NSP

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October 15, 1984

Mr R L Spessard, Director
Division of Reactor Safety
U S Nuclear Regulatory Commission, Region III
799 Roosevelt Road
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PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

In response to your letter dated June 29, 1984 which transmitted Inspection Reports Nos. 282/84-08(DRS) and 306/84-07(DRS), the following information is provided.

VIOLATION

The Prairie Island Nuclear Generating Station, Unit 1 Technical Specification 4.2 states: "Inservice testing of ASME Code Class 1, Class 2 and Class 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g)."

Contrary to the above, five motor operated, Category A Residual Heat Removal System valves are not leak tested or trended per the requirements of IWV-3420 of Section XI of the ASME Code.

This is a Severity Level V violation. (Supplement II).

RESPONSE

A commitment was made during the inspection to make procedure changes to correct the identified noncompliance. In response to that commitment, techniques for quantitative determination of leakrate using the testing methods described in Prairie Island Inservice Testing (IST) Program Request for Relief #59 were evaluated. A technique for quantitatively determining the leakrate of valves MV-32066, 32165 and 32231 has been developed and will be incorporated into the appropriate procedures. The engineering analysis is available at the site for your review.

Even though a quantitative measurement technique has been developed, we still believe that IST Program Request for Relief #59, as approved by the Commission, exempted Prairie Island from testing valves MV-32066, 32165 and 32231 to the requirements of IWV-3420(d). Items 2.a.1 and 2.b.3 of Request for Relief #59 were submitted because it was found at the time to be impractical to quantitatively determine the leakage from these valves. We believe the Commission stated their agreement with this position in Section 3.2.1 of the IST Program SER dated January 4, 1983.

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However, portions of the same SER imply that leakrates would be measured by the alternate test procedure described in Request for Relief #59 and compared against established leakage limits. It was never the intent of the test procedure described in Request for Relief #59 to provide quantitative leakrate data or to impose specific leakrate criteria. The described procedure was only intended to provide a qualitative assessment of the integrity of the subject valves. If this qualitative determination indicated that leakage was occurring, further investigative actions would be taken.

We believe that the violation described in the inspection report was not a result of a failure to comply with regulations. Rather, it resulted from differing interpretations of Request for Relief #59 and the IST Program SER. On this basis we feel that it is appropriate to request that the citation be withdrawn and left as an open matter to resolve with NRR.

In order to avoid further confusion with respect to the intent of Request for Relief #59, the present wording will be revised to clarify the intent of the request for relief and to include the recently developed quantitative measurement technique. This revision will be submitted for NRR review and approval.

Please note that contrary to item 3.b of the subject inspection report, valves MV-32064 and 32065 are category B valves and are not subject to the requirements of IWV-3420(d).

Please contact us if you require further information related to our response.

C E Larson

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Vice-President, Nuclear Generation

CEL/bd

c: Resident Inspector, NRC
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