Wisconsin Electric POWER COMPANY

231 W. Michigan, P.O. Box 2046, Milwaukee, WI 53201 (414) 221-2345

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U. S. NUCLEAR REGULATORY COMMISSION
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Washington, D.C. 20555

Gentlemen:

DOCKETS 50-266 AND 50-301
TECHNICAL SPECIFICATION CHANGE REQUEST 127
INCREASED ALLOWABLE CORE POWER PEAKING FACTORS
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

In accordance with the requirements of 10 CFR 50.59(c), 50.90, and 50.4, Wisconsin Electric Power Company (Licensee) hereby requests amendments to Facility Operating Licenses DPR-24 and DPR-27 for Point Beach Nuclear Plant, Units 1 and 2, respectively, to incorporate changes in the plant Technical Specifications. The proposed changes, which are discussed in detail below, provide for the design and operation of the Point Beach fuel cycles with certain upgraded core features outlined below and at higher core power peaking factors (F_Q and $F_{\Delta H}$) than are currently permitted by the plant Technical Specifications. Attachment 1 provides a safety evaluation of these changes. The specific changes are identified with margin bars on the attached proposed Technical Specification pages (Attachment 2). Attachment 3 provides a significant hazards evaluation in accordance with 10 CFR 50.91(a) and 50.92 for each of the proposed Technical Specification changes. These evaluations demonstrate that no significant hazards consideration is involved in any of the proposed changes. After approval, the upgraded core features and technical specification changes will require revision of the Final Safety Analysis Report (FSAR) Chapter 14 accident analyses. This reanalysis program was originally described in our December 22, 1987 letter to the NRC and in our February 4, 1988 meeting with the NRC staff. All of the upgraded core features and Technical Specification changes have been input to the affected Chapter 14 accident analyses with satisfactory results. The revised accident analyses will be included in a formal FSAR update after both units have incorporated the higher peaking factors and upgraded core features.

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The majority of the changes in this request deal with increasing the core power peaking factors. These higher peaking factors will allow the use of a low-low leakage loading pattern (L4P) fuel management strategy, which will result in decreased neutron fluence to the reactor vessel. The higher peaking factors will also allow additional measures to be pursued (e.g., the use of peripheral power suppression assemblies), which will result in additional reduction of fluence to the reactor vessel walls. This fluence reduction will help address current reactor vessel irradiation damage issues dealing with pressurized thermal shock, low upper shelf material toughness, and heatup and cooldown pressure-temperature restrictions, and will maintain the option to extend the useful life of the reactor vessels at the Point Beach Nuclear Plant (PBNP) beyond the current 40-year operating license.

In addition to the increased allowable peaking factors, the analyses described in Attachment 1 also bound future operation of PBNP with any combination of the following fuel modifications incorporated into the current Westinghouse 14X14 Optimized Fuel Assembly (OFA) fuel design:

1. Removable Top Nozzles (RTN)
2. Integral Fuel Burnable Absorbers (IFBA)
3. Axial Blankets
4. Debris Filter Bottom Nozzles (DFBN)
5. Extended Burnup Geometry

The upgraded fuel features are, with the exception of the DFBN, a subset of the Westinghouse VANTAGE 5 design features generically approved by the NRC in a Safety Evaluation Report dated July 1985 on WCAP-10444-P-A, "VANTAGE 5 Reference Core Report VANTAGE 5 Fuel Assembly." The bottom nozzle used for the PBNP design differs from the VANTAGE 5 bottom nozzle in that it is fabricated from stainless steel rather than Inconel and the size and pattern of the flow holes have been changed. However, the DFBN meets all other design requirements.

Additional core design features were also addressed in this reanalysis program for use at PBNP. These include:

1. Use of Peripheral Power Suppression Assemblies (PPSAs)
2. Removal of fuel assembly thimble plugging devices
3. Elimination of the third line segment of the K(z) curve, (Figure 15.3.10-3).

These features are reflected where appropriate in the proposed Technical Specification changes.

The Technical Specification Change Request is supported by the results of the reanalysis of the PBNP FSAR Chapter 14 design-basis accidents. The reanalysis falls into three categories: large-break Loss of Coolant Accident (LOCA), small-break LOCA, and non-LOCA transients.

The large-break LOCA analysis is being conducted jointly with Northern States Power Company (NSP) as part of the resolution of the Upper Plenum Injection (UPI) issue using the Westinghouse best-estimate W COBRA/TRAC model. NSP's Prairie Island is the lead plant in this joint effort. The NSP large-break LOCA analysis was submitted to the NRC in WCAP-10924, "Westinghouse Large-Break LOCA Best-Estimate Methodology." A Safety Evaluation Report on the methodology and the Prairie Island analysis is expected from the NRC in September 1988. The PBNP-specific large-break LOCA analysis using the best-estimate methodology is expected to be submitted in support of this Technical Specification Change Request in October 1988.

The small-break LOCA analysis of PBNP used the NOTRUMP code described in WCAP-10054-P-A, "Westinghouse Small Break LOCA ECCS Evaluation Model Using the NOTRUMP Code." The use of NOTRUMP for small-break LOCA analyses of Westinghouse reactors was accepted by the NRC in an SER dated May 21, 1985. For this small-break LOCA analysis, only the 4-inch cold leg break was considered. The use of a single break size and location is considered adequate because in NOTRUMP analyses done for NSP's Prairie Island Plant, and the NOTRUMP generic studies of two-loop plants described in WCAP-11145, "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code," submitted to the NRC on June 11, 1986, the 4-inch cold leg break was always limiting. The decision to analyze a single break size and location is further explained in Section 7.2.2 of Attachment 1. The NRC Staff tentatively agreed during our February 4, 1988 meeting that analysis of a single break size and location would be adequate.

The final analysis category comprises the reanalysis or re-evaluation of all the non-LOCA transients presented in Chapter 14 of the PBNP FSAR. The most recent transient reanalysis for PBNP was performed for the licensing of Westinghouse Optimized Fuel Assemblies (OFA), which was approved by the NRC on October 5, 1984 (Unit 2 License Amendment No. 90) and May 22, 1985 (Unit 1 License Amendment No. 86). The only new methodologies used in the present reanalysis were the Westinghouse Revised Thermal Design Procedure (RTDP), WCAP-11397, submitted by Westinghouse to the NRC on March 16, 1987, and the Westinghouse Owners' Group (WOG) Dropped Rod Generic Methodology, WCAP-11394, submitted by the WOG to the NRC on May 22, 1987. NRC SERs for these generic methodologies are expected in the third quarter of 1988.

It is anticipated that most of the reactor core changes described herein will be implemented for the PBNP Unit 1 Cycle 17 start-up in the spring of 1989, and for the PBNP Unit 2 Cycle 16 start-up in the fall of 1989. Unit 1 is expected to shut down for refueling about March 31, 1989. We will need to know no later than the end of February, 1989, if the requested Technical Specification Change Request will be approved. This schedule provides the six-month review period you requested in our February 4, 1988 meeting on this subject.

With the exception of the change to the basis of 15.3.3, these proposed changes will initially apply only to Unit 1. When the upgraded fuel features are incorporated in Unit 2 during its fall 1989 refueling outage, these changes will apply to both Units. We request that if these changes are approved, the effective dates be May 1, 1989 for Unit 1 and November 1, 1989 for Unit 2. The change to 15.3.3-Basis applies to both units now and can be made effective on May 1, 1989.

We have enclosed a check in the amount of $150.00 for the application fee as prescribed in 10 CFR 170. Please contact us if you have any questions regarding this submittal.

Very truly yours,

C. W. Fay

C. W. Fay
Vice President
Nuclear Power

Attachments

Copies to NRC Regional Administrator, Region III
NRC Resident Inspector
R. S. Cullen, PSCW

Subscribed and sworn to before me
this 26th day of August, 1988.

Deloes A. Gueztkowski
Notary Public, State of Wisconsin

My commission expires: 5-27-90