NMC
Committed to Nuclear Excellence

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Point Beach Nuclear Plant
6610 Nuclear Road
Two Rivers, WI 54241

NPL 2000-0472 10 CFR 50.36

October 24, 2000

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

Ladies/Gentlemen:

DOCKETS 50-266 AND 50-301
TECHNICAL SPECIFICATION BASES REVISIONS
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

Nuclear Management Company (NMC), licensee for the Point Beach Nuclear Plant (PBNP) Units 1 and 2, hereby submits a revision to the Bases for Technical Specifications (TS) 15.3.1.B, "Reactor Coolant System Pressure/Temperature Limits" and 15.3.15, "Low Temperature Overpressure Protection System". The change to the TS 15.3.1.B bases added a reference to ASME Code Case N-641, which was approved by the NRC for Point Beach on October 6, 2000. A revised calculation using this code case demonstrated that the existing Pressure-Temperature curves and Low Temperature Overpressure (LTOP) setpoints contained within the Technical Specifications are valid to 23.6 effective full-power years (EFPY) for both units. The change to the TS 15.3.15 bases included this new EFPY value.

These changes have been screened for evaluation pursuant to the requirements of 10 CFR 50.59 in accordance with approved PBNP procedures and were determined to be acceptable.

Attached are mark-up and clean copies of the affected Technical Specification Bases pages indicating the necessary changes.

Please contact me if you have any questions or require any additional information.

Sincerely,

D. Cole
Manager, Site Assessment

JG/tat

Attachments

cc: NRC Regional Administrator
NRC Resident Inspector
NRC Project Manager
PSCW

A001

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Marked-up Technical Specifications Bases Pages

of the vessel is computed to be 2.5 x 10^{19} neutrons/cm^2 for 40 years of operation at 1518.5 MWt and 80 percent load factor.[2] This maximum fluence is the exposure expected at the inner reactor vessel wall. 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. In addition, an exemption from the requirements of 10 CFR 50.60 and Appendix G to 10 CFR 50 has been approved by the NRC (Reference 7). This exemption allows the use of American Society of Mechanical Engineers (ASME), Section XI, Code Case N-641, for Point Beach Nuclear Plant (PBNP), Units 1 and 2.

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 320°F. 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.

References

(1) FSAR, Section 4.1.5
(2) Westinghouse Electric Corporation, WCAP-12794, Rev. 3/12795, Rev. 3
(3) Westinghouse Electric Corporation, WCAP-8743
(4) Westinghouse Electric Corporation, WCAP-8738
(5) Babcock & Wilcox, BAW 1803
(6) Regulatory Guide 1.99, Revision 2
(7) NRC Letter dated October 6, 2000, "Point Beach Nuclear Plant, Units 1 and 2 – Exemption from the Requirements of 10 CFR 50.60 (TAC Nos. MA9680 and MA9681)"

relief utilizes the pressurizer power operated relief valves (PORV's). The PORV's are made operational for low pressure relief by utilizing a dual setpoint where the low pressure circuit is energized and de-energized by the operator with a keylock switch depending on plant conditions. The logic required for the low pressure setpoint is in addition to the existing PORV actuation logic and will not interfere with existing automatic or manual actuation of the PORV's. The OPERABILITY of the PORVs is determined on the basis of their being capable of automatically mitigating an overpressure event during low temperature operation. The LTOP setpoint of 440 psig is valid ~~through an inside surface neutron fluence of the limiting reactor vessel material of less than or equal to 2.05×10^{19} n / cm^2 (E > 1.0 Mev).~~ for the operation of either Point Beach unit to 23.6 Effective Full Power Years.

During plant cooldown prior to reducing reactor coolant system temperature below 355°F, the operator under administrative procedures shall place the keylock switch in the "Low Pressure" position. This action enables the Low Temperature Overpressure Protection System. The redundant PORV channels shall remain enabled and operable while the LTOP system is required to be in operation.

The reactor coolant system is defined as vented if there is an opening in the reactor coolant system pressure boundary to atmosphere or the pressurizer relief tank that has an equivalent system pressure relieving capability as a PORV. Some examples of such openings include an open or removed PORV, open steam generator or pressurizer manways, a removed pressurizer safety valve, and the top of the reactor vessel when the reactor vessel head has been unbolted or removed.

The mass input transient used to determine the PORV setpoint assumes a worse case transient of a single high pressure safety injection pump discharging to the reactor coolant system while the system is solid. Therefore, when LTOP is required to be enabled, only one high pressure safety injection pump shall be operable at any time except when the reactor coolant system is open to the atmosphere.

The heat input transient used to determine the PORV setpoint assumes a temperature difference between the reactor coolant system and the steam generator of 50°F. Therefore, before starting a reactor coolant pump when the reactor coolant system is solid, the operator shall ensure that the secondary temperature of each steam generator is less than 50°F above the temperature of the reactor coolant system unless a pressure absorbing volume has been verified to exist in the pressurizer or steam generator tubes.

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Clean Technical Specifications Bases Pages

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