Wisconsin Electric POWER COMPANY
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August 7, 1980

Mr. Richard Snaider
Generic Issues Branch
Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Snaider:

COMMENTS ON NUREG-0577
DATED OCTOBER 1979

Mr. D. Eisenhut's letter of May 19, 1980 transmitted a NRC proposed implementation plan for NUREG-0577. The letter requested comments by July 7, 1980 on the plan and NUREG-0577, "Potential For Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports".

Our letter to you of July 3, 1980 requested that the comment period be extended until September 15, 1980. As this was not acceptable to NRC, we have condensed our review to provide the following comments at this time.

Proposed Emplementation Plan

1. The plant classification system appears to be without a firm basis (see comment 6) and accordingly the proposed implementation plan is discriminatory.

2. The requirement to immediately inform the regional NRC office (on page 3) connotes an importance associated to this issue which seems to conflict with NRC's position on continued operation. The NRC is aware that older plants may have difficulty in finding documentation sufficiently precise to satisfy more recent requirements. It would be more appropriate to specify that a preliminary evaluation be provided within 60 days, or other time period.

3. The requirement to include the reactor vessel supports (item 6 on page 4) seems to be, in some aspects, a duplication of effort with respect to the PWR Reactor Vessel Support Owner's Group

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effort. This group has been evaluating various LOCA loading situations and has included some evaluation of the primary system equipment supports in their Phase C study (NRC-TAP-TOPIC A-Z). This work should be thoroughly reviewed before additional efforts are required.

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4. Page 1, second paragraph: the words ". . . or damage" seem to be inappropriate in this context. In the case of PBNP it was a design requirement (Westinghouse E-Spec. G-677188, section 3.4.13) that each support foot (there are three) be capable of supporting the entire pump assembly. While it is doubtful that this applied beyond the normal operating condition (the stress analysis has not been reviewed), the support configuration provided margins of safety deemed acceptable when this plant was designed and constructed.

5. The inclusion of additional temperature margins beyond that of the NDTT (NDTT + 1.3 sigma + adjustment from figure 1) is again additional conservatisms that seem to be unwarranted in light of the following statement on page 5:

   > "Therefore, at temperatures above NDT, rapid crack propagation from relatively large cracks, such as those resulting from the growth of small cracks in locally embrittled regions, will be arrested."

6. The classification of plants (identified on pages 8 and 9; and page C-49) seems to be arbitrary but is used to require actions by some utilities while exempting others. On page C-46 the statement is made that the Point Beach Nuclear Plant should remain in Group I primarily because the main columns of their supports were fabricated from A-53 steel which is characterized by the report as a material with very loose specifications. However, ASTM A-53 required that each length of pipe be hydrostatic tested to a prescribed pressure (a minimum of 2400 psi for 12 inch schedule 100 pipe); ASTM A-53 also required that piping not classified as double extra strong should be subjected to a "cold" flattening test, an evaluation of ductility. While this may have been an older form of material evaluation, credit should be

given for having a ductile material. Furthermore, at the top of page 11, it is acknowledged that "section sizes under one inch have a relatively low susceptibility to brittle failure. . .". Twelve inch diameter, schedule 100 pipe has a wall thickness of 0.843 inches.

7. The evaluations in section 4.1 (page C-26) should include and discuss the flattening tests mentioned previously.

8. The evaluations contained in section 4.9 of Appendix C all basically consider a tensile type of loading. Obviously, if a tensile load is trying to pull a piece of steel apart, a crack or defect could propagate. However, the basic loading of the support structures as indicated in Figure 2.1 and disregarding moments, is that of compression. It is not clear that cracks or defects would propagate in a compressive loading situation as they might in a tensile situation. In addition, it should be noted that there are horizontal supports vertically along these pieces of equipment (as discussed later in the report) such that even during accident type loadings, the predominant loading made on the primary support elements remains compressive.

We trust that our comments will be given consideration and further effort on our part will await your review of the comments received and appropriate revisions to the NRC action plan.

Very truly yours,

C. W. Fay, Director
Nuclear Power Department