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R.O. WILLIAMS, JR.
VICE PRESIDENT
NUCLEAR OPERATIONS

September 20, 1988
Fort St. Vrain
Unit No. 1
P-88341

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Docket No. 50-267

SUBJECT: NRC Inspection Report 88-13

REFERENCE: 1) NRC Letter, Callan to Williams dated August 19, 1988 (G-88335)

2) PSC Letter, Williams to NRC dated September 15, 1988 (P-88337)

Gentlemen:

This letter is in response to the Notice of Violation received as a result of the inspection conducted by Messrs. R. E. Farrell and R. P. Mullikin during June 1-30, 1988 (Reference 1). The following response to the items contained in the Notice of Violation is hereby submitted.

Identification of Violation:

Technical Specification 7.4.a requires that written procedures shall be established, implemented, and maintained for activities including surveillance testing.

Technical Specification 7.4.b requires that these procedures be periodically reviewed.

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Contrary to the above:

1. The licensee failed to maintain Procedure SR-5.2.16.a-Q, Issue 34, "PCRV Closure Leakage Determination," current with a Technical Specification change issued on March 18, 1982. The out-of-date procedure resulted in the licensee erroneously placing themselves in the Technical Specification Limiting Conditions for Operation 4.2.9 action statement that could have resulted in a plant shutdown.

2. The licensee failed to establish and implement proceduralized controls for the shedding of electrical DC loads following a turbine trip with a loss of offsite power in accordance with the plant Reference Design Manual, Station Batteries Design Criteria DC-92-1.

3. Special Instruction to Operators No. 85-15, Issue 4, dated June 23, 1988, which was subsequently issued on an interim basis to establish procedural controls for shedding of DC loads, failed to require shedding of Computer Inverter N-9234 consistent with DC-92-1.

This is a Severity Level IV violation. (Supplement I)(267/8813-01)

(1) <u>The Reason For The Violation If Admitted</u>:

The violation for items 1 and 2 identified above is admitted; the violation for item number 3 is not admitted.

Item 1 noted in the Notice of Violation was due to the fact that calculations included in SR-5.2.16.a-Q had not been updated to reflect the March 18, 1982 change to Fort St. Vrain's Technical Specifications. This revision was intended to accommodate an operating mode where the Loop II Steam Generator penetration could be operated at a pressure slightly above cold reheat pressure.

SR-5.2.16.a-Q, "PCRV Closure Leakage Determination," is used to determine the leakage rate from the Prestressed Concrete Reactor Vessel (PCRV) closure interspace. The surveillance is performed on a quarterly basis or in response to an unanticipated alarm for interspace pressurization gas flow.

The methodology of SR-5.2.16.a-Q is a sequential process for determining if the leak rate is in violation of Technical Specifications and for identifying the exact location and pathway of a leak. If conservative acceptance criteria specified in the test are exceeded, the test conductor is directed to utilize procedure SR-RE-151-X, "Penetration Interspace Leakage Pressure Decay Tests," for final assessment of the leak rate. SR-RE-151-X is the more definitive method for determining PCRV penetration leakage when the acceptance criteria of SR-5.2.16.a-Q have been exceeded. When used, SR-RE-151-X is attached to the SR-5.2.16.a-Q being performed and becomes a part of that surveillance.

During the incident noted, it was determined that typing errors and errors in transposition had occurred during the preparation of SR-RE-151-X. SR-RE-151-X had only been performed on one occasion previous to the noted incident.

SR-RE-151-X had been developed in early 1987 as part of an overall revision to Station Betterment Engineering administrative controls and implementing procedures. SR-RE-151-X replaced a previously used procedure, primarily for the purpose of reformatting information. During this reformatting, errors were introduced into the procedure which resulted in the overly-conservative calculation and unnecessarily entering the Action Statement of LCO 4.2.4.

Item 2 noted in the Notice of Violation was due to the fact that Public Service Company of Colorado (PSC) has not developed an operating procedure to address a loss of all AC power event (Station Blackout).

Fort St. Vrain's (FSV) Final Safety Analysis Report (FSAR), Section 8.2.3.4, states that either Station Battery 1A or 1B is adequate to supply its service load requirements, as defined in Station Battery Design Criteria DC-92-1, to a minimum voltage of 105 Volts DC for a four hour discharge. The FSAR Section 8.2.3.4 further states "Battery 1A or 1B is adequate to supply the required shutdown DC loads for four hours following a loss of all AC power." Station Battery Design Criteria DC-92-1, Table 1 (attached), assumes that several loads, which are not automatically shed from the DC buses, are manually shed when their function is no longer required. If these loads were not shed at or before the times assumed in the load profile analysis during a loss of all AC power, Station Batteries 1A and 1B may not be capable of providing an adequate power supply to the loads which are required to function for four hours. These loads are assumed to function in DC-92-1.

A Loss of All AC Power emergency procedure would be the logical document to direct manual shedding of the DC loads. At present, Fort St. Vrain's Emergency Procedures address Loss of Outside Power and Turbine Trip (Emergency Procedure EP F-3), and Loss of Outside Power and Turbine Trip with Failure of One Standby Diesel Generator to Start (Emergency Procedure EP F-4). The FSAR, Section 10.3.2, assesses the latter event. Fort St. Vrain's

Emergency Procedures do not address, nor does the FSAR assess, a loss of all AC power event.

PSC has not previously developed an emergency procedure for a loss of all AC power event as it was not considered to be a credible event in the FSAR. PSC submitted Amendment 5 to the FSV Preliminary Safety Analysis Report (PSAR) in a letter dated October 13, 1967. Question IX.I from the Atomic Energy Commission's (AEC) Director of the Division of Reactor Licensing was addressed in that letter. Question IX.I asked, in part, for PSC to "Analyze the consequences of a total loss of electric power, including the standby diesel generators but not the station battery." PSC responded that "the total loss of electric power is not considered credible, the reasoning for this being summarized in Section A below." PSC then proceeded to discuss the effects of a loss of all AC power on reactor plant equipment.

PSC recognizes that 10CFR50.63, issued June 21, 1988, requires nuclear power plant licensees to develop operating procedures for station blackout events for specified durations. Although 10CFR50.63 is not specifically applicable to FSV, PSC has committed to develop an interim procedure for station blackout (Reference 2). This procedure will address the requirements for manual shedding of DC loads during an interval when AC power is not available. The procedure is discussed further in the corrective steps which will be taken portion of this response.

FSV's current Emergency Procedures do not address, for the most part, events which are not considered to be credible. PSC is in the process of developing new, symptom oriented, Emergency Operating Procedures (EOP) which will be implemented prior to startup following the fourth refueling. The new EOP's are being developed in accordance with the requirements of Supplement 1 of NUREG 0737. These procedures will address credible as well as non-credible emergency events at FSV. PSC will ensure that those accidents assessed in the FSAR, which are not considered to be credible, will be addressed by the new EOP's.

Exception is taken to item 3 noted in the Notice of Violation. Special Instruction (SI) to the Operators No. 85-15 incorporated the load shedding requirements of page B.2.9 of Change Notice (CN) 2672, "Replacement of Batteries 1A, 1B and 1C and required shedding of Computer Inverter N-9234. Included in CN-2672 were provisions for revising DC-92-1, which revised the time requirement for securing the DC load for Computer Inverter N-9234 from 30 minutes to two hours. CN-2672 was implemented in accordance with administrative controls which govern the design change process at FSV.

(2) The Corrective Steps Which Have Been Taken And The Results Achieved:

For item 1, Non-Conformance Report (NCR) 88-172 was initiated against SR-RE-151-X to recheck the calculations for the leak rate. SR-5.2.16.a-Q and SR-RE-151-X were reviewed for validity and revisions to the procedures were initiated. Previous documentation of completed SR-RE-151-X was reviewed, recalculated and found to be within Technical Specification Limits. NCR 88-172 was closed.

For item 2, SI 85-15, Issue 4, was issued effective June 23, 1988. This SI provided interim instructions for the operating staff to secure the necessary loads. The development of a new Emergency Procedure to address a loss of all AC power event was also initiated at this time.

No corrective actions are necessary for item 3.

(3) The Corrective Steps Which Will Be Taken To Avoid Further Violations:

For item 1, errors identified in SR-RE-151-X and SR-5.2.16.a-Q will be corrected. Calculations included in SR-5.2.16.a-Q will be updated to reflect the current Technical Specifications.

For item 2, corrective steps have been initiated in several areas to avoid further violations. An interim procedure, addressing plant operations with only DC power available for up to one hour, is being prepared. An Engineering Evaluation, analyzing the DC load profiles required to support plant operations for one hour without AC power being available, is being prepared. A letter, from R. O. Williams to the NRC dated September 15, 1988 (Reference 2), has been submitted which describes PSC's justification for operating with only DC power available for up to one hour if AC power is unavailable. The station battery load profile technical specification surveillance will be revised to reflect a one hour battery load profile.

No additional corrective steps will be taken for item 3.

(4) The Date When Full Compliance Will Be Achieved:

For item 1 full compliance will be achieved following the updating of SR-5.2.16.a-Q. The revised procedure will be implemented by October 4, 1988.

For item 2 full compliance will be achieved following the issuance of an interim procedure which addresses plant operations with only DC power available. The new procedure will be issued prior to restart following the current circulator outage at FSV.

Should you have any further questions, please contact Mr. M. H. Holmes at (303) 480-6960.

Sincerely,

R. O. Williams, Jr.
Vice President,
Nuclear Operations

ROW:DLW/djc

cc: Regional Administrator, Region IV
ATTN: Mr. T. F. Westerman, Chief
Projects Section B

Mr. Robert Farrell
Senior Resident Inspector
Fort St. Vrain