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10 CFR 50.61
10 CFR 50, Appendix H

October 15, 1991

Document Control Desk
U. S. NUCLEAR REGULATORY COMMISSION
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Gentlemen:

DOCKETS 50-266 and 50-301
REACTOR VESSEL SURVEILLANCE CAPSULE TEST REPORT
AND RT_{PTS} SUBMITTAL
POINT BEACH NUCLEAR PLANT, UNIT 2
AND CHARPY UPPER-SHELF ENERGY STATUS,
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

Surveillance Capsule S was removed from Point Beach Nuclear Plant Unit 2 on October 24, 1990. The capsule testing results are submitted herewith in B&W Nuclear Service Company (BWNS) Report BAW-2140, dated August 1991, as required by 10 CFR 50 Appendix H. A summary of the capsule test results is provided in Attachment A. Three copies of this report are enclosed for your information.

The surveillance capsule test results support continued use of the current Point Beach Technical Specification operating pressure-temperature limit curves. The results indicate a temperature shift less than that predicted by Regulatory Guide 1.99 Revision 2, which was used to calculate the curves. Additionally, in calculating the heatup and cooldown curves, no credit was taken for the neutron flux reductions implemented in 1989. Therefore, the actual vessel fluence will be less than predicted through the expiration date of the current curves.

NRC rule 10 CFR 50.61 requires that licensees submit projected values of RT_{PTS} for reactor vessel beltline materials. Because Point Beach does not project the value for any material in the beltline region to exceed the PTS screening criteria, the assessment must be submitted with the next update of the pressure-temperature limits, or the next reactor vessel material surveillance report, or five years from the effective date of this rule, whichever comes first. Please note that this submittal also fulfills the RT_{PTS} evaluation requirement for Point Beach Unit 2 (See Table 7.6 of BAW-2140).

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Point Beach Unit 1's PTS submittal will be forthcoming once additional, accurate, fluence projection data is available from our cavity dosimetry program. Preliminary calculations indicate that the RT_{PTS} values for all materials in the Unit 1 beltline region will remain below the applicable screening criteria. The Unit 1 submittal will occur within the time requirements specified in 10 CFR 50.61.

On September 30, 1991, Mr. Barry Elliot of NRR's technical staff, Mr. Robert Samworth, NRC Project Manager for the Point Beach Nuclear Plant, and members of the Wisconsin Electric staff participated in a conference call regarding our projections of Charpy upper-shelf energy for the reactor vessel materials in both units. Mr. Elliot requested that we include a discussion of that issue with this letter. The Unit 2 upper-shelf energy projection is discussed in both BAW-2140 and Attachment A. That discussion is generally applicable to Unit 1 also. Based on current end-of-life fluence projections, best estimate material chemistries, surveillance data, and the prediction techniques described in BAW-1803, Revision 1, "Correlations for Predicting the Effects of Neutron Irradiation on Linde 80 Submerged-Arc Welds," we predict that the mean value for the controlling vessel weld metal upper-shelf energy will not decrease below 50 ft-lbs during the vessel design life for either unit. BAW-1803 was developed specifically to address the need for an estimating method for the Automatic Submerged-Arc Mn-Mo-Ni Wire/Linde 80 Flux class of weld metals. BAW-1803 Revision 1 was transmitted to Mr. Barry Elliot directly from BWNS on October 4, 1991. Attachment B presents a listing of our overall reactor vessel integrity program.

Please contact us if additional information is required.

Very truly yours,

J. J. Zach
Vice President
Nuclear Power

Attachments

Copy to: NRC Resident Inspector
NRC Regional Administrator

Attachment A

SUMMARY OF CAPSULE S
REACTOR VESSEL MATERIALS TESTING
POINT BEACH NUCLEAR PLANT, UNIT 2

BAW-2140: ANALYSIS OF CAPSULE S WISCONSIN ELECTRIC POWER COMPANY POINT BEACH NUCLEAR PLANT UNIT NO. 2

Unit 2 was shut down on October 6, 1990 for its sixteenth refueling shutdown. Capsule S was removed from the reactor vessel on October 24, 1990. It had resided in the reactor vessel for approximately 14.8 effective full power years of operation. This capsule represents approximately 119% of peak (inside surface) reactor vessel fluence for estimated lifetime radiation embrittlement considerations.

Capsule S received an average fast neutron fluence (E > 1.0 Mev) of 3.47×10^{19} n/cm^2. The predicted peak fast fluence for the reactor vessel T/4 location at the end of the sixteenth cycle (14.8 EFPY) is 1.06×10^{19} n/cm^2 (E > 1 Mev). Based on the calculated fast flux at the vessel wall, an 80% load factor, and the continued use of the current fuel management techniques, the projected peak fast fluence that the Unit 2 reactor pressure vessel inside surface will receive in 40 calendar years of operation is 2.92×10^{19} n/cm^2 (E > 1 Mev) and the corresponding T/4 fluence is calculated to be 1.93×10^{19} n/cm^2 (E > 1 Mev). (See Table Below)

Peak Fast Neutron Exposure for Point Beach Unit 2 (E > 1.0 MeV) n/cm^2			
	EOC 16 14.8 EFPY	Inside Surface 32.0 EFPY	T/4 Wall Location 32.0 EFPY
Circumferential Weld (SA-1484)	1.59E+19	2.56E+19	1.69E+19
Intermediate Shell Forging (123V500)	1.60E+19	2.92E+19	1.93E+19
Lower Shell Forging (122W195)	1.59E+19	2.66E+19	1.76E+19

The results of the tension tests indicated that the materials exhibited normal behavior relative to neutron fluence exposure. The behavior of the tensile properties as a function of neutron irradiation is an increase in both ultimate and yield strength and a decrease in ductility. The weld metal exhibited greater sensitivity to neutron irradiation than the base metal. However, the difference in tensile properties are insignificant relative to the conservative analysis of the reactor vessel materials at this time period in the reactor vessel service life.

The Charpy impact data results for the base metal forging materials, the weld metal, and the correlation material exhibited the characteristic shift to higher temperature for the 30 ft-lb transition temperature and a decrease in upper-shelf energy (USE). The 30 ft-lb transition temperature shift is in

relatively good agreement with the values predicted in Regulatory Guide 1.99, Rev. 2, and the predicted value is conservative when the margin is added. The maximum end-of-life RT_{NDT} is 280°F for the inside surface intermediate shell to lower shell circumferential weld (SA-1484). The most limiting end-of-life RT_{PTS} is 283 F, also for SA-1484. This RT_{PTS} value is below the screening criteria of 300°F listed in 10 CFR 50.61.

The decrease in Charpy USE with irradiation showed good agreement with predicted values for the base metals. However, the weld metal decrease in Charpy USE was less than predicted. This is probably due to the lack of data available for developing the estimating curves for material with similar copper contents as Point Beach reactor vessel surveillance materials. BWNS has developed a method to evaluate the radiation induced decrease in upper-shelf energy for the Automatic Submerged-Arc, Mn-Mo-Ni Wire/Linde 80 Flux. We believe this approach is more accurate than Regulatory Guide 1.99 Revision 2 for vessels similar to ours. BWNS approach is reported in BAW-1803, Revision 1. Based on surveillance capsule data and the prediction techniques presented in BAW-1803, it is predicted that the controlling vessel weld metal upper-shelf energy will not decrease below 50 ft-lbs during the vessel design life.

Attachment B

POINT BEACH NUCLEAR PLANT
REACTOR VESSEL INTEGRITY PROGRAM
1984 - PRESENT

	Project	Date Complete
1.	Neutron exposure evaluation of Point Beach reactor vessels.	December 1984
2.	Tested Unit 1 Surveillance Capsule T.	December 1984
3.	10 CFR 50.61 - Pressurized Thermal Shock (PTS) Submittal.	January 1986
	Correction to PTS submittal.	March 1986
	Safety evaluation report received from NRC.	July 1986
4.	Reactor Vessel Life Extension Study.	
	Initiated study in May 1986.	
	Evaluation of fuel management techniques and internals modifications (shielding) to meet flux reduction goals.	September 1987
	Identification of critical components in NSSS, including the reactor vessel, and compilation of transient data associated with these components.	October 1987
	Comprehensive scoping risk assessment to examine Point Beach specific concerns and the propriety of the flux reduction goals.	December 1987
	Developed bases and specifications for a plantwide on-line fatigue monitoring system.	December 1987
5.	Inservice Inspection	
a.	Second Unit 1 Reactor Vessel Ten-Year Exam:	
	Performed ASME Code exam utilizing SWRI standard data acquisition system, including 50/70 tandem near surface search units.	May 1987
	Performed exam using NES/Dynacon Ultrasonic Data Recording and Processing System (UDRPS) concurrent with ASME Code exam above.	May 1987
b.	Second Unit 2 Reactor Vessel Ten-year Exam:	
	SWRI Enhanced Data Acquisition System (EDAS) was utilized.	October 1989

6.	Joined Babcock and Wilcox Owner's Group (BWOG) Materials Committee.	August 1988
	Full participant in BWOG Reactor Vessel Integrity Program (RVIP).	August 1988
	Participant in BWOG Reactor Vessel Life Extension Surveillance Program (RVSP).	1989
	Developing master integrated reactor vessel surveillance program to include Westinghouse utilities with Linde 80 welds in their reactor vessels. (BAW-1543)	March 1989
	Submitted BAW-1543 Revision 3 to NRC.	October 1989
	Safety evaluation report received from NRC for BAW-1543.	June 1991
7.	Installation of excore neutron dosimetry (radiometric monitors and solid state track recorders) over one octant of each unit's reactor vessel. Analysis of sensor sets and correlation of cavity measurements with transport calculations will be performed after each fuel cycle for first three sets. Thereafter, a three year interval will be used until sufficient data is obtained to increase the interval.	
	Install mounting hardware and first set of dosimetry in Unit 2.	November 1988
	Install mounting hardware and first set of dosimetry in Unit 1.	May 1989
	First sensor set analyzed for Unit 2.	November 1990
	First sensor set analyzed for Unit 1.	December 1990
	Second sensor set analyzed for Unit 2.	October 1991
8.	Pilot project: On-line fatigue monitoring of Unit 2 pressurizer surge nozzle (related to reactor vessel life extension study fatigue evaluation).	November 1988
9.	Implement super Low Leakage Loading Pattern (L4P) cores and axially-zoned hafnium inserts in the guide tubes of peripheral assemblies.	
	Unit 1	May 1989
	Unit 2	November 1989

10.	Performed image enhancement of selected radiographs of important reactor coolant system components (reactor vessels, piping, steam generators, etc.) and retained radiograph image on media more permanent than original media.	1989
11.	Submit revised heatup and cooldown curves using the guidance of Regulatory Guide 1.99, Revision 2.	August 1989
	Technical Specification change approved by NRC.	January 1990
12.	Tested Unit 2 Surveillance Capsule S.	August 1991