

FEB 17 1983

Docket File
DCS MS-016

Docket Nos. 50-266
and 50-301

Mr. C. W. Fay
Vice President-Nuclear Power
Wisconsin Electric Power Company
231 West Michigan Street
Milwaukee, Wisconsin 53201

Dear Mr. Fay:

By letter dated August 31, 1982, the NRC transmitted Amendment Nos. 63 and 68 to Facility Operating Licenses Nos. DPR-24 and DPR-27 for Point Beach Nuclear Plant Unit Nos. 1 and 2, respectively. These amendments revised the language of the Technical Specifications relating to inservice inspection requirements and of safety class components and granted relief to some examination requirements and were in partial response to your applications transmitted by letters dated February 17, 1977 and November 27, 1978.

By telephone conversation on September 14, 1982 and letter dated September 29, 1982 you pointed out errors in our interpretation of your relief requests and in the transmitted Technical Specification pages. Two of the relief requests for inservice testing of safety class components, reactor vessel nozzle-to-vessel welds and safety injection nozzle-to-safe end welds, items B1.4 and B1.6 respectively, were requested for both Point Beach Units 1 and 2 but were thought by the staff to have been requested for Unit 1 only.

The staff has reviewed the reliefs from examination requirements for Unit 1 and finds them equally applicable to Unit 2. The bases for granting the reliefs are contained in our Safety Evaluation transmitted with our August 31, 1982 letter. During the September 14, 1982 telephone conversation, members of your staff pointed out that, although our Safety Evaluation indicated relief from the requirement to visually inspect reactor vessel cladding patches (Item B1.14) was not necessary as this examination was no longer required by the updated versions of the ASME Code, you still requested approval of this request as it was not your intention to update your program to the later versions of the code. We have reviewed your request and as this examination is no longer required under updated versions of the ASME Code Section XI, we find your proposed alternative schedule acceptable. Enclosed is a revised Table 1 from our August 31, 1982 Safety Evaluation.

As stated in our August 31, 1982 Safety Evaluation, the proposed Technical Specification changes for inservice testing of pumps and valves was beyond the scope of that review. However, a portion of those proposed Technical Specifications relating to inservice testing of pumps and valves was inadvertently included in our transmittal. Corrected pages of the Technical Specifications are enclosed. We apologize for any inconvenience this may have caused

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Mr. C. W. Fay

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Your September 29, 1982 letter also requested an additional relief from the requirements for frequency of visual examination of the reactor vessel interior surfaces. Your justification for this request was that only a small portion of the reactor vessel interior surface is accessible for examination during a normal refueling when the core barrel is in place. Removal of the core barrel requires a complete defueling of the reactor causing significant radiation exposure and potential contamination problems. Your proposed alternative was to perform this inspection when the core barrel was removed but at a frequency not greater than that specified in the ASME Code Section XI.

The code requirement is for a visual examination of the accessible reactor vessel interior surfaces every three years during a normal refueling outage. The purpose of the examination is to determine the general mechanical and structural conditions of components and their supports such as the presence of loose parts, debris, or abnormal corrosion products, wear, erosion, corrosion, and loss of integrity at bolted or welded connections. For component supports and component interiors, the visual examination may be performed remotely with or without optical aids to verify the structural integrity. The staff, therefore, has determined that the requirement is practical to perform and should be performed with the core barrel in place and at the required frequency.

The staff therefore, is not granting relief from the required visual examination frequency.

Sincerely,

Original signed by:

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosures:

- 1. Corrected SE Table 1 to Amendments 63 and 68
- 2. Corrected TS pages 15.4.2.2

cc: See next page

- Docket File ASLAB
- NRC PDR
- Local PDR
- ORB #3 Rdg
- DEisenhut
- PMKreutzer
- TColburn
- OELD
- LJHarmon (2)
- SECY
- TBarnhart (4)
- LSchneider (1)
- DBrinkman
- ACRS (10)
- OPA (Clare Miles)
- RDiggs
- ~~RBattard~~
- NSIC

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DATE	2/1/83	2/1/83	2/17/83				

Wisconsin Electric Power Company

cc:

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Table 1 Class 1 Components

IWB-2600 item no.	IWB-2500 exam. cat.	System or component	Area to be examined	Required method	Licensee proposed alterna- tive exam.	Relief request status
B1.4	B-D	Reactor vessel nozzles (6)	Nozzle-to- vessel welds and inside radiused sections	Volumetric at frequency below: 1st period - 2 welds 2nd period - 1 or 2 welds 3rd period - remaining welds	Volumetric - all nozzles once every 10 years when core barrel is removed	Granted
B1.6	B-F	Safety Injection nozzle- to-safe end	Weld	Volumetric & surface at frequency in IWB-2411.	Volumetric only once every 10 years when core barrel is removed	Granted
B1.12	B-H	Reactor vessel	Integrally- welded supports	Volumetric at frequency below: 1st period- 25% 2nd period - 25% 3rd period - remainder	Volumetric- 100% of weld when core barrel is removed during interval	Granted
B1.14 (Applies to Unit 1 only)	E-I-1	Reactor vessel	Cladding patches	Visual at frequency below: 1st period - 25% 2nd period - 25% 3rd period - remainder	Visual - 100% when core barrel is removed	Granted
<p>B303090268 B30217 PDR ADOCK 05000266 P PDR</p>						
B3.7	B-H	Regenera- tive heat exchanger	Integrally - welded sup- ports	Volumetric (10% of weld)	Visual	Granted
B5.4	B-K-1	Reactor coolant pumps	Integrally - welded supports	Volumetric	Visual	Granted
B5.6	B-L-1	Reactor coolant pumps	Pump casing welds	Volumetric	Examine Weld To 1977 S78 Section XI Code	Granted

2. Containment isolation valves will be tested in accordance with Technical Specification 15.4.4 instead of Section IWV-3420, Valve Leak Rate Test.

Bases

The proposed inspection program is, where practical, in compliance with the recommendations of ASME Boiler and Pressure Vessel Code, Section XI, Summer 1971 Addenda. It must be recognized, however, that equipment and techniques to perform the inspection are still in development. It is recognized, however, that examinations in certain areas are necessary and therefore a schedule is proposed that includes areas and frequencies that are believed practical at this time for this reactor. In most areas scheduled for test, a detailed pre-service mapping will be conducted using techniques which can be used for post-operation inspections. The areas indicated for inspection represent those of relatively high stress and therefore will serve to indicate potential problems before significant flaws develop there or at other areas. As more experience is gained in operation of pressurized-water reactors, the recommended time schedule and location of inspection might be altered, or should new techniques be developed, consideration will be given to incorporate these new techniques into this inspection program.

The use of conventional non-destructive, direct visual and remote visual test techniques can be applied to the inspection of all primary loop components except for the reactor vessel. The reactor vessel presents special problems because of the radiation levels and remote underwater accessibility to this component. Because of these limitations on access to the reactor vessel, several steps have been incorporated into the design and manufacturing procedures in preparation for non-destructive test techniques which may be available in the future. ⁽¹⁾

The techniques for in-service inspection include the use of visual inspections, volumetric (ultrasonic or radiographic) and surface (dye penetrant or magnetic particle) testing of selected parts during refueling periods.

The intent of the inspection is the detection of flaws large enough to initiate fast fracture and gross leakage prior to subsequent inspection. At this time it is judged that such a flaw is substantially larger than 1/2 inch by 1 inch which is the degree of detectability. The inspection method is designed to detect flaws of this magnitude.

(1) FSAR - Section 4.4

- 2: Containment isolation valves will be tested in accordance with Technical Specification 15.4.4 instead of Section IWV-3420, Valve Leak Rate Test.

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(1) FSAR - Section 4.4