Wisconsin Electric
POWER COMPANY

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NPL 97-0275

10 CFR 50.4
10 CFR 50.90

May 15, 1997

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

Gentlemen:

DOCKETS 50-266 AND 50-301
SUPPLEMENT TO TECHNICAL SPECIFICATIONS
CHANGE REQUEST 193
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

In a letter dated January 24, 1997, Wisconsin Electric requested Technical Specifications Change Request (TSCR) 193. This TSCR proposes to provide for storage of fuel assemblies of higher enrichment than is currently authorized at the Point Beach Nuclear Plant.

In a letter dated April 3, 1997, the NRC staff requested additional information related to TSCR 193. Attached is our response to that request. Also attached is a revision to the "No Significant Hazards Consideration" that supersedes in whole the previous "No Significant Hazards Consideration."

We have determined that the additional information does not involve a significant hazard, authorize a significant change in the types or total amounts of any effluent release, or result in any significant increase in individual or cumulative occupational exposure. Therefore, we conclude that the proposed amendments meet the requirements of 10 CFR 51.22(c)(9) and that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared.

Please contact us if you have any questions.

Sincerely,

Thomas G Malkmus

for Douglas F. Johnson
Manager-Regulatory Services
and Licensing

RF/kmc

Attachment

cc: NRC Resident Inspector
NRC Regional Administrator
PSCW

Subscribed and sworn before me on
this 15th day of May, 1997.

Jacqueline Gorski
Notary Public, State of Wisconsin

My commission expires 10/26/2000.

RESPONSE TO
REQUEST FOR ADDITIONAL INFORMATION FOR TSCR 193
INCREASED ENRICHMENT LIMITS FOR FUEL AT POINT BEACH NUCLEAR PLANT

1. Provide a summary of the results of the September 1996 blackness tests at Point Beach including the number of Boraflex panels tested, number of gaps identified, gap sizes, and gap distribution.

   RESPONSE: A summary of the 1996 Boraflex blackness testing performed at Point Beach is enclosed in Attachment 1.

2. Provide a copy of Westinghouse criticality analysis CAA-96-146.

   RESPONSE: A copy of Westinghouse Report CAA-96-146, "Criticality Analysis of the Point Beach Nuclear Plant Spent Fuel Storage Racks Considering Boraflex Gaps and Shrinkage with Credit for Integral Fuel Burnable Absorbers," is enclosed.

3. Please explain the statement included in the No Significant Hazards Analysis the "The proposed changes do not involve a change to plant design."

   RESPONSE: The No Significant Hazards Consideration has been rewritten to account for the fact that, while there are no actual physical changes to the plant under this TSCR, an increase in allowable fuel assembly enrichment may be considered a change to the plant design. The modified No Significant Hazards Consideration is enclosed.

4. Please provide information on planned changes in burnup to accommodate the 18-month cycle and any effects on spent fuel pool cooling capability.

   RESPONSE: Although higher enrichments might be seen to imply higher burnups, we are still limited by current burnup limits. We will continue to meet our current burnup limit in both annual and 18-month cycles. This limit is 60 GWD/MTU based on the lead rod, as delineated in WCAP-10125, Extended Burnup Evaluation of Westinghouse Fuel; and 62 GWD/MTU based on the lead rod on a cycle-specific basis, as delineated in Westinghouse generic letter NTD-NSA-SAII-94-482, Lead Rod Average Burnup to 62,000 MWD/MTU. The conservative methodology for determining decay heat, as defined in ANSI 5.1-1979, assumes full power operation over an extended period of at least 100 days. Decay heat is then primarily dependent on power level and time after shutdown. Under the proposed changes, power level will not increase. Our decay heat calculation, therefore, will not be affected. In a conservative decay heat calculation, decay heat is not dependent on fuel assembly enrichment.

Attachment 1

SUMMARY OF 1996 BORAFLEX BLACKNESS TESTING

Blackness testing of Point Beach spent fuel pool Boraflex panels was conducted by Holtec International in September 1996. Calibration test runs showed that gaps of 1/2-inch width or larger could be detected in the Boraflex absorber material of the actual racks. The Boraflex blackness testing yielded the following results:

Number of panels tested:	22
Number of gaps identified:	0
Gap sizes:	No gaps found.
Gap distribution:	No gaps found.

There was no measurable shrinkage or gap formation in the Boraflex panels analyzed. Testing of three of these panels yielded results that were not considered reliable, but with no obvious indications of gaps. These panels are currently scheduled to be retested in August 1997.

TECHNICAL SPECIFICATION CHANGE REQUEST 193
NO SIGNIFICANT HAZARDS CONSIDERATION

In accordance with the requirements of 10 CFR 50.91(a), Wisconsin Electric Power Company (Licensee) has evaluated the proposed changes against the standards of 10 CFR 50.92 and has determined that the operation of Point Beach Nuclear Plant, Units 1 and 2, in accordance with the proposed amendments does not present a significant hazards consideration. The analysis of the requirements of 10 CFR 50.92 and the basis for this conclusion are as follows:

1. Operation of this facility under the proposed Technical Specifications will not create a significant increase in the probability or consequences of an accident previously evaluated.

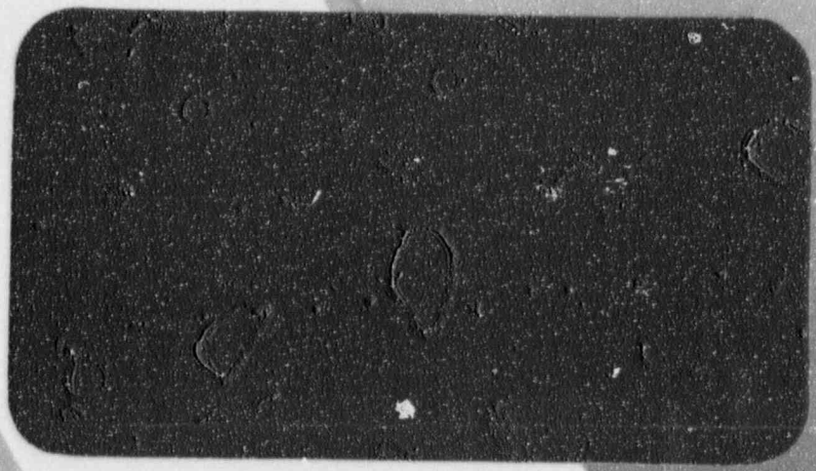
The proposed changes do not involve a change to structures, systems, or components that would affect the probability or consequences of an accident previously evaluated in the PBNP Final Safety Analysis Report (FSAR). The only relevant concern with respect to increasing enrichment limits in the spent fuel pool and new fuel storage racks is one of criticality. The proposed changes use the same criticality limit used in the current Technical Specifications. Therefore, margin to safe operation of Units 1 and 2 is maintained. The probability and consequences of an accident previously evaluated are dependent on this criticality limit. Because the limit will not change, the probability and consequences of those accidents previously evaluated will not change.

2. Operation of this facility under the proposed Technical Specifications change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not involve a change to the physical structure of the spent fuel pool or of the plant. The proposed increase in spent fuel pool and new fuel storage racks fuel assembly enrichment limits maintains the margin to safe operation of Units 1 and 2 because the criticality limit for the spent fuel pool and new fuel storage racks will not change. The enrichment increase does not affect any of the parameters or conditions that contribute to the initiation of any accidents. Because the criticality limit remains the same, these changes have no effect on plant operation or on the initiation of any accidents. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of this facility under the proposed Technical Specifications change will not create a significant reduction in a margin of safety.

The proposed changes maintain the margin to safe operation of Units 1 and 2. The margin of safety is based on the criticality limit of the spent fuel pool and the new fuel storage racks. Because this limit will not change, the margin of safety will not be affected. Therefore, the proposed changes will not create a significant reduction in a margin of safety.



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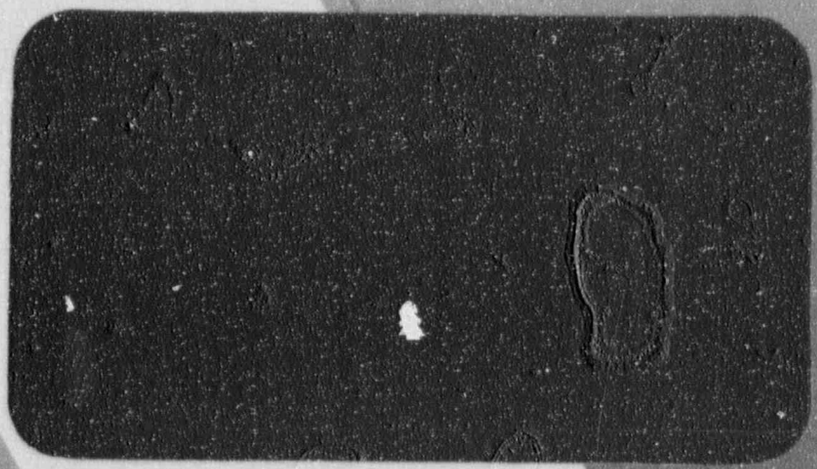
Commercial Nuclear Fuel Division



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