APPENDIX B

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-267/89-02 License: DPR-34

Docket: 50-267

Licensee: Public Service Company of Colorado (PSC)

Facility Name: Fort St. Vrain Nuclear Generating Station (FSV)

Inspection At: FSV, Platteville, Colorado

Inspection Conducted: January 1-31, 1989

Inspectors: R. E. Farrell, Senior Resident Inspector (SRI)
P. W. Michaud, Resident Inspector (RI)

Approved: T. F. Westerman 1-17-89
T. F. Westerman, Chief, Project Section B Date
Division of Reactor Projects

Inspection Summary

Inspection Conducted January 1-31, 1989 (Report 50-267/89-02)

Areas Inspected: Routine, unannounced inspection of licensee action on previously identified inspection findings, operational safety verification, monthly surveillance observation, monthly maintenance observation, and core safety limit.

Results: Continued high moisture levels in the reactor coolant system resulted in a decision to shut down the reactor after 1 week of critical operation (paragraph 3).

The licensee continues to experience breakdowns in the area of procedural compliance. One event, a failure to follow detailed instructions contained in a surveillance procedure, resulted in a violation (paragraph 4). Additionally, plant startup has been delayed for 2 months with an extensive recovery effort due to this event (paragraph 3).

No discrepancies were noted in the maintenance (paragraph 5) or reactor engineering (paragraph 6) areas.

DETAILS

1. Persons Contacted

D. Alps, Supervisor, Security
*M. Block, Systems Engineering Manager
L. Brey, Manager, Nuclear Licensing and Resources
*M. Cappelio, Central Planning & Scheduling Manager
*O. Clayton, Emergency Planning Coordinator
R. Craun, Nuclear Site Engineering, Manager
*M. Deniston, Superintendent, Operations
*J. Eggebroten, Technical Projects Manager
D. Evans, Operations Manager
*M. Ferris, QA Operations Manager
*C. Fuller, Manager, Nuclear Production
*J. Gramling, Supervisor, Nuclear Licensing Operations
*M. Holmes, Nuclear Licensing Manager
M. Niehoff, Nuclear Design Manager
*F. Novachek, Nuclear Support Manager
*H. O'Hagan, Outage Manager
*S. Smith, Staff Assistant, NED
*L. Scott, QA Services Manager
*N. Snyder, Maintenance Manager
*P. Tomlinson, Manager, QA
D. Warembourg, Manager, Nuclear Engineering
R. Williams, Jr., Senior Vice President, Nuclear Operations

The NRC inspectors also contacted other licensee and contractor personnel during the inspection.

*Denotes those attending the exit interview conducted February 2, 1989.

2. Licensee Action on Previously Identified Inspection Findings (92701)

(CLOSED) Open Item (267/8632-02): No Procedure for Technical Specification (TS) Surveillance Requirements SR 4.1.9.D.3 and SR 4.1.9.D.4. The licensee issued Procedure SR 4.1.9.D.4-R, "Functional Test of Reserve Shutdown Assemblies," on April 22, 1988. Procedure SR 4.1.9.D.3-RX, "Refueling Penetrations Piping Examination," was issued on January 26, 1989. The NRC inspectors reviewed these procedures to verify that they meet the requirements of the respective TS. These actions are sufficient to close this item.

(CLOSED) Open Item (267/8722-01): Nuclear Instrumentation Startup Channels Energized Until 10^{-2} Percent Power. The startup channels of nuclear instrumentation are automatically deenergized at 10^{-2} percent power (increasing) on the associated wide-range nuclear instrumentation channel. This power level corresponds to approximately 10^{6} CPM, which is at the high end of the startup channel indication. The NRC inspectors

questioned whether operation of these instruments in this manner might damage the detectors. The licensee does not consider this a problem, and has indicated that any degradation would be noticed during calibrations. In addition, the limited number of future startups and, consequently, the limited time these instruments will in fact operate at high levels, present a sufficient basis to close this item.

(CLOSED) Open Item (276/8820-02): Inconsistency in 120 VAC Breaker Labels. The NRC inspectors discovered two instances where the cable numbers listed on breaker labels were not in agreement with those listed on the associated key diagram. The licensee determined the breaker labels were incorrect and replaced the labels. The NRC inspectors verified all breaker labels on the 120 VAC breakers are correct in accordance with the respective drawings. No discrepancies were noted during a walkdown of this system. This item is closed.

3. Operational Safety Verification (71707)

The NRC inspectors made daily tours of the control room during normal working hours and at least once per week during backshift hours. Control room staffing was verified to be at the proper level for the plant conditions at all times. Control room operators were observed to be attentive and aware of plant status and reasons why annunciators were lit. The NRC inspectors observed the operators using and adhering to approved procedures in the performance of their duties. A sampling of these procedures by the NRC inspectors verified that current revisions and legible copies were available. During control room tours, the NRC inspectors verified that the required number of nuclear instrumentation and plant protective system channels were operable. The operability of emergency AC and DC electrical power, meteorological, and fire protection systems was also verified by the NRC inspectors. The reactor operators' and shift supervisor logs were reviewed daily, along with the TS compliance log, clearance log, operations deviation report (ODR) log, temporary configuration report (TCR) log, and operations order book. Shift turnovers were observed at least once per week by the NRC inspectors. Information flow was consistently good, with the shift supervisors soliciting comments or concerns from the reactor operators, equipment operators, auxiliary tenders, and health physics technicians. The licensee's station manager, operations manager, and superintendent of operations were observed to make routine tours of the control room.

The NRC inspectors made tours of all accessible areas of the plant to assess the overall conditions and verify the adequacy of plant equipment, radiological controls, and security. During these tours, particular attention was paid to the licensee's fire protection program, including fire extinguishers, fire fighting equipment, fire barriers, control of flammable materials, and other fire hazards.

A walkdown of the 120 VAC and 125 VDC essential power distribution systems, emergency diesel generator air starting system, control room ventilation system, and service water system was performed by the NRC

inspectors. Valve and breaker positions were verified, where possible. When affected by a clearance, the valves or breakers were verified to be positioned in accordance with the clearance requirements. Power supplies for components in these systems were verified, but were also subject to clearances in some cases. During these system walkdowns, the NRC inspectors verified the operability of standby or backup equipment when components or portions of systems were inoperable due to clearances.

The NRC inspectors observed health physics technicians performing surveys and checking air samplers and area radiation monitors. Contamination levels and exposure rates were posted at entrances to radiologically controlled areas and in other appropriate areas, and were verified to be up to date by the NRC inspectors. Health physics technicians were present to provide assistance when workers were required to enter radiologically controlled areas. The NRC inspectors observed workers following the instructions on radiation work permits concerning protective clothing and dosimetry, and observed workers using proper procedures for contamination control, including proper removal of protective clothing and whole body frisking upon exiting a radiologically controlled area.

The NRC inspectors randomly verified that the number of armed security officers required by the security plan were present. A lead security officer was on duty to direct security activities on each shift. The NRC inspectors verified that search equipment, including an x-ray machine, explosive detector, and metal detector, was operational or a 100 percent hands-on search was conducted. The protected area barrier was surveyed by the NRC inspectors to ensure it was not compromised by erosion or other objects. The NRC inspectors observed that vital area barriers were well maintained and not compromised. The NRC inspectors also observed that persons granted access to the site were badged and visitors were properly escorted.

The reactor was taken critical at 4:14 p.m. on January 5, 1989. The NRC inspectors observed the licensee's preparations and approach to criticality. The reactor was maintained at less than 2 percent power with the interlock sequence switch in the startup position. Training starts for all licensed operators and trainees were conducted over the following week. The NRC inspectors observed several of these training starts, which consisted of pulling Group 2B Rods (critical rod group) from their fully inserted position until the reactor was made critical. Group 2B Rods were then fully inserted, and a different licensed operator or trainee would repeat the procedure.

On January 13, 1989, the licensee performed an evaluation of the time it would take to remove moisture in the reactor coolant system to levels which would permit unrestricted operation. This evaluation showed that with the reactor shut down, the dew point could be allowed to peak and the moisture removal rate would be double that achievable maintaining the reactor critical. The licensee shut down the reactor at 2:54 p.m. on January 13, 1989. The NRC inspectors observed the shutdown and verified it was completed in an orderly manner. The reactor coolant temperature

was subsequently allowed to increase to approximately 400°F in order to elevate the dew point in the reactor coolant and maximize the moisture removal rate.

The NRC inspectors reviewed the licensee's evaluation of inoperable control rod drive (CRD) Temperature Recorder TR 1262-3. TS LCO 3.1.1.B and Surveillance Requirement 4.1.1.A deal with CRD motor temperature requirements. Though not specifically addressed in the TS, TR 1262-3 normally provides CRD motor temperature indications in the control room. The licensee determined that according to the TS, if Temperature Recorder TR 1262-3 is inoperable, it must be repaired or replaced within 24 hours or the digital thermocouple indicator must be used to determine, monitor, and manually record CRD motor temperatures. The NRC inspectors reviewed the licensee's evaluations and conclusions as documented in memo NLOS-22-098, dated December 30, 1988, and found them acceptable.

An engineering evaluation and 10 CFR 50.59 safety evaluation was performed by the licensee to determine the feasibility of securing all cooling water flow to the core support floor (CSF) when shut down. The purpose of securing all cooling water flow is to eliminate the potential for water ingress via any potentially leaking core support floor tubes. The NRC inspectors reviewed Engineering Evaluation EE-46-006, Revision A, dated January 28, 1989, and the associated safety evaluation (10 CFR 50.59 review), dated January 30, 1989. Both of these evaluations by the licensee were thorough and well documented. The results of these evaluations showed that the CSF cooling water could be isolated as long as: (1) average core outlet temperature and operating circulator inlet temperature are maintained between 110°F and 150°F, (2) PCRV pressure is maintained greater than or equal to 9.5 psia, and (3) the reactor is in the "shutdown" or "refueling" mode. The requirements of the engineering and safety evaluations were implemented in Special Instruction to Operators (OP-Order) 85-15, Issue 4. The NRC inspectors reviewed this OP-Order and noted it contained an additional restriction if a total loss of primary coolant flow was experienced while operating under the prescribed conditions, that primary coolant flow or CSF cooling water flow must be reestablished within 1 hour. The licensee met the specified requirements and isolated cooling water to the core support floor on January 30, 1989. On January 31, however, the licensee discovered the allowable 40°F temperature band could not be maintained, and CSF cooling water flow was restored.

The NRC inspectors reviewed the licensee's TS Clarification No. 003, dated January 24, 1989. This document provided guidance for determining the minimum required primary coolant flow rate during reactor shutdowns. These minimum flow rates are to assure a sufficient flow over the core region outlet thermocouples so they provide a representative temperature which may then be used to calculate bulk core temperature. TS LCO 4.1.9, "Core Inlet Orifice Valves/Minimum Helium Flow and Maximum Core Region Temperature Rise," is not applicable when the calculated bulk core temperature is less than or equal to 760°F with the reactor shut down. In accordance with this TS clarification, the minimum primary coolant flow rate is determined from Procedure CMG-25, "Shutdown Helium Flow

Requirements for Verification of Bulk Core Temperature." Procedure CMG-25, Issue 1, was reviewed by the NRC inspectors and found to contain sufficiently conservative assumptions so that the calculated minimum flow rate will preclude local flow stagnation or reverse flow. The minimum flow rate as determined using Procedure CMG-25 will assure representative temperatures which can then be used to verify whether the requirements of LCO 4.1.9 are applicable (greater than 760°F calculated bulk core temperature) or not.

On January 19, 1989, at approximately 6:50 a.m., the seven-region group of the reserve shutdown system was mistakenly actuated during the performance of a surveillance test. The events associated with this actuation are more fully described in paragraph 4 of this report. This event posed no direct safety concerns, but did cause another delay to plant startup, which is now scheduled for early March.

The licensee formulated a recovery plan and immediately proceeded in an effort to remove the reserve shutdown material from the reactor. Portions of this effort involve evolutions similar to a refueling outage; i.e., removal of individual control rods from the reactor. Other aspects involve unique operations and equipment, such as the reserve shutdown vacuum tool.

The NRC inspectors verified that the licensee's administrative requirements and controls of the recovery operations were in place. Shift crews were verified to contain key personnel with the skills and training required to perform their respective tasks. Clear lines of supervision and control were established. The NRC inspectors verified that QA/QC and health physics personnel were included in each shift crew. At the end of this report period, the licensee had not completed training for all involved personnel and had not finalized several necessary procedures. The NRC inspectors will monitor these efforts and verify all necessary and proper controls are implemented prior to commencing in-core work.

No violations or deviations were identified in the review of this program area.

4. Monthly Surveillance Observation (61726)

The NRC inspectors reviewed Surveillance Procedure SR 4.1.4.B-P-X, "Core Reactivity Status Check During Shutdown," Issue 3, prior to the reactor startup on January 5, 1989. This procedure provides a verification of the reactor shutdown margin as well as the sequence and predicted rod heights for criticality. The NRC inspectors independently verified the accuracy of the calculations and observed the actual critical rod height of 131 inches on Group 2B was within the predicted range of 126 to 164 inches. No discrepancies were noted during the review of this surveillance procedure.

Prior to the reactor startup on January 5, 1989, the NRC inspectors observed the performance of several precritical surveillance tests. These

surveillance tests were verified to conform to the requirements of applicable TS. The NRC inspectors observed qualified personnel utilizing approved procedures in performing these tests. Good communications between persons performing the surveillance tests and the control room operators was observed whenever alarms or indications were expected to change during the performance of the tests. Portions of each of the following surveillance tests were observed by the NRC inspectors:

- SR 4.1.1.D-X, "Full Stroke Scram Test," Issue 4, dated April 29, 1988

- SR 5.4.1.1.8b-M, "Reheat Steam Temperature Scram Test," Issue 21, dated August 5, 1988

- SR 5.4.8-R, "Power-to-Flow Instrumentation Calibration," Issue 23, dated September 23, 1988

On January 19, 1989, while performing Surveillance Procedure SR 4.1.8.C.1/2/3-Q, "Reserve Shutdown Hopper, ACM Disconnect, and Low Pressure Alarm Test," the reactor operator performing the test erroneously actuated the seven-region group of the reserve shutdown (RSD) system.

The RSD system provides an independent means of reactor shutdown and is divided into two subsystems: one for seven regions and the other for the remaining 30 regions of the reactor. The seven-region group is designed to provide a 1 1/2 percent negative reactivity worth, which will compensate for the worst reactivity accident. The 30-region group would be utilized only if all regular means of reactor shutdown were ineffective.

The TS surveillance requirements for this system include tests to verify the RSD system's operability and availability to perform its design function. A portion of these TS surveillance tests are provided in Procedure SR 4.1.8.C.1/2/3-Q. Step 5.5 of this procedure provides instructions to verify the operation of the actuating circuitry for the seven-region group. This is accomplished by turning a keylock switch located on the control board above the respective RSD actuating switches, which breaks the final leg of the actuating circuitry. The circuitry can then be tested without actually moving the firing valves.

Surveillance Procedure SR 4.1.8.C.1/2/3-Q, step 5.5.1, specifies turning Key Switch HS 11219-1 (seven-region RSD group test switch), located on "Board I-03, vertical section, immediately above HS 1102-1 and 1104-1." The reactor operator conducting the surveillance test actually placed the key into and turned Switch HS 11219-2 (same function as HS 11219-1, but for the 30-region RSD group), which is located <u>below</u> HS 1102-1 and 1104-1 (seven-region RSD actuating switches). Step 5.5.2 requires that the two yellow continuity lights corresponding to the subsystem being tested are verified off. The yellow lights for the 30-region group, which had erroneously been placed in test, were verified off. Following this, Step 5.5.3 of the procedure, which specifies turning and holding HS 1102-1 and 1104-1 (seven-region group actuation switches) in the OPEN position,

was correctly followed. Because the actuating circuitry for the seven-region group had not been placed in test in accordance with Step 5.5.1 and verified in accordance with Step 5.5.2, the seven-region group of the RSD system was actuated and did in fact release reserve shutdown material into the reactor. This was verified by observation of a step decrease in wide-range nuclear instrumentation indication (chart recorder) and subsequently by the inability to pressurize the RSD hoppers in the seven regions, indicating the rupture discs were not intact. The failure to follow detailed steps of Section 5.5 of Surveillance Procedure SR 4.1.8.C.1/2/3-Q is an apparent violation of NRC regulations (267/8902-01).

The licensee took immediate steps to prevent a recurrence of this event. Key members of the operations crew onshift at the time of this event have been disciplined. The entire Operations Department was counseled on procedural adherence and the use of caution during high risk operations. The licensee reviewed the surveillance procedure and will revise it to require both reserve shutdown subsystems be placed in "test" before performing continuity checks on either subsystem. The revised procedure will also include independent verification of the test switches' positions prior to test activities. The licensee is also reviewing the entire surveillance procedure for other possible human factor improvements.

One violation was identified in the review of this program area.

5. Monthly Maintenance Observation (62703)

During a plant tour on January 4, 1989, the NRC inspectors identified an apparent leak from Steam Generator Module B-1-4. The insulation around the cold reheat steam inlet pipe to this module was soaking wet and dripping water. The licensee determined that the leak was from the "Marmon" flange connecting the cold reheat steam inlet pipe to the steam generator module. The licensee initiated Station Service Request (SSR) 89500091 to repair the leak.

The "Marmon" flange connection involves a knife edge compressed into a groove with a collar providing compression to keep the joint leak tight. The licensee has experienced leaks from this type of connection in the past. There are 12 steam generator modules, each having a "Marmon" flange connection to a cold reheat steam inlet pipe. All of the connections were seal welded in 1984 via Nonconformance Reports (NCR) 84-60, 84-97, and 84-102, which were generated due to leaks from these connections. Since seal welding these connections in 1984, the licensee has repaired one of these connections on Steam Generator Module B-2-5. This work was done in April 1988 on NCR 88-0039 and SSR 88502312. At the time the seal weld on Steam Generator Module B-2-5 was repaired, the licensee documented the apparent cause of the seal weld failure. The apparent cause was the crown of the seal weld interfering with the compression collar. An uneven crown on the circumferential seal weld caused uneven pressure around the flange

from the collar. The licensee ground out the old seal weld, rewelded the connection, and ground the new seal weld flush with the pipe prior to installing the compression collar.

The NRC inspectors visually examined the leaking seal weld on the cold reheat steam pipe connection to Steam Generator Module B-1-4. The weld exhibited three separate center-line cracks, ranging from approximately 2 inches to 5 inches in length. The weld was crowned sufficiently in the NRC inspector's judgment to interfere with the even compression of the flange by the collar.

The licensee ground out the existing cracked seal weld, replaced the weld, and ground the new weld flush with the pipe prior to reinstalling the flange collar. The weld passed nondestructive examinations and there have been no subsequent leaks detected with the auxiliary boilers providing reheat steam to the steam generators.

On January 4, 1989, the licensee discovered a problem with the current readings across the scram contactors during a routine performance of a prestartup surveillance test. This test was to also serve as a postmaintenance verification of modifications made under Change Notice (CN) CN-2737. This CN replaced the original scram contactors, which had become obsolete and difficult to obtain parts for, with new scram contactors.

The NRC inspectors monitored the licensee's efforts to troubleshoot and resolve this problem with the current readings across the new scram contractor. Work was performed under SSR 88507944, which was reviewed by the NRC inspectors. The problem was found to be that the replacement scram contactors had so much less contact resistance that the difference in the lengths of wires in the various legs of the contacts circuitry (a 2 of 3 logic circuit) provided enough impedance to unbalance the circuitry. This resulted in the unbalanced current readings found during the surveillance test. The licensee's solution was to make all legs of the logic circuitry of equal lengths of wire. The NRC inspectors reviewed CN-2737A, which analyzed and authorized these changes, and observed their installation. The postmaintenance verification showed the repairs were successful. The NRC inspectors verified the associated design documentation, including controlled plant drawings, was updated to reflect these changes.

The NRC inspectors observed various stages of work associated with control rod drive and orifice assembly refurbishment. Procedure FHWP-100-20, Issue 1, "Replacement of Control Rod Cables and Refurbishment of Control Rod Drive and Orificing Assembly #20," was reviewed by the NRC inspectors. The required administrative approvals, quality control hold points, and detailed instructions were included in this work package. The NRC inspectors observed licensee personnel adhering to the requirements of radiological work permits and using good health physics practices at all times. At one point during this work, it was determined that a discrepancy existed between the instructions provided in Step 34.14.14 of

Procedure FHWP-100-20 and the associated design Drawing D1201-301, Issue F. The discrepancy concerned whether a gap in the RSD rupture disc housing was acceptable after assembly. The NRC inspectors observed the quality control inspector stop the work activity and obtain a resolution prior to proceeding again. The gap was allowed in accordance with the disposition of NCR 85-224. The NRC inspectors reviewed this NCR and its disposition and found it acceptable.

No violations or deviations were identified in the review of this program area.

6. Core Safety Limit (59700)

The NRC inspectors reviewed the licensee's TS Safety Limit 3.1, "Reactor Core - Safety Limit," and associated TS Surveillance Requirement SR 5.1.6, "Core Safety Limit Surveillance." The core safety limit for the high temperature, gas-cooled reactor, limits the total power-to-flow ratio integrated over time for the lifetime of each fuel segment. The TS contains a graph which shows limiting power-to-flow ratios and limiting times at each power-to-flow ratio as a function of core power. Allowable power-to-flow ratios range from 1.05 at 15 percent core thermal power to a maximum allowable of 1.17 at 40 percent core thermal power decreasing to 1.05 at 100 percent core thermal power.

The licensee is required by TS Safety Limit 3.1 and Surveillance Requirement SR 5.1.6 to record and monitor the total time each fuel segment is above the permitted range of power-to-flow ratios. The concern here is migration of fuel particle kernels through the carbide coating at elevated temperature due to carbon sublimation.

The licensee performs Surveillance Procedure SR 5.1.6 weekly. During the current fuel cycle, which began in June 1984, there has been no operation with power-to-flow ratios that have counted towards the total allowable time at such power-to-flow ratios listed in Safety Limit 3.1.

The licensee's related TS governing total primary coolant flow at low power, power peaking, and primary coolant flow orifice position effectively require operations with a power-to-flow ratio less than 1.0. The power-to-flow ratio increases towards 1.0 but does not reach 1.0 as core power is increased to the licensee's authorized maximum power of 82 percent of licensed design power.

The licensee's Procedure SR 5.1.6-W, Issue 29, "Core Safety Limit," requires in Step 5.4.4 that the technical services department supervisor be notified immediately if a core safety limit may have been violated. The licensee's TS 7.2.a requires an immediate reactor shutdown with restart only after NRC authorization, should a safety limit be violated.

The NRC inspectors concluded that the licensee's observed operational practice and procedures precluded violation of the core safety limit. Surveillance of this limit as required by TS SR 5.1.6 is being performed.

The NRC inspectors routinely monitor power-to-flow ratios when the reactor is operating and have detected no times when the power-to-flow ratio exceeded 1.0.

No violations or deviation were identified in the review of this program area.

7. Exit Meeting (30703)

An exit meeting was conducted on February 2, 1989, and was attended by those identified in paragraph 1. At this meeting, the NRC inspectors reviewed the scope and findings of the inspection.