APPENDIX B

U. S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-267/88-12 License: DPR-34

Docket: 50-267

Licensee: Public Service Company of Colorado (PSC)

Facility Name: Fort St. Vrain Nuclear Generating Station

Inspection At: Fort St. Vrain (FSV) Nuclear Generating Station, Platteville, Colorado

Inspection Conducted: May 1-31, 1988

Inspectors: R. E. Farrell 6-10-88
R. E. Farrell, Senior Resident Inspector (SRI) Date

R. E. Farrell for 6-10-88
P. W. Michaud, Resident Inspector (RI) Date

T. F. Westerman for 6-20-88
K. L. Heitner, NRR Project Manager Date

Approved: T. F. Westerman 6-21-88
T. F. Westerman, Chief Date
Reactor Projects Section B

Inspection Summary

Inspection Conducted May 1-31, 1988 (Report 50-267/88-12)

Areas Inspected: Routine, unannounced inspection of operational safety verification, licensee event report review, monthly maintenance observation, monthly surveillance observation, radiological protection, and monthly security observation.

Results: Within the six areas inspected, no violations were identified. One deviation was identified in paragraph 3.

DETAILS

1. Persons Contacted

PSC

D. Alps, Supervisor, Security
*M. Block, Systems Engineering Manager
*F. Borst, Nuclear Training Manager
*L. Brey, Manager, Nuclear Licensing and Resources
*M. Cappello, Central Planning and Scheduling Manager
*R. Craun, Nuclear Engineering Manager
D. Evans, Superintendent, Operations
*M. Ferris, QA Operations Manager
*C. Fuller, Manager, Nuclear Production
*J. Gramling, Supervisor, Nuclear Licensing Operations
*M. Holmes, Nuclear Licensing Manager
*F. Novachek, Nuclear Support Manager
*R. Sargent, Assistant to Vice President, Nuclear Operations
*L. Scott, QA Services Manager
*N. Snyder, Maintenance Department Manager
*P. Tomlinson, Manager, QA
R. Walker, Chairman of the Board and CEO
*D. Warembourg, Manager, Nuclear Engineering
*R. Williams Jr., Vice President, Nuclear Operations
W. Woodard, Health Physicist

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

*Denotes those attending the exit interview conducted June 9, 1988.

2. Plant Status

The reactor was operating at 80 percent power level at the close of the inspection period. The reactor was critical 31 percent of the inspection period with turbine generator capacity factor of 13.8 percent for the inspection period. The reactor scrammed on May 6, 1988, following a helium circulator trip caused by a control systems malfunction. The reactor was again taken critical on May 18. The turbine generator was returned to service May 26. The reactor was run below turbine generator service levels from May 18-26, 1988, for reactor coolant cleanup.

During the inspection period the licensee implemented a reorganization of its nuclear operations. This has been a much talked about reorganization, which has been in the planning stage for almost 2 years. The NRC inspectors will closely monitor licensee's activities as the reorganization takes effect.

3. Operational Safety Verification (71707)

The NRC inspectors reviewed licensee activities to ascertain that the facility is being operated safely and in conformance with regulatory requirements and that the licensee's management control system is effectively discharging its responsibilities for continued safe operation.

The NRC inspectors toured the control room on a daily basis during normal working hours and at least twice weekly during backshift hours. The reactor operator and shift supervisor logs and Technical Specification compliance logs were reviewed daily. The NRC inspectors observed proper control room staffing at all times and verified operators were attentive and adhered to approved procedures. Control room instrumentation was observed by the NRC inspectors and the operability of the plant protective system and nuclear instrumentation system were verified by the NRC inspectors on each control room tour. Operator awareness and understanding of abnormal or alarm conditions was verified. The NRC inspectors reviewed the operations order book, operations deviation report (ODR) log, clearance log, and temporary configuration report (TCR) log to note any out-of-service safety-related systems and to verify compliance with Technical Specification requirements.

The licensee's management representatives were observed in the control room on a daily basis prior to the beginning of the day shift.

The NRC inspectors verified the operability of a safety-related system on a weekly basis. The reserve shutdown system, helium purification system, prestressed concrete reactor vessel (PCRV) penetration purge flow, and control rod drive motors and purge flows were verified operable by the NRC inspectors during this report period. During plant tours, particular attention was paid to components of these systems to verify valve positions, power supplies, and instrumentation were correct for current plant conditions.

Shift turnovers were observed at least weekly by the NRC inspectors. The information flow appeared to be good, with the shift supervisors routinely soliciting comments or concerns from reactor operators, equipment operators, and auxiliary tenders.

The NRC inspectors responded to the control room following a reactor scram on May 6, 1988. The scram occurred after "B" helium circulator tripped due to an upset in the Loop 1 bearing water surge tank. Approximately 2 minutes later a reactor scram on high hot reheat steam temperature occurred. This was due to a failure of the cold reheat steam attemperation flow to automatically increase following the circulator trip. Following the trip it was discovered that the hot reheat temperature signal supplied to the overall plant control system had drifted such that it was reading 35°F below actual hot reheat steam

temperature. The plant protective system, which was verified to have been reading actual hot reheat steam temperature, thus saw the actual hot reheat steam temperature reach the scram setpoint before the control system detected anything abnormal.

Because of recent problems with radioactive releases via the core support floor vent following a plant trip, the reactor operators were attempting to reduce primary coolant pressure as fast as reasonably possible following the trip. The rate at which the primary coolant was being vented exceeded the capacity of the low temperature absorber in the helium purification system. This caused the low temperature absorber to heat up and off-gas into the coolant which was being vented into the helium storage bottles. The off-gassing of radioactive noble gases resulted in radiation levels as high as 125 mr/hr in the helium storage bottle area, which was quickly posted as a high radiation area. The NRC inspectors verified the dose rate at the posted boundaries was within the requirements of 10 CFR 20.202. The situation was quickly recognized and the rate of primary coolant depressurization was reduced. It was subsequently determined that the corrective actions taken to address the problem with the core support floor vent were successful in precluding the need to rapidly depressurize the PCRV.

During the PCRV depressurization, the operating purified helium compressor tripped. This resulted in a loss of buffer helium makeup which, in turn, resulted in a small amount of primary coolant flowing down the shaft of "C" and "D" helium circulators. This primary coolant then mixed with bearing water and ended up in the reactor building. It was subsequently released to the atmosphere via the plant vent stack. The reactor building atmosphere was measured at 3.6 E-8 microcuries per CC, and the accumulated dose at the site boundary was calculated to be 1.94 E-7 Rems.

The NRC inspectors were in the control room immediately following the reactor scram and remained there until the plant was stabilized. The licensee's operations personnel were observed to be in control of the plant at all times and responded to the event in a professional manner.

The licensee in a power ascension on May 26, 1988, experienced difficulty in stroking valves HV-2292 and HV-2293. These valves are hydraulically operated valves which direct main steam flow from the startup bypass system to the main steam bypass system in preparation for starting the turbine generator. These two valves must stroke closed during power ascension after the main steam temperature reaches 760°F. The hydraulic actuator on each of these valves is equipped with thermal relief valves to protect the actuator against hydraulic pressure surges. It was a thermal relief valve on HV-2292 failing to reset which initially led to the fire experienced October 2, 1987.

The problem experienced during this inspection period with HV-2292 and HV-2293 was with the thermal relief valves associated with the hydraulic actuators. A relief valve on HV-2292 was leaking slowly and was subsequently replaced. This is normal after very few strokes of the

relief valve. The relief valve on HV-2293 failed to reseat after HV-2293 was stroked closed. The oil flow relieving through the thermal relief valve was stopped by reopening HV-2293. The thermal relief valve was replaced, and HV-2293 was again closed. Again the thermal relief valve lifted, failed to reseat, and relieved oil to the oil recovery system. Again the oil flow was stopped by opening HV-2293. The licensee noted that the oil flow through the thermal relief valve on the actuator of HV-2293 was greater than expected. There is a flow orifice 27 mils in diameter installed upstream of these thermal relief valves to restrict this oil flow. Additionally, restricting this oil flow serves to protect the relief valves and prolong service life of the relief valves. In October 1987, it was discovered that the orifice upstream of the thermal relief valve on HV-2292, which led to the fire, was not installed and consequently the oil flow experienced was greater than could be handled by the oil recovery system.

While NRC inspectors watched, the thermal relief valve and associated piping was disassembled on HV-2293. The flow orifice upstream of the relief valve was missing. The licensee recovered the documentation of the disassembly, inspection, and reassembly of HV-2293 following the October 1987 fire and verified that mechanics had installed the flow orifice and that quality control inspectors had witnessed the installation. Additionally, the parties involved who had actually performed the work and inspection in 1987 were interviewed by the licensee and insisted that they had indeed performed this work as documented. The licensee stated to the NRC inspectors that the men involved were considered to be particularly reliable. The mechanics probed the connecting piping of the hydraulic actuator and did find the flow orifice. This flow orifice is a screw which screws into a fitting and has a 27 mil diameter hole drilled in it. Apparently this screw had vibrated out of its fitting and fallen into a portion of the actuator piping where it could not perform its function.

The licensee reinstalled the orifice and reassembled the actuator on HV-2293. HV-2292 and HV-2293 were successfully stroked closed and power ascension continued. The licensee is preparing to disassemble and inspect all hydraulic actuators with this type of orifice and will take corrective steps to assure that the orifices are installed in a manner which precludes vibrating out of position. The NRC inspectors will follow this action and this is considered an open item (267/8812-01).

By letter dated July 10, 1985, the licensee committed to follow certain Interim Technical Specifications concerning reactivity control at FSV. Interim LCO 3.1.1.C states for a control rod drive (CRD) to be considered operable, there must be helium purge flow to each CRD penetration when reactor pressure is above 100 psia. Interim SR 4.1.1.A.2 requires that purified helium flow to each CRD be verified by verifying flow at each subheader. The purpose of these requirements is to limit the upward flow of contaminated helium coolant to the CRD mechanism in the CRD penetrations.

On May 18, 1988, at about 1500 hours and on May 19, 1988, at about 0815 hours, the inspector observed helium flow at the subheaders (FI-11268-3, 4, and 7) was reading zero or below zero. At the same time, flow indication for some individual CRD penetrations read zero or below zero. (These flow indications were read both locally in the reactor building and remotely in the control room.)

At this time the reactor pressure was above 100 psig (172 psig on May 19 at 0815 hours). The inspector observed that the licensee's instrumentation did not indicate compliance with LCO 3.1.1.C. The licensee stated that the instrumentation might not read correctly because of reduced reactor coolant density (approximately 39 percent of full value).

The NRC inspectors requested the licensee provide the surveillance procedure to satisfy the requirements of Interim SR 4.1.1.A.2 and compliance with Interim LCO 3.1.1.C. The licensee provided pages from the reactor building equipment operator round sheet. This sheet required only that the subheader flow be greater than zero in order to satisfy the LCO 3.1.1.C.

The NRC inspector noted that this criteria for surveillance of subheader purge flow does not reflect the reactor's design criteria. Specifically, the licensee's Reference Design Manual, SD-11-6, notes the helium purge flow is to be 5.5 lbs/hr per penetration (at full helium density). The greater than zero criteria would also allow instrumentation error to falsely indicate that there is adequate flow.

The NRC inspectors concluded that the licensee's current surveillance procedure is inadequate.

In subsequent discussions with the inspector, the licensee stated that by reducing all indications to a common basis, approximately 2.5 to 2.8 ACFM of flow was indicated at 170 psia for the total system. By contrast, the control system was set to deliver 7.4 ACFM. Thus, the system was not operating correctly when observed by the NRC inspector. The licensee was informed of the NRC inspector's observation of the apparent malfunction of the control systems for helium purge.

It is not apparent that the licensee has implemented measures to assure compliance with his July 10, 1985, commitment to follow Interim Technical Specification LCO 3.1.1.C. This is an apparent deviation (267/8812-02).

On a tour of the control room, the NRC inspectors found a television set and a connected video tape machine in the kitchen behind the control boards within the control room vital area. The equipment was not in use at the time, but the NRC inspectors immediately interviewed the licensee's operations superintendent as to why this equipment was in the control room. The operations superintendent explained that people on shift could not attend regularly scheduled safety meetings. Consequently, the licensee was video taping the safety meetings and allowing the operating

crew to watch the video [illegible] of the meeting when they could. This activity was taking place [illegible] kitchen behind the control boards in the control room vital area. [illegible] NRC inspectors inquired as to why this activity had to take place in [illegible] ontrol room since the control room operators could not perform t[illegible] technical specification required licensed activities in front [illegible] the control boards and also watch the video tape simultaneously. The operations superintendent agreed and the equipment was removed from the control room vital area to an office area nearby where on shift personnel, who are not required to be in the control room at their stations, can watch the video tapes of the safety meetings during their shifts. The licensee advised the NRC inspectors that persons required by Technical Specifications to be at their stations within the control room will, in the future, watch the video tape of the safety meetings after being relieved from their post.

There has been no time when the NRC inspectors have observed less than minimum Technical Specification required manning in front of the control panels in the control room. At no time have the NRC inspectors observed on-duty licensed reactor operators watching television or participating in other activities that would divert their attention from their duties.

During tours of the facility, the NRC inspectors noted that the average age of deficiency report tags (DRTs) appears to be growing. The DRT system developed by the licensee identifies equipment already logged as requiring maintenance and also allows observers to trace the maintenance requests pending to repair the equipment. However, the NRC inspectors noted that the requests once generated do not appear to be closed in a timely manner. Observed examples are as follows.

- DRT 004061 on valve V-6223 is dated September 3, 1986.
- DRT 004062 on valve V-6222 is dated September 3, 1986.
- DRT 010055 on valve V-91141 is dated January 26, 1988. This is a hydraulic system valve and the DRT documents a missing hand wheel. The hand wheel was still missing on May 6, 1988. The NRC inspector noted that a missing hand wheel on a hydraulic system valve aggravated the fire experienced October 2, 1987. The valve part is on back order.
- DRT 004450 on M-82, a broken box protecting instrument valves, is dated January 25, 1987.
- DRT 005054 on valve HV-2189-8 dated August 5, 1987.

The licensee has been informed of the NRC inspector's observation of the excessive time required to close DRTs. The licensee is considering actions to address this issue. This item is considered an open item (267/8812-03)

On a tour of the 480 V switchgear room, the NRC inspectors noted that some electrical cable conduits were quite warm to the touch. The licensee's

shift supervisor, maintenance manager, and electrical maintenance supervisor were interviewed regarding these conduits. This condition had previously been identified by licensee personnel. The licensee's engineering organization has determined that the cable in these conduits is qualified to a higher temperature than can be tolerated by human touch. Consequently, the cables were within their qualified parameters. The licensee also advised that the particular conduits carried power cables to a bearing water pump motor. This is a large 480 V motor and the conduit temperature is expected to be warm when the motor is running.

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

4. Review of Licensee Event Reports (LERs) (90712)

The NRC inspectors reviewed the LERs listed below during this inspection period. This review verified that each LER was submitted within the required time, the description of the occurrence is accurate, a root cause was established where possible, and the corrective actions taken or proposed are appropriate. The five LERs reviewed were found to be acceptable in these areas. The LERs are:

- LER 88-07, Surveillance Procedure not Performed Within Technical Specification Interval Due to Error in Computer Scheduling Program
- LER 88-06, Expansion Joint Failure Causing Loss of Circulating Water Resulting in a Manual Scram
- LER 88-05, Neutron Flux Rate of Change High Scram (while shut down)
- LER 88-04, Manual Scram Due to Power Grid Fluctuations
- LER 87-23, Revision 1, HV-2292 Oil Leak Caused Fire and Manual Scram

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

5. Monthly Maintenance Observation (62703)

The NRC inspectors monitored the licensee's efforts to troubleshoot the prestressed concrete reactor vessel (PCRV) penetration interspace purge flow indication, FI-11263, located in the control room. This instrument is used to verify compliance with Technical Specification 4.2.7, which requires the interspace between primary and secondary PCRV penetration seals to be pressurized. A purge flow of purified helium maintains this pressurization.

DRT 9282 identified a problem with flow indication FI-11263 cycling between 0 and 3 ACFM. The flow element was removed and cleaned, which returned the instrument to its normal indication of cycling between 1 and 2 ACFM. The licensee's system engineer then performed a test of an

electrical dampening circuit which took a 2-minute average of the normally cycling signal and gave a time averaged, steady indication. This test was performed under test procedure T-362, which the NRC inspectors reviewed and found acceptable. The purpose of this test was to prove the feasibility of modifying the instrument to provide a more steady, but still meaningful, indication. Based on the licensee's evaluation of the test data, a permanent change notice is being developed to modify the circuitry for this instrument. The NRC inspectors will monitor the licensee's progress in this area.

The NRC resident inspector followed the licensee's actions to check the operation of the Loop 1 bearing water surge tank level control system following the plant trip on May 6, 1988. A level excursion in the Loop 1 bearing water surge tank was the cause of the loss of "B" helium circulator, which was followed by a reactor scram. Troubleshooting efforts discovered the Hi-Hi level dump valve on the surge tank was opening at an approximately 18 inch level. The surge tank level is normally controlled at 17 inches with the Hi-Hi level dump valve set at 23 inches. The licensee was not able to establish a reason for the Hi-Hi level dump valve setpoint being as found. The fact that it was operating at that point does explain how a level excursion could have occurred under normal operations. The NRC inspectors verified both the Loop 1 and Loop 2 bearing water surge tank level controls were calibrated prior to returning to power operation.

During power ascension, the fluid in the main steam lines at FSV goes from water to wet steam to superheated steam depending upon the power level at the time. Consequently, the safety relief valves in the main steam lines are designed to handle water, wet steam, or superheated steam. The particular design of these safety relief valves causes them to perform best and to be most leak tight when exposed to superheated steam. The manufacturer does recommend gagging leaking valves when the working fluid is water or wet steam. There are three safety relief valves on each main steam line and one power operated relief valve on each main steam line. Each of the safety relief valves (SRV) will pass approximately 34 percent of the full flow of its associated main steam line.

On May 25, 1988, the NRC inspectors noted that SRV V-2214 was gagged in main steam Loop 1 and SRV V-2245 was gagged in main steam Loop 2. The NRC inspectors interviewed the maintenance supervisor responsible for gagging the valves and reviewed station service requests (SSRs) 88503122, which authorized gagging V-2245, and SSR 88503082, which authorized gagging V-2214. The NRC inspectors noted that in both cases quality control had inspected the installation of the gags on the safety relief valves and that SSRs for gagging safety relief valves stay open until the gag is again removed. The controlled work instructions (part of the SSR) stated in a note to Step 8, "Gag must be removed before going above 30 percent." The NRC inspectors verified by personnel interview, documentation review, and visual inspection that the gags were removed from the safety relief valves when the facility went above 30 percent power.

The NRC inspectors also reviewed the documentation and witnessed portions of the work of SSR 88500302 "Perform Quarterly Inspection." This was the flushing and quarterly maintenance on the "B" Instrument Air Compressor. The preventative maintenance was being performed according to Procedure MP-7055 (Q), Issue 1, "Quarterly Inspection and Preventative Maintenance, Guardner-Denver Instrument Air Compressor." The SSR also incorporated by Reference Procedure ME-2051, Issue 3, "Guardner-Denver Air Compressor, Coolant System, Chemical Cleaning Procedure Using Rydlyme."

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

6. Monthly Surveillance Observation (61726)

During the course of the inspection period, the NRC inspectors monitored the Technical Specification surveillance logs to assure that Technical Specification required surveillances were current. Additionally, they observed performance of parts of the following surveillances:

- Emergency Diesel Generator Weekly Load Test
- Radiation Monitor Operability Test
- Gaseous Radwaste System Surveillance
- Vital Area Door Alarm Test
- Primary Coolant Chemistry Analysis
- Alternate Cooling Method Diesel Generator Weekly Test

The NRC inspectors reviewed the results of the 10-inch scram tests and back-EMF tests on the control rod drives performed during the course of the month.

The NRC inspectors also met with the licensee's technical staff to review licensee use of a new computer code for doing fuel analyses and accountability required by 10 CFR Part 74.13. The NRC inspectors conferred with the NRR Project Manager. Based on the interviews with the licensee and discussions with the NRR Project Manager, the NRC inspectors have no further questions at this time regarding use of this computer code.

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

7. Radiological Protection (71709)

The NRC inspectors verified that required area surveys of exposure rates are made and posted at entrances to radiation areas and in other appropriate areas. The NRC inspectors observed health physics professionals on duty on all shifts, including the backshift. The NRC inspectors observed the health physics technicians checking area radiation

monitors, air samplers, and doing area surveys for radioactive contamination. The NRC inspectors observed the health physics technicians checking primary coolant chemistry for total oxidants.

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

8. Monthly Security Observation (71881)

The NRC inspectors verified that there was a lead security officer (LSO) on duty authorized by the facility security plan to direct security activities onsite for each shift. The LSO did not have duties that would interfere with the direction of security activities.

The NRC inspectors verified, randomly and on the backshift, that the minimum number of armed guards required by the facility's security plan were present. A 100-percent hands-on search was being utilized throughout the inspection period as the licensee was unable to declare the new metal detector operable.

The protected area barrier was surveyed by the NRC inspectors. The barrier was properly maintained and was not compromised by erosion, openings in the fence fabric, or walls, or proximity of vehicles, crates or other objects that could be used to scale the barrier. The NRC inspectors observed the vital area barriers were well maintained and not compromised by obvious breaches or weaknesses. The NRC inspectors observed that persons granted access to the site are badged indicating whether they had unescorted or escorted access authorization.

The NRC inspectors observed armed response force deployment when badged unescorted individuals attempted to enter areas for which they did not have access. No deliberate attempts to violate access levels were observed. Rather, newly badged individuals have recently shown a propensity to confuse the central alarm station door with the reactor building entrance. The NRC inspectors observed that the security force responded according to the security plan.

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

9. Exit Interview (30703)

An exit meeting was conducted on June 9, 1988, attended by those identified in paragraph 1. At this time the NRC inspectors reviewed the scope and findings of the inspection.