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Document Control Desk
U.S. NUCLEAR REGULATORY COMMISSION
Mail Station P1-137
Washington, D.C. 20555

Gentlemen:

DOCKETS 50-266 AND 50-301
RESPONSE TO 10 CFR 50.61
FRACTURE TOUGHNESS REQUIREMENTS FOR PROTECTION AGAINST
PRESSURIZED THERMAL SHOCK (PTS) EVENTS
POINT BEACH NUCLEAR PLANT

10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," requires licensees to submit projected values of RT_{PTS} for reactor vessel beltline materials. Original RT_{PTS} submittals for our Point Beach Nuclear Plant, Units 1 and 2, were provided to the NRC on January 20, 1986, and March 14, 1986. Since these original submittals, we have instituted a super low leakage loading pattern (L4P) core design with part length hafnium absorbers in the guide tubes of the peripheral assemblies, instituted a cavity dosimetry monitoring program, joined the B&W Owners Group Reactor Vessel Working Group, and refined our beltline material properties.

10 CFR 50.61 was revised in 1991 to change the procedure for calculating the amount of radiation embrittlement that a reactor vessel receives. The revised rule requires each pressurized water reactor licensee to submit projected values of RT_{PTS} for each reactor vessel beltline material. This assessment must be submitted within 5 years of the effective date of the rule if no materials in the beltline region are projected to exceed the screening criteria before the expiration of the operating license. The rule further states that these submittals must be updated whenever there is a significant change in the projected values of RT_{PTS}. We provided RT_{PTS} values for Point Beach Nuclear Plant Unit 2 on October 15, 1991, with our Surveillance Capsule S Report. In our response to Generic Letter 92-01, Reactor Vessel Structural Integrity," dated June 25, 1992, we reported beltline material properties and fluence values which have been refined over the past several years. Due to these refinements, we have recalculated RT_{PTS}

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values for our Point Beach Nuclear Plant Units 1 and 2. Although the projected values of RT_{PTS} have not significantly changed, this letter is submitted to provide an updated summary of our RT_{PTS} calculations.

Accordingly, plant specific RT_{PTS} calculations have been performed as of August 1, 1992, and for the expiration date of the operating license for all materials in the beltline region of the Point Beach Nuclear Plant Units 1 and 2 reactor vessels. These calculations indicate that the RT_{PTS} values for all beltline materials in the reactor vessels will not exceed the screening criteria defined in 10 CFR 50.61(b)(2) through the expiration date of the current operating licenses.

The attachment to this letter provides the results of the RT_{PTS} calculations for Point Beach, as well as the bases for the fluence and material properties used in the RT_{PTS} calculations. The fluence projections and the material properties used in the calculations are the same as reported in our response to Generic Letter 92-01, "Reactor Vessel Structural Integrity," dated June 25, 1992.

We believe that, based on comparison of our calculated RT_{PTS} values in Tables 1 and 2 of the attachment to the PTS screening criteria contained in 10 CFR 50.61, Point Beach Nuclear Plant Units 1 and 2 are projected to conform to 10 CFR 50.61 for the duration of the current operating licenses.

If you have any questions or require additional information regarding this report, please contact us.

Sincerely,

Bob Link
Vice President
Nuclear Power

GLM/jg

Attachment

cc: NRC Regional Administrator, Region III
NRC Resident Inspector

ATTACHMENT I
RT_{PTS} CALCULATIONS AND BASES FOR POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

IDENTIFICATION AND LOCATION OF BELTLINE REGION MATERIALS

Figures 1 and 2 identify and indicate the location of all beltline region materials for the Point Beach Nuclear Plant Unit 1 and 2 reactor vessels, respectively. The beltline region is defined in 10 CFR 50 Appendix G to be the "region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."

Since our original PTS submittal dated January 20, 1986, we have decided to include two additional materials in each reactor vessel as part of the beltline region. These two materials are the nozzle belt (NB) forgings and the nozzle belt forging to intermediate shell welds. Our fluence monitoring program indicates that these materials will experience accumulated fluence greater than 1×10^{17} n/cm^2, which is the point where Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials" indicates the materials start to experience radiation damage.

PLANT SPECIFIC MATERIAL PROPERTIES

The pertinent chemical and material properties of the beltline region materials for Point Beach Unit 1 and 2 are provided in Tables 1 and 2. These material properties have been refined over the past several years based on our participation in the B&W Owners Group Reactor Vessel Working Group. All of these material properties have been previously reported in BAW-2166, "B&W Owners Group Response to Generic Letter 92-01," which was forwarded to the NRC by P&W Nuclear Service Company on June 17, 1992. Additionally, we referenced BAW-2166 in our docketed response to Generic Letter 92-01 dated June 25, 1992.

REACTOR VESSEL NEUTRON FLUENCE

In 1989, a reactor cavity measurement program was instituted at both Point Beach Units 1 and 2 to provide continuous monitoring of the neutron fluence of the beltline region of the reactor pressure vessel. When used in conjunction with dosimetry from previously withdrawn internal surveillance capsules and with the results of neutron transport calculations, the reactor vessel cavity neutron dosimetry provides neutron exposure data for the reactor pressure vessel and the embrittlement gradients through the vessel wall.

Additionally, in 1989, we implemented a super low leakage loading pattern (L4P) core design and introduced part length hafnium absorbers in the guide tubes of the peripheral assemblies. This flux reduction approach was designed to reduce the maximum neutron exposure on the limiting reactor vessel beltline materials. In our fluence projections for Point Beach Units 1 and 2, we assume this core design will be maintained through the expiration date of our operating license.

As of August 1, 1992, Point Beach Units 1 and 2 had been operated for a total of 16.4 and 16.3 Effective Full Power Years (EFPY), respectively. Assuming an 80% cumulative capacity factor, the reactor vessel neutron fluence can be projected for the end of the operating license. The fluence projections for August 1, 1992 and the end of our licensed life are provided in Tables 1 and 2.

RT_{PTS} VALUES FOR POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

RT_{PTS} calculations have been performed according to the requirements of 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," and the results are provided in Tables 1 and 2. It is concluded that the Point Beach reactor vessel beltline materials will not exceed the applicable screening criteria through the expiration of the current operating licenses.

TABLE 1: POINT BEACH UNIT 1 REACTOR VESSEL BELTLINE REGION MATERIAL PROPERTIES

Beltline Material	Initial RT_{NDT}	Copper Content	Nickel Content	Fluence 8/1/92	Fluence 32 EFPY	Margin	RT_{PTS} 8/1/92	RT_{PTS} 32 EFPY	Screening Criteria
NB Forging 122P237	+50 (3)	0.15 (3)	0.82 (3)	1.75E+18 (3)	2.95E+18 (3)	+34 (1)	146°F	161°F	270°F
Intermediate Shell (IS) A9811-1	+1 (3)	0.20 (3)	0.056 (3)	1.59E+19 (2)	2.68E+19 (2)	+48 (1)	148°F	160°F	270°F
Lower Shell (LS) C1423-1	+1 (3)	0.12 (3)	0.065 (3)	1.55E+19 (2)	2.33E+19 (2)	+48 (1)	111°F	117°F	270°F
NB to IS Weld SA-1426	0 (1)	0.20 (3)	0.55 (3)	1.75E+18 (3)	2.95E+18 (3)	+66 (1)	148°F	167°F	300°F
IS to LS Weld SA-1101	+10 (3)	0.26 (3)	0.60 (3)	1.55E+19 (2)	2.33E+19 (2)	+56 (1)	268°F	287°F	300°F
IS Longitudinal SA-812 (ID 27%)	0 (1)	0.17 (3)	0.52 (3)	9.87E+18 (2)	1.71E+19 (2)	+66 (1)	204°F	225°F	270°F
IS Longitudinal SA-775 (OD 73%)	0 (1)	0.19 (3)	0.63 (3)	--------	--------	------	------	------	--------
LS Longitudinal SA-847	0 (1)	0.25 (3)	0.54 (3)	9.71E+18 (2)	1.56E+19 (2)	+66 (1)	232°F	254°F	270°F

(1) 10 CFR 50.61, Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.

(2) WCAP-12794, Rev. 1, Reactor Cavity Neutron Measurement Program for Wisconsin Electric Power Company Point Beach Unit 1, March 1992.

(3) BAW-2166, B&W Owners Group Response to Generic Letter 92-01, June 1992.

TABLE 2: POINT BEACH UNIT 2 REACTOR VESSEL BELTLINE REGION MATERIAL PROPERTIES

Beltline Material	Initial RT_{NDT}	Copper Content	Nickel Content	Fluence 8/1/92	Fluence 32 EFPY	Margin	RT_{PTS} 8/1/92	RT_{PTS} 32 EFPY	Screening Criteria
NB Forging 123V352	+40 (3)	0.15 (3)	0.73 (3)	2.06E+18 (3)	3.50E+18 (3)	+34 (1)	139°F	154°F	270°F
Intermediate Shell (IS) 123V500	+40 (3)	0.09 (3)	0.70 (3)	1.72E+19 (2)	2.92E+ (2)	+34 (1)	141°F	149°F	270°F
Lower Shell (LS) 122W195	+40 (3)	0.05 (3)	0.72 (3)	1.69E+19 (2)	2.66E+19 (2)	+34 (1)	110°F	113°F	270°F
NB to IS Weld CE Weld	-56 (1)	0.27 (3)	0.90 (3)	2.06E+18 (3)	3.50E+18 (3)	+66 (1)	144°F	175°F	300°F
IS to LS Weld SA-1484	0 (1)	0.24 (3)	0.60 (3)	1.67E+19 (2)	2.56E+19 (2)	+66 (1)	264°F	283°F	300°F

(1) 10 CFR 50.61, Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.

(2) WCAP-12795, Rev. 1, Reactor Cavity Neutron Measurement Program for Wisconsin Electric Power Company Point Beach Unit 2, October 1991.

(3) BAW-2166, B&W Owners Group Response to Generic Letter 92-01, June 1992.

FIGURE 1

IDENTIFICATION AND LOCATION OF BELTLINE REGION MATERIAL
FOR THE POINT BEACH UNIT NO. 1 REACTOR VESSEL

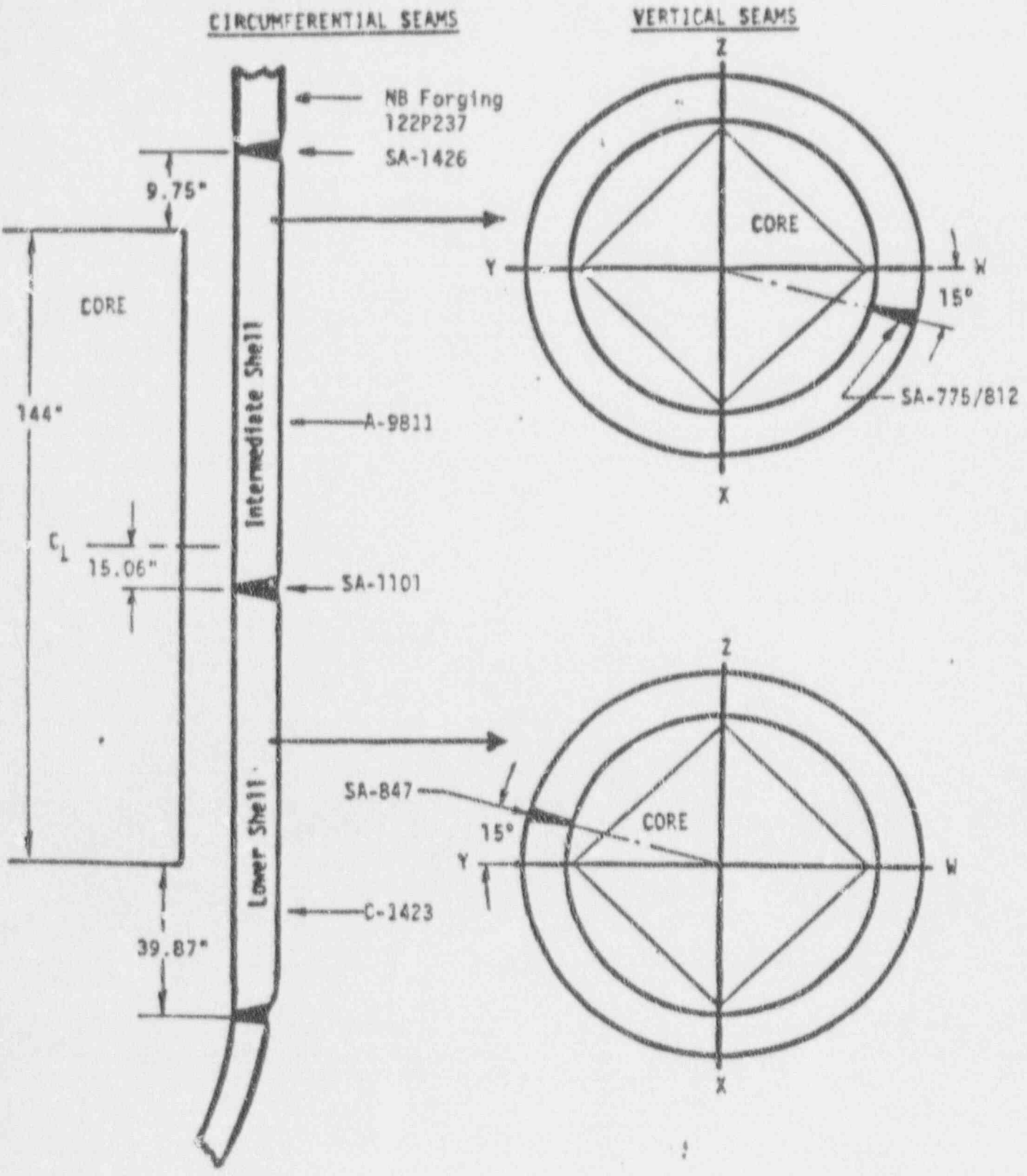

FIGURE 2

IDENTIFICATION AND LOCATION OF BELTLINE REGION MATERIAL
FOR THE POINT BEACH UNIT NO. 2 REACTOR VESSEL