Attachment 1

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THE B&W OWNERS GROUP

MATERIALS COMMITTEE

Pressurized Thermal Shock Evaluations in Accordance with 10 CFR 50.61 for Babcock & Wilcox Owners Group Reactor Pressure Vessels

Babcock & Wilcox
a McDermott company

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PRESSURIZED THERMAL SHOCK EVALUATIONS IN ACCORDANCE WITH 10 CFR 50.61 FOR BABCOCK & WILCOX OWNERS GROUP REACTOR PRESSURE VESSELS

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Key Words: Pressurized Thermal Shock, Screening Criteria, Fracture Toughness, Reference Temperature (Pressurized Thermal Shock), Reference Temperature (Nil Ductility Temperature), Atypical Weld Metal, Linde 80 Weld Metal, Reactor Pressure Vessel

ABSTRACT

Pressurized thermal shock evaluations were performed in accordance with 10CFR50.61, Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," for the eight Babcock & Wilcox Owners Group 177 FA reactor pressure vessels. The projected values of RT_{PTS} for all the materials in the reactor vessel beltline region are below the screening criteria at the expiration date of the operating license of all the plants. The evaluation of the atypical weld metal showed the projected values of RT_{NDT} also are below the screening criteria at the operating license expiration date for the affected reactor vessels.

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1. INTRODUCTION

The Nuclear Regulatory Commission (NRC) amended its regulations for light water nuclear power plants, effective July 23, 1985,[1] to (1) establish a screening criterion related to the fracture resistance of pressurized water reactor (PWR) vessels during pressurized thermal shock (PTS) events; (2) require analyses and schedule for implementation of flux reduction programs that are reasonably practicable to avoid exceeding the screening criterion; and (3) require detailed safety evaluations to be performed before plant operation beyond the screening criterion will be considered. These amendments are intended to produce an improvement in the safety of PWR vessels by identifying those corrective actions that may be required to prevent or mitigate potential PTS events.

Transients and accidents can be postulated to occur in pressurized water reactors (PWRs) that result in severe overcooling (thermal shock) of the reactor vessel concurrent with high pressure. In these pressurized thermal shock (PTS) events, rapid cooling of the reactor vessel internal surface causes a temperature distribution across the reactor vessel wall. This temperature distribution produces a thermal stress on the reactor vessel with a maximum tensile stress at the inside surface of the vessel. The magnitude of the thermal stress varies with the rate of change of temperature and with time during the transient, and its effect is compounded by coincident pressure stresses.

Severe reactor system overcooling events with pressurization of the reactor vessel (PTS events) are postulated to result from a variety of causes. These include system transients, some of which are initiated by instrumentation and control system malfunctions (including stuck open valves in either the primary or secondary system), and postulated accidents such as small break loss-of-coolant accidents, main steam line breaks, and feedwater line breaks. As long as the fracture resistance of the reactor vessel material is relatively high, these events are not expected to cause vessel failure. However, the fracture resistance of the reactor vessel material decreases with the integrated exposure to fast neutrons during the life of a nuclear power plant. The rate of decrease is

dependent on the chemical composition of the vessel wall and weld materials. If the fracture resistance of the vessel is reduced sufficiently by neutron irradiation, severe PTS events could cause small flaws that might exist near the inner surface to propagate into the vessel wall. The assumed initial flaw might be enlarged into a crack through the vessel wall of sufficient extent to threaten vessel integrity and, therefore, core cooling capability.

The toughness state of reactor vessel materials can be characterized by a "reference temperature for nil ductility transition" (RT_{NDT}). At normal operating temperatures, vessel materials are quite tough and resistant to crack propagation. As the temperature decreases, the metal gradually loses toughness over a temperature range of about 100°F. RT_{NDT} is a measure of the temperature range at which this toughness transition occurs. Its value depends on the specific material in the vessel wall and the integrated neutron irradiation received by the vessel. These effects are determined by destructive tests of material specimens. Correlations, based on tests of irradiated specimens, have been developed to calculate the shift in RT_{NDT} as a function of neutron fluence for various material compositions. The value of RT_{NDT} at a given time in a vessel's life is used in fracture mechanics calculations to determine whether assumed pre-existing flaws would propagate when the vessel is subjected to overcooling events.

On the basis of studies of severe overcooling events that have occurred, generic calculations of postulated PTS events that could occur, and vessel integrity calculations, the NRC concluded that a value of RT_{NDT} can be selected so that the risk from PTS events for reactor vessels with smaller RT_{NDT} values is acceptable. (The risk of vessels with higher values of RT_{NDT} might also be shown to be acceptable but the demonstration would require detailed plant-specific evaluations and possibly modifications to existing equipment, systems, and procedures.) The NRC approach to selection of the RT_{NDT} screening criterion is described in detail in SECY-82-465.[2] In summary, the approach was to use a deterministic fracture mechanics algorithm to calculate the value of RT_{NDT} for which assumed pre-existing flaws in the reactor vessel would be predicted to initiate (grow deeper

into the vessel wall) assuming occurrence of one of the severe overcooling events that have been experienced. These "critical" values of RT_{NDT} were related to the expected frequency of the experienced severe overcooling events based on a limited data base, consisting of eight events in 350 reactor-years.

The designation RT_{PTS} (Reference temperature for pressurized thermal shock) is the nil ductility temperature of the material as defined by 10CRF50.61, Paragraph (b)(2) for use as a screening criterion. This designation is used to avoid confusion with the RT_{NDT} used to characterize the toughness state of reactor pressure vessel materials.

On the basis of these studies, the NRC concluded that the PWR reactor pressure vessels with conservatively calculated values of RT_{PTS} less than 270^{o}F for plate and forging material and axial welds, and less than 300^{o}F for circumferential welds present an acceptably low risk of vessel failure from PTS events.

The requirements of 10 CFR 50.61 further state the following:

> "For each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall submit projected values of RT_{PTS} (at the inner vessel surface) of reactor vessel beltline materials by giving values from the time of submittal to the expiration date of the operating license. The assessment must specify the bases for the projection, including the assumptions regarding core loading patterns. This assessment must be submitted by January 23, 1986 and must be updated whenever changes in core loadings, surveillance measurements, or other information indicate a significant change in projected values."

2. SUMMARY

Table 2-1 provides a summary of the pressurized thermal shock evaluations as required by 10 CFR 50.61 for each of the B&W Owners Group reactor pressure vessels.

A brief description of the status of each reactor pressure vessel is as follows:

2.1 Oconee Nuclear Station Unit 1 - (Table 4-1) Projected values of RT_{PTS} for all materials in the reactor vessel beltline region are below the screening criteria at the expiration date of operating license.

2.2 Oconee Nuclear Station Unit 2 - (Table 4-2) Projected values of RT_{PTS} for all materials in the reactor vessel beltline region are below the screening criteria at the expiration date of operating license.

2.3 Oconee Nuclear Station Unit 3 - (Table 4-3) Projected values of RT_{PTS} for all materials in the reactor vessel beltline region are below the screening criteria at the expiration date of operating license.

2.4 Three Mile Island Unit 1 - (Table 4-4) Projected values of RT_{PTS} for all materials in the reactor vessel beltline region meet the screening criteria at the expiration date of operating license.

2.5 Crystal River Unit 3 - (Table 4-5) Projected values of RT_{PTS} for all materials in the reactor vessel beltline region are below the screening criteria at the expiration date of operating license.

2.6 Arkansas Nuclear One, Unit 1 - (Table 4-6) Projected values of RT_{PTS} for all materials in the reactor vessel beltline region are below the screening criteria at the expiration date of operating license.

2.7 Rancho Seco Unit 1 - (Table 4-7) Projected values of RT_{PTS} for all materials in the reactor vessel beltline region are below the screening criteria at the expiration date of operating license.

2.8 Davis Besse Unit 1 - (Table 4-8) Projected values of RT_{PTS} for all materials in the reactor vessel beltline region are below the screening criteria at the expiration date of operating license.

2.9 The evaluation of the atypical weld metal for the three reactor pressure vessels required to be evaluated by the procedure established by the NRC showed that the projected values of RT_{NDT} are below the screening criteria at the expiration date of the operating licenses.

TABLE 2-1. SUMMARY OF PRESSURIZED THERMAL SHOCK EVALUATIONS FOR B&W OWNERS GROUP REACTOR PRESSURE VESSELS

Plant Name	Current License Expiration Date	RT_{PTS} at Current License Expiration	Calendar Year Criteria Exceeded#	Reference
Oconee Nuclear Station, Unit 1	November 6, 2007	231 vs. 270	2032	Table 4-1
Oconee Nuclear Station, Unit 2	November 6, 2007	292 vs. 300	2011	Table 4-2
Oconee Nuclear Station, Unit 3	November 6, 2007	220 vs. 300	2100	Table 4-3
Three Mile Island Unit 1	May 18, 2008	270 vs. 270	2008	Table 4-4
Crystal River Unit 3	September 25, 2008	267 vs. 300	2031	Table 4-5
Arkansas Nuclear One, Unit 1	December 6, 2008	251 vs. 300	2053	Table 4-6
Rancho Seco Unit 1	October 11, 2008	265 vs. 270	2012	Table 4-7
Davis Besse Unit 1	March 24, 2011	217 vs. 300	2126	Table 4-8

#Assumes 0.80 utilization factor (EFPY/Calender Year) and no change in future fuel cycles.

3. BASIS OF INPUT DATA

The pressurized thermal shock regulations require that the data used to perform the specified calculations must be traceable by including the source of all values included in the assessment. The relationship of the material on which any measurements are made to the actual material in the reactor pressure vessel must be described. For the fluence values, the assessment must specify the bases for all projections including the assumptions regarding core loading patterns such as standard vs. low-leakage cores.

The following describes the sources for all data used to evaluate the B&W Owners Group reactor pressure vessels.

3.1 Fluence Estimates

The integrated reactor vessel surveillance program as described in BAW-1543A, Revision 2[3] provides a general description of the interrelationship of the fluence values of the participating reactor pressure vessel. This relationship is the basis for cross referencing of the fluence analysis for reactor vessels without active surveillance dosimetry to those reactor vessels with dosimeters. A summary of this information is presented in Table 3-1.

The maximum, or peak, fluence values for each material at the various locations on the inside surface of each reactor vessel at January 1, 1986, license expiration, and 32 EFPY are listed in Tables 4-1 through 4-8. These values were obtained by first determining the maximum fluence on the inside surface of the reactor vessel and then multiplying by appropriate azimuthal and axial factors. The maximum fluence values with the exceptions noted below were determined as described in the surveillance capsule reports listed in Table 3-1.

The general analytical method uses fluence values obtained from transport theory calculations through the latest fuel cycle included in a DOT code analysis. These calculations are normalized to the most recent measured dosimeter results obtained from the corresponding surveillance capsules.

Fluence values are extended beyond this value by assuming that the flux above 1.0 MeV at the reactor vessel is proportional to the flux above 1.85 MeV in the baffle region at the edge of the core. This method is described in detail in BAW-1485.[4] Baffle flux values for the completed fuel cycles were used explicitly and the flux value for the last cycle was used to extrapolate future cycles.

Exceptions to this approach were made for Three Mile Island Unit 1 (TMI-1) and Rancho Seco. The last available DOT calculation for TMI-1 is for Cycle 1. The referenced report (BAW-1439) extrapolates on the basis of Cycle 1 only. In the analysis used in this evaluation, Cycles 2 through 5 were included using the ratio of PDQ baffle flux values to that in Cycle 1. Cycle 6 was assumed to be the first low-leakage core with a baffle flux ratio to Cycle 5 of 0.82. Cycle 7 was assumed to be a second low-leakage core with a baffle flux ratio to Cycle 5 of 0.72. These ratios are based on PDQ calculations for similar cores. Cycles 8 and beyond were assumed to be very-low-leakage cores. The ratio of flux values at specific locations for very-low-leakage to low-leakage cores were obtained from a recently completed vessel fluence reduction study.[5] This ratio is 0.665 at zero degrees from a major axis (azimuthal angle for critical weld) and was used to adjust TMI-1 Cycle 7 to obtain the fluence values for Cycles 8 and beyond.

The general method described above based on extrapolating fluence values using baffle flux ratios was used to obtain the fluence for Rancho Seco through Cycle 7. Cycles 8 and beyond were assumed to be very-low-leakage cores. The ratio of flux at 14 degrees from a major axis (azimuthal angle for most critical weld) for very-low-leakage to low-leakage cores was obtained from the same fluence reduction study used for TMI-1. A value of 0.661 was obtained at the 14 degree location and was used to adjust Rancho Seco Cycle 7 to obtain the fluence for Cycles 8 and beyond.

Spatial factors for specific weld locations within the reactor pressure vessel were obtained in the following way. Axial factors were obtained from BAW-1485 except those for Davis Besse (which was not included in BAW-1485). The values for Davis Besse were obtained by comparing weld

locations to those for Oconee Unit 1. Azimuthal factors were obtained from the most recent DOT calculation for each reactor vessel except for TMI-1, for which the azimuthal factors were obtained from the DOT calculation for Oconee Unit 1 Cycles 3 through 7. It is believed that these values are more representative of future fuel cycles since DOT calculations are not available for TMI-1 beyond Cycle 1.

3.2. Chemical Compositions

The bases of the chemical composition of the materials in the beltline region of the reactor vessels is BAW-1820[6] which is supplemented by the data and information in BAW-1799[7] (Nonproprietary version of BAW-1500P[8]).

3.3 Material Properties

The bases of the material properties which represent actual measured properties of the beltline region materials is BAW-1820. In the cases where the NRC regulations did not provide a generic initial value of RT_{NDT} for either SA-533 Grade B plate, or SA-508 Class 2 forging material, the statistical average value of these materials was calculated using the data base presented in BAW-10046P.[9] These values are as follows:

Plate Material, SA-533, Grade B = +1F
Forging Material, SA-508, Class 2 = +3F

TABLE 3-1. REACTOR PRESSURE VESSEL CORE LOADING SCHEMES AND BASES FOR FLUENCE ESTIMATES

JANUARY 1, 1986

Plant Name	Core Loading	Current Fuel Cycle	Conversion To LL Fuel Cycle	Bases of Fluence Estimates
Oconee Nuclear Station, Unit 1	Low-Leakage	9	6	Capsule OC1-A (BAW-1837)[10]
Oconee Nuclear Station, Unit 2	Low-Leakage	8	5	Capsule OC2-A (BAW-1699)[11]
Oconee Nuclear Station, Unit 3	Low-Leakage	9	6	Capsule OC3-B (BAW-1697)[12]
Three Mile Island Unit 1	Standard	5	6,8*	Capsule TMI1-E (BAW-1439)[13]
Crystal River Unit 3	Low-Leakage	6	4	Capsule CR3-C (BAW-1898)[14]
Arkansas Nuclear One, Unit 1	Low-Leakage	7	4	Capsule AN1-A (BAW-1836)[15]
Rancho Seco	Low-Leakage	7	4,8*	Capsule RS1-D (BAW-1792)[16]
Davis Besse	Low-Leakage	5	5	Capsule TE1-A (BAW-1882)[17]

*Assumed Conversion to Very Low-Leakage Fuel Cycle

4. REACTOR VESSEL SPECIFIC CALCULATIONS

4.1 Pressurized Thermal Shock

For the purpose of comparison with the PTS criterion, the value of RT_{PTS} for each of the reactor pressure vessel materials must be calculated as described in the following paragraphs. The calculation must be made for each weld, plate, and forging in the reactor vessel beltline. For each material, the RT_{PTS} is the lower of the results given by Equations 1 and 2. Equation 1 was applicable to all the materials in B&W Owners Group reactor pressure vessels.

Equation 1: $RT_{PTS}=I+M+[-10+470Cu+350CuNi]f^{0.270}$

Equation 2: $RT_{PTS}=I+M+283f^{0.194}$

a. "I" means the initial reference temperature of the unirradiated material measured as defined in the ASME B&PV Code Section III, Paragraph NB-2331. If a measured value is not available, the following generic mean value must be used: 0 ^{o}F for weld made with Linde 80 flux.

b. "M" means the margin to be added to cover uncertainties in the values of initial RT_{NDT}, copper and nickel content, fluence and the calculational procedures. In Equation 1, $M=48^{o}F$ if a measured value of I was used and $M=59^{o}F$ if the generic mean value of I was used.

c. "Cu" and "Ni" mean the best estimate weight percent of copper and nickel in the material.

d. "f" means the best estimate neutron fluence, in units of $10^{19}n/cm^2$ (E greater than or equal to 1 MeV), at the clad-base metal interface on the inside surface of the vessel at the location where the material in question receives the highest fluence for the period of service considered.

The results of the reactor pressure vessel specific PTS calculations using Equation 1 and the data sources described in Section 3 which meet the

requirements as described in 10CFR50.61 are included in Tables 4.1 through 4.8.

4.2 Atypical Weld Metal

The NRC letter of December 12, 1979 transmitting the staff review of BAW-10144A,[18] "Evaluation of the Atypical Weldment," stated in part: "... conclude that in calculating pressure-temperature operating limits for these vessels, the properties of atypical material should be considered." The effect of pressurized thermal shock on reactor pressure vessel integrity was not an issue at the time. The NRC letter further stated: "... the probability that atypical weld metal was used in fabricating the subject vessel is very low." However, any evaluation of reactor vessel integrity involving the mechanistic approach using fracture mechanics dictates that the properties of the atypical weld metal be evaluated for those locations which contain weld metal designated as WF-70.

The NRC evaluation of the atypical weld metal determined that the material was uniquely different from other materials used in the fabrication of reactor pressure vessels. The limited irradiation data then available indicated that the material would also exhibit a uniquely different response to neutron irradiation. Therefore, the NRC evaluation prescribed a procedure for evaluating the response of the atypical weld metal to neutron radiation damage. This procedure is based on Regulatory Guide 1.99, Revision 1,[19] however, the procedure is adopted by Regulatory Guide 1.99, Revision 2[20] for those cases where actual irradiation results are available for a given material. The available surveillance data for the atypical weld metal and the evaluation of change in RT_{NDT} resulting from neutron radiation are presented in Appendix A.

An evaluation of the atypical weld metal in the three reactor pressure vessels which have weld metal WF-70 at the inside surface in the beltline region (i.e. Crystal River 3, Three Mile Island Unit 1, Rancho Seco) was performed for the neutron exposure periods designated by 10CFR50.61 using the formulation specified for the atypical weld metal. The results of these evaluations are presented in Table 4-9.

TABLE 4-1. EVALUATION OF OCONEE UNIT 1 REACTOR PRESSURE VESSEL IN ACCORDANCE WITH PRESSURIZED THERMAL SHOCK CRITERION

1. Operating License Expiration Date: November 6, 2007
2. Calendar Years to Screening Criteria: 46
3. EFPY at Screening Criteria: >32 (Best Est. 58)
4. Fluence at Screening Criteria: 2.0E19 n/cm^2

Material Description			Chemical Composition, w/o		Constants for RT_{PTS} Calculations, F		Inside Surface Fluence, n/cm^2			PTS Screening Criteria	Calculated RT_{PTS}	
Reactor Vessel Beltline Region Location	Heat Number	Type	Copper	Nickel	Initial RT_{NDT}	Margin	1 Jan 1986	32 EFPY	License Expire		32 EFPY	License Expire
Lower Nozzle Belt	AHR 54	SA508, Cl2	0.18	0.65	(+3)*	59	5.7E17	1.9E18	1.5E18	270	126	123
Intermediate Shell	C2197-2	SA302B, Mod.	0.15	0.50	(+1)*	59	2.7E18	8.8E18	7.5E18	270	144	140
Upper Shell	C3278-1	"	0.12	0.60	(+1)*	59	3.6E18	1.2E19	9.6E18	270	136	131
Upper Shell	C3265-1	"	0.10	0.50	(+1)*	59	3.6E18	1.2E19	9.6E18	270	118	114
Lower Shell	C2800-1	"	0.11	0.63	(+1)*	59	3.6E18	1.2E19	9.6E18	270	130	126
Lower Shell	C2800-2	"	0.11	0.63	(+1)*	59	3.6E18	1.2E19	9.6E18	270	130	126
Interim. Circum. Weld (100%)	SA-1135	ASA/Linde 80	0.25	0.54	0	59	5.6E17	1.8E18	1.5E18	300	158	152
Upper Circum. Weld (I.D. 61%)	SA-1229	"	0.26	0.61	0	59	2.7E18	8.8E18	7.3E18	300	221	213
Upper Circum. Weld (O.D. 39%)	WF-25	"	0.35	0.68	0	59	N/A	N/A	N/A	N/A	N/A	N/A
Middle Circum. Weld (100%)	SA-1585	"	0.21	0.59	0	59	3.6E18	1.2E19	9.6E18	300	198	190
Lower Circum. Weld (100%)	WF-9	"	0.21	0.59	0	59	2.0E18	6.5E18	5.4E18	270	93	92
Interm. Longit. Weld (100%)	SA-1073	"	0.29	0.64	0	59	2.1E18	7.0E18	5.8E18	270	233	224
Upper Longit. Weld (100%)	SA-1493	"	0.29	0.55	0	59	2.8E18	9.0E18	7.5E18	270	236	228
Lower Longit. Weld (100%)	SA-1430	"	0.29	0.55	0	59	3.0E18	9.7E18	8.1E18	270	240	231
Lower Longit. Weld (100%)	SA-1426	"	0.29	0.55	0	59	3.0E18	9.7E18	8.1E18	270	240	231

*Values estimated per Paragraph 3.3

TABLE 4.2. EVALUATION OF OCONEE UNIT 2 REACTOR PRESSURE VESSEL IN ACCORDANCE WITH PRESSURIZED THERMAL SHOCK CRITERION

1. Operating License Expiration Date: November 6, 2007
2. Calendar Years to Screening Criteria: 25
3. EFPY at Screening Criteria: 29
4. Fluence at Screening Criteria: 1.1E19 n/cm^2

Material Description			Chemical Composition, w/o		RT_{PTS} Constants for Calculations, F		Inside Surface Fluence, n/cm^2			PTS Screening Criteria	Calculated RT_{PTS}	
Reactor Vessel Beltline Region Location	Heat Number	Type	Copper	Nickel	Initial RT_{NDT}	Margin	1 Jan 1986	32 EFPY	License Expire		32 EFPY	License Expire
Lower Nozzle Belt	AMX77	SA508, Cl2	0.08	0.76	(+3)*	59	2.4E18	9.1E18	7.1E18	270	96	94
Upper Shell	AAW163	"	0.04	0.75	+10	48	3.2E18	1.2E19	9.4E18	270	79	77
Lower Shell	AAW164	"	0.02	0.08	+20	48	3.2E18	1.2E19	9.4E18	270	73	73
Upper Circum. Weld (100%)	WF-154	ASA/Linde 80	0.31	0.59	0	59	2.4E18	9.1E18	7.1E18	300	254	241
Middle Circum. Weld (100%)	WF-25	"	0.35	0.67	0	59	3.2E18	1.2E19	9.4E18	300	308	292
Lower Circum. Weld (100%)	WF-112	"	0.31	0.59	0	59	1.8E18	6.7E18	5.3E16	300	111	108

*Values estimated per Paragraph 3.3

TABLE 4-3. EVALUATION OF OCONEE UNIT 3 REACTOR PRESSURE VESSEL IN ACCORDANCE WITH PRESSURIZED THERMAL SHOCK CRITERION

1. Operating License Expiration Date: November 6, 2007
2. Calendar Years to Screening Criteria: 114
3. EFPY at Screening Criteria: >32 (Best Est. 94)
4. Fluence at Screening Criteria: 4.7E19 n/cm^2

Material Description			Chemical Composition, w/o		Constants for RT_{PTS} Calculations, F		Inside Surface Fluence, n/cm^2			PTS	Calculated RT_{PTS}	
Reactor Vessel Beltline Region Location	Heat Number	Type	Copper	Nickel	Initial RT_{NDT}	Margin	1 Jan 1986	32 EFPY	License Expire	Screening Criteria	32 EFPY	License Expire
Lower Nozzle Belt	RDM 4680	SA508, Cl2	0.20	0.91	(+3)*	59	3.0E18	1.2E19	9.1E18	270	218	206
Upper Shell	AWS 192	"	0.01	0.73	40	48	3.9E18	1.6E19	1.2E19	270	90	89
Lower Shell	ANK 191	"	0.02	0.76	40	48	3.9E18	1.6E19	1.2E19	270	94	93
Upper Circum. Weld (100%)	WF-200	ASA/Line 80	0.24	0.63	0	59	3.0E18	1.2E19	9.1E18	300	223	211
Middle Circum. Weld (I.D. 75%)	WF-67	"	0.24	0.60	0	59	3.9E18	1.6E19	1.2E19	300	233	220
Middle Circum. Weld (O.D. 25%)	WF-70	"	0.35	0.59	0	59	N/A	N/A	N/A	N/A	N/A	N/A
Lower Circum. Weld (100%)	WF-169-1	"	0.18	0.63	0	59	2.2E16	8.7E16	6.7E16	300	91	89

*Values estimated per Paragraph 3.3

TABLE 4-4. EVALUATION OF THREE MILE ISLAND UNIT 1 REACTOR PRESSURE VESSEL IN ACCORDANCE WITH PRESSURIZED THERMAL SHOCK CRITERION

1. Operating License Expiration Date: May 18, 2008
2. Calendar Years to Screening Criteria: 23
3. EFPY at Screening Criteria: 21
4. Fluence at Screening Criteria: 7.0E18

Material Description			Chemical Composition, w/o		Constants for RT_{PTS} Calculations, F		Inside Surface Fluence, n/cm^2			PTS Screening Criteria	Calculated RT_{PTS}	
Reactor Vessel Beltline Region Location	Heat Number	Type	Copper	Nickel	Initial RT_{NDT}	Margin	1 Jan 1986	32 EFPY	License Expire		32 EFPY	License Expire
Nozzle Belt	ARY 059	SA508, Cl2	0.08	0.72	(+3)*	59	1.4E18	7.5E18	5.3E18	270	106	103
Upper Shell	C2789-1	SA302B, Mod	0.09	0.57	(+1)*	59	1.9E18	9.9E18	7.0E18	270	110	106
Upper Shell	C2789-2	"	0.08	0.57	(+1)*	59	1.9E18	9.9E18	7.0E18	270	110	106
Lower Shell	C3307-1	"	0.12	0.55	(+1)*	59	1.9E18	9.9E18	7.0E16	270	130	123
Lower Shell	C3251-1	"	0.11	0.50	(+1)*	59	1.9E18	9.9E18	7.0E18	270	121	116
Upper Circum. Weld (100%)	WF-70	ASA/Linde 80	0.35	0.59	0	59	1.4E18	7.5E18	5.3E18	300	269	250
Middle Circum. Weld (100%)	WF-25	"	0.35	0.68	0	59	1.9E18	9.9E18	7.0E18	300	296	275
Lower Circum. Weld (I.D. 50%)	WF-67	"	0.24	0.60	0	59	1.1E16	5.5E16	3.9E16	300	97	94
Lower Circum. Weld (O.D. 50%)	WF-70	"	0.35	0.59	0	59	N/A	N/A	N/A	N/A	N/A	N/A
Upper Longit. Weld (100%)	WF-8	"	0.29	0.55	0	59	1.9E18	9.9E18	7.0E16	270	241	225
Lower Longit. Weld (O.D. 60%)	SA-1494	"	0.19	0.63	0	59	N/A	N/A	N/A	N/A	N/A	N/A
Lower Longit. Weld (I.D. 37%/100%)	SA-1526	"	0.35	0.68	0	59	1.7E18	9.1E18	6.4E18	270	291	270

*Values estimated per Paragraph 3.3

TABLE 4-5. EVALUATION OF CRYSTAL RIVER UNIT 3 REACTOR PRESSURE VESSEL IN ACCORDANCE WITH PRESSURIZED THERMAL SHOCK CRITERION

1. Operating License Expiration Date: September 25, 2008
2. Calendar Years to Screening Criteria: 45
3. EFPY at Screening Criteria: >32 (Best Est. 42)
4. Fluence at Screening Criteria: 1.2E19 n/cm^2

Material Description			Chemical Composition, w/o		Constants for RT_{PTS} Calculations, F		Inside Surface Fluence, n/cm^2			PTS Screening Criteria	Calculated RT_{PTS}	
Reactor Vessel Beltline Region Location	Heat Number	Type	Copper	Nickel	Initial RT_{NDT}	Margin	1 Jan 1986	32 EFPY	License Expire		32 EFPY	License Expire
Lower Nozzle Belt	AZJ 94	SA508, Cl2	0.10	0.72	(+3)*	59	1.5E18	7.3E18	5.5E18	270	120	115
Upper Shell	C4344-1	SA533, GrB	0.20	0.54	+20	48	2.0E18	9.6E18	7.2E18	270	189	180
Upper Shell	C4344-2	"	0.20	0.54	+20	48	2.0E18	9.6E18	7.2E18	270	189	180
Lower Shell	C4347-1	"	0.12	0.58	-10	48	2.0E18	9.8E18	7.2E18	270	108	103
Lower Shell	C4347-2	"	0.12	0.58	+45	48	2.0E18	9.6E18	7.2E18	270	163	158
Upper Circum. Weld (I.D. 40%)	SA-1769	ASA/Linde 80	0.28	0.61	0	59	1.5E18	7.3E18	5.5E18	300	213	202
Upper Circum. Weld (O.D. 60%)	WF-169-1	"	0.18	0.63	0	59	N/A	N/A	N/A	N/A	N/A	N/A
Middle Circum. Weld (100%)	WF-70	"	0.35	0.59	0	59	2.0E19	9.8E18	7.2E18	300	284	267
Lower Circum. Weld (100%)	WF-154	"	0.31	0.59	0	59	1.1E18	5.4E16	4.0E16	300	108	105
Upper Longit. Weld (100%)	WF-8	"	0.29	0.55	0	59	1.8E18	8.9E18	6.6E18	270	236	223
Upper Longit. Weld (100%)	WF-18	"	0.29	0.55	0	59	1.8E18	8.9E18	6.6E18	270	236	223
Lower Longit. Weld (100%)	SA-1580	"	0.29	0.55	0	59	1.7E18	8.1E18	6.0E18	270	231	218

*Values estimated per Paragraph 3.3

TABLE 4-6. EVALUATION OF ARKANSAS NUCLEAR ONE-UNIT 1 REACTOR PRESSURE VESSEL IN ACCORDANCE WITH PRESSURIZED THERMAL SHOCK CRITERION

1. Operating License Expiration Date: December 6, 2008
2. Calendar Years to Screening Criteria: 67
3. EFPY at Screening Criteria: >32 (Best Est. 60)
4. Fluence at Screening Criteria: 2.0E19 n/cm^2

Material Description			Chemical Composition, w/o		Constants for RT_{PTS} Calculations, F		Inside Surface Fluence, n/cm^2			PTS Screening Criteria	Calculated RT_{PTS}	
Reactor Vessel Beltline Region Location	Heat Number	Type	Copper	Nickel	Initial RT_{NDT}	Margin	1 Jan 1986	32 EFPY	License Expire		32 EFPY	License Expire
Nozzle Belt	AYN-131	SA508, Cl2	0.03	0.70	(+3)*	59	2.1E18	8.4E18	6.5E18	270	73	73
Upper Shell	C5120-2	SA533,GrB	0.17	0.55	-10	48	2.8E18	1.1E19	8.8E18	270	144	137
Upper Shell	C5114-2	"	0.15	0.52	-10	48	2.8E18	1.1E19	8.6E18	270	128	123
Lower Shell	C5120-1	"	0.17	0.55	-10	48	2.8E18	1.1E19	8.6E18	270	144	137
Lower Shell	C5114-1	"	0.15	0.52	0	48	2.8E18	1.1E19	8.6E18	270	138	133
Upper Circum. Weld (100%)	WF-182-1	ASA/Linde 80	0.24	0.83	0	59	2.1E18	8.4E18	6.5E18	300	208	198
Middle Circum. Weld (100%)	WF-112	"	0.31	0.59	0	59	2.8E18	1.1E19	8.8E18	300	264	251
Lower Circum. Weld (100%)	SA-1788	"	0.25	0.54	0	59	1.8E16	6.2E16	4.8E16	300	99	98
Upper Longit. Weld (100%)	WF-18	"	0.29	0.55	0	59	1.9E18	7.6E18	5.9E18	270	229	217
Lower Longit. Weld (100%)	WF-18	"	0.29	0.55	0	59	2.0E18	8.0E18	6.3E18	270	231	220

*Values estimated per Paragraph 3.3

TABLE 4-7. EVALUATION OF RANCHO SECO REACTOR PRESSURE VESSEL IN ACCORDANCE WITH PRESSURIZED THERMAL SHOCK CRITERION

1. Operating License Expiration Date: October 11, 2008
2. Calendar Years to Screening Criteria: 26
3. EFPY at Screening Criteria: 27
4. Fluence at Screening Criteria: 7.9E18 n/cm^2

Material Description			Chemical Composition, w/o		Constants for RT_{PTS} Calculations, F		Inside Surface Fluence, n/cm^2			PTS Screening Criteria	Calculated RT_{PTS}	
Reactor Vessel Beltline Region Location	Heat Number	Type	Copper	Nickel	Initial RT_{NDT}	Margin	1 Jan 1986	32 EFPY	License Expire		32 EFPY	License Expire
Lower Nozzle Belt	FV4823	SA508, Cl2	0.15	0.68	10	48	2.1E18	6.9E18	5.4E18	270	145	140
Upper Shell	C5062-1	SA533,Gr B1	0.12	0.80	4	48	2.8E18	9.1E18	7.1E18	270	122	118
Upper Shell	C5062-2	"	0.12	0.80	-10	48	2.8E18	9.1E18	7.1E18	270	108	104
Lower Shell	C5070-1	"	0.10	0.58	-20	48	2.8E18	9.1E18	7.1E18	270	84	81
Lower Shell	C5070-2	"	0.10	0.58	-20	48	2.8E18	9.1E18	7.1E18	270	84	81
Upper Circum. Weld (100%)	WF-233	ASA/Linde 80	0.29	0.68	0	59	2.1E18	6.8E18	5.4E18	300	236	225
Middle Circum. Weld (100%)	WF-154	"	0.31	0.59	0	59	2.8E18	9.1E18	7.1E18	300	254	241
Lower Circum. Weld (100%)	WF-233	"	0.29	0.68	0	59	2.1E18	6.9E18	5.4E18	300	236	225
Upper Longit. Weld (100%)	WF-29	"	0.23	0.63	0	59	2.6E18	8.7E18	6.8E18	270	203	193
Lower Longit. Weld (100%)	WF-29	"	0.23	0.63	0	59	2.7E18	8.8E18	6.9E18	270	203	194
Lower Longit. Weld (I.D. 73%)	WF-70	"	0.35	0.59	0	59	2.7E18	8.8E18	6.9E18	270	278	265
Lower Longit. Weld (O.D. 27%)	WF-29	"	0.23	0.63	0	59	N/A	N/A	N/A	N/A	N/A	N/A

TABLE 4-8. EVALUATION OF DAVIS BESSE 1 REACTOR PRESSURE VESSEL IN ACCORDANCE WITH PRESSURIZED THERMAL SHOCK CRITERION

1. Operating License Expiration Date: March 24, 2011
2. Calendar Years to Screening Criteria: 140
3. EFPY at Screening Criteria: >32 (Best Est. 111)
4. Fluence at Screening Criteria: 5.8E19 n/cm^2

Material Description			Chemical Composition, w/o		RT_{PTS} Constants for Calculations, F		Inside Surface Fluence, n/cm^2			PTS Screening Criteria	Calculated RT_{PTS}	
Reactor Vessel Beltline Region Location	Heat Number	Type	Copper	Nickel	Initial RT_{NDT}	Margin	1 Jan 1986	32 EFPY	License Expire		32 EFPY	License Expire
Nozzle Belt	ADB-203	SA508, Cl2	0.04	0.68	+50	48	3.8E17	2.7E18	2.1E18	270	111	110
Upper Shell	AKJ 233	"	0.04	0.77	+20	48	2.4E18	1.7E19	1.3E19	270	91	89
Lower Shell	BCC 241	"	0.02	0.81	+50	48	2.4E18	1.7E19	1.3E19	270	104	104
Upper Circum. Weld (I.D. 9%)	WF-232	ASA/Linde 80	0.18	0.64	0	59	3.8E17	2.7E18	2.1E18	300	140	135
Upper Circum. Weld (O.D. 91%)	WF-233	"	0.29	0.68	0	59	N/A	N/A	N/A	N/A	N/A	N/A
Middle Circum. Weld (100%)	WF-182-1	"	0.24	0.63	+2	48	2.4E18	1.7E19	1.3E19	300	230	217
Lower Circum. Weld (I.D. 12%)	WF-232	"	0.18	0.64	0	59	1.3E18	9.5E18	7.3E18	300	92	90
Lower Circum. Weld (O.D. 88%)	WF-233	"	0.29	0.68	0	59	N/A	N/A	N/A	N/A	N/A	N/A

TABLE 4.9. EVALUATION OF REACTOR PRESSURE VESSELS WITH ATYPICAL WELD METAL

Plant	Reactor Vessel Beltline Region Location	Inside Surface Fluence, n/cm^2		Calculated RT_{NDT}*	
		32 EFPY	License Expire	32 EFPY	License Expire
Crystal River	Middle Circum. Weld (100%)	9.6E18	7.2E18	263	252
Three Mile Island Unit 1	Upper Circum. Weld (100%)	7.5E18	5.3E18	253	242
Rancho Seco	Lower Longit. Weld (I.D. 73%)	8.8E18	6.9E18	259	251

*

Intial Value RT_{NDT} = 90F
Radiation Induced Shift = Per Appendix A
Margin for Uncertainties = 48F

5. CERTIFICATION

This report is an accurate description of the pressurized thermal shock evaluations of the Babcock & Wilcox Owners Group reactor pressure vessels in accordance with 10 CFR 50.61.

A. L. Lowe Jr P.E. 10 Jan 1986

A. L. Lowe, Jr., P.E. — Date
Project Technical Manager

N. L. Snidow Jan 10, 1986

N. L. Snidow — Date
Performance Analysis

This report has been reviewed and is an accurate description of the pressurized thermal shock evaluations in accordance with 10 CFR 50.61.

L B Gross, PE Jan. 10, 1986

L. B. Gross, P.E. — Date
Chemistry, Materials &
Structural Analysis

G. F. Malan Jan. 10, 1986

G. F. Malan — Date
Performance Analysis

6. REFERENCES

1. U.S. Code of Federal Regulations, Title 10, Energy, Part 50, Section 61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," First Published - Federal Register, Vol. 50, No. 141, July 23, 1985.

2. U.S. Nuclear Regulatory Commission Staff Evaluation of Pressurized Thermal Shock, SECY-82-465, November 1982.

3. A. L. Lowe, Jr., et al., Integrated Reactor Vessel Material Surveillance Program, BAW-1543A, Rev. 2, Babcock & Wilcox, Lynchburg, Virginia, May 1985.

4. C. L. Whitmarsh, Pressure Vessel Fluence Analysis for 177-FA Reactors, BAW-1485, Babcock & Wilcox, Lynchburg, Virginia, June 1978.

5. J. R. Rodes, Vessel Fluence Reduction Fuel Cycle Study, BAW-1884, Babcock & Wilcox, Lynchburg, Virginia, December 1985.

6. J. D. Aadland, Babcock & Wilcox Owners' Group 177-Fuel Assembly Reactor Vessel and Surveillance Program Materials Information, BAW-1820, Babcock & Wilcox, Lynchburg, Virginia, December 1984.

7. 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.

8. K. E. Moore and A. S. Heller, Chemistry of 177-FA B&W Owners Group Reactor Vessel Beltline Welds, BAW-1500P, Babcock & Wilcox, Lynchburg, Virginia, September 1978.

9. H. S. Palme, H. W. Behnke, and W. J. Keyworth, Methods of Compliance With Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G, BAW-10046P, Rev. 1, Babcock & Wilcox, Lynchburg, Virginia, March 1976.

10. A. L. Lowe, Jr., et al., Analysis of Capsule OC1-A From Duke Power Company Oconee Unit 1 Reactor Vessel Materials Surveillance Program, Revision 1, BAW-1837, Babcock & Wilcox, Lynchburg, Virginia, August 1984.

11. A. L. Lowe, Jr., et al., Analysis of Capsule OCII-A From Duke Power Company Oconee Nuclear Station, Unit 2, Reactor Vessel Material Surveillance Program, BAW-1699, Babcock 7 Wilcox, Lynchburg, Virginia, December 1981.

12. A. L. Lowe, Jr., et al., Analysis of Capsule OCIII-A From Duke Power Company Oconee Nuclear Station, Unit 3, Reactor Vessel Materials Surveillance Program, BAW-1697, Babcock & Wilcox, Lynchburg, Virginia, October 1981.

13. A. L. Lowe, Jr., et al., Analysis of Capsule TMI-1E From Metropolitan Edison Company Three Mile Island Nuclear Station - Unit 1, Reactor Vessel Materials Surveillance Program, BAW-1439, Babcock & Wilcox, Lynchburg, Virginia, January 1977.

14. A. L. Lowe, Jr., et al., Analyses of Capsule CR3-C Florida Power Corporation Crystal River Unit 3, Reactor Vessel Materials Surveillance Program, BAW-1898, Babcock & Wilcox, Lynchburg, Virginia, January 1986.

15. A. L. Lowe, Jr., et al., Analysis of Capsule AN1-A From Arkansas Power & Light Company Arkansas Nuclear One - Unit 1, Reactor Vessel Materials Surveillance Program, BAW-1836, Babcock & Wilcox, Lynchburg, Virginia, July 1984.

16. A. L. Lowe, Jr., et al., Analyses of Capsule RS1-D Sacramento Municipal Utility District Rancho Seco Unit 1, Reactor Vessel Materials Surveillance Program, BAW-1792, Babcock & Wilcox, Lynchburg, Virginia, October 1983.

17. A. L. Lowe, Jr., et al., Analyses of Capsule TE1-A The Toledo Edison Company Davis-Besse Nuclear Power Station Unit 1, Reactor Vessel Materials Surveillance Program, BAW-1882, Babcock & Wilcox, Lynchburg, Virginia, September 1985.

18. K. E. Moore, et al., Evaluation of the Atypical Weldment, BAW-10144A, Babcock & Wilcox, Lynchburg, Virginia, February 1980.

19. U.S. Nuclear Regulatory Commission, Effect of Residual Elements on Predicted Radiation Damage to Reactor Vessels, Regulatory Guide 1.99, Revision 1, April 1977.

20. U.S. Nuclear Regulatory Commission, Radiation Damage to Reactor Vessel Material, Draft Regulatory Guide 1.99, Revision 2, August 14, 1985.

21. A. L. Lowe, Jr., et al., Analyses of Capsule CR3-B Florida Power Corporation Crystal River Unit 3, Reactor Vessel Materials Surveillance Program, BAW-1679, Rev. 1, Babcock & Wilcox, Lynchburg, Virginia, June 1982.

22. A. L. Lowe, Jr., et al., Analyses of Capsule CR3-D Florida Power Corporation Crystal River Unit 3, Reactor Vessel Materials Surveillance Program, BAW-1899, Babcock & Wilcox, Lynchburg, Virginia, To Be Published.

APPENDIX A

Evaluation Of Atypical Weld Metal

1. Introduction

During 1978, B&W initiated work with the B&W Owners Group on a program for evaluating the material properties of "early vintage" 177-fuel assembly reactor vessel welds. One of the work phases in this program had the objective of characterizing the chemistry of reactor vessel beltline welds. Extensive chemical analyses of the archive sources of reactor vessel welds have been performed as part of this work. Two samples of test weldments made for the Crystal River 3 reactor vessel surveillance program were part of the weld metal archives subjected to chemical analysis. The results of these analyses, performed by the Mt. Vernon Works Quality Assurance Laboratory, indicated that one of these samples had atypical concentrations of nickel and silicon, while the concentrations of the other elements were in the normal range for MnMoNi:Linde 80 submerged-arc weldments. The other sample had the nominal chemistry. The atypical weld was made with weld wire designated by Heat Number 72105. This heat of weld wire was used in the fabrication of 12 reactor vessels.

2. Regulatory Position

To resolve the atypical weld issue, B&W conducted an extensive investigation of records, metallographic examinations, chemical analyses, and fracture mechanics tests on both unirradiated and irradiated atypical weld material. The results of this study are presented in BAW-10144A.

Charpy V-notch tests were performed on both unirradiated and irradiated material. The irradiated specimens were irradiated in the Crystal River 3 reactor vessel. Dynamic and static fracture toughness tests were conducted on one inch thick compact tension specimens at room temperature. Although the dropweight NDT is -20^{o}F, the results of the Charpy tests show that 50 ft-lbs of energy is absorbed at 150^{o}F, therefore the unirradiated value of RT_{NDT} is 90^{o}F (150 minus 60). Using RT_{NDT} equal to 90^{o}F, the toughness properties obtained from the fracture mechanics tests, K_{Ic} (static) and K_{Id} (dynamic), were found to be conservative (i.e., lie above) to the K_{Ic} curve in ASME B&PV Code, Section XI and the K_{IR} curve in ASME B&PV Code, Section III respectively. Using an RT_{NDT} of -20^{o}F (the dropweight NDT), the fracture mechanics data fall within the scatter of the data of normal

material used to obtain the K_{Ic} and K_{IR} curves. This indicates that the RT_{NDT} value of 90^oF is conservative.

The effect of irradiation on the mechanical properties of atypical material have been evaluated, using the test results obtained from the Crystal River 3 surveillance specimens. These specimens were subjected to a fluence of 1.17×10^{18} n/cm^2. This fluence produced an increase in RT_{NDT} of 35^oF.

The NRC staff concluded that the probability that atypical weld metal was used in fabricating any of the vessels is very low. However, they also concluded that in calculating pressure-temperature operating limits for these vessels, the properties of atypical material should be considered. It was determined, "... that an initial value of RT_{NDT} of 90^oF was a very conservative value. The increase in RT_{NDT} due to irradiation should be based on the measured value of 35^oF at a fluence of 1.1×10^{18} n/cm^2 and the damage prediction slopes in Regulatory Guide 1.99."[18]

3. Evaluation of Atypical Weld Metal

Since the NRC staff evaluation of the atypical weld metal was completed, two events have occurred which affect any evaluation of the neutron radiation damage response of the atypical weld metal. The first is the revision of Regulatory Guide 1.99 to reflect a better understanding of radiation damage of reactor vessel materials. This warrants a change from the original approach of evaluating the atypical weld metal based on the slope of the damage prediction curves. The same basic technical relationship is defined in the revised Regulatory Guide but based on a statistical evaluation of available reactor surveillance capsule data. The second event is the testing and evaluation of two reactor vessel material surveillance capsules containing atypical weld metal. These data, added to the previously reported capsule data, provide a basis for evaluation of the atypical weld metal using the procedure described in Regulatory Guide 1.99, Revision 2.

The atypical weld metal data from the three available capsules from Crystal River Unit 3 reactor vessel surveillance program are presented in Table A-1.

Table A-1. Irradiated Atypical Weld Metal Data

Capsule	ΔRT_{NDT},F	Fluence, n/cm^2	Reference
CR3-B	28	1.17E18	BAW-1679[21]
CR3-C	122	6.56E18	BAW-1898[14]
CR3-D	119	7.50E18	BAW-1899[22]

The equation developed for the atypical weld metal using the data in Table A-1 and the procedure described in Regulatory Guide 1.99, Draft Revision 2, Section C.2 is as follows:

$$\Delta RT_{NDT}(\text{Surface}) = 125.8f^{(0.28-0.10\log f)}$$