Wisconsin Electric POWER COMPANY
231 W. MICHIGAN, P.O. BOX 2046, MILWAUKEE, WI 53201

November 9, 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 Reactor Branch 3

Gentlemen:

DOCKET NOS. 50-266 AND 50-301
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
LOCKED ROTOR AND LOCA ANALYSES
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

Your letter dated October 22, 1982 forwarded a request for additional information concerning the loss of reactor coolant flow locked rotor accident and requested an explanation of the differences between the LOCA reanalysis contained in our letter dated November 19, 1979 and the analysis presented in the updated FSAR. We have responded to these requests in the attachment to this letter. This information should satisfy your inquiries in these matters.

Very truly yours,

S. Burstein

Executive Vice President

Sol Burstein

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Attachment

Copy to NRC Resident Inspector

ATTACHMENT

1. With respect to the locked Rotor Analyses in the FSAR, provide the following information.

   a. Submit justification for assuming a 0.9 second time interval between the time of pump seizure and the beginning of control rod motion as described on page 14.1.8-6 of the FSAR.

RESPONSE:

For a locked rotor loss of flow accident a reactor trip is assumed to be initiated by a low flow trip in the affected loop. As stated on page 14.1.8-3 of the FSAR, the time from the initiation of the low flow signal to initiation of control rod motion is 0.6 seconds. During the initial plant startup testing, a check of the circuitry disclosed that there was actually only a 0.45 second delay from the time flow reaches the low flow trip setting until the instant that the rods are released. Figure 14.1.8-7 shows that the low flow trip point setting is reached in less than 0.1 seconds in the affected loop; therefore, the assumption of 0.9 second delay for control rod motion is conservative when compared to the most probable time delay of approximately 0.55 seconds. A 0.9 second time delay is also the time delay typically used by the reactor vendor in conducting this analysis.

   b. Was credit taken for pressurizer sprays during the accident analysis? If so, what is the peak pressure expected without pressurizer spray actuation?

RESPONSE:

The locked rotor analysis presented in the updated FSAR did not assume any credit for actuation of pressurizer spray.

   c. Justify the availability of off-site power during the transient and the continued operations of one reactor coolant pump.

RESPONSE:

Because of the short duration of the transient imposed by a locked rotor accident loss of off-site power has not been assumed for this accident, nor was there a requirement for this assumption at the time this analysis was completed for the Point Beach Nuclear Plant. We understand that the assumption of no loss of off-site power for this transient is true for virtually all presently licensed plants. We further understand that the NRC has previously concluded in NUREG-0138 that the assumption of loss of off-site power for the locked rotor analysis was not necessary. We have also been advised by Westinghouse that recent studies conducted for NTOL plants have confirmed that the locked rotor analysis is not sensitive to a loss of off-site power incident.

   d. Westinghouse report WCAP 8151, dated June of 1973, provides the results of the locked rotor analysis at low pressure and indicated that 63% of the fuel rods reach a DNBR of less than 1.3. This transient lasts for a few seconds. Provide the results of site boundary dose calculation for this condition.

RESPONSE:

WCAP-8151, "Fuel Densification Point Beach Nuclear Plant Unit No. 2 Low Pressure Analysis", conservatively concluded that analysis of the locked rotor accident for operation at reduced system pressure indicated an increase in both the number of rods with a minimum DNB ratio less than 1.30 and in peak cladding temperature; however, the peak cladding temperature for the worst point in the core remains well below the limiting value. The report also concluded that the consequences of the hypothetical locked rotor accident are not significantly affected by operation with reduced system pressure. The report notes that the analysis was very conservative since the reactor coolant pressure increase as a result of the transient was ignored when calculating the number of rods with DNBR below 1.3 and rods for which the fluid conditions were beyond the range of the DNB correlation were conservatively assigned DNB ratios less than 1.3. The number of rods with a DNB ratio less than 1.3 would therefore be expected to be substantially less than the upper bound value assumed.

Because of additional conservatisms, such as assuming for the peak cladding temperature calculation that the rod hot spot is already in DNB at the start of the accident and the short duration of this transient, actual fuel rod failures are not anticipated. Accordingly, specific site boundary dose calculations for this condition have not been calculated.

Disregarding the conservatisms inherent in this analysis, even if all rods in DNB were assumed to have cladding failure, the site boundary dose calculations would still be bounded by the analyses presented in FSAR Section 14.3.5 and the guidelines for site boundary doses given in 10 CFR Part 100 would not be exceeded. Therefore no specific dose calculation for the accident was necessary.

2. A comparison of the Point Beach 1 and 2 LOCA analysis contained in the updated FSAR was made with a NRC previously approved analysis submitted by your staff by letter dated 11/19/79. Some inconsistencies exist. Please fill in the missing numbers in the table below and indicate which analysis is the most current.

RESPONSE:

We have provided below those numbers missing (underlined) from the Table provided with your request. The LOCA analysis contained in the updated FSAR is based on licensee's filing dated March 20, 1979. An error was made in the updated FSAR Table 14.3.2-2 indicating that this analysis was effective for up to 18% steam generator tubes plugged. The March 20, 1979 analysis and the analysis presented in the updated FSAR actually assumed 10% steam generator tubes plugged. The FSAR will be corrected in the first annual update. The most current analyses for up to 18% steam generator tubes plugged were provided with licensee's letter dated November 19, 1979, for operation at 2250 psia, and with licensee's letter dated November 27, 1979, for operation at 2000 psia.

COMPARISON OF POINT BEACH LOCA ANALYSIS

Assumptions	FSAR Analysis	November 19, 1979 Analysis
Power, %	102	102
T_{AVE}	570°	574°
Core Inlet Temp.	540°	544°
Pressure, Psia	2250	2250
% Tubes Plugged	10	18
Peaking Factor	2.32	2.32