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REACTOR PRESSURE VESSEL AND SURVEILLANCE PROGRAM
MATERIALS LICENSING INFORMATION
FOR
SURRY UNITS 1 AND 2

Babcock & Wilcox
a McDermott company

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1. INTRODUCTION

This report provides a review and update of the materials data and information for the reactor pressure vessels of Surry Units 1 and 2 to ensure that they are in compliance with the requirements of 10CFR50, Appendix G.[1] In addition, the reactor pressure vessel surveillance programs were reviewed for compliance with 10CFR50, Appendix H.[2] The reactor pressure vessel surveillance capsule withdrawal schedule was modified to meet the intent of ASTM E185-82[3] as referenced by 10CFR50, Appendix H.

As a result of this review and update the reactor vessels materials data bases for Surry Units 1 and 2 were found to be in compliance with 10CFR50, Appendix G. The surveillance program materials properties data bases are in compliance with 10CFR50, Appendix H and will provide the material data necessary to ensure continued licensibility of the reactor vessels.

A new reactor vessel surveillance capsule withdrawal schedule was developed to meet the requirements of ASTM E185-82 as referenced by 10CFR50, Appendix H. This new schedule will provide needed irradiation materials data in a timely manner.

The reason for this review and update is that the reactor pressure vessels were fabricated and the corresponding surveillance programs were developed prior to the implementation of 10CFR50, Appendixes G and H. These regulations recognized that the older plants could not meet all the requirements and established guidelines to meet the intent, if not the letter of the regulations. In addition, these regulations have been revised as experience, new data and analysis capability related to reactor vessel integrity have developed. A periodic review and update is necessary to ensure continued compliance with the regulations.

2. REACTOR VESSEL DATA BASES

The establishment of the mechanical and toughness properties of reactor pressure vessels in accordance with applicable regulations and standards is an essential aspect of the licensing process. As these rules are improved it is necessary to ensure that the data used for licensing of the reactor vessels are representative of the best information and materials properties available for each specific reactor vessel. The data are also essential in establishing the normal pressure-temperature operating limitations as required by 10CFR50, Appendix G.

2.1. Surry Unit - 1

The materials and chemical composition data for the Surry Unit 1 reactor vessel are presented in Tables 2-1 through 2-6. These data represent an accumulation of information from various sources (References 4, 10 and 11) which include the improved chemical composition data for the Linde 80 submerged arc weld metals as reported in BAW-1799.[5] In addition, the initial reference temperature data represents the best available data as defined in Section 2.4.

The location and identification of the plates and welds within the belt-line region of the Surry Unit 1 reactor pressure vessel are shown in Figure 2-1.

2.2. Surry Unit 2

The data for Surry Unit 2 reactor vessel are presented in Tables 2-8 through 2-13. These data represent an accumulation of information from various sources (References 4, 12 and 13 which include the improved chemical composition data for the Linde 80 submerged arc weld metals as reported in BAW-1799.[5] In addition, the initial reference temperature data represents the best available data as defined in Section 2.4.

The location and identification of the plates and welds within the belt-line region of the Surry Unit 2 reactor pressure vessel are shown in Figure 2-2.

2.3. Surveillance Data Bases

Each reactor vessel has a surveillance program to monitor the neutron radiation damage of the materials in the beltline region. These data for each reactor vessel at Surry were tabulated separately from the main data base as a convenience for easy reference. The data are presented in Tables 2-7 and 2-14.

2.4. Initial Value of Reference Temperature

The initial value of reference temperature is not always available for the materials used to fabricate older reactor vessels because it was not an established requirement of the ASME Code. Even for reactor vessels completed after the establishment of the requirement, the value was often unattainable because no suitable material was available. The necessary drop weight test data were usually obtained for both plate and forging materials and this provided a reliable data base to establish the initial reference temperature for these materials.

The initial reference temperature of weld metals was not obtained until after it was required by the ASME Code. At that time an effort was made to re-evaluate the weld qualification to obtain initial reference temperatures. Subsequently, a statistical evaluation was made of the Linde 80 weld metals fabricated after the establishment of the ASME Code requirements, concurrent with a re-evaluation of old weld metals, to provide a basis for a mean value and standard deviation. A similar approach was used to establish values for plate and forging materials for which the needed actual test data were not available. An acceptable method for establishing the initial reference temperature is presented in the NRC Standard Review Plan, Section 5.3.2.[6] It is recognized that the values recommended in the Standard Review Plan are very conservative.

Previous evaluations of the Surry reactor vessels had established reference temperatures for the base materials which were representative of actual data. However, the weld metals had no data for the Linde 80 submerged-arc weldments made by B&W and only the initial value of Rotterdam welds established per the NRC Standard Review Plan, Section 5.3.2.

A more recent development is the need to establish the standard deviation of the reactor vessel materials to be used in determining the reference temperature shift as a result of irradiation. Standard deviations for measured initial reference temperature have been established from data obtained since the ASME Code requirement for establishing the reference temperature of all reactor vessel materials.

Listed below are the initial reference temperatures and standard deviations used if actual measured values are not available:

Material Description	Initial RT_{NDT}, F	RT_{NDT} Std. Dev.	Reference
SA508, Cl. 2	3	±30F	BAW-10046P[7]
SA533, Gr.Bl	1	±26F	BAW-10046P
Linde 80 Welds	-6	±19F	BAW-1803[8]
RDM Welds	0	±20F	SECY 82-465[9]

Another recent development is the need to establish unirradiated, or initial, Charpy upper-shelf energy (USE) values for the high-copper Linde 80 submerged-arc weldments made by B&W for which no data are available. A statistical evaluation was made of the upper-shelf energy data for the Linde 80 weld metals fabricated after the establishment of the ASME Code requirements; this included a re-evaluation of the limited data from the old weld metals, to provide a basis for a mean value and standard deviation.

Listed below are the initial Charpy upper-shelf energy value and standard deviations to be used if actual measured values are not available:

Material Description	Initial USE, ft-lb.	USE Std. Dev., ft-lbs.	Reference
Linde 80 Welds	70	±6	BAW-1803[8]

Table 2-1. Identification of Reactor Vessel Beltline Region Weld Metals - Surry Unit-1

Weld Identification	Weld Location	Welding Process	Weld Wire		Flux		Reference
			Type	Heat No.	Type	Lot No.	
J726*	Nozzle Shell to Interm. Shell Circle Seam	Sub. Arc	SMIT 40	25017	SAF 89	1197	Docket No. 50-280[10]
SA1494	Interm. Shell Longi- tudinal Seams L3 & L4	Sub. Arc	Mn-Mo-Ni	8T1554	Linde 80	8579	"
SA1585	Interm. to Lower Shell Circle Seam (I.D. 40%)	Sub. Arc	Mn-Mo-Ni	72445	Linde 80	8597	"
SA1560	Interm. to Lower Shell Circle Seam (O.D. 60%)	Sub. Arc	Mn-Mo-Ni	72445	Linde 80	8632	"
SA1494	Lower Shell Longitudinal Seam L1	Sub. Arc	Mn-Mo-Ni	8T1554	Linde 80	8579	"
SA1526#	Lower Shell Longitudinal Seam L2	Sub. Arc	Mn-Mo-Ni	299L44	Linde 80	8596	"

* - Weld made by De Rotterdamsche Droogdok Mu (All other welds made by B&W)

\# - Denotes material included in Reactor Vessel Surveillance Program

Table 2-2. Chemical Composition of Reactor Vessel Beltline Region Weld Metals - Surry Unit-1

Weld Identification	Chemical Composition, Weight Percent									Reference
	C	Mn	P	S	Si	Cr	Ni	Mo	Cu	
J726	0.093	1.67	--	--	0.27	--	[0.10]	0.44	0.33	Docket No. 50-280(10)
SA1494	0.09	1.52	0.015	0.012	0.44	0.08	0.63	0.37	0.18	BAW-1799(5)
SA1585	0.08	1.45	0.016	0.016	0.51	0.09	0.59	0.38	0.21	"
SA1560	0.08	1.45	0.016	0.016	0.51	0.09	0.59	0.38	0.21	"
SA1526#	0.09	1.53	0.013	0.017	0.53	0.09	0.68	0.42	0.35	"

- Denotes material included in Reactor Vessel Surveillance Program

[ ] - Estimate based on review of similar material

Table 2-3. Mechanical Properties of Reactor Vessel Beltline Region Weld Metal - Surry Unit 1

	Toughness Properties				Tensile Properties					
					Strength, Ksi					
Weld Identification	T_{NDT}, F	10F Energy Ft-Lbs	RT_{NDT}, F	USE Ft-Lbs	yield	Ult.	Elong %	RA %	Post Weld Heat Treatment**	Reference
J726	---	54,77,51	0*	---	67.40	82.76	31.0	69.6	S-1130°F - 30 HR-FC	Docket No. 50-280[10]
SA1494	---	54,25,44	-6*	70*	---	81.00	---	---	S-1125 ± 25°F-48HR-FC	"
SA1585	---	50,54,51	-6*	70*	---	81.00	---	---	S-1125 ± 25°F-80HR-FC	"
SA1560	---	46,43,45	-6*	70*	65.00	81.00	30.5	---	S-1125 ± 25°F-48HR-FC	"
SA1526#	---	33,33,33	-6*	70*	---	88.00	---	---	S-1125° ± 25F-48HR-FC	"

* - Estimated per Section 2.4
** - S = Stress Relieved/Furnace Cooled (FC)
\# - Denotes material included in Reactor Vessel Surveillance Program

Table 2-4. Identification of Reactor Vessel Beltline Region Base Materials - Surry Unit-1

Material Identification Heat No.	Type	Component	Supplier	Heat Treatment*	Reference
122V109	SA508 CL.2	Nozzle Shell Forging	Bethlehem	A-1550°F-11HR/WQ T-1200°F-22HR/AC S-1125°F-40HR/FC	Docket No. 50-280 (10)
C4326-1#	SA533,Gr.B1	Inter. Shell Plate	Lukens	A-1675±25°F-9HR/WQ T-1210°F-9HR/AC S-1125±25°F-60HR/FC	"
C4326-2	SA533,Gr.B1	Inter. Shell Plate	Lukens	A-1675±25°F-9HR/WQ T-1210°F-9HR/AC S-1125±25°F-60HR/FC	"
C4415-1#	SA533,Gr.B1	Lower Shell Plate	Lukens	A-1675±25°F-9HR/WQ T-1200-1225°F-9HR/AC S-1125±25°F-60HR/FC	"
C4415-2	SA533,Gr.B1	Lower Shell Plate	Lukens	A-1675±25°F-9HR/WQ T-1200°F-9HR/AC S-1125±25°F-60HR/FC	"

- Denotes material included in Reactor Vessel Surveillance Program

* - A = Austentized/Water Quenched (WQ), Brine Quenched (BQ)
T = Tempered/Air Cooled (AC), Brine Quenched (BQ)
S = Stress Relieved/Furnace Cooled (FC)

Table 2-5. Chemical Composition of Reactor Vessel Beltline Region Base Materials - Surry Unit-1

Material Identification	Chemical Composition, Weight Percent C	Mn	P	S	Si	Ni	Cr	Mo	Co	V	Cu	Reference
122V109	0.22	0.70	0.010	0.011	0.24	0.74	0.36	0.60	0.010	0.01	[0.09]	Docket No. 50-280(10)
C4326-1#	0.23	1.35	0.008	0.015	0.23	0.55	0.069	0.55	0.014	0.001	0.11	"
C4326-2	0.23	1.35	0.008	0.015	0.23	0.55	0.069	0.55	0.014	0.001	0.11	"
C4415-1#	0.22	1.33	0.014	0.014	0.23	0.50	0.078	0.55	0.015	0.001	0.11	"
C4415-2	0.22	1.33	0.014	0.014	0.23	0.50	0.078	0.55	0.015	0.001	0.11	"

- Denotes material included in Reactor Vessel Surveillance Program

[ ] - Estimate based on review of similar material

Table 2-6. Mechanical Properties of Reactor Vessel Beltline Region Base Materials - Surry Unit 1

Material Identification	Toughness Properties			Tensile Properties, RT				Reference
				Strength, Ksi		Elong	RA	
	T_{NDT}, F	RT_{NDT}, F	USE, Ft-Lbs	Yield	Ult.	%	%	
122V109	40	40*	82.5*	76.5	96.5	23.25	65.45	Docket No. 50-280(10)
C4326-1#	10	10*	87.5*	67.5	88.1	26.95	67.70	"
C4326-2	0	0*	94*	67.6	88.1	26.95	67.70	"
C4415-1#	20	20*	82*	72.0	94.6	25.00	65.90	"
C4415-2	0	0*	86.5*	69.5	91.8	25.00	64.10	"

* - Estimated Per NRC Standard Review Plan Section 5.3.2.

\# - Denotes material included in Reactor Vessel Surveillance Program

Table 2-7. Properties of Surveillance Program Plate and Weld Material - Surry Unit-1

Material Identification	Toughness Properties T_{NDT}, F	RT_{NDT}, F	USE, Ft-lbs	Tensile Properties, RT Strength, Ksi Yield	Ult.	Elong %	RA %	Post Weld Heat Treatment**	Reference
C4326-1	10	10	115	68.00	90.47	25.75	70.85	A-1650-1700F-9HR/WQ T-1210F-9HR/AC S-1125F-15-1/2HR-FC	WCAP-7723(11)
C4415-1	20	20	103	71.80	93.77	24.45	69.80	A-1650-1700F-9HR/WQ T-1200F-9HR/AC S-1125F-15-1/2HR-FC	"
SA1526	--	-6*	70	69.67	83.20	26.50	66.70	S-1125F-15-1/2HR-FC	"

Material Identification	Chemical Composition, Weight Percent C	Mn	P	S	Si	Cr	Ni	Mo	Co	V	Cu	Reference
C4326-1	0.23	1.35	0.008	0.015	0.23	0.069	0.55	0.55	0.014	0.001	0.11	WCAP-7723(11)
C4415-1	0.22	1.33	0.014	0.014	0.23	0.078	0.50	0.55	0.015	0.001	0.11	"
SA1526	0.10	1.49	0.011	0.010	0.37	0.08	0.68	0.46	0.001	0.001	0.25	"

* - Estimated per Section 2.4
** - A = Austentized/Water Quenched (WQ)
T = Tempered/Air Cooled (AC)
S = Stress Relieved/Furnace Cooled (FC)

Table 2-8. Identification of Reactor Vessel Beltline Region Weld Metals - Surry Unit-2

Weld Identification	Weld Location	Welding Process	Weld Wire Type	Weld Wire Heat No.	Flux Type	Flux Lot No.	Reference
L737	Nozzle Shell to Interm. Shell Circle Seam	Sub. Arc	S4Mo	4275	SAF 89	02275	Docket No. 50-281[12]
SA1585	Interm. Shell Longitudinal Seams L3 & L4	Sub. Arc	Mn-Mo-Ni	72445	Linde 80	8579	"
R 3008*#	Interm. to Lower Shell Circle Seam	Sub. Arc	S3Mo	0227	Grau Lo	LW320	"
WF 4	Lower Shell Longitudinal Seam L1 (I.D. 63%) Seam L2 (100%)	Sub. Arc	Mn-Mo-Ni	8T1762	Linde 80	8597	"
WF 8	Lower Shell Longitudinal Seam L1 (O.D. 37%)	Sub. Arc	Mn-Mo-Ni	8T1762	Linde 80	8632	"

* - Weld made by De Rotterdamsche Droogdok Mu (All other welds made by B&W)

\# - Denotes material included in Reactor Vessel Surveillance Program

Table 2-9. Chemical Composition of Reactor Vessel Beltline Region Weld Metals - Surry Unit-2

Weld Identification	Chemical Composition, Weight Percent									Reference
	C	Mn	P	S	Si	Cr	Ni	Mo	Cu	
L737	0.084	1.74	--	--	0.35	--	[0.10]	0.38	[0.35]	Docket No. 50-281(12)
SA1585	0.08	1.45	0.016	0.016	0.51	0.09	0.59	0.38	0.21	BAW-1799(5)
R3008#	0.09	1.51	0.017	0.016	0.46	0.10	0.56	0.41	0.19	Docket No. 50-281(12)
WF 4	0.07	1.45	0.015	0.013	0.43	0.12	0.55	0.41	0.29	BAW-1799(5)
WF 8	0.06	1.45	0.010	0.009	0.53	0.12	0.55	0.41	0.29	"

- Denotes material included in Reactor Vessel Surveillance Program

[ ] - Estimate based on review of similar material

Table 2-10. Mechanical Properties of Reactor Vessel Beltline Region Weld Metal - Surry Unit 2

Material Identification	Toughness Properties: T_{NDT}, F	Toughness Properties: 10F Energy Ft-Lbs	Toughness Properties: RT_{NDT}, F	Toughness Properties: USE Ft-Lbs	Tensile Properties, RT: Strength, Ksi Yield	Tensile Properties, RT: Strength, Ksi Ult.	Tensile Properties, RT: Elong %	Tensile Properties, RT: RA %	Post Weld Heat Treatment**	Reference
L737	—	75,62,65.5	0*	—	66.69	82.62	29.0	67.6	S-1130°F - 24HR/FC	Docket No. 50-281 (12)
SA1585	—	50,54,51	-6*	70*	—	81.00	—	—	S-1125 ± 25°F-80HR/FC	"
R3008#	—	65.5,50.5,46	0*	—	79.00	89.90	26.0	—	S-1130°F - 25HR/FC	"
WF 4	—	40,31,34	-6*	70*	65.06	82.25	25.0	64.9	S-1125 ± 25°F-80HR/FC	"
WF 8	—	45,38,30	-6*	70*	71.00	85.50	25.0	—	S-1125 ± 25°F-48HR/FC	"

* - Estimated per Section 2.4
** - S = Stress Relieved/Furnace Cooled (FC)
\# - Denotes material included in Reactor Vessel Surveillance Program

Table 2-11. Identification of Reactor Vessel Beltline Region Base Materials - Surry Unit-2

Material Identification Heat No.	Type	Component	Supplier	Heat Treatment*	Reference
123V303	SA508,CL.2	Nozzle Shell Forging	Bethlehem	A-1550°F-12HR/WQ T-1200°F-22HR/AC S-1125°F-40HR/FC	Docket No. 50-281(12)
C4208-2	SA533,Gr.B1	Inter. Shell Plate	Lukens	A-1600-1650°F-9HR/BQ T-1200-1225°F-9HR/BQ S-1125°F-60HR/FC	"
C4339-1#	SA533,Gr.B1	Inter. Shell Plate	Lukens	A-1600-1650°F-9HR/BQ T-1200-1225°F-9HR/BQ S-1125°F-60HR/FC	"
C4331-1	SA533,Gr.B1	Lower Shell Plate	Lukens	A-1600-1650°F-9HR/BQ T-1200-1225°F-9HR/BQ S-1125°F-60HR/FC	"
C4339-2	SA533,Gr.B1	Lower Shell Plate	Lukens	A-1600-1650°F-9HR/BQ T-1200-1225°F-9HR/BQ S-1125°F-60HR/FC	"

- Denotes material included in Reactor Vessel Surveillance Program

* - A = Austentized/Water Quenched (WQ), Brine Quenched (BQ)
T = Tempered/Air Cooled (AC), Brine Quenched (BQ)
S = Stress Relieved/Furnace Cooled (FC)

Table 2-12. Chemical Composition of Reactor Vessel Beltline Region Base Materials - Surry Unit-2

Material Identification	Chemical Composition, Weight Percent											Reference
	C	Mn	P	S	Si	Ni	Cr	Mo	Co	V	Cu	
123V303	0.20	0.63	0.010	0.010	0.24	0.73	0.36	0.58	0.009	0.02	[0.09]	Docket No. 50-281(12)
C4208-2	0.21	1.28	0.008	0.013	0.24	0.55	--	0.55	0.020	--	0.15	"
C4339-1#	0.23	1.30	0.012	0.014	0.25	0.54	--	0.54	0.010	--	0.11	"
C4331-2	0.23	1.42	0.009	0.015	0.22	0.60	--	0.55	0.015	--	0.12	"
C4339-2	0.23	1.30	0.012	0.014	0.25	0.54	--	0.54	0.010	--	0.11	"

- Denotes material included in Reactor Vessel Surveillance Program

[ ] - Estimate based on review of similar material

Table 2-13. Mechanical Properties of Reactor Vessel Beltline Region Base Materials - Surry Unit 2

	Toughness Properties			Tensile Properties, RT				
				Strength, Ksi				
Material Identification	T_{NDT}, F	RT_{NDT}*, F	USE,* Ft-Lbs	Yield	Ult.	Elong %	RA %	Reference
123V303	30	3C	103	66.37	87.12	25.00	70.20	Docket No. 50-281(12)
C4208-2	-30	-30	93.5	64.00	85.87	26.55	67.35	"
C4339-1#	-10	30	79.0	67.00	90.25	26.95	64.70	"
C4331-2	-10	10	84.0	68.75	92.25	25.80	65.25	"
C4339-2	-20	10	82.5	68.50	92.12	26.60	66.10	"

* - Estimated from data in the major working direction per NRC Standard Review Plan Section 5.3.2.

\# - Denotes material included in Reactor Vessel Surveillance Program

Table 2-14. Properties of Surveillance Program Plate and Weld Material - Surry Unit-2

Material Identification	Toughness Properties			Tensile Properties, RT				Post Weld Heat Treatment*	Reference
	T_{NDT}, F	RT_{NDT}, F	USE, Ft-Lbs	Strength, Ksi Yield	Strength, Ksi Ult.	Elong %	RA %		
C4339-1	-10	11	104	68.15	91.3	26.35	69.55	A-1600-1650F-9HR/BQ T-1200-1225F-9HR/BQ S-1140F-15 1/4HR/FC	WCAP-8085 (13)
R3008	0	0	91	70.85	86.50	26.40	67.80	S-1140F-15 1/4HR/FC	"

Weld Identification	Chemical Composition, Weight Percent									Reference
	C	Mn	P	S	Si	Cr	Ni	Mo	Cu	
C4339-1	0.23	1.30	0.012	0.014	0.25	0.075	0.54	0.54	0.011	WCAP-8085 (13)
R3008	0.09	1.51	0.017	0.016	0.46	0.10	0.56	0.41	0.019	"

* - A = Austentized/Brine Quenched (BQ)
T = Tempered/Brine Quenched (BQ)
S = Stress Relieved/Furnace Cooled (FC)

Figure 2-1. Location and Identification of Materials Used in the Fabrication of the Belt-Line Region of Surry Unit-1 Reactor Pressure Vessel (Ref. 14)

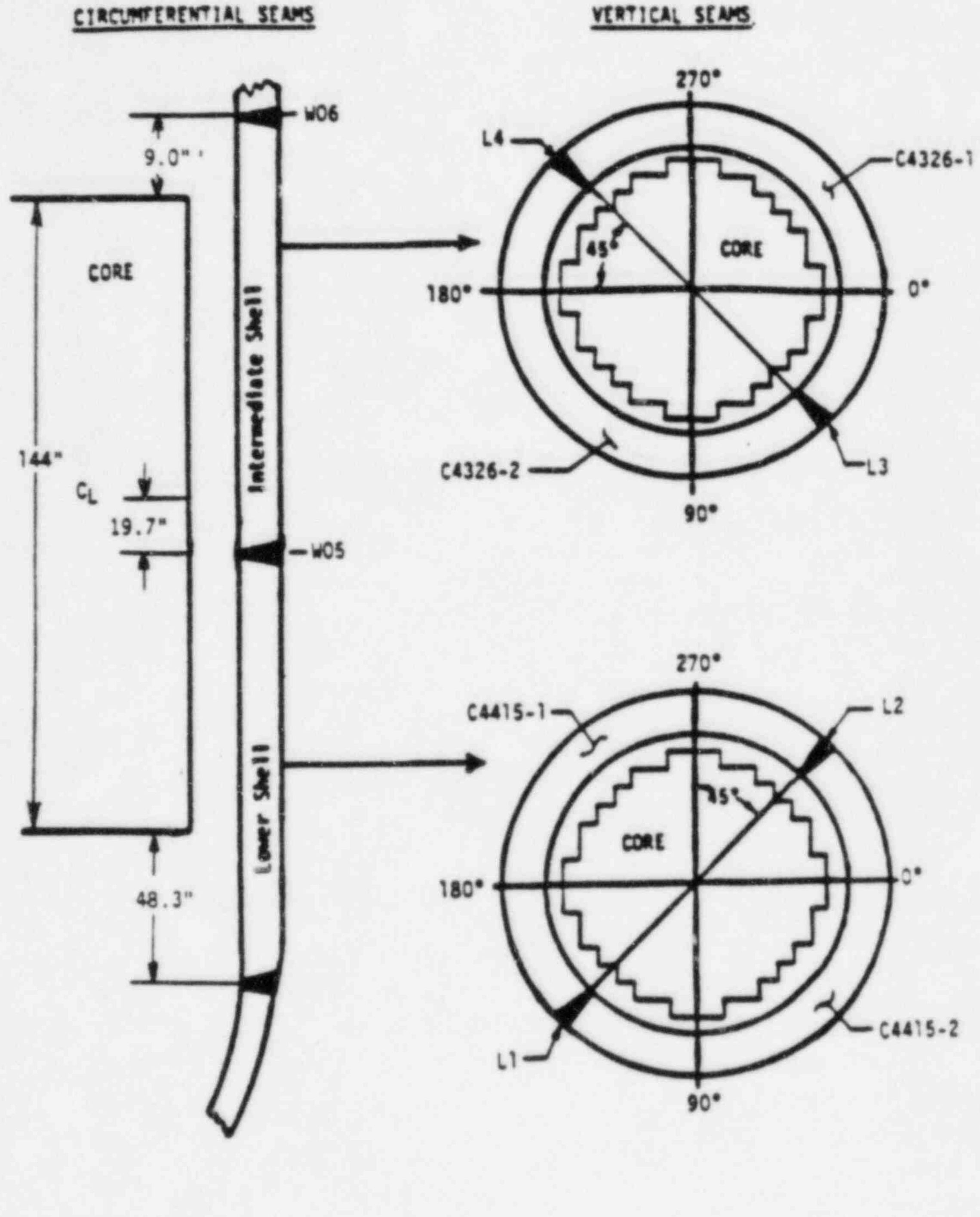
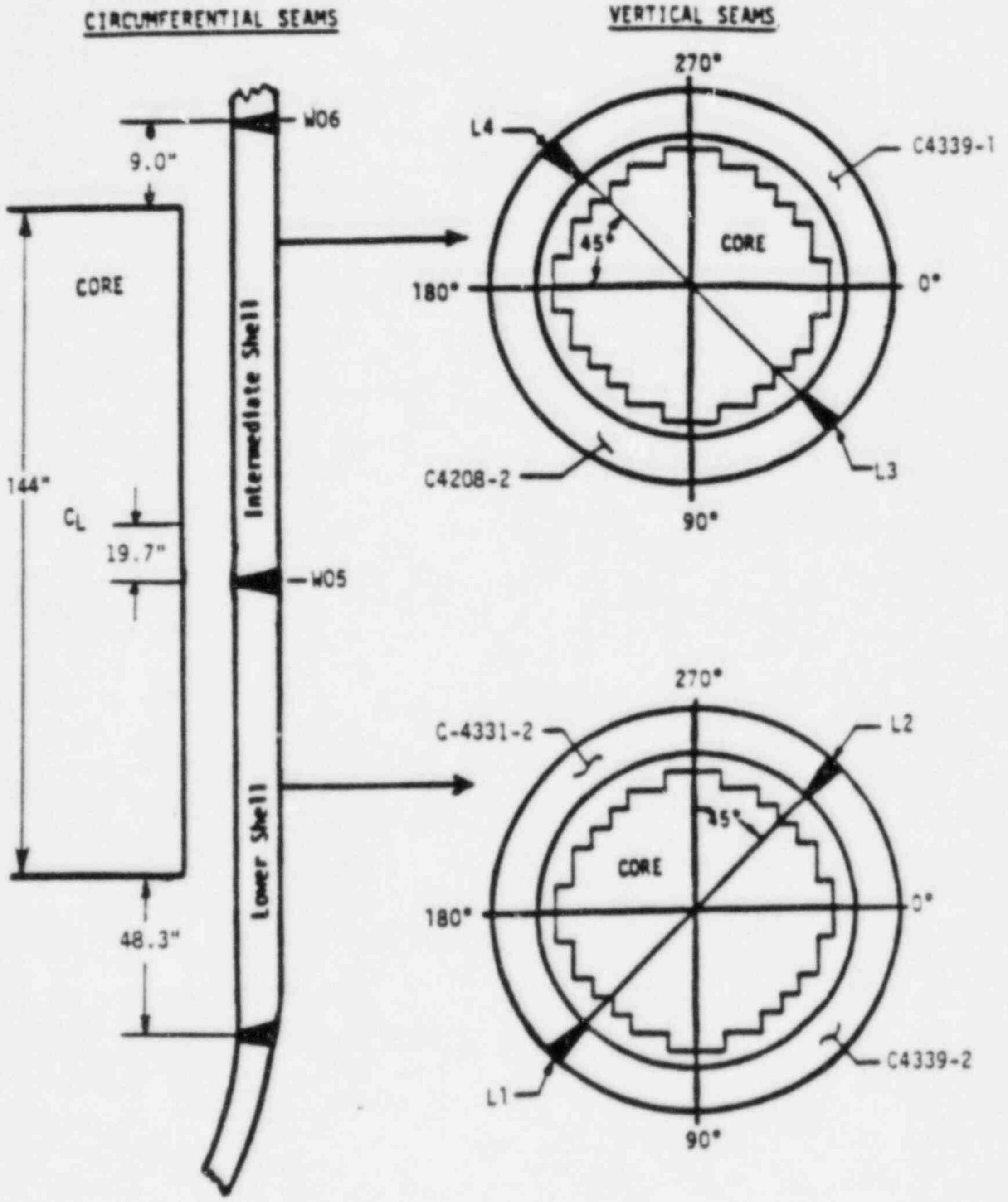


Figure 2-2. Location and Identification of Materials Used in the Fabrication of the Belt-Line Region of Surry Unit-2 Reactor Pressure Vessel (Ref. 14)



3. EVALUATION OF REACTOR VESSEL TOUGHNESS

One aspect of reactor pressure vessel licensibility is the toughness of the materials used in its fabrication. These properties are used to calculate the pressure-temperature operating limits in accordance with the requirements of 10CFR50, Appendix G. The objective of these limits is to prevent nonductile failure of the reactor vessel during any normal operating condition, including anticipated operational occurrences and system hydrostatic tests.

The closure head region, the reactor vessel outlet nozzle, and the beltline region have been identified as the only regions of the reactor vessel that regulate the pressure-temperature limits. Since the closure head region is significantly stressed at relatively low temperatures (due to mechanical loads resulting from bolt preload), this region largely controls the pressure-temperature limits of the first several service periods. The reactor vessel outlet nozzle also affects the pressure-temperature limit curves of the first several service periods. This is due to the high local stresses at the inside corner of the nozzle, which can be two to three times the membrane stresses of the shell. After the first several years of neutron radiation exposure, the RT_{NDT} of the beltline region materials will be high enough that the beltline region of the reactor vessel will start to control the pressure-temperature limits of the reactor coolant pressure boundary (RCPB). For the service period for which the limit curves are established, the maximum allowable pressure as a function of fluid temperature is obtained through the point-by-point comparison of the limits imposed by the closure head region, the outlet nozzle, and the beltline region. The maximum allowable pressure is taken to be the lowest of the three calculated pressures.

The unirradiated toughness properties of each reactor pressure vessel were determined for the belt-line region materials in accordance with 10CFR50, Appendix G. For the other belt-line region materials for which the measured properties are not available, the unirradiated impact properties and residual elements, as originally established for the belt-line region materials, were determined from acceptable data bases using recognized estimating techniques. The adjusted reference temperatures are calculated by adding the predicted radiation-induced shift of the RT_{NDT} to the unirradiated RT_{NDT} including margin. The predicted shift, RT_{NDT}, is calculated using the respective neutron fluence, plus the copper and nickel contents. The neutron fluence values are calculated based on functions derived from adjoint functions using transport calculations. The design curves of Regulatory Guide 1.99, Rev. 2,[15] were used to predict the radiation-induced shift of RT_{NDT}.

The results of the evaluation are presented in Tables 3-1 and 3-2 which show that both reactor pressure vessels have RT_{NDT} values which will permit normal operation to the expiration of current licenses.

Table 3-1. Evaluation of Reactor Pressure Vessel Fracture Toughness - Surry Unit-1

Material Identification		Beltline	Chemical Composition[a]		Initial		Fluence at Current License Expiration[b]		RT_{NDT} at Current License Expiration[c]	
Heat No.	Type	Region Location	Copper w/o	Nickel w/o	RT_{NDT}, F[a]	Margin, F[c]	Inside Surface n/cm^2	T/4 Location n/cm^2	Inside Surface n/cm^2	T/4 Location n/cm^2
122V109	SA508,Cl.2	Nozzle Shell Forging	0.09	0.74	40	34	5.52E18	3.06E18	122	116
C4326-1	SA533,Gr.B1	Interm. Shell Plate	0.11	0.55	10	34	3.54E19	1.97E19	142	130
C4326-2	SA533,Gr.B1	Interm. Shell Plate	0.11	0.55	0	34	3.54E19	1.97E19	132	120
C4415-1	SA533,Gr.B1	Lower Shell Plate	0.11	0.50	20	34	3.54E19	1.97E19	151	139
C4415-2	SA533,Gr.B1	Lower Shell Plate	0.11	0.50	0	34	3.54E19	1.97E19	131	119
J726	Subm. Arc	Noz. to Int. Cir. Weld	0.33	0.10	0	69	5.52E18	3.06E18	196	180
SA1585	Subm. Arc	Int. to low. Cir. Weld	0.21	0.59	−6	68	3.54E19	1.97E19	279	252
SA1494	Subm. Arc	Int. Longit. Weld	0.18	0.63	−6	68	5.71E18	3.17E18	196	180
SA1494	Subm. Arc	Lower Longit. Weld	0.18	0.63	−6	68	5.71E18	3.17E18	196	180
SA1526	Subm. Arc	Lower Longit. Weld	0.35	0.68	−6	68	5.71E18	3.17E18	251	225

NOTE: Weld metal SA1560 is not listed because it is the outside 60% of the intermediate to lower shell circumferential weld.

(a) Per Tables 2-1 through 2-6
(b) Per WCAP-11015[14]
(c) Per Regulatory Guide 1.99, Rev. 2[15]

Table 3-2. Evaluation of Reactor Pressure Vessel Fracture Toughness - Surry Unit-2

Material Identification		Beltline	Chemical Composition (a)		Initial		Fluence at Current License Expiration (b)		RT_{NDT} at Current License Expiration (c)	
Heat No.	Type	Region Location	Copper w/o	Nickel w/o	RT_{NDT}, F (a)	Margin, F (c)	Inside Surface n/cm^2	T/4 Location n/cm^2	Inside Surface n/cm^2	T/4 Location n/cm^2
123V303	SA508,Cl.2	Nozzle Shell Forging	0.09	0.73	30	34	4.74E18	2.64E18	120	114
C4208-1	SA533,Gr.Bl	Interm. Shell Plate	0.15	0.55	-10	34	3.04E19	1.69E19	142	125
C4339-1	SA533,Gr.Bl	Interm. Shell Plate	0.11	0.54	30	34	3.04E19	1.69E19	159	147
C4331-2	SA533,Gr.Bl	Lower Shell Plate	0.12	0.60	10	34	3.04E19	1.69E19	151	138
C4339-2	SA533,Gr.Bl	Lower Shell Plate	0.11	0.54	10	34	3.04E19	1.69E19	139	127
L737	Subm. Arc	Int. to Low. Cir. Weld	0.35	0.10	0	69	4.74E18	2.64E18	196	180
R3008	Subm. Arc	Noz. to Int. Cir. Weld	0.19	0.56	0	69	3.04E19	1,69E19	264	239
SA1585	Subm. Arc	Upper Longit. Weld	0.21	0.59	-6	68	6.27E18	3.48E18	204	187
WF 4	Subm. Arc	Lower Longit. Weld	0.29	0.55	-6	68	6.27E18	3.48E18	222	202

NOTE: Weld metal WF 8 is not listed because it is the outside 37% of the lower shell longitudinal weld.

(a) Per Tables 2-8 through 2-13
(b) WCAP-11015[14]
(c) Per Regulatory Guide 1.99, Rev. 2[15]

4. REACTOR VESSEL SURVEILLANCE PROGRAMS

The design of a reactor vessel materials surveillance program is based on the need to monitor the toughness properties of the controlling radiation sensitive material from which the reactor vessel was fabricated. Of equal importance is the benchmarking, or verification, of the fluence which the reactor vessel experiences.

The extent to which a surveillance program meets these objectives depends on when the reactor vessel was fabricated. This is due to the evolution of surveillance program requirements as more knowledge has been obtained from existing programs. Some of the requirements can be upgraded to meet the current 10CFR50, Appendix H while other reactor vessels will be required to make do with the installed programs. Each of the Surry surveillance programs will be described separately.

4.1. Surry Unit 1

The surveillance program was designed prior to the date when 10CFR50, Appendix H, established surveillance program requirements. The fact that the controlling materials were contained in the program is because the designers were knowledgeable as to the developing requirements. Even though the weld metal is not controlling by current standards, this fact is due to the azimuthal fluence distribution and not by the chemical composition of the weld metal which, if fluence values were equal, would be controlling. The surveillance weld metal was fabricated using the same weld wire as the controlling weldment; therefore, it will monitor the relative neutron radiation sensitivity of the controlling weldment.

The major deficiency of the program is the withdrawal schedule which is not designed to provide timely data as required by the current 10CFR50, Appendix H. A new capsule withdrawal schedule was developed around the current requirements as defined in ASTM E185-82 and the desire to move capsules from low lead factor sites to high lead factor sites only during scheduled ten year reactor vessel inspections.

Two methods are permitted to determine when a capsule is to be removed for evaluation; i.e., EFPY exposure or cumulative fluence. The cumulative fluence was used to establish the new schedule which will match the materials data obtained at each capsule evaluation to the critical times in the reactor vessel design life. The EFPY schedule would produce results with too low cumulative fluence to provide useful irradiation materials data. The new proposed withdrawal schedule for Surry Unit 1 is shown in Table 4-1.

4.2. Surry Unit 2

The surveillance program was designed prior to the date when 10CFR50, Appendix H established surveillance program requirements. This program is similar to the program for Surry Unit 1. The controlling materials are contained in the program. The program is updated by establishing a new withdrawal schedule based on accumulative fluence in accordance with ASTM E185-82. The new proposed withdrawal schedule for Surry Unit 2 is shown in Table 4-2.

4.3. Spare Capsules

Extra Surveillance Capsules not required to meet the current requirements of ASTM E185-82 will remain in the reactor vessel. These capsules can be used in the future to provide data for the verification of reactor vessel fluence calculations or to provide materials data for support of plant life extension. To ensure that the extra capsules will provide useful data in the future they will be moved to maximum lead factor positions during the appropriate inservice inspection as the positions become available, in order to maximize the total accumulated fluence.

Table 4-1. Revised Surveillance Capsule Withdrawal Schedule - Surry Unit-1

1 February 1986

Recommended Capsule Withdrawal Schedule Per 10CFR50, Appendix H, and E185-82			Revised Withdrawal Schedule*				
Capsule Sequence	Vessel EFPY	Capsule Fluence, n/cm^2	Capsule I.D.	EFPY	Estimated Capsule[14] fluence, n/cm^2	Estimated Maximum[14] RV fluence, n/cm^2	Estimated Withdrawal, Yr
First	1.5	5E18 or RT_{NDT}>50F	T**	1.1	2.89E18	1.70E18	--
Second	3	1.00E19	W**	3.4	4.31E18	5.49E18	--
Third	6	1.97E19 (EOL T/4)	V	8.0	1.94E19	1.21E19	1986
Fourth	15	3.54E19 (EOL I.W.)	X***	21.2	3.63E19	2.95E19	2004
Fifth	EOL	>3.54E19	Z***	25.6	4.57E19	3.54E19	2008

NOTE: Remainder of capsules moved to maximum lead factor position during inservice inspections as postions become available in order to maximize total fluence.

* - Reviewed after each fuel cycle and revised as needed after each capsule withdrawal and evaluation. Estimated withdrawals based on 18-month fuel cycles and 0.80 plant capacity.

** - Capsules withdrawn and evaluated.

*** - Transferred from 1.1 lead factor location to 1.8 lead factor location during the 20 year inservice inspection (December 1992/12.53 EFPY).

Table 4-2. Revised Surveillance Capsule Withdrawal Schedule - Surry Unit-2

1 February 1986

Recommended Capsule Withdrawal Schedule Per 10CFR50, Appendix H, and E185-82			Revised Withdrawal Schedule*				
Capsule Sequence	Vessel EFPY	Capsule Fluence, n/cm^2	Capsule I.D.	EFPY	Estimated Capsule(14) fluence, n/cm^2	Estimated Maximum(14) RV fluence, n/cm^2	Estimated Withdrawal, Yr
First	1.5	5E18 or RT_{NDT}>50F	X**	1.1	3.01E18	1.84E18	—
Second	3	9E18	W#	8.5	1.30E19	1.21E19	1986
Third	6	1.7E19 (EOL T/4)	V	8.5	2.02E19	1.21E19	1986
Fourth	15	3.0E19 (EOL I.W.)	Y***	19.7	3.19E19	2.94E19	2001
Fifth	EOL	>3.0E19	U***	25.8	4.38E19	3.04E19	2008

NOTE: Remainder of capsules moved to maximum lead factor position during inservice inspections as positions become available in order to maximize total fluence.

* - Reviewed after each fuel cycle and revised as needed after each capsule withdrawal and evaluation. Estimated withdrawals based on 18-month fuel cycles and 0.80 plant capacity.

** - Capsules withdrawn and evaluated.

*** - Transferred from 1.1 lead factor location to 1.8 lead factor location during the 20 year inservice inspection (April/1993/12.96 EFPY).

\# - Recommended withdrawal of Capsule "V" and leaving Capsule "W" in reactor since Capsule "V" will lead reactor vessel maximum wall fluence and Capsule "W" will only be equivalent to reactor vessel maximum wall fluence. Capsule "W" to be moved to a maximum lead factor position as it becomes available.

5. INTEGRATED REACTOR VESSEL SURVEILLANCE PROGRAM

The idea of an integrated reactor vessel surveillance program develops whenever two or more nuclear plants share the same site, or when owned by the same utility and share a common design. It is readily apparent that a savings can be recognized from reduced capsule evaluations and, of special importance, reduced worker radiation exposure. The requirements set forth in 10CFR50, Appendix H, as applicable to integrated surveillance programs are as follows:

> "C. An integrated surveillance program may be considered for a set of reactors that have similar design and operating features. The representative materials chosen for surveillance from each reactor in the set may be irradiated in one or more of the reactors, but there must be an adequate dosimetry program for each reactor. No reduction in the requirements for number of materials to be irradiated, specimen types, or number of specimens per reactor is permitted, but the amount of testing may be reduced if the initial results agree with predictions. Integrated surveillance program must be approved by the Director, Office of Nuclear Reactor Regulation, on a case-by-case basis. Criteria for approval include the following considerations:
>
> 1. The design and operating features of the reactor in the set must be sufficiently similar to permit accurate comparisons of the predicted amount of radiation damage as a function of total power output.
>
> 2. There must be adequate arrangement for data sharing between plants.
>
> 3. There must be a contingency plan to assure that the surveillance program for each reactor will not be jeopardized by operation at reduced power level or by an extended outage of another reactor from which data are expected.

4. There must be a substantial advantage to be gained, such as reduced power outages or reduced personnel exposure to radiation, as a result of not requiring surveillance capsules in all reactors in the set."

The four requirements as set forth above can be met by the Virginia Power plants. The plants are similar in design and power ratings, especially as to current knowledge as to flux rate effects on neutron radiation damage. Since the plants have the same owner no problem would exist as to data sharing. Each plant could continue to have a plant specific surveillance program as a backup in case the other plants experienced a protracted shutdown. Finally, there would be a gain from the reduced personnel exposed to radiation.

The fault of this approach is the wording in the first paragraph, "... must be an adequate dosimetry program for each reactor". In recent years, the monitoring of the neutron fluence the reactor vessel receives has developed to be of equal importance with monitoring material damage. In fact, because of the large uncertainties that can be assigned to fluence analysis, if not properly verified (i.e., dosimeters in removed surveillance capsules), when combined with materials property changes can produce restricted pressure-temperature operations which could more than offset savings to be realized from a surveillance capsule evaluation. On the other hand, if a capsule is removed to benchmark a fluence determination a major portion of the cost is associated with the dosimeter and fluence evaluation. Therefore, it would be more practical to perform the complete capsule evaluation.

In addition, the new revision of Regulatory Guide 1.99, gives credit for obtaining surveillance data for the controlling materials. When two or more credible data points become available from a reactor they may be used to determine the adjusted reference temperature and decrease in Charpy upper shelf energy. With the exception of Surry Unit 1, all the plants have the controlling materials in their surveillance programs. Thus, the data should provide the best evaluation of material damage to minimize the effect on operating limitations. In the case of Surry Unit 1, the weld metal in the surveillance program is similar to the controlling weld metal and may be used to provide a high degree of confidence that prediction techniques are not unduly restricting the operating limits.

In summary, while an integrated reactor vessel surveillance program for Surry Units 1 and 2 may be acceptable from a regulatory viewpoint, it would not be practical, since capsules would have to be withdrawn from each unit in order to provide a fluence benchmark. However, this option may become practical in the future and should be re-evaluated after additional capsules have been removed and evaluated.

6. SUMMARY

As a result of this review and update the reactor vessels materials data bases for Surry Units 1 and 2 were found to be in compliance with 10CFR50, Appendix G. The surveillance program materials properties data bases and capsule withdrawal schedules are in compliance with 10CFR50, Appendices G and H and will provide the material data necessary to ensure continued compliance with these appendices.

7. REFERENCES

1. U.S. Code of Federal Regulations, Title 10, Energy, Part 50, "Domestic Licensing of Production and Utilization Facilities, Appendix G, Facture Toughness Requirements."

2. U.S. Code of Federal Regulations, Title 10, Energy, Part 50, "Domestic Licensing of Production and Utilization Facilities, Appendix H, Reactor Vessel Material Surveillance Program Requirements."

3. ASTM Standard E185-82, "Practice for Conducting Surveillance Tests for Light Water-Cooled Nuclear Power Reactor Vessels," ASTM Standards 03.01, August 1985.

4. Surry Power Station, Units 1 and 2, Updated Final Safety Analysis Report, Virginia Electric and Power Company, July 16, 1982, as amended.

5. K. E. Moore and A. S. Heller, "B&W 177-FA Reactor Vessel Beltline Weld Chemistry Study," BAW-1799, Babcock & Wilcox, Lynchburg, Virginia, July 1983.

6. United States Nuclear Regulatory Commission, Standard Review Plan Branch Technical Position 5-2, Revision 1 NUREG-0800, July 1981.

7. H. S. Palme, et al., "Methods of Compliance With Fracture Toughness and Operational Requirements of 10CFR50, Appendix G," BAW-10046P, Babcock & Wilcox, Lynchburg, Virginia, March 1976.

8. A. S. Heller and A. L. Lowe, Jr., "Correlations for Predicting the Effects of Neutron Radiation on Linde 80 Submerged-Arc Welds," BAW-1803, Babcock & Wilcox, Lynchburg, Virginia, January 1984.

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10. Letter from C. M. Stallings, Virginia Electric and Power Company to E. G. Case, Office of Nuclear Reactor Regulation, Subject: Reactor Vessel Material Surveillance Program. Docket No. 50-280, January 23, 1978, Public Document Accession No. 780260154.

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13. S. E. Yanichko and D. J. Lege, "Virginia Electric and Power Co. Surry Unit No. 2 Reactor Vessel Radiation Surveillance Program," WCAP-8085, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, June 1973.

14. E. L. Furchi et al., "Surry Units 1 and 2 Reactor Vessel Fluence and RT_{PTS} Evaluations," WCAP-11015, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, December 1985.

15. "Effect of Residual Elements on Predicted Radiation Damage to Reactor Vessels," U. S. NRC Regulatory Guide 1.99, Revision 2, Draft dated August 14, 1985.