Wisconsin Electric POWER COMPANY
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May 3, 1982

Mr. H. R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555

Attention: Mr. R. A. Clark, Chief
Operating Reactors, Branch 3

Gentlemen:

DOCKET NOS. 50-266 AND 50-301
TECHNICAL SPECIFICATION CHANGE REQUEST NO. 72
REDUNDANT DECAY HEAT REMOVAL
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

On January 25, 1982, we received your letter responding to our license amendment application dated November 16, 1981. The application was submitted in response to your request dated August 14, 1981 and transmitted proposed changes to the Point Beach Nuclear Plant Technical Specifications to provide additional assurance of the availability of redundant reactor decay heat removal capabilities. Your January 25 letter provided six comments on our amendment application and requested that we supply additional information and modify the application if necessary.

The items from your January 25 letter have been reviewed and the following actions are being taken:

1. In Item 1, the NRC requested that the action statement in proposed Specification 15.3.1.A.3.b(3) be applied to the conditions described in proposed Specification 15.3.1.A.3.a(4) as well. An appropriate modification has been made to incorporate this comment in the proposed Specification 15.3.1.A.3.a(4) of the attached Technical Specification page change.

2. In Item 2, NRC requested that the existing specification which requires one steam generator to be operable whenever the average reactor coolant temperature is above 350°F remain in the proposed specifications. Accordingly, this specification has been reinserted in 15.3.1.A.2.

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3. In Item 3 of the January 25 letter, the NRC staff questioned the basis for the proposed Technical Specifications which states that a single residual heat removal (RHR) loop provides sufficient heat removal capacity for removing the reactor core decay heat. The Staff based this question on Table 6.2-7 of the Point Beach FFDSAR which lists the heat removal capacity of a single RHR heat exchanger as 24.15×10^6 BTU/hr or about 0.47% of rated core thermal output and the bases of existing Technical Specification 15.3-3 which predicts the decay heat load 48 hours following a shutdown after a full power run as 0.62% rated core thermal output.

   One must first recognize that these references are conservative design figures. The actual RHR system performance and present day conservatisms concerning decay heat loads demonstrate a single RHR heat exchanger is able to remove the reactor core decay heat as soon as RHR can realistically be initiated and well before the steam generators and reactor coolant loops could be taken out of service for inspection, maintenance, or other purposes. This conclusion is based on the following observations:

   a. The FFDSAR stated design heat capacity of a single RHR heat exchanger of 24.15×10^6 BTU/hr or 0.47% rated power assumes a RHR inlet temperature from RCS of 160°F and a service water inlet temperature of 80°F to the component cooling water heat exchanger, which cools the component cooling water to the RHR heat exchangers. A revised analysis was done assuming a primary coolant inlet temperature to the RHR heat exchangers of less than 200°F, which satisfies the temperature requirements for a cold shutdown condition, and a service water inlet temperature of 65°F, which is consistent with the assumption in FFDSAR Table 9.3-3. This calculation showed that a single RHR loop and a single component cooling water heat exchanger could handle a heat load of 42×10^6 BTU/hr or 0.80% rated power.

   b. A more accurate method for determining decay heat in terms of percent of rated power is presented in American National Standard ANS 5.1, 1979, "Decay Heat Power in Light Water Reactors". The best estimate for reaching 0.80% full power based on this standard is six hours. The method used in the

standard assumes that all the fuel in the core has been irradiated for three consecutive cycles just prior to shutdown and is, therefore, actually overly conservative.

c. The design heat capacity of a single RHR loop was verified in a test conducted March 26-27, 1982. Unit 1 was shut down for a steam generator inspection after sustained operation at 87% power and a core average burnup of 2740 MWD/MTU. Following reactor coolant system cooldown to 208°F, achieved about 18-1/2 hours after shutdown, the heat removal capability of a single RHR train and a single component cooling loop was tested. At that time, in addition to the reactor decay heat input, a reactor coolant pump was running. The minimum number of service water pumps (3) were running and service water temperature was 37°F. The reactor coolant system was cooled down to about 200°F over a 2-1/2 hour period with the RHR heat load at 42 x 10^6 BTU/hr. About 36 hours after shutdown, the other RHR train and component cooling water loop were tested. At that time, with reduced decay heat input and a reactor coolant pump still running, the RHR system was able to maintain the primary system at 158°F. The RHR heat load at that time was 32 x 10^6 BTU/hr. Within the parameters of the test (slightly less decay heat due to reduced power but significantly more heat load was added by the reactor coolant pump), this test demonstrates the fact that a single RHR loop can remove core decay heat significantly before the existing Technical Specification basis would lead one to believe.

In light of the above discussion, we maintain that the Technical Specification basis statement regarding the capacity of a single RHR loop to remove core decay heat is, as a practical fact, correct. We, therefore, have proposed no additional change to the suggested specification for this item.

4. This item requested that, in order for Wisconsin Electric to take credit for steam generators as a redundant decay heat removal system without addressing the operability of the reactor coolant pumps, we should show that adequate thermal mixing will take place down to the lowest cold leg temperature anticipated for this cooling mode and that no void formation and possible core damage will take place prior to establishment of sufficient thermal driving

head to initiate natural circulation flow. Engineering judgment would cause us to believe that natural circulation would occur precluding core damage prior to exceeding normal operating condition clad temperature limits. With the minimal heat flux in these conditions, DNB would not occur. Nonetheless, after reviewing the available information concerning these considerations, we have determined that a documented analysis to support this judgment would be both costly and time consuming and is not available at this time. Accordingly, we have modified the attached Specification 15.3.1.A.3.a to include either reactor coolant loops A or B together with its associated steam generator and a reactor coolant pump as a decay heat removal method. Qualifying statements for starting a reactor coolant pump at low temperatures are specified in Technical Specification Section 15.3.15.B.2.

5. In Item 5, NRC requested that Technical Specification 15.3.3.A.2 be clarified to require that one accumulator isolation valve be checked open prior to shutting the other accumulator isolation valve when taking an accumulator out of service to conduct repairs. This clarification has been included on the attached proposed Technical Specification pages at 15.3.3.A.2.

6. In Item 6, NRC suggested additional changes to proposed Specification 15.3.A.2.b(3) to replace the words "in the reactor decay heat load" with "in reactor coolant system temperature". The NRC reasoning was that decay heat load is uncontrollable while temperature is controllable. We believe to the contrary that in this circumstance decay heat load is in fact controllable by removal of irradiated fuel assemblies, if necessary, and temperature is in fact not controllable because of the postulated failure of the decay heat removal capability. We therefore do not believe that the changes suggested to Specification 15.3.3.A.1(d) are necessary, or appropriate.

7. As discussed in Item 3.b, ANS 5.1, 1979, "Decay Heat Power in Light Water Reactors", presents a more accurate and acceptable method for predicting reactor core decay heat. Accordingly, the table in the basis of Specification 15.3.3 has been revised to list the ANS 5.1 predicted decay heat with time values.

Pursuant to the Commission's regulations, we have enclosed three signed originals and 40 copies of this modification to our November 16, 1981 license amendment application. We have, for your convenience, enclosed both the changes discussed in this letter and those contained in our original application. As discussed with Mr. Colburn of your staff, we were unable to completely respond to this information request within the time period specified in your letter. We trust this delay in our response has not significantly affected your review of this topic.

Very truly yours,

C W Fay

Assistant Vice President

C. W. Fay

Enclosures

Copy to NRC Resident Inspector

Subscribed and sworn to before me this 6th day of May 1982.

Dorothy Fleischmann
Notary Public, State of Wisconsin

My Commission expires July 1, 1984.