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

September 29, 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
ASME SECTION XI RELIEF REQUESTS
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

In Mr. Clark's letter dated August 31, 1982, the NRC transmitted to Wisconsin Electric Power Company license amendments which revised the Technical Specifications for inservice inspection of safety-class components and granted relief from specific ASME Section XI Code requirements. During our review of this transmittal letter and the attached Technical Specification changes, we observed several inconsistencies. The first of these inconsistencies concerned two of the 10 CFR Section 50.55a reliefs granted in the letter for Point Beach Unit 1 but not for Point Beach Unit 2. In a telephone call to Mr. Tim Colburn and other members of your staff on September 14, 1982, we discussed this problem and requested that this letter be revised to grant these reliefs for both units. We also noted a second inconsistency between the letter and the Technical Specification revision concerning inservice testing of pumps and valves. Mr. Colburn acknowledged these concerns and indicated they would be resolved in a revision to the license amendment transmittal.

During the September 14 telephone call, we also discussed Wisconsin Electric's letter to Mr. Denton dated February 23, 1982. In this letter we provided the NRC with a detailed listing of all the examinations performed during the Point Beach Unit 1 first ten-year inspection interval. Enclosure 4 to that letter listed the areas where we did not comply with the inservice inspection program and identified the reasons for not meeting the program. During the

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review of the August 31 NRC letter, we discovered that a relief request had not been filed for one of these items. As discussed in the February 23 letter, this item concerns the frequency of the reactor vessel interior examination (Cat BN1) which was not practical at Point Beach Nuclear Plant. Relief requests were previously submitted to cover all the other items listed in the February 23 letter.

Accordingly, enclosed with this letter is a relief request from the frequency of the reactor vessel interior examination (Cat BN1). This relief request is applicable to Point Beach Units 1 and 2 for the first ten-year interval.

It is our understanding that, with NRC acceptance of the relief requests previously submitted and herein provided, the first ten-year inservice inspection interval requirements for Point Beach Unit 1 are complete. It is expected that all the first interval inservice inspection requirements for Unit 2 will be completed during the spring refueling outage in 1983.

Very truly yours,

CW Fay

Assistant Vice President

C. W. Fay

Enclosure

Copy to NRC Resident Inspector

UNITS 1 AND 2 RELIEF REQUEST

COMPONENT

Reactor pressure vessel.

EXAM AREA

Reactor vessel interior surfaces.

ISOMETRIC OR COMPONENT DRAWING

None.

ASME SECTION XI CATEGORY

BN1 (1974/S75)

ASME SECTION XI ITEM NUMBER

B1.15 (1974/S75)

ASME SECTION XI EXAMINATION REQUIREMENT

A visual examination (VT) is required every 3 years of the accessible areas of the vessel interior surfaces during a normal refueling outage.

ALTERNATIVE EXAMINATION

A visual examination (VT) of the reactor vessel interior will be performed when the core barrel is removed but not at a frequency greater than that specified in the Code. The core barrel will not be removed specifically for this examination.

REASON FOR LIMITATION

As can be seen from Figure 1, only a small portion of the reactor vessel interior surfaces are accessible with the core barrel in place. There is approximately 10" of the vessel interior surface accessible from the top of the core barrel flange to the reactor vessel flange during a normal refueling outage. A meaningful examination cannot be performed unless the core barrel is removed. Removal of the core barrel requires a complete defueling of the reactor and significant ALARA impacts including exposure and contamination problems.

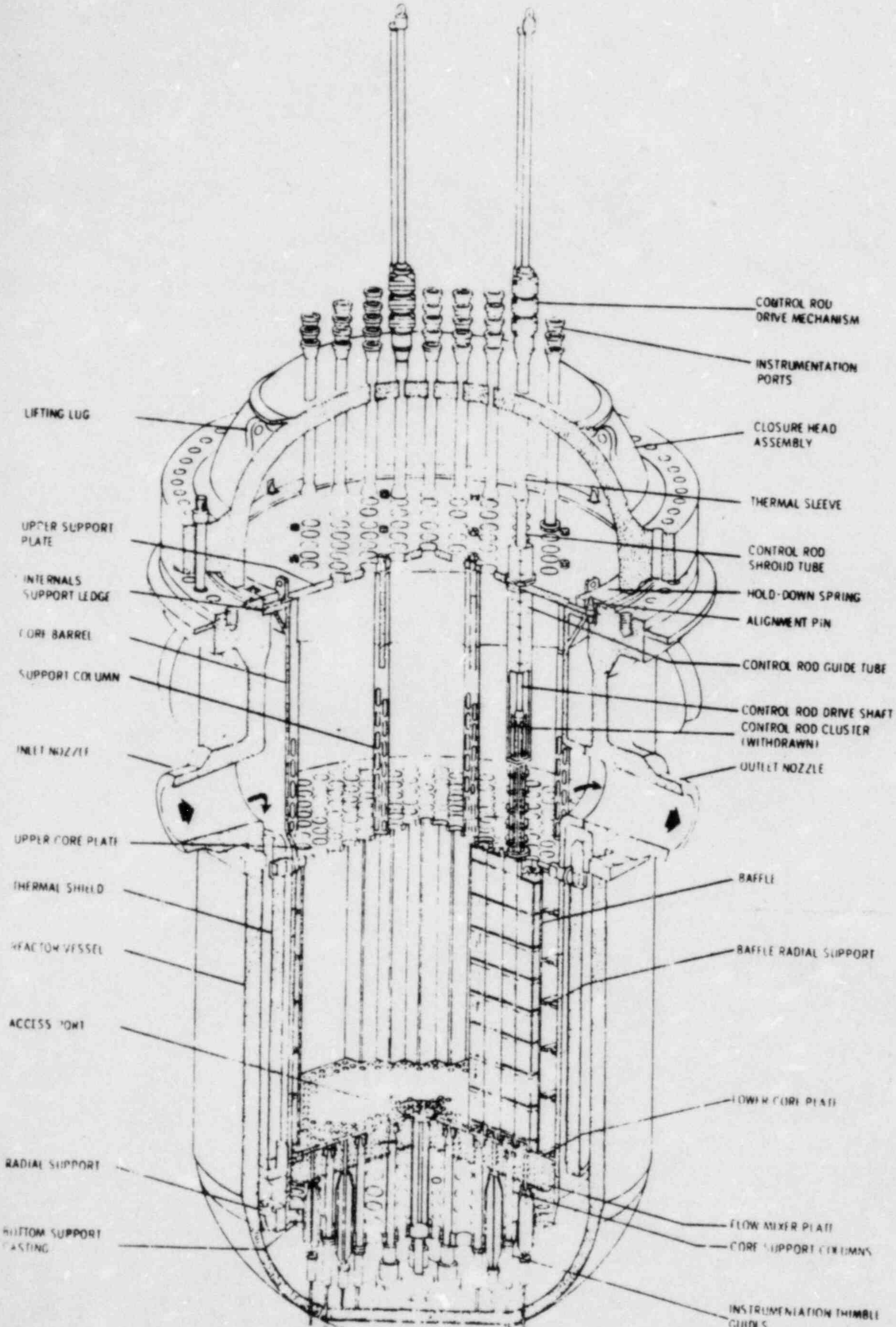

FIGURE 1

REACTOR VESSEL INTERNALS
FIGURE 3.2.3-2