PRESSURIZED THERMAL SHOCK EVALUATION
IN ACCORDANCE WITH 10CFR50.61
FOR THE REACTOR VESSELS IN
BYRON UNITS 1 & 2
AND
BRAIDWOOD UNITS 1 & 2

January 13, 1986

B&W Contract #583-7497
Task 001

Prepared by
Babcock & Wilcox Company
Nuclear Power Division
P. O. Box 10935
Lynchburg, Virginia 24506-0935

Babcock & Wilcox
a McDermott company

REPORT DOCUMENTATION

DOCUMENT IDENTIFIER: 77-1159832-00

TITLE: PRESSURIZED THERMAL SHOCK EVALUATION IN ACCORDANCE WITH 10CFR50.61 FOR THE REACTOR VESSELS IN BYRON UNITS 1 & 2 AND BRAIDWOOD UNITS 1 & 2

DATE: JANUARY 13, 1986

CUSTOMER: COMMONWEALTH EDISON

CUSTOMER ORDER NO.: P. O. #303654

B&W CONTRACT NO.: 583-7497, TASK 001

TABLE OF CONTENTS

List of Tables

Babcock & Wilcox
Nuclear Power Division
Lynchburg, Virginia

K. E. Moore, S. H. Campbell, and A. L. Lowe, Jr.

Pressurized Thermal Shock Evaluations in Accordance with 10CFR50.61 for the Reactor Vessels in Commonwealth Edison Company's Byron Units 1 & 2 and Braidwood Units 1 & 2

ABSTRACT

Pressurized thermal shock evaluations were performed in accordance with 10CFR50.61, Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," for the reactor vessel (RV) beltline region materials in the Commonwealth Edison Company's Byron Units 1 & 2 and Braidwood Units 1 & 2. The projected values of RT_{PTS} for all these materials are below the screening criteria for fast neutron fluences projected to 32 effective full power years (2.80×10^{19}; E > 1 MeV).

1. INTRODUCTION

The Nuclear Regulatory Commission's pressurized thermal shock (PTS) rules for pressurized water reactors (PWRs) are contained in 10CFR50.61[1]. This document requires that licensees submit projected values of reference temperature for each of the reactor vessel beltline materials. These values, (RT_{PTS}), as determined for the Commonwealth Edison Byron Units 1 and 2 and Braidwood Units 1 and 2, are presented in this report. It also contains PTS background information, a description of the reactor vessel beltline materials, the source of the materials, neutron fluence data, and a review of the calculational methods employed.

2. BACKGROUND

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 in 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 depends 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 the 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 10CFR50.61, 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°F for plate and forging material and axial welds, and less than 300°F for circumferential welds present an acceptably low risk of vessel failure from PTS events.

The requirements of 10CFR50.61 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 the 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."

3. INPUT DATA

The pressurized thermal shock regulations require that the data used to perform the specified calculations 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 vessel (RV) 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 R.V. beltline materials in the Byron and Braidwood units.

3.1. Materials Data

The R.V. beltline materials of all four of the Byron and Braidwood units met the requirements of Appendix G of 10CFR50[2]. This included the use of materials with prescribed levels of copper and fracture toughness properties. The chemical compositions and reference temperature data shown in Tables 1-4 were obtained from the Quality Assurance records available at The Babcock & Wilcox Company, the manufacturer of these vessels. Either SA 508 Cl 2 mod. or SA 508 Cl 3 forgings were used in the beltline of these plants. The test data were obtained from coupons of the actual forgings in accordance with Section III, Article NB-2000 of the 1971 Edition of the ASME Code and the following Addenda:

Byron I	All Addenda through Summer 1972
Byron II, Braidwood I, II	All Addenda through Summer 1973

The test data shown in these tables for the beltline welds were obtained from weld metal qualification test samples which also met the requirements of Section III, Article NB-2000 of the ASME Code. As can be seen, measured values of RT_{NDT} and copper and nickel concentrations were available for each of the beltline materials.

3.2. Neutron Fluence Estimates

The peak fast neutron flux at the inside surface of each reactor vessel is 2.77×10^{10} nvt as presented in the FSAR for each plant. The estimated peak neutron fluence is $2.77 \times 10^{10} \times 32$ effective full power years or 2.80×10^{19} n/cm^2 (E > 1 MeV). The value of 32 EFPY is based on assumed 40-year licensed operating period and 80% full power operation during this period. This fluence was applied to the upper and lower shell forgings and the circumferential weld joining these shells.

The projected peak neutron fluences at other R.V. beltline locations were based on the following:

- The relationship for relative axial variation of fast neutron flux and fluence within the pressure vessel wall (E > 1 MeV) for Zion Units 1 & 2[3].

- The relative positions of the beltline materials with respect to the core midplane. These positions are virtually identical for all four reactor vessels.

As shown in Tables 1-4, this value was applied to this weld and the nozzle belt forging.

Table 1. Evaluation of Byron 1 Reactor Pressure Vessel in Accordance with Pressurzized Thermal Shock Criteria

Material Description Reactor Vessel Beltline Region Location	Heat Number	Type	Chemical Composition, w/o		Constants for RT_{PTS} Calculations, F		Inside Surface Fluence, n/cm^2	PTS Screening Criteria, F	Calculated RT_{PTS}, F
			Copper	Nickel	Initial RT_{NDT}	Margin	32 EFPY		32 EFPY
Lower Nozzle Belt	123J218	SA 508 Cl 2 mod.	.05	.72	+20	48	6.30E18	270	91
Upper Shell	5P-5933	SA 508 Cl 2 mod.	.05	.73	+40	48	2.80E19	270	123
Lower Shell	5P-5951	SA 508 Cl 2 mod.	.04	.64	+10	48	2.80E19	270	81
Upper Circumferential Weld	WF501	ASA/Linde 80	.028	.63	+10	48	6.30E18	300	66
Middle Circumferential Weld	WF336	ASA/Linde 80	.031	.46	-30	48	2.80E19	300	31
Lower Circumferential Weld	WF472	ASA/Linde 80	.23	.57	+10	48	< E17	300	--

Table 2. Evaluation of Byron 2 Reactor Pressure Vessel in Accordance with Pressurzized Thermal Shock Criteria

Material Description Reactor Vessel Beltline Region Location	Heat Number	Type	Chemical Composition, w/o Copper	Chemical Composition, w/o Nickel	Constants for RT_{PTS} Calculations, F Initial RT_{NDT}	Constants for RT_{PTS} Calculations, F Margin	Inside Surface Fluence, n/cm^2 32 EFPY	PTS Screening Criteria, F	Calculated RT_{PTS}, F 32 EFPY
Lower Nozzle Belt	4P-6107	SA 508 Cl 2 mod.	.05	.74	+10	48	6.30E18	270	81
Upper Shell	49D329) 49C297)-1-1	SA 508 Cl 3	.01	.70	-20	48	2.80E19	270	30
Lower Shell	49D330) 49C298)-1-1	SA 508 Cl 3	.05	.73	-20	48	2.80E19	270	63
Upper Circumferential Weld	WF562	ASA/Linde 80	.03	.65	+40	48	6.30E18	300	98
Middle Circumferential Weld	WF447	ASA/Linde 80	.053	.62	+10	48	2.80E19	300	93
Lower Circumferential Weld	WF614	ASA/Linde 80	.18	.54	+40	48	< E17	300	--

Table 3. Evaluation of Braidwood 1 Reactor Pressure Vessel in Accordance with Pressurzized Thermal Shock Criteria

Material Description Reactor Vessel Beltline Region Location	Heat Number	Type	Chemical Composition, w/o		Constants for RT_{PTS} Calculations, F		Inside Surface Fluence, n/cm^2 32 EFPY	PTS Screening Criteria, F	Calculated RT_{PTS}, F 32 EFPY
			Copper	Nickel	Initial RT_{NDT}	Margin			
Lower Nozzle Belt	5P-7016	SA 508 Cl 2 mod.	.04	.71	+10	48	6.30E18	270	75
Upper Shell	49C344) 49D383)-1-1	SA 508 Cl 3	.05	.73	-30	48	2.80E19	270	53
Lower Shell	49C867) 49C813)-1-1	SA 508 Cl 3	.03	.73	-20	48	2.80E19	270	44
Upper Circumferential Weld	WF645	ASA/Linde 80	.033	.50	-30	48	6.30E18	300	28
Middle Circumferential Weld	WF562	ASA/Linde 80	.03	.65	+40	48	2.80E19	300	102
Lower Circumferential Weld	WF653	ASA/Linde 80	.19	.56	-40	48	< E17	300	--

Table 4. Evaluation of Braidwood 2 Reactor Pressure Vessel in Accordance with Pressurzized Thermal Shock Criteria

Material Description Reactor Vessel Beltline Region Location	Heat Number	Type	Chemical Composition, w/o Copper	Chemical Composition, w/o Nickel	Constants for RT_{PTS} Calculations, F Initial RT_{NDT}	Constants for RT_{PTS} Calculations, F Margin	Inside Surface Fluence, n/cm^2 32 EFPY	PTS Screening Criteria, F	Calculated RT_{PTS}, F 32 EFPY
Lower Nozzle Belt	5P-7056	SA 508 Cl 2 mod.	.04	.90	+30	48	6.30E18	270	97
Upper Shell	49D963) 49C904)-1-1	SA 508 Cl 3	.03	.71	-30	48	2.80E19	270	33
Lower Shell	50D102) 50C97)-1-1	SA 508 Cl 3	.06	.75	-30	48	2.80E19	270	63
Upper Circumferential Weld	WF645	ASA/Linde 80	.033	.50	-30	48	6.30E18	300	28
Middle Circumferential Weld	WF562	ASA/Linde 80	.03	.65	+40	48	2.80E19	300	102
Lower Circumferential Weld	WF696	ASA/Linde 80	.038	.60	-16	48	< E17	300	--

4. RT_{PTS} CALCULATIONS

For the purpose of comparison with the PTS criterion, the value of RT_{PTS} for each of the reactor vessel beltline 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 the beltline materials in the four reactor vessels in the Byron and Braidwood plants.

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°F for weld made with Linde 80 flux (as stated in Part 3, measured values were available for all materials).

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°F if a measured value of I was used and M=59°F if the generic mean value of I was used. (Since measured values were available, M=48°F was employed in these calculations.)

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 vessel specific PTS calculations using Equation 1 and the data sources described in Part 3 are included in Tables 1 through 4.

5. CONCLUSIONS

The Byron Units 1 and 2 and Braidwood Units 1 and 2 reactor vessel beltline materials met the requirements of Appendix G to 10CFR50. The projected 40-year RT_{PTS} values for these materials are well under the PTS screening criteria. All of the calculated RT_{PTS} values were $\leq$ 123 at the estimated peak neutron fluence of 2.80×10^{19} n/cm^2 (E > 1 MeV).

6. REFERENCES

1. 10CFR50.61, "Analysis of Potential Pressurized Thermal Shock Events," July 23, 1985.

2. Appendix G, 10CFR Part 50, "Fracture Toughness Requirements," March 1, 1973.

3. S.L. Anderson, Plant Specific Neutron Fluence Evaluation for Zion Units 1 and 2," WCAP-10902, August 1985.