TECHNICAL SPECIFICATIONS CHANGE REQUEST 171
SAFETY EVALUATION

INTRODUCTION

Wisconsin Electric Power Company (Licensee) is applying for amendments to Facility Operating Licenses DPR-24 and DPR-27 for Point Beach Nuclear Plant, Units 1 and 2. The proposed changes extend the operation of both units with the current heatup and cooldown limit curves in the Technical Specifications to 23.6 effective full power years (EFPY). The Basis section for Specification 15.3.1.B, "Pressure/Temperature Limits," is also being revised to reflect the methodology for curve compilation.

EVALUATION

Technical Specifications Figures 15.3.1-1, "Heatup Limitations," and 15.3.1-2, "Cooldown Limitations," are revised to reflect their applicability through 23.6 effective full power years (EFPY), or approximately January 1, 2001. Exposure of reactor vessel materials to neutron radiation throughout operating life results in a change in the reference temperature (RT_{NDT}) of the materials due to neutron embrittlement. This change in reference temperature is calculated periodically and appropriate limits revised. Because of flux reduction measures that we have implemented, the rate of neutron embrittlement of the Point Beach reactor vessels has been reduced significantly over what was projected when the heatup and cooldown curves were previously submitted on October 3, 1989. Therefore, the heatup and cooldown limitation curves being submitted remain unchanged from the curves presently contained in the Technical Specifications.

We are monitoring vessel fluence with a reactor cavity neutron measurement program. The results of this measurement program are recorded in Westinghouse Electric Corporation WCAP-12794, "Reactor Cavity Neutron Measurement Program for Wisconsin Electric Power Company, Point Beach Unit 1," Revision 2, and WCAP-12795, "Reactor Cavity Neutron Measurement Program for Wisconsin Electric Power Company, Point Beach Unit 2," Revision 2. Based upon these results, we have determined that the reactor vessel fluence will not reach the previously analyzed limit of 2.05×10^{19} n/cm^2 (E > 1 MeV) until 23.6 EFPY, or approximately January 2001.

The proposed heatup and cooldown curves are identical to the current heatup and cooldown curves except for their projected expiration dates. The curves were calculated using the most limiting weld and fluence information from either unit as input to the acceptable methodology of Regulatory Guide 1.99, Revision 2.

CONCLUSIONS

The proposed revisions will ensure the safe and reliable operation of Point Beach Nuclear Plant.

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NO SIGNIFICANT HAZARDS CONSIDERATION

We have evaluated these proposed amendments in accordance with the requirements of 10 CFR 50.91(a), against the standards of 10 CFR 50.92, and have determined that these modifications will not result in a significant hazards consideration. A proposed amendment will not involve a significant hazards consideration if it does not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

The proposed heatup and cooldown curves are identical to the current heatup and cooldown curves except for their projected expiration. The curves were calculated using the most limiting weld and fluence information from either unit as input to the acceptable methodology of Regulatory Guide 1.99, Revision 2. The consequences or probability of a previously evaluated accident will, therefore, not significantly be increased or a margin of safety reduced.

The underlying purpose of these curves is to define an acceptable operating range of pressures and temperatures to protect the reactor vessels against non-ductile failure. Since this purpose remains unchanged, a new or different kind of accident cannot be created.

stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

During cooldown the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stress at the outside wall.

The heatup and cooldown curves are composite curves which are prepared by determining the most conservative case with either the inside or outside wall controlling for any heatup or cooldown rate up to 100°F in any one hour.

In developing these curves, an initial unirradiated RT_{NDT} of -6°F was utilized as reported in BAW-1803 dated January 1984. (Reference 5) This value is based upon a statistical evaluation of Linde 80 weld material test data consisting of measured reference temperatures, drop weight data, and related pre-irradiated Charpy data. A standard deviation (σ_I) of 19°F was also calculated for this data set. Both the initial RT_{NDT} and standard deviation values in BAW-1803 may be revised as additional data are obtained.

As a result of fast neutron irradiation, there will be an increase in the RT_{NDT} with nuclear operation. The maximum integrated fast neutron exposure of the vessel is computed to be ~~3.5×10^{19}~~ 2.5×10^{19} neutrons/cm^2 for 40 years of operation at

1518 MWt and 80 percent load factor.[2] This maximum fluence is the exposure expected at the inner reactor vessel wall. ~~, which will be reduced when flux reduction measures are implemented.~~ However, the neutron fluence used to predict the ΔRT_{NDT} shift is the one-quarter shell thickness neutron exposure. The relationship between fluence at the vessel ID wall and the fluence at the one-quarter and three-quarter shell thickness locations is as presented in Regulatory Guide 1.99 Revision 2, "Radiation Damage to Reactor Vessel Materials." (Reference 6)

Once the fluence is determined, the adjusted reference temperature used in revising the heatup and cooldown curves is obtained by utilizing the method in Section 1.1 of Regulatory Guide 1.99 Revision 2 (Reference 6) for the limiting weld material of both Unit 1 and Unit 2.

The heatup and cooldown curves presented in Figure 15.3.1-1 and 15.3.1-2 were calculated based on the above information and the methods of ASME Code Section III (1974 Edition), Appendix G, "Protection Against Nonductile Failure", and are applicable up to the operational exposure indicated on the figures.

The regulations governing the pressure-temperature limits (10 CFR 50 - Appendix G and ASME Code Section III - Appendix G) do not require additional margins for instrumentation uncertainties be added to the heatup and cooldown curves. This is because the inclusion of instrumentation uncertainties, in addition to other conservatisms in the methods for calculating the pressure temperature limits, is not necessary to protect the vessel from damage.

The actual temperature shift of the vessel material will be established periodically during operation by removing and evaluating reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are identified by a specified lead factor, the measured temperature shift for a sample is an excellent indicator of the effects of power operation on the adjacent section of the reactor vessel. If the experimental temperature shift (at the 30 ft-lb level) does not substantiate the predicted shift, new prediction curves and heatup and cooldown curves must be developed.

The pressure-temperature limit lines shown on Figure 15.3.1-1 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 for reactor criticality and for inservice leak and hydrostatic testing.

The spray should not be used if the temperature difference between the pressurizer and spray fluid is greater than 320F°. This limit is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit.

The temperature requirements for the steam generator correspond with the measured NDT for the shell.

The reactor vessel materials surveillance capsule removal schedules have been developed based upon the requirements of the Code of Federal Regulations, Title 10, Part 50, Appendix H and with consideration of ASTM Standard E-185-82. When the capsule lead factors are considered, the scheduled removal dates accommodate the weld data needs of all the participants in the Babcock and Wilcox Master Integrated Reactor Vessel Surveillance Program. Additionally, the schedule will provide plate/forging material data as well as fluence data corresponding to the expiration of the current licenses and of any future license extension.

<u>References</u>

(1) FSAR, Section 4.1.5
(2) Westinghouse Electric Corporation, ~~WCAP-10638~~ WCAP-12794, Rev.2/12795, Rev.2
(3) Westinghouse Electric Corporation, WCAP-8743
(4) Westinghouse Electric Corporation, WCAP-8738
(5) Babcock & Wilcox, BAW 1803
(6) Regulatory Guide 1.99, Revision 2

Figure 15.3.1-1/PBNP Units 1 & 2
Heatup Limitations Applicable to
~~18.1~~ 23.6 Effective Full Power Years
(Approximately January ~~1995~~ 2001)

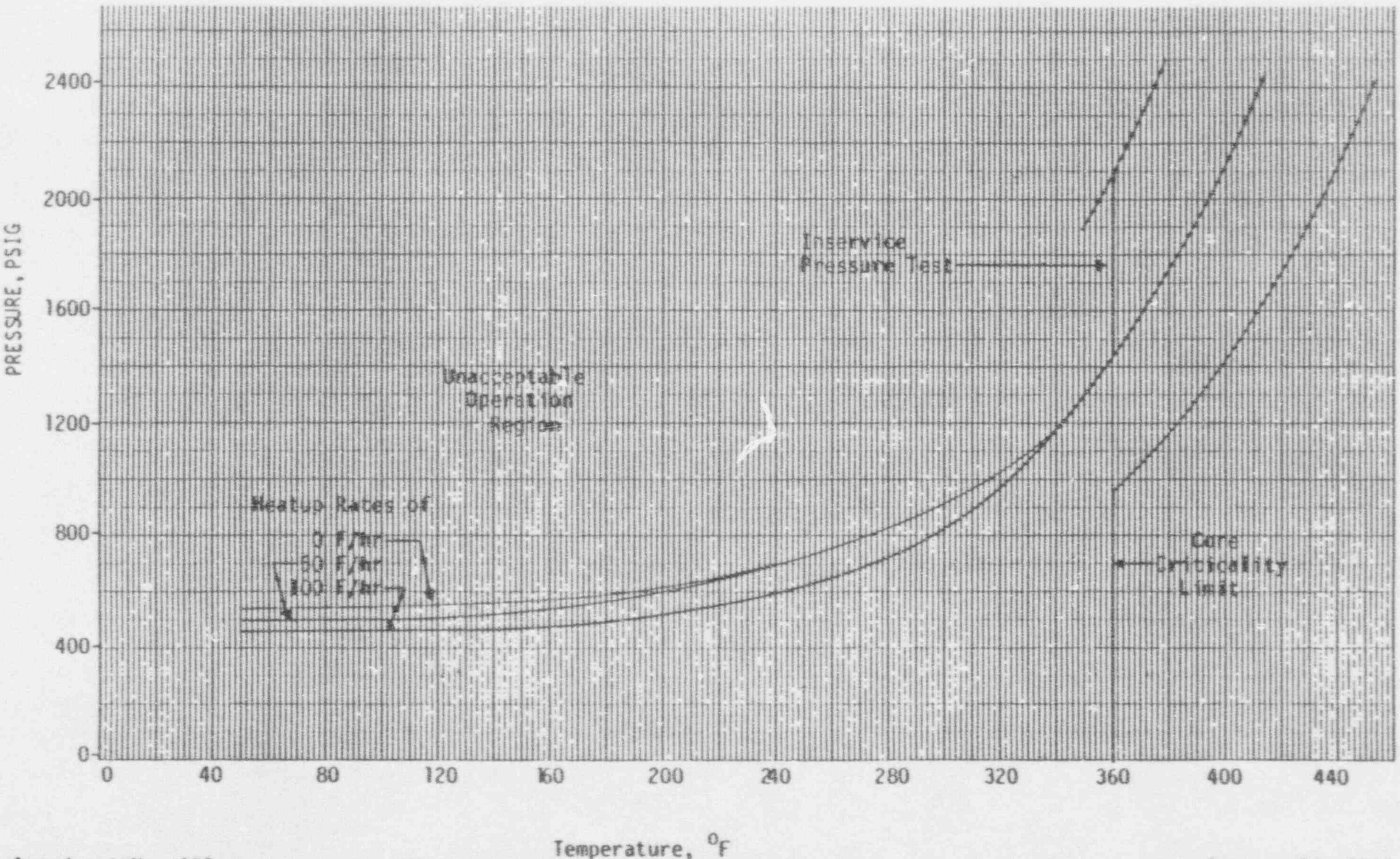

Figure 15.3.1-2/PBNP Units 1 & 2
Cooldown Limitations Applicable to
~~18.1~~ 23.6 Effective Full Power Years
(Approximately January ~~1995~~ 2001)