APPENDIX

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-267/86-16 License: DPR-34

Docket: 50-267

Licensee: Public Service Company of Colorado (PSC)
P. O. Box 840
Denver, Colorado 80201

Facility Name: Fort St. Vrain Nuclear Generating Station

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

Inspection Conducted: May 6-9, 1986

Inspector: for J. P. Jaudon 5/22/86
R. E. Farrell, Senior Resident Inspector (SRI) Date

Approved: J. P. Jaudon 5/22/86
J. P. Jaudon, Chief, Project Section A Date
Reactor Projects Branch

Inspection Summary

Inspection Conducted May 6-9, 1986 (Report 50-267/86-16)

Areas Inspected: Special, unannounced inspection of operation in excess of authorized power limit.

Results: Within the areas inspected, one violation was identified (paragraph 2).

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DETAILS

1. Persons Contacted

Principal Licensee Employees

*H. Brey, Manager, Nuclear Licensing and Fuels
*M. Deniston, Shift Supervisor
*D. Evans, Superintendent Operations
*J. Gahm, Manager Nuclear Production
*D. Rodgers, Planning & Scheduling Manager
*J. McLotter, Lawyer, Kelly, Stansfield, and O'Donnell
*R. Walker, President and Chief Executive Officer

NRC/NRR Personnel

*F. Allenspach, Reactor Systems Engineer
*R. Farrell, Senior Resident Inspector
*J. Gagliardo, Chief, Reactor Projects Branch
*R. Ireland, Chief, Engineering Section
*J. Jaudon, Chief, Reactor Projects Section A
*E. Johnson, Director, Division Reactor Safety Project
*D. Powers, Enforcement Officer
*M. Skow, Project Engineer

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

*Denotes those in attendance during the Enforcement Conference held in Arlington, Texas on May 9, 1986.

2. Operation in Excess of Authorized Power Limit

At 1009 MDT, on May 6, 1986, the NRC SRI was contacted by FSV operations management and informed that there had been a small perturbation with the reactor. The SRI was informed that based on preliminary information then available, the reactor power level had "drifted" above 35% power to a possible maximum of 40% power. The SRI, was additionally told, that the time above 35% power was approximately 10 minutes. The following is a narrative description of the actual event as determined from plant personnel interviewed, computer printouts, recorded plant parameters, and inspection of control room strip chart recorders.

At 0820 MDT, on May 6, 1986, the reactor was in steady-state operation with the following parameters:

- core power = 34.7%
- generator output = 90 megawatts electric

- core outlet temperature = 1207^{o} F
- reactor pressure = 618 psia
- reactor coolant flow = 47%
- reactor coolant dewpoint = -42.6^{o} F

The licensee had been experiencing a hydraulic fluid leak on a hydraulically operated steam isolation valve. The valve was on a main steam bypass line which is used in startup, prior to admitting steam to the high pressure turbine. With the turbine on line, this valve is normally closed, and main steam follows its normal flow path through the high pressure turbine. With the valve open, steam is bypassed to the desuperheater and the flash tank and hence to the cold reheat steam line without passing through the high pressure turbine. Cold heat steam drives the circulators and then is reheated in the steam generator. A decision had been made to remove hydraulic oil pressure from this valve and to repair the hydraulic fluid leak, utilizing a downstream pressure control valve as the steam isolation valve because the licensee anticipated that the isolation valve could drift open when hydraulic fluid pressure was removed.

The pressure control valve in the main steam by-pass line was closed and hydraulic fluid pressure to the isolation valve in this line was removed to facilitate repair of the hydraulic fluid leak. Following removal of hydraulic fluid pressure from the main steam by-pass isolation valve, the control room noted a step increase in feed water flow. Concurrently, the control room received a Delta Temperature Alarm between reactor Loop I and reactor Loop II. That is, the steam coming from the steam generator for one loop was at a temperature different than that from the other steam generator. A shift supervisor who was acting as operations superintendent was in the control room and directed the reactor operators to increase helium circulator speed on the loop with the lower temperature to equalize the steam outlet temperatures of the two steam generators. This increased helium flow to the core, increasing heat removal from the core and raising core power. At this point, the reactor was already above its authorized power level of 35%, and operator actions taken were tending to increase power level even further. The reactor was in the remote automatic control mode, in which all control systems follow main turbine steam demand. Following the step increase in feedwater flow, turbine generator power increased, the center control rod (regulating rod) stepped out to match turbine demand, and reactor power increased peaking between 42.6% 43.3% reactor power at approximately 0840 MDT. The shift supervisor (acting operations superintendent) in the control room made the decision to bring the plant down below its authorized power level of 35% in a "slow controlled manner," rather than by scram of the reactor or by shifting the control mode to local automatic to reduce power rapidly.

When the reactor operators noticed the turbine generator output had increased from 90 megawatts electric to approximately 108 megawatts electric, the shift supervisor directed them to reduce turbine generator load demand in an attempt to reduce reactor power to within the authorized operating limit of 35%. As the reactor operators reduced turbine generator demand, the regulating rod inserted, but reactor power continued to increase as did feed water flow. The negative thermal coefficient of reactivity inherent in the reactor core served to increase reactor power as feedwater flow increased, and additional heat was removed from the core.

Control room personnel who were analyzing the event to determine why feedwater flow and reactor power did not follow turbine generator demand, identified the work on the main steam by-pass line valves as the apparent cause. At this time maintenance was contacted, and hydraulic oil pressure was restored to the main steam by-pass line isolation valve, causing that valve to close. Reactor power dropped as did feedwater flow to correspond to the power demanded by the turbine generator. At 0915 MDT, reactor power was again below the authorized limit of 35%.

With the reactor in remote automatic control, the control loops in the plant regulate equipment to match turbine generator load demand, taking the signal from the high pressure turbine throttle pressure. When the hydraulic pressure was removed from the main steam by-pass line isolation valve, that valve partially opened, passing high pressure steam to the by-pass line. The pressure control valve in the main steam by-pass line was being relied upon to function as an isolation valve, even though it is not designed to provide such service; this pressure control valve either leaked or was not completely seated, and it passed high pressure steam. The steam flow through the by-pass line reduced turbine throttle pressure causing the control systems to demand more steam and power from the reactor. Additionally, the steam passing through the flash tank and into the reheat lines, increased reheat steam flow. Since a reheat steam flow signal is utilized to control feed water flow, an increase in reheat steam flow caused an increase in feed water flow. Consequently, the feedwater system was getting an increase flow, signal from the turbine generator control system, which was sensing an inadequate turbine throttle pressure and was getting an increase feed water flow signal from reheat steam flow. At the same time the turbine generator control system was demanding more feedwater flow, it caused the regulating rod to withdraw, increasing reactor power to match the turbine generator load demand. The increase in feedwater flow cooled the reactor core and caused an increase in reactor power due to the negative thermal coefficient of reactivity.

The effect of the negative thermal coefficient of reactivity held reactor power above 35% even as turbine generator load demand was reduced by the reactor operators. Reactor power returned to authorized levels when the flow through the main steam by-pass line was terminated, and the turbine generator remote automatic controls were again sensing all of the steam flow provided from the reactor.

The licensee, while operating at 34.7% reactor power with an authorized reactor power limit of 35%, chose to do maintenance on a valve which had the potential for effecting reactor power level. While doing this work, the licensee relied upon a pressure control valve, not designed as an isolation valve, to perform an isolation function. The pressure control valve utilized as an isolation valve was not tested for isolation capability prior to relying upon it for isolation. When the first alarm, the Delta T on main steam alarm, was received, the reactor operators were directed to take action which increased reactor power in order to clear the alarm.

The reactor was operated at power levels in excess of the authorized power level from 0830 MDT, May 6, 1986, to 0915 MDT, May 6, 1986. Although there was no apparent intention to exceed the authorized power level, the licensee did choose to do maintenance on a valve which had the potential to affect reactor power level without reducing reactor power level to provide a margin of error in the event power level did increase. Additionally, the licensee relied upon an untested valve to perform a function for which it was not designed. When the licensee realized that the reactor was not operating within the authorized power limit, the first actions caused reactor power level to increase further. Additionally, the licensee chose to remain at an unauthorized power level for 45-minutes rather than manually scramming the reactor or quickly reducing reactor power, risking an automatic scram.

Operating the reactor at power levels above 35% is an apparent violation of the NRC Order of November 26, 1985, authorizing operation of the FSV reactor at power levels not to exceed 35% (50-267/8616-01).

3. Enforcement Conference

An Enforcement Conference was conducted on May 9, 1986, at the NRC Region IV offices in Arlington, Texas. At that time, the events of May 6, were reviewed and potential actions to prevent reoccurrence were discussed. The meeting was attended by those indicated in paragraph 1.