration in water shall not exceed 20 microcuries of total iodine per ml of water.	1. a. A reactor coolant sample shall be taken at least every 96 hours and analyzed for radioactivity content. b. Isotopic analysis of a reactor coolant sample shall be made at least once per month.

TABLE 4.6.3
REACTOR VESSEL MATERIAL
SURVEILLANCE PROGRAM WITHDRAWAL SCHEDULE

Capsule Number	Effective Full Power Years (EFPY)
1	4.17
2	15 (approx.)
3	32 (End of Life)

Attachment C to BECo

Response to Generic Letter 92-01

CHARPY V-NOTCH IMPACT RESULTS FOR PILGRIM BASE METAL

Specimen Identification	Test Temperature, F	Impact Energy, ft-lb	Lateral Expansion, mils	Fracture Appearance, Percent Shear
Y2/C	-100	3.1	2.4	1
Y1/L	-50	8.3	9.0	5
Y2/4	0	24.0	23.2	25
Y3/2	25	52.0	43.2	30
Y2/D	30	44.0	37.4	25
Y1/2	50	74.9	54.2	45
Y3/Y	50	71.3	61.2	40
Y2/1	50	56.4	50.0	45
Y1/D	80	83.8	67.0	60
Y2/2	115	110.0	85.2	85
Y2/5	150	131.9	75.0	100
Y3/M	250	130.3	92.0	100
Y1/7	300	130.2	84.2	100

CHARPY V-NOTCH IMPACT RESULTS FOR PILGRIM WELD METAL

Specimen Identification	Test Temperature, F	Impact Energy, ft-lb	Lateral Expansion, mils	Fracture Appearance, Percent Shear
Y4/B	-200	11.0	9.4	0
Y5/M	-150	14.5	17.0	1
Y6/U	-125	6.8	6.8	5
Y6/5	-125	37.0	31.6	10
Y6/Y	-100	41.0	54.0	20
Y4/5	-50	62.0	54.8	35
Y6/7	0	73.4	68.2	65
Y5/T	50	85.8	79.0	75
Y4/D	60	81.0	72.8	80
Y5/U	60	98.0	83.8	85
Y4/7	60	89.0	77.8	85
Y6/1	80	96.9	86.6	90
Y4/A	150	111.1	93 °	100
Y4/3	230	114.8	85.4	100
Y6/6	300	112.2	75.0	100

CHARPY V-NOTCH IMPACT RESULTS FOR PILGRIM HAZ METAL

Specimen Identification	Test Temperature, F	Impact Energy, ft-lb	Lateral Expansion, mils	Fracture Appearance, Percent Shear
YA/J	-200	4.5	3.8	0
Y7/P	-165	21.5	16.0	5
Y7/A	-135	13.0	10.8	1
YB/1	-100	13.0	12.4	5
Y7/U	-50	37.5	30.0	15
YA/3	-40	68.6	53.4	35
YA/P	-35	74.5	58.4	45
YA/T	0	68.0	60.4	75
Y7/2	35	97.5	84.0	85
YB/2	50	135.7	79.4	100
YB/6	81	107.9	77.2	90
Y7/7	150	99.9	84.8	100
YB/E	300	112.0	87.8	100

TENSILE TEST RESULTS FOR PILGRIM UNIRRADIATED BASELINE SPECIMENS

Material	Specimen Number	Test Temp, F	0.2 Percent Offset Yield Strength, ksi	Ultimate Tensile Strength, ksi	Fracture			Reduction in Area, percent	Elongation, %	
					Load, lb	Strength, ksi	Stress, ksi		Uniform	Total
Base	YC-5	72	64.5	86.4	2800	55.7	185	69.90	12.4	25.5
Base	YC-6	550	57.7	62.8	2925	58.2	166	65.00	10.6	21.4
Weld	YD-T	72	68.8	78.4	2450	49.6	178	72.02	14.5	28.6
Weld	YD-K	550	54.6	78.3	2725	54.7	148	63.04	12.0	23.0
HAZ	YE-D	72	61.7	79.6	2475	49.7	178	72.17	8.5	22.9
HAZ	YE-E	550	57.2	78.1	2410	48.4	136	64.48	7.9	18.9

Attachment D to BECo

Response to Generic Letter 92-01

Attachment D

LIST OF PIECE, CODE AND HEAT NUMBERS

Bottom Head

Pc. No.	Code No.	Heat No.
336-02	G-3114	C-2888-3
336-03	G-3113-1, 2, 3	C-2913-3, A-2222-2, A-222
336-04	G-3111	C-2851-2

Lower Shell Assembly

Pc. No.	Code No.	Heat No.
337-01	G-3109-1, 2, 3	C-2957-1, C-2957-2, C-2973-1,
337-03	G-3108-1, 2, 3	C-2921-2, C-2945-1, C-2945-2

Upper Shell Assembly

Pc. No.	Code No.	Heat No.
337-02	G-3109-4, 5, 6	C-2561-2, C-2973-2, C-3301-2
337-04	G-3107-1, 2, 3	A-2094-1, A-2094-2, C-2906-1
339-02	G-3101	2V-545
339-06	G-3142	Certification

Vessel Support Skirt

Pc. No.	Code No.	Heat No.
340-02	S-8529	VIEF-1225
340-04	G-3154	A-2933-5
340-06	G-3153	A-3118-2

PILGRIM REACTOR VESSEL BELTLINE
WELD DATA SUMMARY

DESCRIPTION	WELD NO.	FILLER HEAT NO.	WIRE TYPE	FLUX TYPE	FLUX LOT	WIRE CHEMISTRY CU	WIRE CHEMISTRY NI	WIRE CHEMISTRY REF. SOURCE	WIRE R.G. 1.99 CF	WELD CHEMISTRY CU	WELD CHEMISTRY NI	WELD CHEMISTRY REF. SOURCE	WELD R.G. 1.99 CF	GENERIC CHEMISTRY CU	GENERIC CHEMISTRY NI	GENERIC CHEMISTRY REF. SOURCE	GENERIC R.G. 1.99 CF
LONGITUDINAL WELD LOWER INTERMEDIATE SHELL	1-338 A,B,C	27204	B-4 (MOD)	LINDE	3774	0.12	1.06/ 1.07	4	161					0.32 0.24 0.22 0.18	0.82 1.00 1.10 0.96	2 & 3	234 239 245 209
		12008	B-4 (MOD)	1092	3774	0.13	0.99	4	172					0.13 0.22	0.99 0.83	3	172 205
GIRTH WELD LOWER INTERMEDIATE TO LOWER SHELL	1-344	21935	B-4 (MOD)	LINDE 1092	3869	0.13	0.71	4	151	0.22 0.20	*0.71* *0.71*	2 & 4 4	184 177	0.21 0.22	0.68 0.83	2 & 3	177 205
LONGITUDINAL WELD LOWER SHELL	2-338 A,B,C	27204	B-4 B-4	LINDE 1092	3774 3714	0.13	1.06/ 1.07	4	173					0.21	0.98	2 & 3	226
TEST CAPSULE SEAM WELD	1-366	13253	B-4	LINDE 1092	3774	0.13	1.06/ 1.07	4	173	0.17	0.81	2	185				
COMPOSITE WELD	ALL									*0.22*	*1.07*		241	0.32	*1.07*		275
"WORST CASE" WELD	ALL									*0.35*	*1.07*		284				

REFERENCES: (1) SWRI REPORT
(ATTACHED) (2) EPRI DATA NP-4797 (EXCERPTS& T. GRIESBACH)
(3) G. E. DATA (T.CAINE)
(4) C. E. DATA (R.HILLIS)

PILGRIM REACTOR VESSEL BELTLINE
WELD DATA SUMMARY