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10 CFR Part 2
Appendix C

U S Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Response to Notice of Violation
NRC Inspection Report Nos. 282/94005(DRS) and 306/94005(DRS)
Inservice Testing of Pumps and Valves (IST) Inspection Report

Your letter of June 2, 1994, which transmitted Inspection Report Nos. 282/94005(DRS) and 306/94005(DRS), requested responses: (1) to a violation; (2) to an inspection follow-up item; and (3) to comments, regarding the inservice testing (IST) program, contained in the cover letter. This letter provides these responses; responses (1) and (2) are included as attachments.

The following is an assessment of the IST Program and implementation issues identified in Inspection Report Nos. 282/94005(DRS) and 306/94005(DRS).

Prairie Island acknowledges that the ASME Section XI code requirements have not been adequately implemented in all cases. During the Design Basis Document (DBD) project and during the preparation of the third 10-year IST Program submittal, in 1993, deficiencies were identified in the scope and implementation of the existing IST Program. The deficiencies in scope were corrected with the development of the third 10-year IST Program submittal. In addition, informal training was provided to affected system engineers by the IST Coordinator, on the requirements of the O&M Standards and the scope of the third 10-year IST Program.

As a result of the comments and questions, in the NRC's December 8, 1993 SER, concerning the documentation to support the third 10-year IST Program submittal, Plant Management requested that Nuclear Generation Services' Design Group conduct a thorough review of the NRC's SER and Prairie Island's IST Program submittal and determine what actions were necessary to implement the third 10-year IST Program for Unit 1. Due to the preparation for and conduct of Unit 1's May 1994 refueling outage, the IST Program is in a state of transition. As a result of some reorganization within the Plant Engineering



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Department, the individual responsible for the review of the NRC's SER and the third 10-year IST Program transferred to and assumed the position of Superintendent Technical Programs. Under this individual's direct guidance, a number of initiatives had been planned to improve the IST Program.

These initiatives include:

(1) Preparation of a revised IST Program for submittal later in 1994.

(2) Development of an IST basis document to define safety functions and design acceptance criteria for components in the program.

(3) Development of more comprehensive test procedures to adequately implement ASME Section XI requirements.

(4) Meeting with NRR IST Engineering personnel on April 12, 1994, to discuss the NRC's SER and Prairie Island's actions to address the issues identified.

(5) Increased management oversight and attention on the development and implementation of the IST Program.

(6) Preparation of training for System Engineers on the specific requirements of ASME Section XI and the O&M Standards.

Plant Management recognizes that with the amount of work associated with developing the third 10-year IST Program, revising the submittal and implementing the program, a part-time IST Coordinator will not be able to accomplish these tasks. Plant Management will provide additional resources and/or designate a full-time IST Coordinator to ensure timely implementation of the IST Program. Coupled with the initiatives listed above, Prairie Island is confident that the revised third 10-year IST Program will be adequately documented and effectively implemented.

Attachment 1 provides our response to the violation.

Attachment 2 provides our response to the inspection follow-up item.

New commitments to the NRC are indicated by italics in the attachments to this letter.

Please contact Jack Leveille (612-388-1121, Ext. 4662) if you have any questions related to our response to the subject inspection report.

E J Watt for
Roger O Anderson
Director
Licensing and Management Issues

c: Regional Administrator III, NRC
Senior Resident Inspector, NRC
NRR Project Manager, NRC
J E Silberg

Attachment 1: Response to Notice of Violation
Attachment 2: Response to Inspection Follow-up Item

Attachment 1

RESPONSE TO NOTICE OF VIOLATION

Notice of Violation

During an NRC inspection conducted on April 18-22 and May 2-6, 1994, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C, the violation is listed below:

Prairie Island Technical Specification 4.2.A.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(f), except where specific written relief has been granted by the NRC.

A. ASME Code, Section XI, IWP-3100, "Inservice Test Procedure," states that, "The resistance of a system shall be varied until...the measured flow rate equals the corresponding reference value."

   Contrary to the above, as of May 2, 1994, the test procedures for the residual heat removal (RHR) pumps did not vary the measured flow rate to equal the corresponding reference value.

B. ASME Code, Section XI, IWV-3427, "Corrective Action," states "Valves with leakage rates exceeding either the values specified by the Owner, or those rates given in IWV-3426 shall be replaced or repaired."

   Contrary to the above, as of May 2, 1994, no corrective actions were taken when the containment sump isolation valves exceeded the leak rate limits established in the local leak rate test procedures.

C. ASME Code, Section XI, IWV-3522, "Exercising Procedure," states "Check valves should be exercised to the position required to fulfill their function...." The IST program identified the refueling water storage tank to RHR pump suction check valves (SI-7-1 and SI-7-2) have an open safety function.

   Contrary to the above, as of May 2, 1994, the test procedures for check valves SI-7-1 and SI-7-2 contained inadequate acceptance criteria to exercise the open safety function.

This is a Severity Level IV violation (Supplement I.D.3).

Response

Background

A description and background on the event can be found in NRC Inspection Report Nos. 282/94005(DRS) and 306/94005(DRS).

1. Reason for the Violation

The primary reason for the violation is less than a full understanding by the engineering staff of the purpose and methods required by ASME Section XI and the O&M standards, and a secondary reason is inadequate instrumentation.

a. In the case of the RHR pump test line, the piping resistance was considered to be constant since the geometry was fixed and the isolation valve was in the full open position.

   NUREG-1482, Guidelines for Inservice Testing at Nuclear Power Plants addresses allowable variances from reference points in section 5.3. In systems where flow can be varied, the flow is to be set at the reference point, and the differential pressure measured. The NUREG recognizes that the flow point may not fall precisely at the same point each test. The NRC Recommendation is that the tolerance be set as low as possible, but in no case exceed +/- 2%. In the RHR pump quarterly surveillance test, SP1089, the mini-flow has a normal band of 131 to 137 gpm, which is 134 gpm +/- 2%. The problem occurs when this is read as differential pressure. Fluctuations in the indicated differential pressure results in a range of 139 to 152 psid which is 145.5 psid +/- 4.5%. Although this was incorrectly identified as the NORMAL range, the proper methodology was embodied in the procedure.

   Since flow in the system cannot be adjusted (i.e. no flow paths are changed or valve positions changed), the variation in indicated flow from test to test could be caused by one of three possibilities:

   1. Pump performance is changing
   2. Instrumentation is not accurate
   3. Orifice is wearing

   1. The indicated mini-flow varied high and low in the normal range. There is not a discernable trend in either direction. This indicates that the pump is not degrading. Full flow testing during the Unit 1 refueling outage confirmed that the Unit 1 RHR pumps have not degraded.

   2. Controlling the pressure pulsations in the mini-flow line has been an ongoing problem in obtaining consistent gauge readings.

Efforts were moderately successful; however, some pulsations continued. An assumption was made that the installed plates were orifice plates. The Unit 1 plates were removed for inspection during the Unit 1 refueling outage. They were not orifice plates but each was a 1/8" plate with a drilled hole. This configuration was a contributor to the pressure pulsations.

3. The inspection of the orifice verified the dimensions of the hole still met the original specifications, and there was no physical degradation of the orifice (i.e., no erosion or other damage).

From the three items above, it is concluded that the major problem is instrument inadequacy due to pressure pulsations causing fluctuations in the indicated differential pressure.

b. Corrective action for repair of containment isolation valves was not considered necessary because it was believed that 10CFR50 Appendix J was the controlling document and that containment composite leakage criteria were always met.

c. Procedures to full stroke exercise SI-7-1 and SI-7-2 used acceptance criteria based upon the calculated flow necessary to fully open the valves. It was not recognized that full accident flow in this case was required.

2. <u>Corrective steps that have been taken and the results achieved</u>

The following corrective actions were taken on Unit 1:

a. 1. The RHR pump surveillance procedure was revised, requiring flow rate be recorded and the pump data taken at a known reference value. The RHR pumps were tested with the new procedure at the recent refueling outage; new baseline data was obtained.

2. The orifices were removed and inspected. The plate was not an orifice plate, but rather a hole drilled through a 1/8" plate. Inspection of the orifice verified the dimensions of the hole still met original specifications and there were no signs of physical degradation. An alteration was performed to properly bevel the downstream side of each plate.

3. Hydraulic snubbers were installed in the tubing to the differential pressure indicators to dampen the pressure pulses. The flow indication was calibrated and added to the I&C calibration schedule.

The changes in a.2. and a.3. above did not significantly reduce the pressure-induced fluctuations in indicated differential pressure.

b. The containment leakage rate testing procedure was revised to assign specific acceptance criteria for each valve and require corrective action if the criteria are exceeded. The procedure was performed at the recent Unit 1 refueling outage; no problems were encountered.

c. The test procedure for SI-7-1 and SI-7-2 was revised to require full accident flow of 1800 gpm to assure the valve obturator travels to the full open position. The procedure was performed at the recent Unit 1 refueling outage; no problems were encountered.

3. <u>Corrective steps that will be taken to avoid further violations</u>

The following corrective actions will be taken to prevent recurrence:

a. *The system engineers will be trained on the specific requirements of ASME Section XI and the O&M Standards by November 1, 1994.*

b. *Procedures for pumps and valves in other systems will be reviewed to assure the deficiencies identified in the Inspection Report are not present in those procedures. Revised surveillance procedures to meet the third 10-year IST interval will be completed by October 31, 1994 for Unit 1 and December 21, 1994 for Unit 2.*

c. *Unit 2 containment leak rate test procedures will be updated prior to their next use to reflect the changes made to the Unit 1 procedures.*

d. The actions taken in 2.a.2. and 2.a.3. did not significantly reduce the pressure induced fluctuations in indicated differential pressure. A search is being made to determine if there is available replacement instrumentation that will provide stable indication. If replacement instrumentation is not available, relief from the ASME Section XI code will be pursued due to the RHR system hydraulic performance when in the recirculation mode.

4. <u>Dates when Full Compliance will be Achieved</u>

Full compliance has been achieved except for the RHR pump testing. *We plan to achieve compliance for RHR pump testing by one of the two approaches below:*

*A determination of replacement RHR instrumentation will be completed by December 16, 1994. If suitable instrumentation is available, it will be installed by the next refueling for each unit.*

*If suitable replacement RHR instrumentation is not available, a request for relief from ASME Section XI requirements will be submitted by December 16, 1994.*

Attachment 2

RESPONSE TO INSPECTION FOLLOW-UP ITEM

The following is a response to the Inspection Follow-up Item concerning the weakness associated with the development and implementation of an adequate check valve program as recommended by SOER 86-03.

Prairie Island acknowledges that the existing check valve program implementation must be improved to meet the recommended actions in industry guidance documents. Prairie Island concurs that more appropriate management oversight and attention should have been given to assure a well developed, documented, and implemented check valve program.

The existing check valve program will be re-evaluated against the recommendations in SOER 86-03 and the guidance contained in EPRI report NP-5479. Risk assessment and results-centered maintenance (RCM) will be utilized to determine the scope of the check valve program. A thorough design application review will be a part of the revised check valve program. This review in conjunction with RCM on check valves will determine the basis for the check valve PM program.

Management oversight and attention on the revision and implementation of the check valve program will be increased. In addition, dedicated engineering resources will be assigned to the development and implementation of the check valve program.

Scheduled Completion Dates

1. *The re-evaluation of the existing check valve program against SOER 86-03 and EPRI report NP-5479 will be completed by January 1, 1995.*

2. *The check valve program document and associated PM program will be completed by January 1, 1995.*