PDR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OCT 16 1984

Mr. R. F. Walker, President
Public Service Company of Colorado
P. O. Box 840
Denver, Colorado 80201

Dear Mr. Walker:

In early July I directed my staff, with assistance from Region IV, to perform an audit of Fort St. Vrain operations including problem areas associated with the June 23, 1984 event regarding the failure of a number of control rods to insert.

A preliminary report (copy attached) has been developed by the staff which documents the results of our assessment. The report contains findings that the staff believes should be implemented before and after station restart. These findings are contained in the Executive Summary and at the end of each section in the body of the report. The report is preliminary in that various options to solve staff's findings are available to the licensee and need to be discussed prior to final resolution.

Public Service Company of Colorado should review and evaluate the report and determine what followup actions are appropriate. We intend to schedule a meeting to discuss your proposed actions to resolve these findings and will be in contact with you to schedule such a meeting.

Sincerely,

HR Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc: See next page

Public Service Company of Colorado

C. K. Millen
Senior Vice President
Public Service Company
of Colorado
P. O. Box 840
Denver, Colorado 80201

Mr. David Alberstein, 14/159A
GA Technologies, Inc.
P. O. Box 85608
San Diego, CA 92138

J. K. Fuller, Vice President
Public Service Company
of Colorado
P. O. Box 840
Denver, Colorado 80201

G. L. Plumlee
NRC Senior Resident Inspector
P. O. Box 640
Platteville, Colorado 80651

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

R. D. Martin, Regional
Administrator, Region IV
Parkway Central Plaza Building
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

Chairman, Board of County Comm.
of Weld County, Colorado
Greeley, Colorado 80631

Regional Representative
Radiation Programs
Environmental Protection Agency
1860 Lincoln Street
Denver, Colorado 80203

J. W. Gahm
Nuclear Production Manager
Public Service Company of Colorado
P. O. Box 368
Platteville, Colorado 80651

Albert J. Hazle, Director
Radiation Control Division
Department of Health
4210 East 11th Avenue
Denver, Colorado 80220

Kelley, Stansfield & O'Donnell
Public Service Company Building
Room 900
550 15th Street
Denver, Colorado 80202

O. R. Lee, Vice President
Public Service Company
of Colorado
P. O. Box 840
Denver, Colorado 80201

PRELIMINARY REPORT

RELATED TO THE RESTART AND CONTINUED OPERATION OF

FORT ST. VRAIN NUCLEAR GENERATING STATION

Docket No. 50-267

Public Service Company of Colorado

OCTOBER 1984

~~Dup 8410190430 XA~~

ABSTRACT

This preliminary report documents the NRC's special assessment of the operation of Fort St. Vrain Nuclear Generating Station (FSV). This assessment was requested by the Director, Office of Nuclear Reactor Regulation as a result of his review of several operating problems that occurred at Fort St. Vrain. This plant, which is located in Weld County, Colorado, is owned and operated by the Public Service Company of Colorado (PSC). Based on the results of this assessment, which found problems in each audit area, the staff has concluded that Fort St. Vrain not be restarted until all control rod drives have been inspected and refurbished or until modifications have been made or other appropriate corrective actions have been taken, that will provide reasonable assurance that the control rods will scram automatically on receipt of a scram signal. This report is preliminary in that various options to solve the staff's findings are available to the licensee and need to be discussed prior to final resolution.

In order to correct other deficiencies confirmed by this special assessment, it is required that the licensee develop a comprehensive program for determining the underlying cause for these deficiencies and correcting them. This program should be aimed at reviewing the Public Service Company of Colorado management structure and practices relative to the operation of FSV, and should use a third party, consulting group to perform the analysis. The effort should be independent to the extent that the consulting group completes its evaluation and recommendations and submits them to PSC and the NRC simultaneously. The effort should be completed in a manner whereby the consulting group's effort is not biased by PSC. The structure and overall schedule for this comprehensive program should be committed to by the licensee prior to restart.

TABLE OF CONTENTS

APPENDICES

LIST OF FIGURES

LIST OF TABLES

EXECUTIVE SUMMARY

This report documents the U.S. Nuclear Regulatory Commission's (NRC) assessment of the operation of Fort St. Vrain Nuclear Generating Station. The assessment was initiated by the Director, Office of Nuclear Reactor Regulation, as a result of his review of several operating problems that occurred at Fort St. Vrain.

In addition to various reviews and visits to the plant by Region IV on July 9-11, 1984, the NRC staff audited the overall operation of Fort St. Vrain Nuclear Generating Station. A further plant visit was conducted on August 1-3, 1984. The staff defined as areas for review: (1) the failure of 6 of 37 rod pairs to automatically insert on a scram signal on June 23, 1984; (2) the overall conduct of operations; including maintenance and housekeeping; (3) assessment of existing Technical Specifications; (4) the continued water ingress problem; and (5) the construction and utilization of Building 10. Each area is described in detail in the body of this report with the exception of item 5, construction and utilization of Building 10. This item will be addressed at a later date as a specific licensing and/or inspection issue. The August visit was performed to audit the performance of control rod instrumentation in response to observed anomalies.

The staff concluded that FSV shall not be restarted until weaknesses in the areas audited are addressed. Specifically, prior to restart, PSC should complete the following:

1. <u>Actions Required for Control Rod Problems (See Sections 2 and 3 for Details)</u>

   a. Ensure that future scram signals will result in all rods automatically being inserted into the core. The licensee must identify the failure mechanism and take corrective action for the rods that did not scram; or if the cause cannot be positively identified through examination or analysis of the drive mechanisms and the circumstances of the failure, other compensatory measures must be taken to provide assurance of reliability of control rods. These measures could reasonably include refurbishing all drive mechanisms. Regardless, of any other measures taken to remedy the failure to scram problem prior to reactor restart, PSC must outline and commit to periodic inspection/preventive maintenance and surveillance programs for control rod drives and associated position instrumentation.

   b. Implement procedures to prevent overdriving the control rods past the rod-in limit.

   c. One 30-weight percent and one 40-weight percent reserve shutdown hopper should be functionally tested to assure that the reserve shutdown capability is fully available.

   d. Until the long term corrective actions are completed, the licensee should develop a procedure that will require a reactor shutdown under conditions where purge flow is lost or when high levels of moisture exists in the coolant.

   e. Implement a procedure for recording representative samples of CRDM temperatures at all operating conditions until continuous recording capability is available.

2. Actions Required to Correct Weaknesses Noted in the Area of Overall Conduct of Operations (See Section 4 for Details)

In the area of overall conduct of operations, the staff confirmed the deficiencies noted in various Region IV inspection reports and in the last two SALP reports. The staff has concluded that PSC must develop a comprehensive program for identifying the underlying causes for the deficiencies and for appyling corrective measures. This program should be conducted by a third party consulting organization and should be aimed at reviewing the PSC management structure and practices relative to the operation of FSV with emphasis on correcting deficiencies noted in the various Region IV inspection reports, the last two SALPs and programmatic weaknesses identified in Section 4 of this report. PSC should submit the scope and schedule for this program prior to reactor restart.

3. Actions Required for the Upgrade of Technical Specifications (See Section 5 for Details)

   a. A high priority effort should be undertaken to review and proposes revisisions to the existing Technical Specifications to reduce the likelihood of operator error and/or misinterpretation and correct omissions. The staff has determined that a schedule should be developed by PSC which will reflect completion of the review, revision, and submittal of the proposed Technical Specifications by April 1, 1985.

   b. To improve control rod and reserve shutdown reliability, the licensee shall propose the following changes to Technical Specifications, and implement interim procedures until the Specifications are approved:

      (1) A weekly control rod exercise surveillance program for all partially or fully withdrawn control rods;

      (2) A Limiting Condition for Operations defining control rod operability and the minimum requirements for rod position indication; and

      (3) A Limiting Condition for Operations and a corresponding surveillance test to define and confirm reserve shutdown system operability.

4. Actions on Continued Water Ingress (See Section 2 for Details)

In the area of continued water ingress the staff has determined that PSC must develop a plan to carry out any of those modifications recommended by the PSC "Moisture Ingress Committee" that are determined by PSC to have a high potential to significantly reduce the frequency and severity of upsets involving injection of circulator bearing water into the helium coolant. Any significant reduction would clearly reduce the frequency of plant transients; improve the reliability of overall plant operations; and might, if moisture has an effect, improve the performance of control rod drives. This plan should include a status report to the NRC as part of the annual report on the progress in implementing modifications.

In addition to the above items required for restart, the staff noted several weaknesses that should be corrected on a longer term basis. The licensee must submit schedules within 60 days of restart for completing these items:

<u>Actions Required Following Restart</u>:

a. Provide continuous recording of a representative sample of CRDM temperatures at all operating conditions to provide part of the data necessary for the longer term program noted below (Section 2).

b. Determine whether compensating design and/or operational modifications are needed to minimize moisture ingress to the CRDM cavities and minimize temperatures in the vicinity of the rod drives. In the event that temperatures recorded during plant operation prove to be higher than those for which the assembly was initially qualified, take immediate steps to perform environmental requalification testing of a CRDM assembly or hold temperatures to that for which the CRDM has been qualified (Section 2).

c. The present Watt-meter testing of the shim motor during drive-in and drive-out is not a reliable method to verify full insertion or withdrawal of control rods. This test should be refined or an alternative, reliable test for control rod position verification, must be developed (Section 3).

d. Investigate a design change to provide a positive stop on the CRDM position indicator potentiometer shaft to prevent overtravel (Section 3) and provide the results to the NRC.

e. Conduct an integrated systems study to resolve rod position indication, maintenance and operability questions (Section 3).

f. Establish procedures for verification and sign-off by the Maintenance Quality Control (MQC) of key steps in Technical Specification surveillance procedures (Section 5).

g. Establish a procedure for review and concurrence by the QA organization of safety-related procedures and changes thereto (Section 5).

h. At the time of the audit, the MQC group was reviewing each completed surveillance procedure. The staff concluded that this practice should continue.

i. A review by the QA organization of the content and adequacy of the Technical Specification procedures is important, and the staff has determined that this should be implemented.

1 INTRODUCTION AND CHARTER

Early on the morning of June 23, 1984, following a scram signal at the Fort St. Vrain Nuclear Generating Station (FSV), 6 of 37 rod pairs failed to scram. FSV is operated by the Public Service Company of Colorado (PSC). This event was reported to the NRC by the licensee through the normal reporting channels. During one of the routine briefings on operating reactors following the event, the Director, Office of Nuclear Reactor Regulation (NRR), ordered that an assessment of the overall operation of FSV be performed. Prompting this assessment was the fact that several recent events have high lighted potential safety concerns at FSV. These events include the June 23, 1984, failure to scram; the continued water ingress problem; the maintenance and housekeeping situation highlighted by Commissioner V. Gilinsky; and the question of whether NRC authorization should have been obtained prior to construction of a new building (Building 10) to house vital equipment. NRC staff members from NRR, Region III, Region IV, and consultants from Los Alamos National Laboratory conducted audits of the operations at Fort St. Vrain on July 9-11, 1984, and August 1-3, 1984. This report documents the audits and presents the staff's conclusions and recommendations regarding the restart and continued operation of the plant.

A copy of the Director's memorandum is contained in Appendix A. Appendix B summarizes recent events and Region IV initiatives related to Fort St. Vrain.

2 CONTROL ROD INSERTION FAILURE AND MOISTURE INGRESS

2.1 Introduction

On June 23, 1984, six of 37 control rod pairs failed to insert into the Fort St. Vrain reactor core following a scram signal. The reactor was shutdown without the insertion of these six rod pairs and they were driven into the core without difficulty within about 20 minutes of the scram signal. The failure of control rods to insert automatically is a very serious safety matter and this event was a principal reason for the staff review. In this section, the sequence of events leading to the scram signal is summarized and the control rod drive mechanism (CRDM) failure is reviewed from the perspective of material, thermal-hydraulic, and mechanical properties.

On June 22, 1984, several hours before the scram signal, loss of a 480V vital bus caused an upset of the helium circulator auxiliary system and a consequent pressure imbalance in the helium buffer seal of a circulator; this imbalance permitted circulator bearing water to rise along the circulator shaft, thus introducing moisture into the primary helium coolant. Reactor power was reduced to hold the system temperature low enough to prevent graphite oxidation and moisture removal commenced through use of one of the two helium purification trains. (The other train was unavailable because it was being regenerated.)

Because of the high moisture content in the helium coolant, icing occurred in the nitrogen-cooled, low-temperature adsorber in the purification train causing it to shutdown. This resulted in the loss of all helium purge flow to the CRDM penetrations. Normal helium letdown stopped. However, helium continued to enter the reactor by way of the circulator seals and contributed to a programmed high-pressure trip because of a slowly increasing helium coolant inventory, and the programming down of the reactor pressure setpoint as the reactor power was reduced. The high-pressure trip resulted in reactor scram during which six (Regions 6, 7, 10, 14, 25, and 28) of the 37 control rod pairs did not enter the core (Region 25 control rod pair fell 5.2 inches before stopping). Additional efforts on the part of the operator to cause scram by the removal of power fuses, thus cutting power to individual CRDM brakes, were unsuccessful. The operator then engaged the CRDM motor (normally off) to drive the drum holding the control rod cables, allowing the six control rod pairs to be successfully inserted into the core. The shift supervisor on duty reported the scram to the operations officer at the NRC headquarters in accordance with 10 CFR 50.72 requirements and plant procedures, about one half hour after it occurred. However, the fact that six rods had failed to insert was not reported until several hours later. The failure to report a circumstance of obvious interest and concern to NRC raised questions as to whether there was a violation of reporting requirements.

A special inspection by Region IV personnel disclosed that the shift supervisor had precisely followed operating procedures for scram and subsequent actions for insertion of stuck rods.

Table 2.1 summarizes the overall operating conditions and pertinent information at the time of the scram for those rod pairs not scrammed. The licensee noted that four drive mechanisms (Regions 6, 7, 25, and 28) had been tested the week before the failure. The test consisted of a rod drop of 6 inches followed by withdrawal of the rod pair to its original position. After the failure, retesting of the six assemblies resulted in four successful tests and two (Regions 14 and 25) unsuccessful ones. After further testing, all six control rod drive mechanisms scrammed successfully.

The licensee also noted that two control rod pairs (Regions 7 and 28) had failed to insert in 1982 under comparable environmental conditions of high moisture content and loss of purge flow. The combination of high moisture content in the primary helium and loss of purge flow to the CRDM cavity appears to be a common link between the failure in 1982 and the current CRDM failures.

Since the CRDM failures on June 23, 1984, Fort St. Vrain has been in a shutdown condition. On June 26, 1984 the Administrator of Region IV issued a confirmatory action letter to PSC confirming that the licensee shall not restart the reactor until the control rod problem is resolved and authorization is received from the NRC. The licensee has committed to remove and inspect the six faulty CRDMs, including the drum, gearing, motor/brake assembly, and control rods. If necessary, additional CRDMs may be removed for inspection. As of July 30, the CRDMs in Regions 7 and 14 had been removed, visually inspected to various degrees, and reinstalled.

2.2 Control Rod Drive Mechanism

The CRDM, as described in the Fort St. Vrain's "Updated Final Safety Analysis Report," (Ref. 1) is basically an electrically powered winch that raises and lowers the control rod pair by means of flexible steel cables. A CRDM is housed in the upper cavity of each of the 37 refueling penetrations in the top head of the prestressed concrete reactor vessel (PCRV). Figure 1, as reproduced from the FSAR, shows the general arrangement of the CRDM, which raises and lowers the control rods by winding or unwinding the control rod suspension cables on a common drum having two winding grooves. The principal components of interest in the CRDM include the drive motor, the motor brake, and the reduction gearing.

The CRDM functions are dependent on the operation of the motor/brake assembly and the reduction gear train assembly that actually rotates the cable drum. The brake, a modified Bendix multiple-disk EFL model, is shown in Figure 2 energized and engaged, as it would be in retaining the control rod pair in a stationary position. The energized electromagnet draws the armature plate toward the magnet body, thereby clamping the friction disk, which is constrained from rotation by the spline on the magnet body. The friction disk is keyed by three slots that engage with the arms of the spider, which is keyed to the rotor shaft through the motor drive. When the electromagnet is deenergized, spring plungers push the armature plate away from the magnet body, disengaging the brake and allowing the friction disk, spider, and rotor shaft to rotate freely as the control rods enter the core by gravity.

Figure 3 shows the relation between the motor brake and drive motor assembly. The drive motor is a three-phase, four-pole induction motor. The rotor shaft through the motor connects the brake assembly on one end with the pinion gear and reduction gear train on the other end. Normally, the motor is energized only to move the control rods in our out of the core and is deenergized while the brake is engaged.

The pinion gear on the end of the rotor shaft drives the set of reduction gears shown in Figure 4, which turn the cable drum. The gear ratio between the motor and the drum is 1150. All bearings and gears in the drive assembly are coated with a dry-film molybdenum disulfide lubricant (MoS_2). According to the FSAR, the coefficient of friction for MoS_2 is also not affected by a dry helium atmosphere.

2.3 Control Rod Drive Mechanisms Inspection

The motor/brake assembly from Region 14 showed adherent corrosion on all parts. Water drop patterns were observable on the top of the motor housing and water-staining appeared at the intersection of the center disk and spider in the brake. PSC personnel stated that several other CRDMs had previously shown similar water marks, implying that the CRDMs being examined were probably not very different in appearance from the remainder of the CRDMs in the reactor.

A videotape of the Region 7 mechanism showed the disassembly of the motor/brake assembly and the gears in the reduction train. Rust-type corrosion was evident on most of the CRDM components, but gear teeth surfaces were shiny and appeared to be meshing properly. From the amount of corrosion and its appearance, it can be concluded that the corrosion process has been long term.

An unknown quantity and composition of dusty debris, appearing to be "old" rust, blanketed the CRDM and the housing and was found between the rotor and stator in the motor drive. The black, dusty debris was somewhat magnetic; there was no evidence that it was graphitic. Magnetic debris could be from Fe_3O_4 (a type of corrosion product) or particles worn from the gears. The nonmagnetic grit was possibly from the dry lubricant or from the bronze-impregnated brake plate, if the assumption is made that graphitic material was not present. However, when asked whether the dry lubricant could contribute to the debris, PSC stated that the lubricant was burnished on the CRDM components in a very thin layer, providing lubrication up to 600°F. The effects of moisture on the dry lubricant are being investigated.

The lower portion of the 19 foot long refueling penetration housing (Region 7), as seen in the FSV hot cell facility, showed significant corrosion over the outer cylindrical surface. A very apparent ring of corrosion was on the bottom edge of the housing, as if water had been sitting on the seal between the housing and the refueling penetration liner.

2.4 Metallurgical Evaluation of Control Rod Drive Mechanisms

The gears in the reduction train are made of Nitralloy (0.42% C, 1.60% CR, 0.37% Mo and 1.0% Al), which is a relatively tough alloy. When used for gears, the gear surfaces are generally nitrided to provide greater wear resistance. Bearings of either tungsten carbide (WC) or type 440 C stainless steel were used in a steel race. Type 440 C is a martensitic stainless steel containing nominally 16% to 18% Cr, 0.75% Mo, and 1.0% each of C, Mn, and Si. It is readily heat-treated to very high hardness, and remains hard up to approximately 600°F. The gears and bearings are also lubricated with molybdenum disulfide (MoS_2.), a dry lubricant designed for elevated temperature use. To restrain gear movement, brake pads of bronze Rabestos were used. These pads are made to controlled porosity by powder metallurgy techniques and the pores are filled with the lubricant, in this case MoS_2.

The generally corroded appearance of the CRDM components showed that air and water had been present in the CRDM cavity at one or more times in the past. However, the degree of corrosion does not seem sufficient to affect CRDM operation. PSC commented that there had been no planned preventive maintenance performed on the CRDMs since their installation in 1974.

The particulate matter found in the CRDM cavity may be directly responsible for the CRDM failures. It is not known whether high moisture content in the primary helium could have led to condensation on or above the CRDM, washing down particulate matter onto the gear mechanism, brake/motor, or bearings. If of sufficient size, the particulate matter, possibly agglomerated, could produce a resisting torque sufficient to prevent the motor shaft from turning. However, when PSC deliberately introduced particulate matter and/or moisture into the drive mechanism and bearings, no interference with normal CRDM operation was observed.

Another materials aspect is the possibility that at elevated temperatures, a reduced spring constant coupled with increased remanence of the magnetized disc may have kept the brake from disengaging (remanence is the induced magnetism remaining in a material when the magnetizing force is reduced to zero). This is not considered a likely contributor to the control rod insertion failure.

Because no material failures, nor any consistent material shortcomings were observed, the staff concludes that by itself the adherent corrosion found on the CRDM components does not specifically inhibit the CRDM ability to function. However, the particulate matter, whether originating from corrosion, lubricant, or graphite matter, may be involved in the failure process.

2.5 Thermal-Hydraulic Evaluation of Control Rod Drive Mechanisms

The actual environment experienced by the CRDMs, when the control rods failed to insert, was not considered normal. Prior to scram, the core was being "overblown", i.e. the circulators were being run faster than normal to provide additional cooling; also the primary helium coolant had a high moisture content. The moisture in the coolant caused the helium purification train to ice-up, resulting in a total loss of cool, dry helium purge flow to CRDM cavities. The loss of purge flow allowed reactor coolant to enter the CRDM cavities, raising the moisture levels and possibly the ambient temperatures in the vicinity of the drive mechanisms to abnormally high levels.

Under normal operation, the purge flow (approximately 4 lbm/hr) is introduced at the top of each CRDM cavity, and supposedly flows directly onto the mechanism. However, the analyses previously performed by Los Alamos National Laboratory on the CRDM over-temperature problem, Ref. 3, indicated that even though directed at the CRDM, the purge flow would tend to flow down the water-cooled wall of the CRDM cavity, where it would join heated primary helium leaking in from the upper reactor plenum at the control rod seals and orifice valve motor (OVM) penetrations. The analysis indicated that the heated gas would continue to flow up past the CRDM to form a naturally convective loop. The net result may be that cooling of the drive mechanisms by purge flow is less effective than expected in the design.

If purge flow were eliminated, then primary helium coolant inleakage would increase because of the increased differential pressure across the reactor plenum and CRDM ends of the housing. If inleaking helium coolant had a high moisture content, as it did at the time of the event, then moisture could condense above the CRDM on the cooler top access plate. Condensation and dripping on to the CRDM components would explain the water drop marks found on the inspected CRDM mechanisms.

The loss of purge flow in conjunction with increased inleakage of "hot" primary coolant could cause a temperature increase of the CRDM components. The qualification testing on the CRDM was performed in a 180°F helium atmosphere, resulting in motor temperatures averaging 215°F. The analyses performed by GA Technologies predicted maximum motor safe operating temperatures of 272°F, and a safe operating ancillary CRDM component temperature of at least 250°F. The Los Alamos analysis (Ref 3) indicated that the overtemperature problem was fundamentally caused by the ingress of primary helium into the refueling penetrations, and then through OVM penetrations into the CRDM cavity. The analysis indicated that small increases in in-leakage flow can result in relatively large increases in the CRDM temperatures. The synergistic effects in the CRDM motor of temperatures above 215°F are not known.

The staff noted that the licensee is installing thermocouple instrumentation on each mechanism that is removed for inspection and refurbishment. The staff believes that the licensee should record temperatures under both steady state and transient reactor conditions to acquire temperature data on the CRDMs. Such data would be useful to determine whether the mechanisms may be exposed to excessive temperatures under some conditions.

2.6 Mechanical Evaluation of Control Rod Drive Mechanisms

During normal operation, the electro-magnetic brake assembly is engaged, retaining the control rods in a fixed position. During a scram, the brake is disengaged, freeing the rotor shaft to rotate, and allowing the offset force of the control rod weight (120 lbs per cable) to drop the rods into the core by gravity. However, according to the FSAR, only 16-20 inch-ounces torque at the motor shaft is sufficient to prevent a scram from occurring on command. According to PSC, the free rotational torque of the Region 14 and 7 motor shaft ranged between 9 and 11 inch-ounces during the current inspection program. For this reason, it appears that the introduction of particulate matter at the intersection of the motor shaft pinion gear and the reduction gearing, or the introduction of particulate matter in the motor bearings could produce a resisting torque great enough to prevent motor rotation when the brake is released by a scram signal.

As previously mentioned, the loss of purge flow, and the ingress of hot primary helium into the CRDM cavity could have caused increased CRDM component temperatures. The increased temperature effects could lead to differential thermal expansion problems between components; if sufficient interference develops this also could forestall rod insertion.

In summary, the effects of particulate matter, moisture, and overtemperature on the CRDM appear to be common elements of the failure to insert problem.

2.7 SUMMARY

To date, there is no clearly discernible, common-mode failure to explain why 6 of 37 control rod pairs failed to insert on the scram signal. PSC is still inspecting the remaining CRDMs, but the link between control rod insertion failure and loss of purge flow to the CRDM cavity during a high moisture event cannot be ignored.

From the information available to date, it is believed that the problem lies somewhere between the motor brake assembly, the drive motor assembly, and the pinion gear leading to the reduction gearing, because of the very small resistance needed to overcome the torque imposed at the motor pinion gear by the control rod weight. In the motor brake assembly, moisture, temperature and/or particulate matter could affect the force applied through the brake friction pad and could interfere with brake plates closing or opening. In the motor drive, particulate matter in the bearing raceways could produce sufficient resistance to prevent the start of free motor rotation. There is the possibility that the lubricating properties of MoS_2 may change in the presence of moisture. Particulate matter between the pinion gear and the first mating gear in the reduction train could also present sufficient torque resistance to prevent free rotation. However, particulate matter interference in other gears of the reduction train seems unlikely to cause a problem with free rotation of the gear train because of the much larger torque forces transmitted by rod weight to these gears. All CRDM components may be susceptible to overtemperature problems through differential thermal expansion.

All of the 37 CRDMs are likely to show a degree of component corrosion comparable to that seen on the mechanisms from Regions 7 and 14. The potential for moisture ingress into the CRDM cavities, the effects of moisture on CRDM component behavior and corrosion, especially when related to overtemperature problems, and the effects of particulate matter on and in the CRDMs are being investigated.

2.8 CONCLUSIONS

As a result of its review, the staff has determined that the licensee must complete short-term remedial measures and commit to the scheduling and completion of long-term remedial measures. The staff recognizes that the licensee's investigation is not complete and that the licensee must demonstrate to the NRC, prior to restart of the reactor, that there is reasonable assurance that all control rods will insert automatically following a scram signal.

The staff has concluded that the following actions must be completed prior to restart:

1. Until the failure mechanism is fully understood and long-term corrective actions are taken, the licensee should develop a procedure which will require a reactor shutdown under conditions where purge flow is lost or when high levels of moisture exist in the coolant.

2. A preventive maintenance program should be developed and initiated on all CRDMs and associated cavities to assure inspection, cleaning, and refurbishment as each region is refueled.

3. One 30-weight percent and one 40-weight percent reserve shutdown hopper should be functionally tested prior to reactor restart to assure that the reserve shutdown capability is fully available.

4. A control rod exercise program should be developed for use during reactor operation that requires one complete revolution of the first gear driven by the rotor shaft pinion gear; initially, the frequency should be once a week.

5. The licensee should develop a plan to carry-out circulator system modifications proposed to mitigate moisture ingress by the licensee's Moisture Ingress Committee, which are determined to have a potential to reduce the frequency and severity of upsets involving injection of circulator bearing water into the helium coolant.

6. In addition, the staff concludes that various long-term corrective actions are necessary. These long-term actions are as follows:

   a. Representative instrumented CRDM motor, plate and ambient temperatures should be continuously recorded, regardless of reactor power level, in order to establish a baseline operational history.

   b. The licensee should determine whether compensating design and/or operational modifications are needed to minimize moisture ingress to the CRDM cavities and minimize temperatures in the vicinity of the rod drives. In the event that temperatures recorded during plant operation prove to be higher than those for which the assembly was initially qualified, perform environmental requalification testing of a CRDM assembly.

7. Chemically analyze and evaluate the debris and "rust" on the CRDMs as to particulate size and composition. Likewise, the debris recently found during ultrasonic cleaning of motor bearings needs to be characterized and analyzed.

2.9 References

1. Fort St. Vrain Nuclear Generating Station, "Updated Final Safety Analysis Report", Public Service Co. of Colorado.

2. Gulf General Atomic Drawing No. D1201-217, "Motor and Brake Assembly".

3. Meier, Karl, "Fort St. Vrain Reactor Control Rod Drive Mechanism Over-Temperature Problem", TN-FSV-3, Los Alamos National Laboratory, July 1982.

Table 2.1. Regional and Overall Core Parameters at Scram

Region	Orifice Position, (% open)	Region Outlet Temperature (°F)	Region Peaking Factor
6	18.5	1065	0.75
7	16.6	1087	0.77
10	64.7	988	1.27
14	35.8	980	1.12
25	21.1	1130	1.07
28	45.2	1045	1.38

Core Power	19.8 %
Core Avg Inlet Temperature	606 °F
Core Flow, Total	60.1 %
Reactor Pressure	642 psig
Coolant Moisture Level	1000-1200 ppm (estimated)

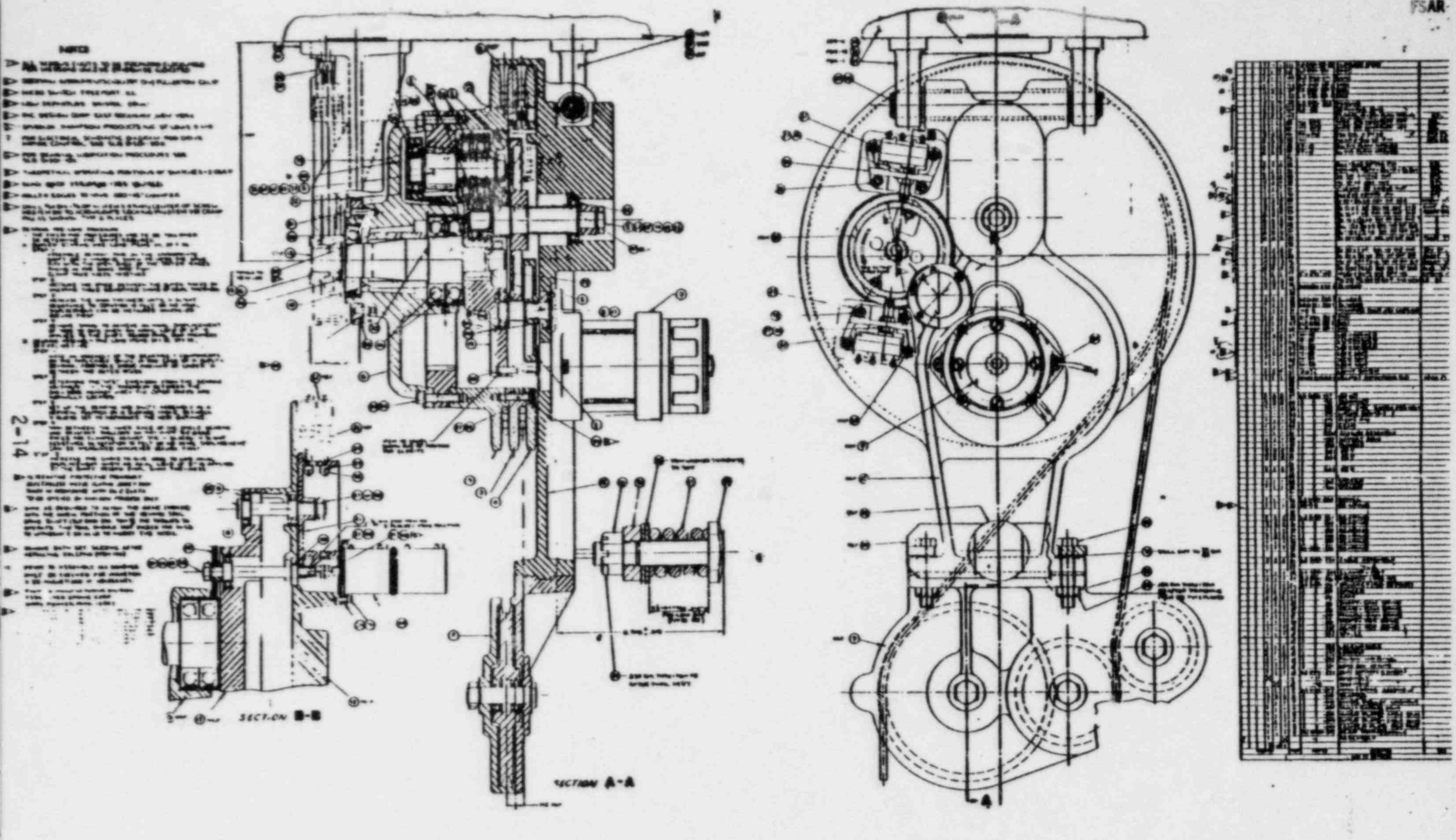


FIGURE 1 CRDM

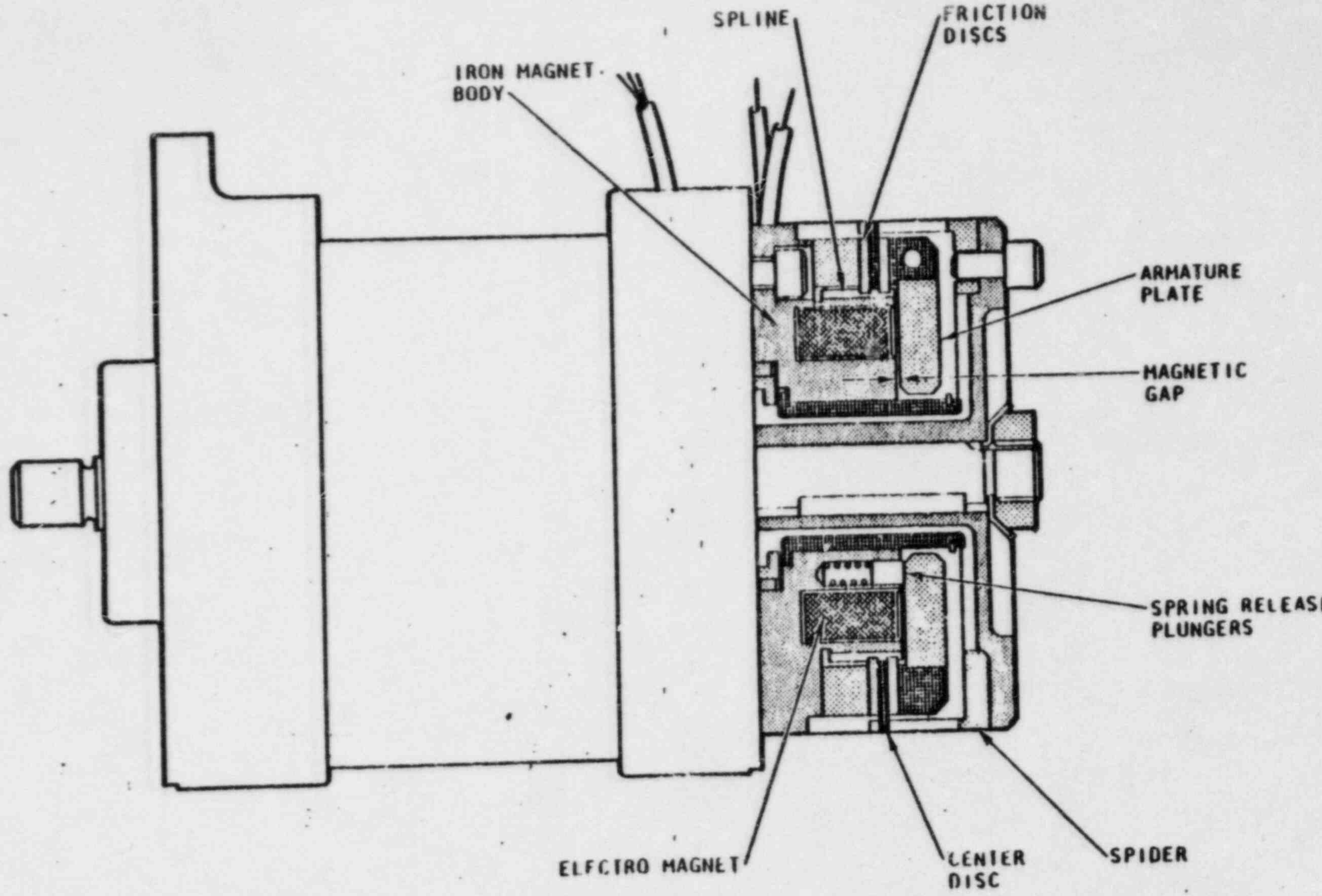


FIGURE 2 **Control Rod Drive Brake**

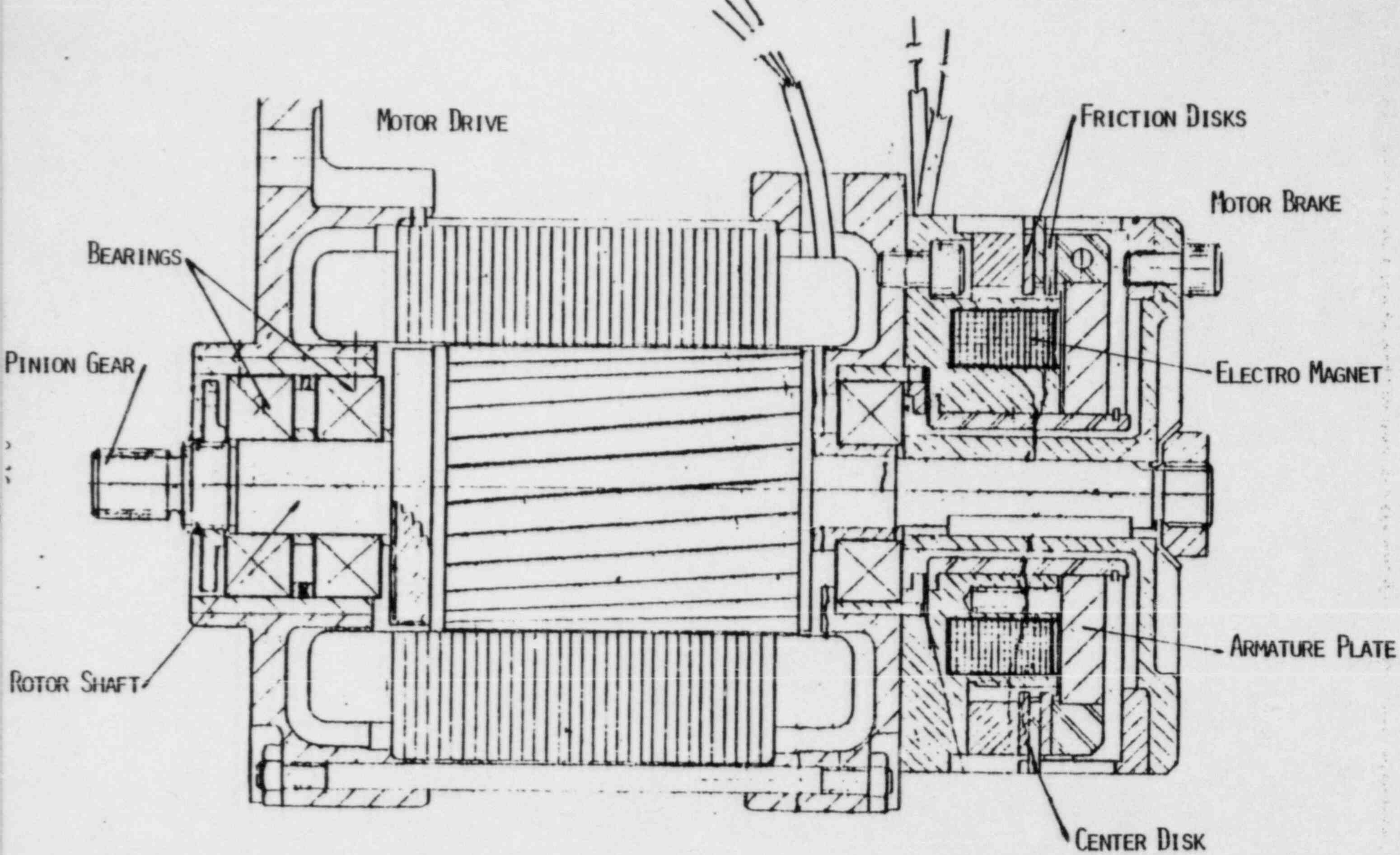


Figure 3 Motor & Brake Ass'y

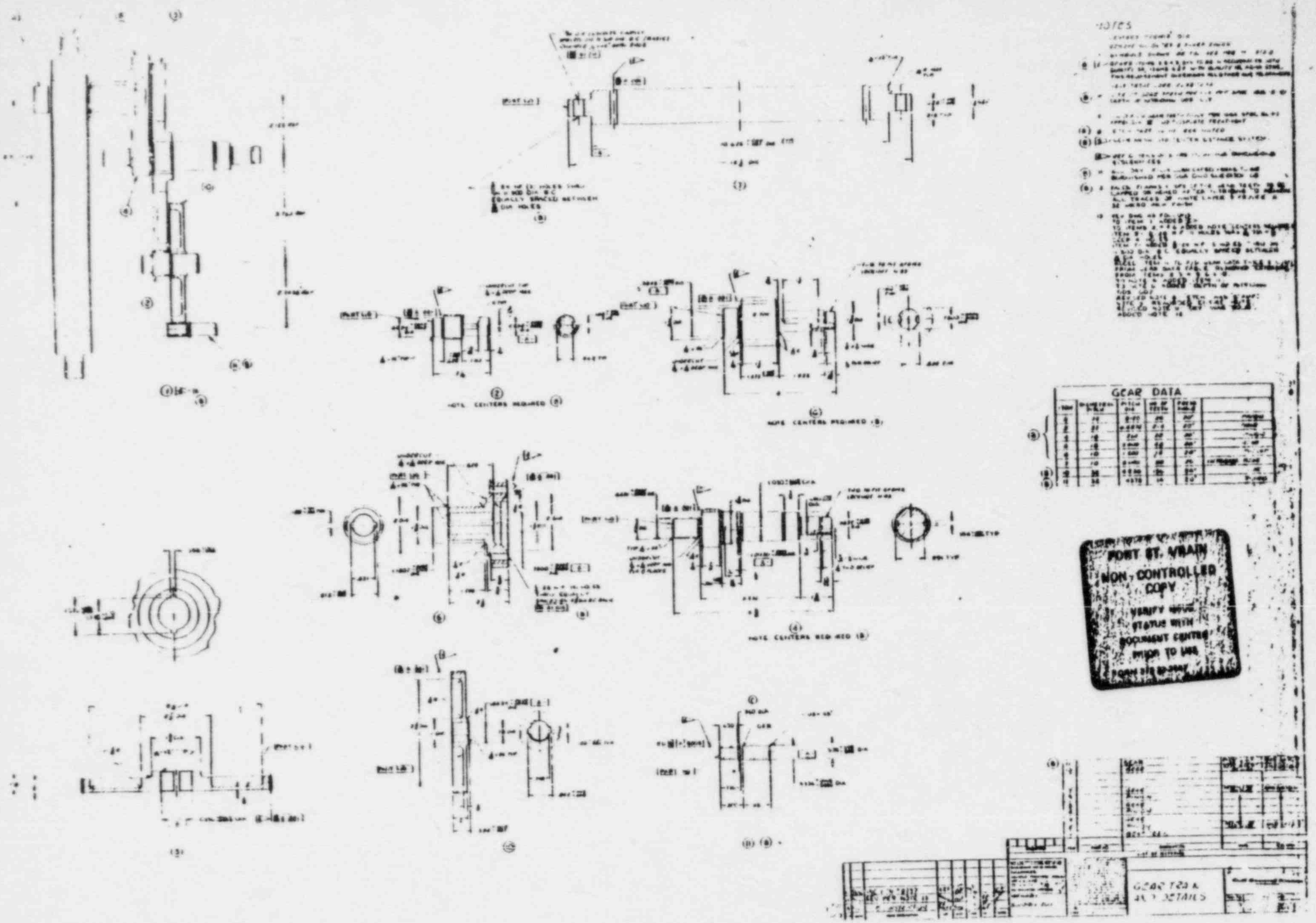


FIGURE 4 REDUCTION GEAR TRAIN

3 CONTROL ROD INSTRUMENTATION ANOMALIES

3.1 Introduction

On July 30, 1984, the NRC was informed of numerous and various control rod instrumentation anomalies in several refueling regions in the reactor. The eleven anomalies included: simultaneous rod-in and rod-out indications, out-limit switch lights remaining lighted, indications of partial rod withdrawal, no position signals, disparity between analog and digital rod position information, and a slack-cable indication. A team of NRC technical personnel and their consultants from Los Alamos National Laboratory, visited the plant site on August 1-3, 1984, to review the instrumentation problems.

This Section reports the results of that plant visit. This Section includes a description of the Control Rod Drive instrumentation characteristics and the anomalies observed, a preliminary evaluation of the anomalies and their effect on verifying control rod position, and recommendations. The results are included in the assessment report because they relate to the overall performance of the CRDMs.

3.2 Control Rod Instrumentation

As described in Section 2 of this report, the principal mechanical components of the CRDM are the shim motor and motor brake assembly, the reduction gearing to the cable drum, and the control rod pair suspended by cables from the drum. Integrated into the CRDM and used to determine control rod positions are the instrumentation-related components that include the rod position potentiometers, rod-in and rod-out limit switches, limit switch cams and gear reducer, and the slack-cable indication device. The relative locations of these components are shown on Figure 3.1.

The control rod potentiometers are intended to provide continuous monitoring of control rod position. As shown in Figure 3.2, both of the ten-turn, potentiometers are directly coupled to the rotation of the cable drum. The coupling is provided by connection of the potentiometer shaft to the cable drum hub. The potentiometer shaft passes through the drum support, and a pinion gear on the shaft drives the limit switch cam wheel. Beyond the pinion, a multi-jaw coupling drives the two potentiometers. Output from the potentiometers is provided to the operator through separate analog and digital readouts in the control room.

As previously mentioned, "rod-in" and "rod-out" position indication is provided by cam-actuated limit switches (the potentiometer shaft shown in Figure 3.2 also drives the cam wheel). At the full rod-in position, the cable anchor on the cable drum and the cam are at the 6 o'clock position, and both rod-in limit switches are actuated. Full rod travel (190 $\pm$ 1 inches) to the rod-out position causes the cam wheel to rotate clockwise 3/4 turn, actuating the pair of rod-out limit switches. Limit switch actuation is indicated by steadily-lighted lamps in the control room.

The slack-cable switch assembly shown in Figure 3.3 is provided for monitoring the tension in the cables supporting the control rod pair. The spring plunger exerts an upward force on the underside of the cable drum spindle, which counteracts the downward force caused by the control rod weight. In the event a control rod cable becomes slack, the upward force overcomes the control rod weight, and microswitches are actuated. Slack cable indication is signalled in the control room by both an alarm and a light.

3.3 Control Rod Drive Instrumentation Review

In response to the information received about instrumentation anomalies the NRC staff and Los Alamos personnel reviewed the overall control rod assembly as it relates to control rod drive performance and control rod position indication. The team requested and received extensive, detailed information on the CRDM and interfacing instrumentation. The major items discussed were:

1. the instrumentation anomalies currently being experienced;

2. the CRDM mechanical interface to the rod position instrumentation, which includes the potentiometer drive gear and shaft acting as the

pinion for the cam wheel drive and the rod-in and rod-out limit switches, a multi-jaw coupling, and the two potentiometers; and

3. the electrical capabilities of the CRDM instrumentation to determine control rod position under normal and adverse conditions.

The instrumentation anomalies observed at the time of the plant visit are summarized in Table 3.1.

From a mechanical perspective, the CRDM instrumentation is indeed directly coupled to the motion of the cable drum, which dictates the movement of the control rods. However, under certain conditions, the mechanical aspects can actually inhibit the performance of the instrumentation. The common shaft controlling both the rod position potentiometers and the limit switches is susceptible to damage by overdriving the control rod pair past the rod-in position. This problem has been encountered when, for some reason, a given control rod pair, when supposedly fully inserted, may not actuate the rod-in limit switch. In an attempt to actuate the switch, the operator would typically try to overdrive the control rods to attain full insertion (even though the operation manuals (Ref. 1) strongly advise against such a maneuver). In this case, the in-limit cam can rotate past the in-limit switch, and the out-limit cam can rotate to interfere with other mechanical components resulting in damage to potentiometer shaft. This damage can occur because neither a positive stop is provided to restrict cam wheel overrotation nor is there sufficient clearance for the cam to overrotate. If the shaft is damaged by overtorquing or shearing, or the multi-jaw coupling is displaced, the two rod position potentiometers at the end of the shaft can give: the same erroneous signal about the actual position of the control rods on both readouts, different analog and digital outputs for the same rod position, or no output signals at all. On the other hand, if the control rods are overdriven by 1/4 turn of the cable drum (about 10 inches), the cable anchor can become wedged in the cable groove of the drum, and then upon withdrawal, the control rod cables may not feed onto the drum correctly, or can become entangled. It is clear that this type of integration of mechanical and instrumentation components can result in a single point failure, can potentially result in damage to the instrumentation, and does not provide an independent indication of control rod full-in position.

Discussions were held with the licensee regarding the electrical interfacing and control of the CRDM and instrumention. Electrical schematics from the "Rod Control System Equipment I-9303, Operation and Maintenance Manual", Ref. 1 and the "Installation, Operation and Maintenance Manual for the Control and Orificing Assembly for the Fort St. Vrain Reactor" Ref. 2, were reviewed by the staff. The licensee stated that none of the control and indication circuitry for the CRDM is classified as safety-related, except for the bypass circuitry used during a scram to energize the shim motor to drive in the control rods.

A number of questions were raised as to the effective redundancy in circuits such as rod-in and rod-out limit switch indications. According to the electrical schematic (Figure 3.4), the rod position limit switches may not be, in effect, redundant, because the loss of one switch of the pair is not detectable. In other words, a single circuit switch failure would go undetected. If this occurs the loss of the second switch will result in complete loss of full-in or full-out indication. The rod-in and rod-out limit switches are integrated into the mechanical aspects of the CRDM to the extent that the same switches are used for both rod drive control and rod position indication.

Purchase specifications were reviewed to determine how the CRDM instrumentation should be expected to perform in normal and adverse environmental conditions. That review indicated that all CRDM instrumentation is of commercial grade, and that no special quality or safety-related specifications are required. No housing or shielding is provided to protect CRDM instrumentation from the existing environment.

Under relatively normal operating conditions, such as just prior to the control rod insertion failure, control rod position instrumentation had been generally operable. Within two days after the control rod insertion failure, all control rods were exercised, and rod position instrumentation was operable. Some two weeks after the insertion failure event the core was depressurized. It was after the depressurization that the broad array of instrumentation anomalies were observed during subsequent control rod exercising. The staff, therefore, believes that the rod position instrumentation is quite likely to become unreliable when subjected to depressurization following exposure to a hot, moist environment. It is likely that condensation of moisture and wetting of electrical components occurred after reactor pressure was reduced.

The staff focused its attention on the specific problems regarding the CRDM located in Region 19, where both the analog and digital readouts indicated that the control rod was withdrawn about 40 inches. When asked if the instrumentation should be believed, the PSC responded that they did not believe the installed rod position readouts, but that they had verified to their satisfaction that the control rod was indeed fully inserted. This conclusion was based on signature traces from watt-meter testing of the shim motor. According to PSC, at some point after depressurization, the Region 19 control rod pair was being exercised, and when the rod pair was fully inserted, the rod-in limit switch indication did not come on. The operator drove the rod pair in further so as to make contact with the limit switch. Again, no contact was indicated. At some point during this attempt to get full rod-in indication, the 40-inch offset was noticed. Therefore, a watt-meter test was performed on the Region 19 shim motor. The wattage reading of the shim motor was recorded as the rod pair was "yo-yo-ed," i.e., a repetitive withdrawal and insertion sequencing. Based on past experience and the Region 19 traces, PSC concluded that the control rod pair was fully inserted. However, when the staff examined the Region 19 traces, the evidence was inconclusive. The inconclusive watt-meter traces in conjunction with analog/digital indications prompted the staff to request that a more definitive verification of the Region 19 control rod position be performed.

The verification technique selected by the licensee consisted of manually retracting the control rod pair from the region, as if in preparation for refueling removal--i.e., the control rods are completely drawn up into the refueling penetration housing. A completely separate limit switch, indicates full control rod retraction to the refueling position. If the control rod were completely inserted, then full retraction would require an additional 50 inches of travel over the normal 190 inches of rod travel (for a total of 240 inches) from the rod-in position, corresponding to a total of 330 turns of the manual rewind tool. On the night of August 2, 1984, the Region 19 control rod pair was withdrawn 330 turns when the full retract position indication was obtained, proving that indeed the rod had been fully inserted. However, it also demonstrated that the installed rod position instrumentation alone is insufficient to determine control rod positions under adverse conditions, such as when mechanical damage may have occurred to rod position instrumentation.

It is evident that control rod analog and digital position indications can be in agreement without reflecting the true rod pair position.

Statistically, no particular instrumentation anomaly tends to be prevalent under these adverse circumstances. A summary of the current status of the control rod drives, instrumentation and orificing, as prepared by PSC, shows that the anomalies tend to be evenly distributed among rod limit switching problems, digital and/or analog readout problems from rod position potentiometers, with the exception of the single slack-cable related problem. This pattern is in contrast to the historical distribution, where rod position potentiometer problems had been dominant. In reviewing past instrumentation anomalies as recorded on PTRs (Plant Trouble Report) since plant startup, approximately 47% of the anomalies are rod position potentiometer problems, about 17% are limit-switch related, less than 2% are related to slack-cable indication, 17% are connected to orifice valve problems that are not germaine to this review, and 19% are related to motor/brake and other anomalies. In a majority of the cases, the "faulty" component was simply replaced.

3.4 SUMMARY

In general, the instrumentation anomalies are believed to be the result of mechanical damage or exposing the CRDMs to a hot, moist atmosphere and a subsequent core depressurization. As stated in Ref. 1, proper CRDM and instrumentation performance requires maintaining purge flow into the CRDM cavity, and maintaining drive mechanism temperatures below 250°F. Loss of purge flow certainly contributed to the ingress of moist helium into the CRDM cavities. The effects of depressurization on instrumentation performance are not well understood, but there is significant evidence that depressurization in conjunction with a moist environment can tend to increase instrumentation problems. These problems are likely caused by condensation of moisture on electrical components. In contrast, the Region 19 instrumentation anomalies were most likely caused by overdriving the control rods, thereby damaging the potentiometer shaft, and resulting in ambiguous and erroneous position instrumentation readings.

3.5 CONCLUSIONS

The following conclusions are based on the possibility of unreliable performance of the CRDM rod position instrumentation under adverse conditions and/or mechanical damage from overdriving, questions concerning instrumentation redundancy, and the lack of independent rod full-in position verification.

The staff has determined that the following actions must be completed prior to restart:

1. To prevent CRDM damage and to protect rod position potentiometers and limit switches, plant procedures should be changed to prevent overdriving the control rods past the rod-in limit (yo-yo-ing).

2. Periodic surveillance of rod position potentiometers and switches should be developed and implemented in interim procedures and be proposed for inclusion in the plant Technical Specifications. This surveillance should include verification of limit switch operability and confirmation that redundancy has not been lost.

The following actions should be taken in the long-term:

1. Damage due to overtravel should be precluded either by the installation of a positive mechanical stop or by providing sufficient clearance to prevent damage.

2. An appropriate, independent and definitive means of verification of control rod full-in position should be provided because the installed rod position instrumentation can be inadequate to verify control rod position. In the present form, Watt-meter testing of the shim motor is considered inadequate to verify full insertion of control rods. It is therefore concluded that the Watt-meter method be refined or an alternate method be developed to achieve sufficient resolution of rod position and then formalized into a plant procedure.

3. Conduct an integrated systems study to resolve rod position indication maintenance and operability questions.

3.6 REFERENCES

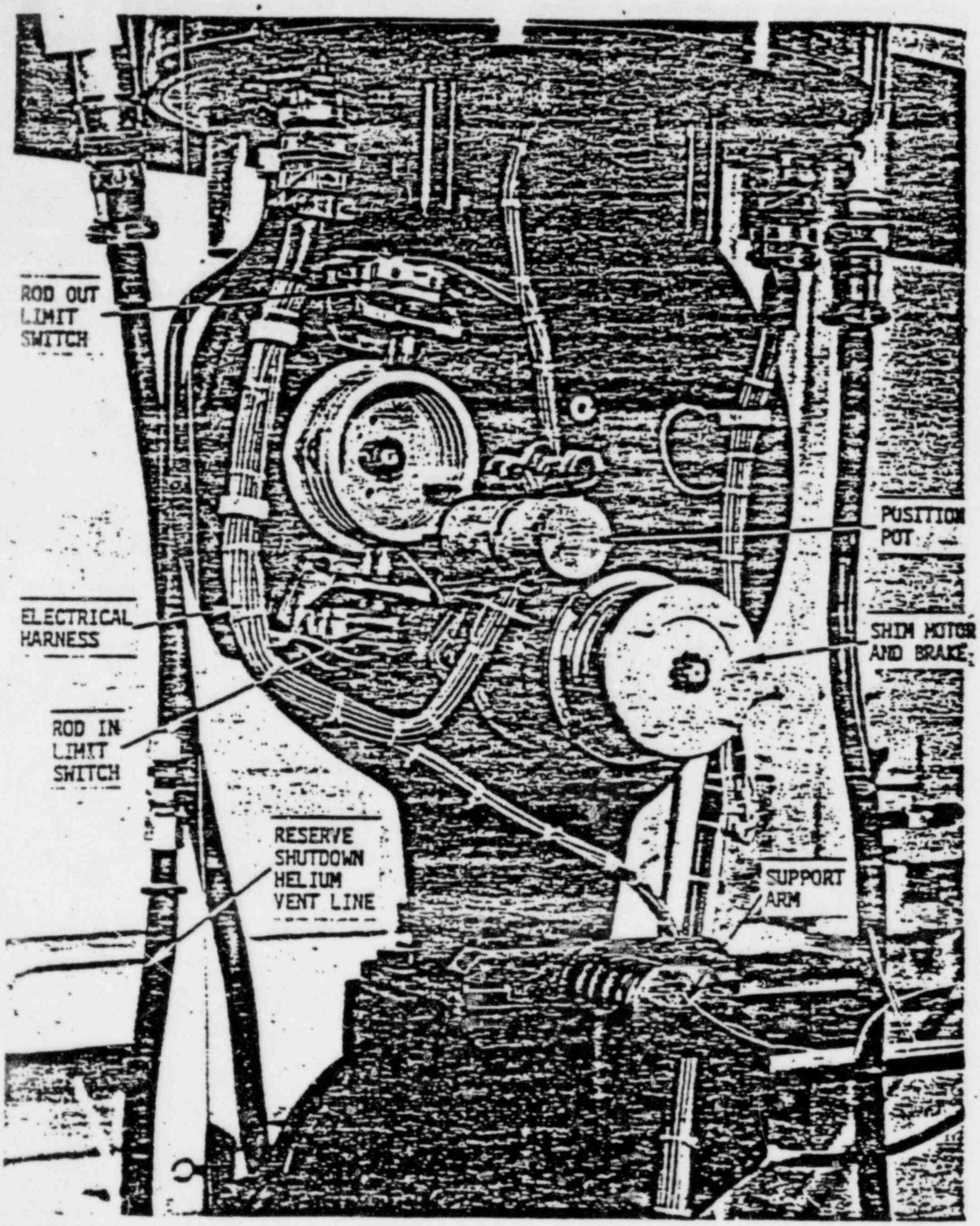
1. "Installation, Operation, and Maintenance Manual for the Control and Orificing Assembly for the Fort St. Vrain Reactor", GA-9806, May 1977.

2. "Rod Control System Equipment I-9303. Operation and Maintenance Manual", E-115-265 (REV. 3), August 1979.

Table 3.1 Current CRDM and Instrumentation Anamalies

CRDM No.	Current Region	Instrumentation Problem
6	1	1 of 2 rod-out limit switches inoperable.
12	15	Incorrect Control Room (CR) analog position indication.
13	19	Faulty rod-in limit switch, incorrect CR analog and digital indication.
25	7	Faulty slack-cable switch.
26		Faulty rod-out switch.
29	ESW5*	Faulty rod-out switch.
33	16	Faulty analog position indication.
37	3	Faulty rod-out switches.
39	23	Incorrect CR analog position indication, faulty rod-in limit switch.

* Equipment Storage Well



HT87275

Fig. 3.1 Control rod drive mechanism (rear view)

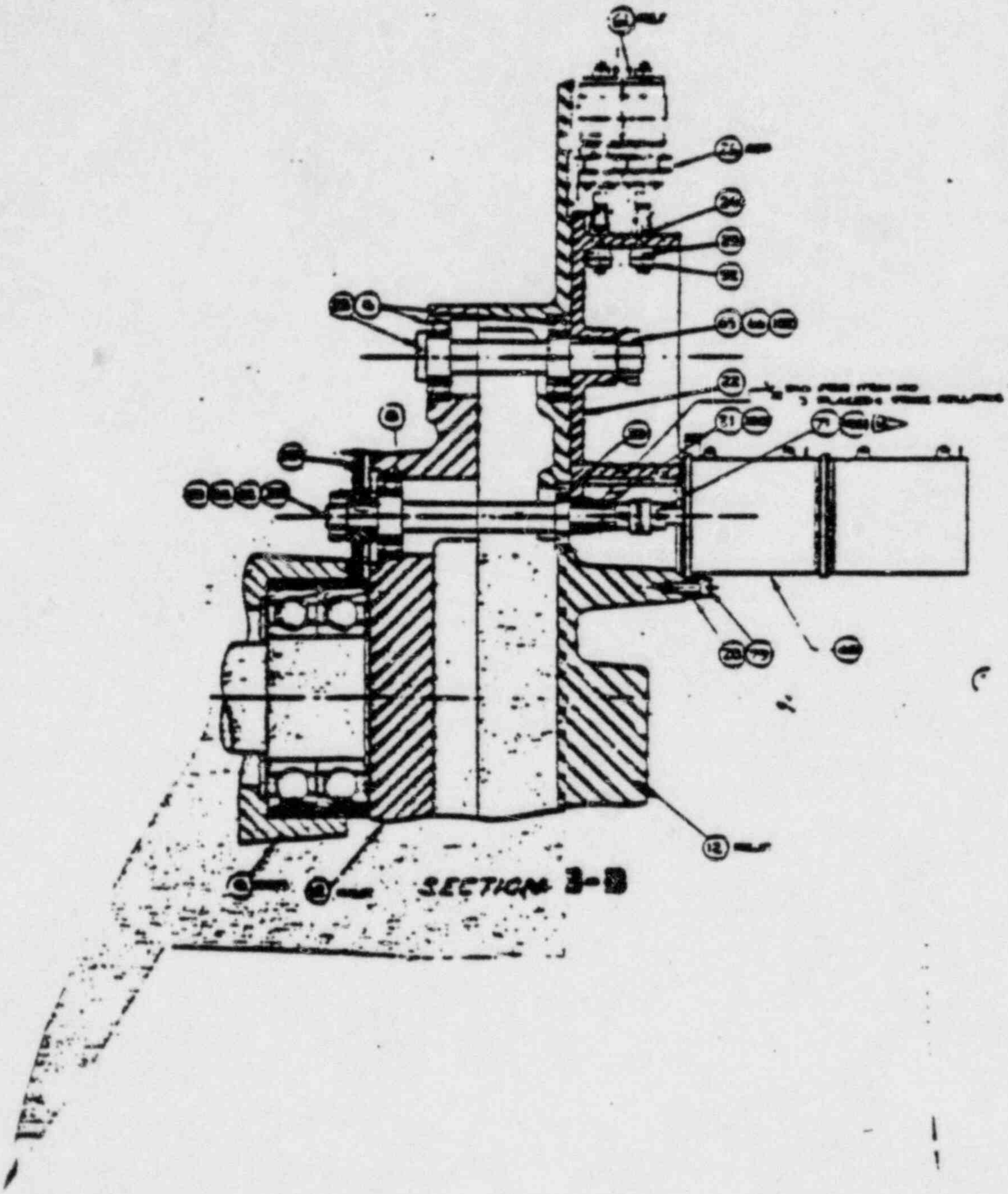


Fig. 3.2 Section BB of Drawing D-1201-200, Control Rod Assembly.

CONTROL ROD DRIVE MECHANISM

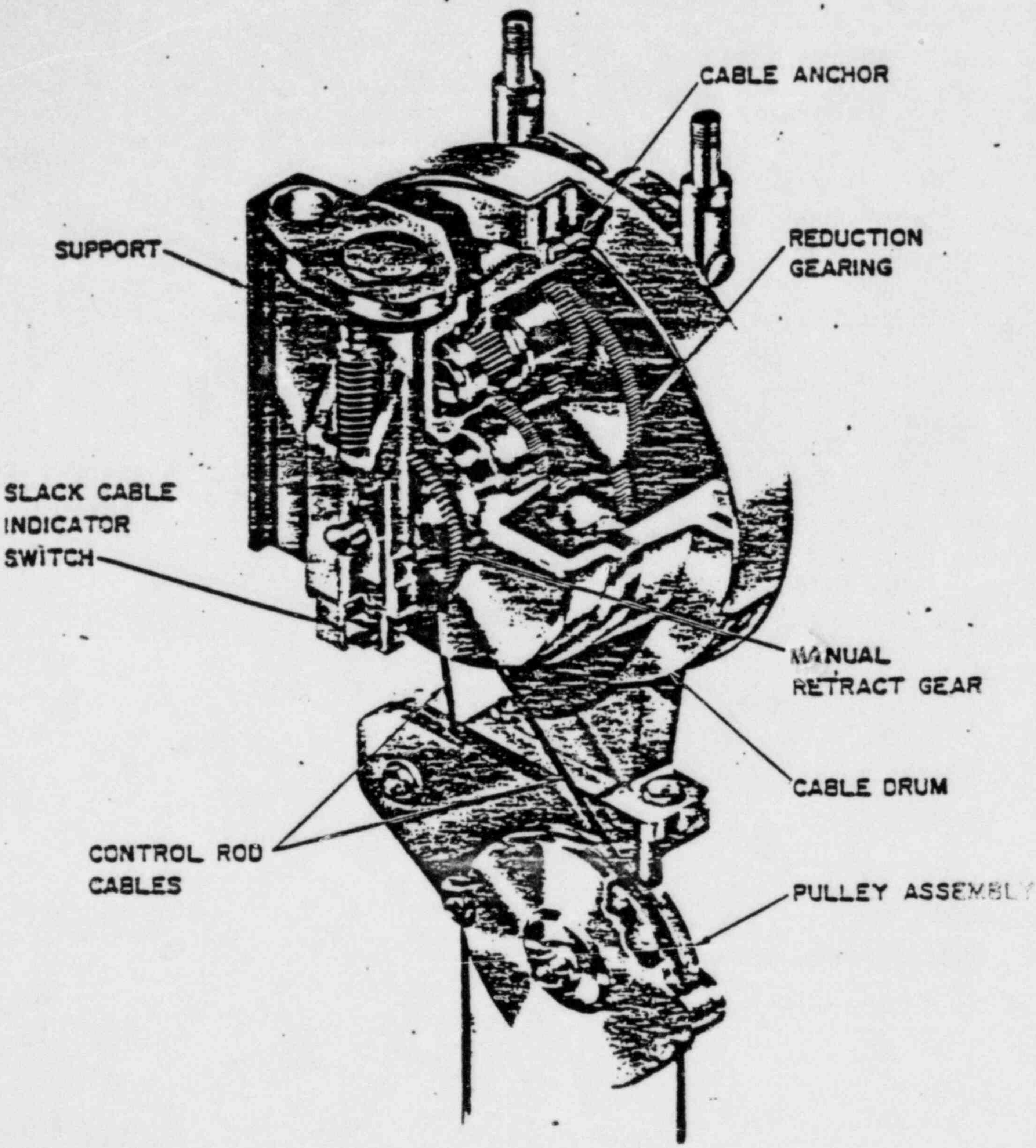


Figure 3.3 Control Rod Driver Mechanism

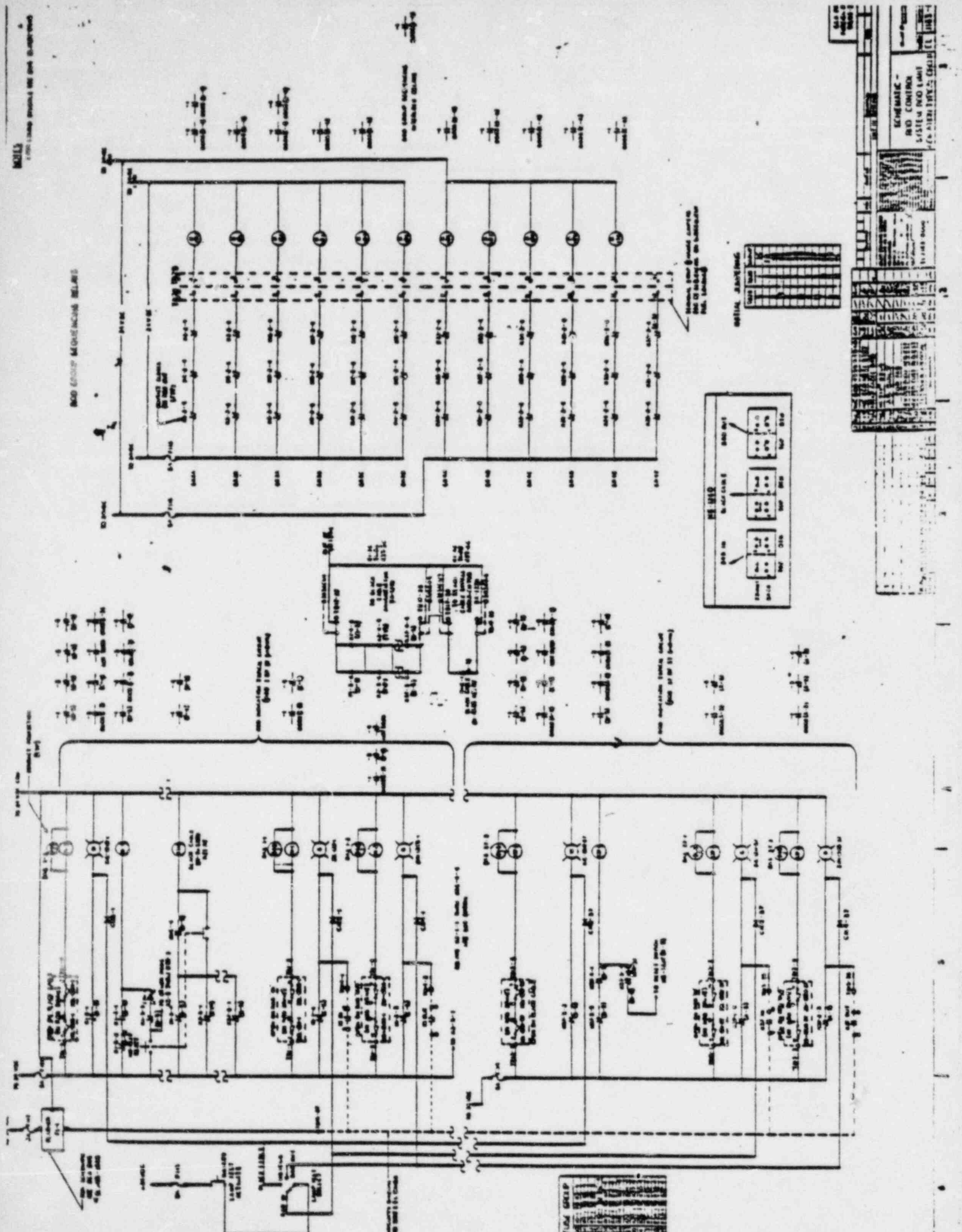


Figure 2.4 Schematic, Rod Control Limit Switches   Page 2-13

CONDUCT OF PLANT OPERATIONS

4.1 INTRODUCTION

The staff's review of the conduct of operations at Fort St. Vrain addressed recent operating history, operational activities, and maintenance. The review disclosed weaknesses in the conduct of operations which confirm previous observations by NRC in its Systematic Assessment of Licensee Performance (SALP) review. The staff was informed that the findings of the Institute of Nuclear Operations (INPO) are similar. These weaknesses may be the result of an operating philosophy that appears to subscribe to less formality and less rigid control of operations in terms of the use of procedures, detail and verification addressed in the procedure, and adherence to the procedures than is common at other commercial nuclear power plants.

In this area, the Fort St. Vrain licensee appears to need strengthened management controls through the adoption of standard industry practices such as those developed and recommended by INPO.

4.2 RECENT OPERATING HISTORY

This review of recent operating history covering the preceding 3 years is based on a compilation of data from the following sources:

1. Licensee Event Reports filed from January 1, 1982, to May 31, 1984

2. 1982 and 1983 reports published by the Systematic Assessment of Licensee Performance Board (SALP Reports)

3. licensee's response to Generic Letter 83-28 (Salem anticipated transients without scram)

4. an analysis of the actual electrical generating capacity of the plant from 1981 to 1983 (1984 is not included because the plant has been shut down for most of 1984)

5. conduct of operations

6. maintenance practices

Table 4.1 compares Fort St. Vrain's performance with that of other operating reactors of similar age.

These data were supplemented by information gathered through interviews with licensee personnel during a site visit on July 9-11, 1984. This review has revealed a pattern of events and operational practices. The staff's assessment of the data is presented as conclusions and recommendations in the following sections.

4.2.1 Review of Licensee Event Reports

The Licensee Event Reports (LERs) for the time period of January 1, 1982, to May 31, 1984, were categorized into five problem areas: dirty electrical contacts, human errors, plant trips, moisture in the primary coolant, and shortcomings in the Technical Specification or other documentation shortcomings.

The staff analyzed 166 LERs. Slightly more than half (53%) could be attributed to normal equipment wear out and a few unusual occurrences. Of the remaining 47%, the largest problem area (17.5%) was due to human error. The overwhelming majority of these errors was caused by operators and technicians failing to follow established procedures. The second largest cause of trouble was attributable to dirt, primarily in electrical contacts, or poor housekeeping at the plant (13.3%). High moisture in the helium primary coolant accounted for 9% of the events and the remainder (7.2%) were categorized as improperly written Technical Specifications, procedures, or related documentation.

There were LERs for five reactor trips during the analysis period. After one of the trips, which occurred on February 22, 1982, two rod pairs failed to insert on receipt of a trip signal. Cause of the failure was not identified by the licensee. Exercising the rod mechanism eliminated the sticking. This event closely duplicates the symptoms of the event on June 23, 1984, during which six rod pairs failed to insert.

4.2.2 Systematic Assessment of Licensee Performance Reports

SALP reports for 1982 and 1983 were reviewed for pertinent data. Of the 13 functional areas evaluated in the reports, 8 were considered relevant. The written remarks are presented rather than the numerical score in order to be more informative. The findings are as follows:

(1) Plant Operations - Most violations stem from failure to follow established procedures or from personnel errors. (This finding closely tracks the LER review, which also revealed a high percentage of human errors.)

(2) Radiological Controls - Because of its unique fuel design, this plant had very low personnel exposure compared with that at other plants. Although no significant problems were noted from inadvertant effluent releases, most of the releases that did occur were attributable to operator error. The plant exhibited very good performance in terms of radiochemistry.

(3) Maintenance - New maintenance procedures were written in 1982, but many violations occurred because these procedures were not followed. The significant backlog of safety-related maintenance actions continues.

(4) Surveillance - Violations cited failure to follow procedures. Personnel involved were apparently not providing sufficient attention to detail. Procedures still need improvement and more management attention is required .

(5) Licensing Activities - The SALP reports indicate that the licensee has a tendency to take the position that new NRC requirements apply only

to water-cooled reactors and not to high temperature gas reactors. Establishing a central point of contact for licensing activity was recommended by the SALP Board.

(6) Quality Assurance - Violations cited failure to follow procedures. Weaknesses in the control of purchased material were noted. Quality assurance audits were considered thorough and complete. The licensee has been very responsive to NRC directives in the QA area.

(7) Design Changes and Modifications - Evidence exists that newly modified systems were returned to operation without the required updating of documentation and without the shift supervisor's verification of cold checkout or functional tests.

(8) Management Control - The licensee was trying to improve operation personnel's conformance to established procedures. In some instances, management's response to NRC initiatives was superficial and required considerable NRC followup.

4.2.3 Licensee's Response to Generic Letter 83-28

The area of primary interest for this assessment is the licensee's description of the programs for post-trip review and for vendor interface.

The post-trip review is intended to provide a systematic methodology for determining causes of reactor trips and for ascertaining that safety-related and other important equipment and systems functioned correctly and that restart of the plant can be safely undertaken. This review is conducted by the licensee's Transient Review Committee (TRC). Typically, this committee consists of the Superintendent of Nuclear Engineering, and the engineering staff, technical advisory staff, operations, training, and other staff as necessary. The station manager has overall responsibility for ensuring that an appropriate analysis has been performed and makes the decision to restart the reactor. The post-trip review process consists of the following five steps: data collection, trip investigation, trip investigation review, restart decision, and identification of lessons learned followup action. Data collection is aided by the plant process computer and a hard copy of the chronological sequence of major plant alarms, trips, and actuations.

The licensee's response to Generic Letter 83-28 is very complete for post-trip reviews. However, the licensee does not appear to understand completely what is intended by the vendor interface program. One of the major objectives of this program is the establishment by the licensee of a procedure for periodic communication with equipment vendors concerning maintenance or replacement of components or other technical information necessary to ensure the continued reliable operation of the equipment in its service environment. This is directly related to maintaining the qualified life of equipment and ensuring a feedback mechanism to alert the licensee to generic problems or design improvements.

The licensee's statement that because the plant is a one-of-a-kind design and, therefore, the need for vendor interface as at other plants does not exist, indicates that the issue of qualified life has not been addressed. Furthermore, during the interview at the site a member of senior management indicated that no response to Generic Letter 83-28 had been submitted because Fort St. Vrain does not have reactor trip breakers. This is an example of breakdown in communication within the management's organization that should be corrected.

4.2.4 Record of Power Generation

Fort St. Vrain began commercial operations in 1979. Since that time it has run 5 days at 100% power for testing purposes. Design problems (core outlet temperature fluctuations and cable separation) limited the plant to 70% power. This limit was removed in 1982. Administrative rules now limit the power to 85% pending completion of initial rise-to-power testing.

The power limitations notwithstanding, the plant has often been run at reduced power or has been forced to shutdown due to operational or equipment problems. The plant has been beset by problems. Some of the most lengthy outages have been caused by high moisture in the helium coolant. The plant was at power 62%

of the time in 1981, 58% in 1982, and 61% in 1983. The plant has a lifetime average (not counting 1984) of under 21% capacity.

4.2.5 Conduct of Operations

4.2.5.1 Shift Turnover Procedures - At the present time the licensee does not have a formal procedure to control shift turnover. This item was the subject of a recent violation identified by the Region IV staff; the licensee is currently taking corrective actions in this area. Of significant concern to the staff is the fact that the lack of a shift turnover procedure and an associated shift turnover checklist deviates from the short term improvements contained in NUREG-0737, specifically Items I.C.2 and I.C.3. Even though there is no formal procedure, it should be noted that a practice exists in which the control room operators review all control room logs during the conduct of shift turnovers.

4.2.5.2 Administrative Controls - The licensee has no procedural controls requiring shift supervisors or licensed reactor operators to read and review the various logs when they return to the control room after an absence for extended periods due to training, leave, or sickness. There is no procedural requirement for the shift supervisors or the Operations Superintendent to conduct periodic plant tours. Discussions with the Operations Superintendent and shift supervisors, however, indicate that they do conduct plant tours.

4.2.5.3 Operator Aids - While conducting a plant tour, it was noted by the staff that the licensee provides several locations throughout the plant where drawings are kept to assist the operators during the performance of their duties. Not all drawings at these stations are controlled. Only one station located in the control room is considered up to date in that changes are marked on the drawings subsequent to the implementation of a plant modification until the drawings are updated. The licensee informed the staff that concerns in this area are being addressed as a result of similar INPO findings.

In addition to drawings, the licensee keeps system operating procedures at some local panels. These procedures are controlled copies, but there is no receptacle or designator to identify the panels to which the control procedures apply. Instead the procedures are inserted between the control switch pistol grip knobs and the panel.

4.2.5.4 Tagging Practices - During the plant tour the staff identified a tag that has been in place since 1982. The information on the tag is not legible. Discussions with the shift supervisor revealed that the tag was one of eight hung to provide instruction on valve positioning for the reactor building air handler units. The licensee does not have a program to review outstanding valve lineup tags to determine which should be incorporated into system procedures. This is part of a larger problem in which the licensee has been using status and clearance tags in place of design modifications.

The licensee permits a practice where tags used for maintenance are written in pencil and erased for subsequent reuse. The licensee's status tagging system is confusing because only one tag with appropriate information and a group of auxiliary tags with only a reference number and no explanatory data are used.

In the auxiliary cable spreading room the back door of the electrical cabinet "Bus 1-2 Dist. Panel," identified with a "High Voltage" sticker was found open. The cabinet was not tagged.

The staff noted additional examples indicative of laxity in tagging practices throughout the plant.

4.2.5.5 Shift Supervisor's Office Location - The shift supervisor's office is currently located in the turbine building next to the control room access door. The office is equipped with an annunciator panel that provides alarm functions to indicate an actuation of the reactor protection system and trips of the helium circulators. The shift supervisor does not have a key to gain access to the control room in case the normal process of inserting an identification card in the card reader becomes inoperative.

Locating the shift supervisor's office outside the control room area inhibits access to the control room for the performance of the shift supervisor's duties. To our knowledge for a one-unit site this is a unique situation.

4.2.5.6 Verification of the Correct Performance of Operating Activities - The licensee's program for the verification of correct operating activities is limited.

The staff could not determine that double verification of maintenance activities in safety-related systems, for example, maintenance activities for the emergency diesel generator was always performed. It was also noted that double verification of jumpered and lifted leads in the plant protective system is not required by procedures.

4.2.6 Maintenance Practices

4.2.6.1 Observations - The maintenance and surveillance functions are carried out at FSV by two separate organizations: (a) the maintenance department, which includes the crafts of welding, millwright, machinists, fuel handlers, and electricians; and (b) the nuclear betterment department, which has the I&C functions.

Maintenance as a whole is under the supervision of the maintenance superintendent, and each craft is headed by a craft supervisor. There is also a scheduling group. The nuclear betterment engineering superintendent has responsibility for two I&C functions, one for repair and another for surveillance; the functions of the results engineering group include reviewing and functional testing of engineering changes. Each group is headed by a supervisor.

The maintenance department has 49 crafts people. The workload for June 1984 included a total of 480 items. The betterment department has 23 technicians and usually completes 200 items in 1 month.

An independent Maintenance Quality Control (MQC) Group is responsible for assuring satisfactory performance by the maintenance and betterment engineering departments. MQC has seven technicians headed by a supervisor. The functions of the MQC group include checking instrument calibrations and the quality of maintenance work, verification part traceability, and preparation of nonconformance reports. The nonconformance log book recorded 214 items for 1984 up to the date of the staff visit. The MQC group also prepared nonconformance TRENDING; i.e., analyzing whether nonconformance was due to procedures, equipment design, or human error. An example of their effectiveness is shown by the fact that disciplinary action was initiated for a craftsman for repeated nonconformance work.

Maintenance contractor services are used periodically. Contractor personnel work under a supervisor who reports to the plant maintenance superintendent. During refueling, 70 to 75 contractor personnel may be working in maintenance, and 5 to 7 technicians in betterment engineering.

Equipment history data, completion of plant trouble reports (PTRs), and the preventive maintenance items are stored in a computer system called STAIRS. A full-time administrative maintenance clerk is employed to update the records. Spot check observations disclosed satisfactory recordkeeping. The STAIRS system is to be replaced with a new system called PPMIS (Power Plant Maintenance Information System). No firm date was available about the installation of the PPMIS. The PPMIS is used in the fossil plants, and consequently, the FSV system will be compatible with the rest of the plants operated by PSC.

4.2.6.2 Housekeeping - Contractor personnel are being used to accomplish the large-scale cleanup that has been initiated at the plant in response to the June 1984 special inspection by Region IV personnel. The Vice President for Power Generation has approved the following commitments for painting and cleaning: for 1984 - 10,000 labor-hours in the reactor building and, 10,000 labor-hours in the turbine building; for 1985 - 25,000 labor-hours.

The massive cleanup before this audit resulted in bringing the plant's housekeeping to what must be considered an average level.

4.2.6.3 Assessment Team Concerns -

(1) Scheduling

a. Prioritization

A formalized priority system is not in use for maintenance and I&C work. There is an informal understanding between the craft supervisors (first-level supervision), the scheduling group, and the superintendents about the importance and urgency of maintenance work requests (PTRs). The work that is judged most important by the individuals concerned is done first. As a result, some work is assigned a low priority and is significantly delayed.

b. Daily Work Designation

The scheduling group prepares a daily and weekly work assignment for each craft supervisor. However, the interviews with the craft supervisors indicated that the work assigned and the amount completed during a reporting period is often left to their discretion. Those items which are not completed slip to the next period without being identified as late.

(2) Preventive Maintenance - A printout of scheduled preventive maintenance (PM) items that includes about 690 elements is provided at the plant. However, the majority of the PM items are surveillance and calibration items and lubrication-type maintenance work. The list was developed by one individual 5 years ago. The system appears to be a scheduling aid rather than a true PM program. Effective PM programs normally include the following features not found at Fort St. Vrain: an engineering analysis, means to evaluate its effectiveness, and methods for modifying or updating the program. The PM program appeared to use a static, nontechnical approach, rather than a dynamic, technology-based, engineered method for ensuring equipment readiness. During the audit, the staff did not observe

that any new approaches to PM were to be included in the upcoming computerized PPMIS.

(3) Spare Parts Management - For most components there is no shelf-life program at Fort St. Vrain. The latest directive on shelf-life (letter NDG-84-0405) dated May 18, 1984, from the Nuclear Design Manager to Manager, Quality Assurance recommends only visual/physical inspection of parts by the maintenance technician and states that traditional aging effects are not applicable at the plant because of the unique design of the reactor. Further review of the validity of this approach is recommended (see previous discussion on licensee's response to Generic Letter 83-28).

The term "quality-rated" applies at Fort St. Vrain to any component, safety related or not, that has Technical Specifications associated with it. This usage of the quality-rated designation is unique in the industry. The adequacy of a parts management system without a special designation for safety-related items is questionable and is recommended for further review by the staff.

(4) Maintenance Procedures - An audit review of maintenance procedures indicated that they were adequately identified on the plant trouble reports and surveillance and calibration orders. The craft supervisors have ready access to the procedures. However, the audit of procedures also showed that some of them are not precise. For example, a surveillance test procedure (SR 5.6.2a-W, Issue 22) stated that "if data collected is not within the acceptance criteria, initiate PTR" but did not designate the individual responsible for the action. The craft supervisor was uncertain about whether the technician or the supervisor would initiate the action, and after some vacillation decided that it is the supervisor's responsibility. An indepth review of maintenance procedures should be included in Region IV's follow-up report.

(5) Maintenance Testing - The purposes of postmaintenance testing are to (1) verify that the maintenance work was done correctly and (2) to assure that

plant safety is not jeopardized by reliance on unproven equipment that may not perform its required function. It is the staff's opinion that all safety related equipment should be functionally tested before it is returned to service following maintenance work.

Despite incidents where improperly maintained equipment was returned to service, the licensee's response to Generic Letter 83-28 does not endorse the concept of 100% postmaintenance testing. The licensee proposes to limit testing to only those items whose normal operating mode does not demonstrate their capability to perform their safety functions. The risk of this approach is that, as stated above, the tested piece of equipment may cause operational problems as soon as it is returned to service. Furthermore, a piece of equipment may perform satisfactorily under normal plant conditions, but may not necessarily be able to perform its safety function under plant upset conditions. Procedures should include requirements that functional testing be performed upon completion of equipment maintenance and that these tests be independently witnessed by the quality control organization.

The staff audit was not intended to review the qualifications and training of maintenance personnel. However, some interview questions concerned with training were asked in order to reveal the need for upgrading training programs. Mechanics receive 1-week of classroom training and 6 months of on-the-job training to qualify for classification upgrading. I&C technicians receive 2 weeks of classroom training and 12 months of on-the-job training in each classification category. Vendor courses are also provided. It appears that training programs are adequate.

(6) Backlog - None of the people interviewed provided a list of backlog maintenance items. The oral estimates varied from a comment that "we have no backlog after a refueling" to items numbering from 50 to 200. Most nuclear plant maintenance departments have the means to display the status of backlog items to management. This feature is missing at Fort St. Vrain.

4.3 CONCLUSIONS

Based on its review of the area of plant operations, maintenance and housekeeping, the staff has determined that PSC undertake initiatives that are designed to strengthen overall management control and that will improve plant operations. This will require an assessment of the root causes of operational problems and identification of broad-ranged corrective measures. Such assessment should be performed by a third party consulting group, the scope and schedule for which must be determined prior to restart. Additional items of concern noted in this section should be taken account of in the third party review to determine if improvements are necessary in those areas.

Table 4.1 Comparison of Fort St. Vrain with other plants

Plant(vendor)	Operating License date	Size, MWe	Licensee Event Reports*
Fort St. Vrain (GA)	12/21/73	330	6
Arkansas Nuclear One (B&W)	5/21/74	836	4
Calvert 1 (CE)	7/31/74	845	6
Pilgrim (GE)	6/8/72	655	8
Brunswick 2 (GE)	12/27/74	790	7
Zion 1 (W)	4/6/73	1040	14
Zion 2 (W)	11/14/73	1040	13
Indian Pt. 2 (W)	9/28/73	873	4
Palisades (CE)	10/72	798	6
Oconee 1 (B&W)	2/6/73	860	2
Oconee 2 (B&W)	10/6/73	860	0
Oconee 3 (B&W)	7/19/74	860	2
Turkey Point 3 (W)	7/19/72	728	16
Turkey Point 4 (W)	8/10/73	728	7
Hatch 1 (GE)	8/6/74	768	8
Cook 1 (W)	10/25/74	1050	7
Duane Arnold (GE)	2/27/74	545	19

*January 1, 1984, to approximately May 23, 1984
NOTE: B&W=Babcock and Wilcox Co.; CE=Combustion Engineering; GE=General Electric; W=Westinghouse; GA=General Atomic

5. TECHNICAL SPECIFICATIONS

5.1 Introduction

During the week of July 9, 1984, an audit was made of the Fort St. Vrain Technical Specifications. This audit included an onsite visit by members of the NRC staff. The purpose of this audit was to assess the adequacy of the Technical Specifications and their implementation. In assessing the content of the specifications, the staff reviewed selected limiting conditions for operation (LCOs) and their corresponding surveillance requirements and bases for completeness, clarity and correctness. The specifications were compared against the Fort St. Vrain Updated Final Safety Analysis Report (FSAR) and the NRC Standard Technical Specifications (STS) for light-water reactors (LWR)s. In assessing implementation, the staff reviewed the administrative and procedural controls, internal audits, and recordkeeping used by the licensee to ensure compliance with the Technical Specifications.

Experience over the past several years in the review and use of the Fort St. Vrain Technical Specifications has indicated that problems exist in certain specifications in the areas of technical adequacy and clarity. These problems have been experienced by both the licensee and the NRC and have led to many revisions and proposed revisions to the Technical Specifications. Moreover, because of the extent of the problems still existing, the licensee developed and issued on June 1, 1984, a schedule to upgrade the entire set of Technical Specifications beginning in June 1984 and ending in December 1985. However, at the time of the audit, the licensee was behind in meeting this schedule. Accordingly, as part of the followup to the June 23, 1984 incident at Fort St. Vrain during which 6 out of 37 control rods failed to insert on receipt of a scram signal, NRC, decided to include in its review, an audit of the Fort St. Vrain Technical Specifications to reassess the extent of the problems and the priority that should be applied to the upgrading effort. This review was part of the overall look at the conduct of operations at the plant.

5.2: Technical Specification Content

The scope of the staff's review of the content of the Fort St. Vrain Technical Specifications addressed the following areas:

(1) Completeness

(a) Are all systems, components and parameters requiring a Technical Specification adequately addressed?

(b) How do the Technical Specificatons compare with the NRC Standard Technical Specifications for LWRs?

° What are the differences in LCO content and format from those used for LWRs, especially in areas similar to those at an LWR?

° Is there an action statement for each LCO?

° Is there a surveillance requirement for each LCO and vice versa?

° How do the surveillance frequencies compare with analagous LWR Standard Technical Specification surveillance frequencies?

(2) Clarity

Are the limits, actions, surveillances and bases written in a clear, unambiguous fashion?

(3) Correctness

(a) Are the Technical Specifications consistent with the FSAR?

(b) Is there a good basis for each limit?

The LCOs in the Fort St. Vrain Technical Specifications are divided into nine sections (see Table 5.1). Each of these sections contains LCOs and their corresponding bases. Each of these nine sections also has a corresponding section covering surveillance and the bases for the surveillance. In the audit the staff did not completely review all sections. Instead the staff selected three sections (4.1, 4.2, and 4.7) for examination regarding content. All nine sections were reviewed for the existence of action and surveillance requirements.

The results of the audit indicate that some problems exist in all three areas of the review. These problems are summarized below:

(1) Completeness

Major areas where the Technical Specifications were found to be incomplete are:

(a) Not all LCOs have a corresponding Technical Specification surveillance requirement, and conversely not all surveillance requirements correspond to an LCO (see Table 5.2 for comparison).

(b) In the surveillance sections there are many instances where the acceptance criteria for the surveillance are not in the Technical Specifications but rather are in the surveillance procedures. In these instances changes to surveillance requirements and associated acceptance criteria could be made through procedure change without NRC review.

(c) Many items that should be included as LCOs (and typically are in LWR Technical Specifications) are not in the Fort St. Vrain Technical Specifications. The absence of these items leaves gaps in the control and monitoring of key safety parameters. Major items found in this category were:

(i) Section 4.1

- A limit on the maximum worth of the power/temperature defect is missing. This limit is required to ensure that the reserve shutdown system is capable of shutting down the plant as designed.

- A limit on maximum control rod drive temperature and/or on minimum control rod drive purge flow is required to ensure the control rod drive mechanisms are operated in an environment for which they are qualified.

- A limit on the maximum number of inoperable control rods (based on the concern for common-cause failure) should be specified. This should be an absolute limit, not just a limit based on maintaining an adequate shutdown margin.

(ii) Section 4.2

- A limit on the maximum heatup and cooldown rate of the primary system may be needed to protect the integrity of the prestressed concrete reactor vessel, liner, and core support blocks.

- In addition to limits on liner cooling system water temperatures, a specification on system flow rate or maximum liner temperature may be required to ensure satisfactory operation.

- The limit on liner cooling system surge tank pressure and atmosphere composition should be specified consistent with FSAR section 9.7.2.

(iii) Section 4.7

- To ensure proper fuel handling an LCO on the operability of the fuel-handling machine device that verifies the orientation of the fuel block should be provided.

- The limits on the composition of the atmosphere in the fuel storage wells should be specified.

(d) In some instances existing surveillance requirements are found to be incomplete. Incomplete surveillance requirements can lead to concerns on the operability of safety systems. Examples of incomplete surveillance requirements are:

(i) Surveillance Requirement 5.1.1 only requires periodic exercising of fully withdrawn control rods. Partially inserted rods receive no test. Exercising should be provided for all rods.

(ii) Surveillance Requirement 5.1.2 does not require inspection of the boron balls in the reserve shutdown system for degradation. Such inspection should be required to ensure the system will function as designed. In fact, in NRC's approval of License Amendment No. 13 (letter from R. A. Clark to the licensee dated June 18, 1976) such inspection was called for following certain conditions of high moisture in the primary coolant. However, this request has not been implemented in the Technical Specifications, but rather has been implemented by surveillance procedures and thus is not necessarily mandatory.

(2) Clarity

There is a lack of clarity in certain areas of the Technical Specifications. This lack of clarity leads to the following problems:

(a) In some cases it is not clear when an LCO applies. There are no defined operational modes applied to the Fort St. Vrain Technical Specifications such as in the LWR Standard Technical Specifications. Therefore, in certain LCOs (for example, LCO 4.4.1, "Plant Protection System Instrumentation"), the times when equipment must be operable or when parameters must be maintained within limits are not clear.

(b) Some LCOs have been subject to varying or case-by-case interpretations (for example, LCO 4.1.2 regarding when a control rod is considered operable). In several instances this has led to disagreements between NRC and the licensee over whether or not the licensee was in compliance with the Technical Specifications. For example, in assessing compliance with Surveillance Requirement 5.1.1, there has been disagreement between NRC and the licensee over which control rods require periodic exercising while at power.

(c) Some limits that should be LCOs are contained in the "Bases" sections (for example, the "Bases" section to LCO 4.1.3 contains the control rod worth limits). All limits should be in the specifications, not in the "Bases" section .

(3) Correctness

(a) Certain items in the Technical Specifications are incorrect. Examples are:

(i) LCO 4.1.9, "Core Region Temperature Rise," contains limits on allowable power to flow and power to maximum core region delta T that are not conservative and that do not provide for transition from a mode of equal core region flow operation to equal core region delta T operation. It should be noted, however, that approximately 6 months ago, the licensee submitted a proposed change to this LCO to NRC for approval and has implemented selfimposed operating procedural changes to compensate for the errors, until this can be resolved.

(ii) The "Bases" section for LCO 4.2.11 "Loop Impurity Levels - Low Temperature," contains incorrect values for the Technical Specification limits on shutdown margin and allowable reactivity anomaly, although the LCO itself appears correct.

(b) In Section 4.2 it is not clear that the specification of minimum operable equipment provides for sufficient redundancy of safety-related equipment.

(c) The surveillance frequencies listed in the Fort St. Vrain Technical Specifications are, in some cases, not consistent with the surveillance frequencies for similar LCOs listed in the LWR Standard Technical Specifications.

To assess whether or not the above problems are also being experienced by the operating crews, the staff interviewed five members (two shift supervisors and three senior operators) of two operating crews. In discussing the problems with the existing Technical Specifications, the operators unanimously indicated that the Technical Specifications should be clarified and simplified.

This would remove the potential for incorrect or inconsistent interpretations that now exists.

5.3 <u>Technical Specification Implementation</u>

The scope of the staff's review of the implementation of the Fort St. Vrain Technical Specifications consisted of the following areas:

(1) Administration

(a) Are responsibilities for Technical Specification implementation, changes, and reporting defined?

(b) Are required approvals defined?

(c) Are actions to be taken concerning noncompliance defined?

(2) Compliance

(a) How is the plant's compliance with the Technical Specifications verified?

(b) What procedures are used?

(c) Is there an adequate internal audit program?

(c) Is there adequate recordkeeping?

(3) Operator Perspective

(a) What training do the operators receive on Technical Specifications?

(b) What is the operators' level of knowledge on Technical Specifications?

(c) What problems do the operators have with the Technical Specifications?

Overall, the current program in place for implementing the Technical Specifications appears satisfactory. Some areas that could be improved are discussed below. A separate group (Technical and Administrative Services) within the licensee's Nuclear Production Division acts as the focal point for matters related to the Technical Specification. The group's responsibilities are defined in PSC Procedure Q-1, "FSV Organization and Responsibilities." The group prepares and processes changes to the Technical Specifications, reviews logs and records for compliance, writes reports and evaluations on potential or actual Technical Specification problems, violations, and corrective actions, and takes the lead in resolving questions on problems with the Technical Specifications. The licensee stated that reporting was in accordance with 10 CFR 50.72 and 50.73. The procedure, documentation, and required approvals for making changes to the Technical Specifications are defined in PSC Procedure TSP-15 "Facility License Amendment and 10 CFR 170 Fees".

To ensure compliance with the Technical Specifications, a combination of administrative procedures, operating procedures, surveillance procedures, and a Technical Specification compliance log are used; however, a complete cross reference between all the requirements in the Technical Specifications and the procedures that implement them does not exist. The staff considers the cross referencing important and that it should be developed since it provides a documented check to ensure all Technical Specifications are implemented. Surveillance procedures are written for each surveillance requirement included in the Technical Specifications. These generally are requirements on equipment condition and operability. In addition, a computerized surveillance log sheet is used by the operating crews to determine compliance with the Technical Specifications at least once per shift. This log sheet procedure is defined in PSC Procedure SR-OP-20-W, "Technical Specification Compliance Log". The staff checked approximately five surveillance procedures to see if they adequately reflected the Technical Specification requirements and found no discrepancies.

The operators' narrative log and plant trouble reports are also reviewed each working day by the Technical and Administrative Services group (in accordance with PSC Procedure TSP-9 "Identification and Preparation of Reportable Occurrences") to determine compliance and problems. The licensee has indicated that they are also planning to implement a procedure in the near future to formally require the technical advisors to check the computerized Technical Specification surveillance logs, although such checks have been occurring on an informal basis.

Independent checks of Technical Specification compliance such as this are important and should continue. Additional steps currently not in place but under active consideration by the licensee to help ensure compliance are:

(1) Having the MQC group witness and sign off key steps in the performance of surveillance procedures.

(2) Having the quality assurance (QA) organization review the content of and concur in all safety-related procedures, including future changes. Such review and concurrence by the QA organization on Technical Specification procedures has not been the practice in the past.

In addition to checking the Technical Specification logs and surveillance records by the Technical and Administrative Services group and the Maintenance Department's QC group, the PSC QA organization formally audits the Technical Specification compliance program. These audits consist of (1) a yearly audit under the charter of the Nuclear Facility Safety Committee, (2) an audit of the implementation of each approved Technical Specification change, and (3) periodic audits of various plant organizations and functions. The periodic audits are done on a rotating basis with each area covered once every two years. Formal reports are issued for the yearly and rotating audits. Formal findings, corrective action and tracking of corrective action are required. PSC indicated that the yearly and rotating audits are extensive, and each audit takes approximately 3 person-weeks of effort. The audit of implementation of changes is conducted in accordance with PSC Procedure QAMP-Q2-3, "Technical Specification Amendments."

Recordkeeping associated with the Technical Specifications appeared satisfactory to the extent reviewed. The records checked were available and appeared correctly filled out.

Operator requalification training in Technical Specifications is given as part of systems reviews and by self-study guides on Technical Specifications given to each operator. Licensee requalification and NRC operator examinations include questions on Technical Specifications to verify the operators' knowledge. Additionally, if changes are made to the Technical Specifications, implementing procedures or other related items, a system exists to provide this information to the operating crews in a timely manner.

5.4 CONCLUSIONS

1. The review of FSV Technical Specifications revealed that they need substantial improvement to (a) correct deficiencies in content (omission of limits and surveillance requirements), (b) improve clarity (LCOs subject to various interpretations), and (c) correct errors. A high priority effort should be undertaken to achieve such improvement on a schedule committing the licensee to completion of the review, revision, and submittal of the Technical Specifications by April 1, 1985. Although the existing Technical Specifications need substantial improvement, the licensee's system for implementing Technical Specifications and ensuring compliance with them appears satisfactory.

2. To provide immediate improvement in the control drive and reserve shutdown systems, the licensee should propose the following Technical Specification changes and in the interim implement them by operating procedures:

    a. A weekly control rod exercise surveillance program for all partially or fully withdrawn control rods;

    b. A Limiting Condition for Operations defining control rod operability and the minimum requirements for rod position indication; and

    c. A Limiting Condition for Operations and a corresponding surveillance test to define and confirm reserve shutdown system operability.

3. Additional improvement in the use of Technical Specifications can be achieved by implementing the following procedures:

   a. Verification and sign-off by the Maintenance Quality Control of key steps in Technical Specification surveillance procedures.

   b. Review and concurrence by the QA organization of safety-related procedures and changes thereto. This review should include a check to determine the adequacy of procedures.

   c. At the time of the audit, the MQC group was reviewing each completed surveillance procedure. The staff concludes that this practice should continue.

   d. A review by the QA organization of the content and adequacy of the Technical Specification procedures is important, and the staff has determined that this should be implemented.

Table 5.1 Fort St. Vrain's limiting conditions for operation

Section	Title
4.1	Reactor Core and Reactivity Control
4.2	Primary Coolant System
4.3	Secondary Coolant System
4.4	Instrumentation and Control Systems
4.5	Confinement System
4.6	Auxiliary Electric Power System
4.7	Fuel Handling and Storage Systems
*	-
4.9	Fuel Loading and Initial Rise to Power
4.10	Fire Suppression Systems

* Section 4.8, "Radioactive Effluent Disposal Systems," has been incorporated into the Environmental Technical Specifications.

Table 5.2 Comparison of Fort St. Vrain's limiting conditions for operation and Corresponding Surveillance Requirements

Limiting Condition for Operation	Subject	Surveillance Requirement
4.1.1	Core Irradiation	None
4.1.2	Operable Control Rods	5.1.1
4.1.3	Rod Sequence	5.1.5
4.1.4	Partially Inserted Rods	None
4.1.5	Reactivity Change With Temperature	5.1.3
4.1.6	Reserve Shutdown System	5.1.2
4.1.7	Core Inlet Orifice Valves	5.1.7
4.1.8	Reactivity Status	5.1.4
4.1.9	Core Region Temperature Rise	None
None	Core Safety Limit Surveillance	5.1.6
4.2.1	Number of Operable Circulators	None
4.2.2	Operable Circulator	None
4.2.3	Turbine Water Removal Pump	None
4.2.4	Service Water Pumps	None
4.2.5	Circulating Water Makeup System	5.2.24
4.2.6	Fire Water System/Fire Suppression Water System	5.2.10
4.2.7	PCRV Pressurization	5.2.15
4.2.8	Primary Coolant Activity	5.2.11
4.2.9	PCRV Closure Leakage	5.2.16
4.2.10	Loop Impurity Levels, High Temperature	5.2.12
4.2.11	Loop Impurity Levels, Low Temperature	5.2.12
4.2.12	Liquid Nitrogen Storage	None
4.2.13	PCRV Liner Cooling System	None
4.2.14	PCRV Liner Cooling Tubes	None
4.2.15	PCRV Cooling Water System Temperatures	None
4.2.16	Deleted	-
4.2.17	Diesel-Driven Generator for ACM	5.2.20
4.2.18	Primary Coolant Depressurization	None
4.2.19	Firewater Booster Pumps	5.2.23
None	PCRV and PCRV Penetration Overpressure Protection	5.2.1
None	Tendon Corrosion and Anchor Assemblies	5.2.2
None	Tendon Load Cell	5.2.3
None	PCRV Concrete Structure	5.2.4
None	Liner Specimen	5.2.5
None	Plateout Probe	5.2.6
None	Water Turbine Drive	5.2.7
None	Bearing Water Makeup Pump	5.2.8
None	Helium Circulator Bearing Water Accumulators	5.2.9
None	PCRV Concrete Helium Permeability	5.2.13
None	PCRV Liner Corrosion	5.2.14
None	Helium Circulators	5.2.18
None	Hand Valve and Transfer Switch	5.2.21

Table 5.2 (Continued)

Limiting Condition Operation	Subject	Surveillance Requirement
None	PGX Graphite	5.2.22
None	Core Support Block	5.2.25
None	Region Constraint Devices	5.2.26
None	Helium Shutoff Valves	5.2.27
None	PCRV Penetrations and Closures	5.2.28
4.3.1	Steam Generators	None
4.3.2	Boiler Feed Pumps	None
4.3.3	Steam/Water Dump Tank Inventory	None
4.3.4	Emergency Condensate and Emergency Feedwater Headers	None
4.3.5	Storage Ponds	None
4.3.6	Instrument Air System	5.3.6
4.3.7	Hydraulic Power System	5.3.5
4.3.8	Secondary Coolant Activity	5.3.7
4.3.9	Deleted	-
4.3.10	Shock Suppressors (Snubbers)	5.3.8
None	Steam/Water Dump System	5.3.1
None	Main and Hot Reheat Steam Stop Check Valves	5.3.2
None	Bypass and Pressure Relief Valves	5.3.3
None	Safe Shutdown Cooling Valves	5.3.4
None	Safety Valves	5.3.9
None	Secondary Coolant System Instrumentation	5.3.10
None	Steam Generator Bimetallic Welds	5.3.11
4.4.1	Plant Protective System Instrumentation	5.4.1
4.4.2	Control Room Temperature	5.4.7
4.4.3	Area Radiation Monitors	5.4.9
4.4.4	Seismic Instrumentation	5.4.10
4.4.5	Analytical System Primary Coolant Moisture Instrumentation	5.4.12
4.4.6	Room Temperature, 480 Volt Switchgear	5.4.13
None	Control Room Smoke Detector	5.4.2
None	Core Region Outlet Temperature Instrumentation	5.4.3
None	PCRV Cooling Water System Temperature Scanner	5.4.4
None	PCRV Cooling Water System Flow Scanner	5.4.5
None	Core Delta P Indicator	5.4.6
None	Power to Flow Instrumentation	5.4.8
None	PCRV Surface Temperature Indication	5.4.11
4.5.1	Reactor Building	5.5.1
4.5.2	Reactor Vessel Internal Maintenance	None
None	Reactor Building Pressure Relief Device	5.5.2

Table 5.2 (Continued)

Limiting Condition for Operation	Subject	Surveillance Requirements
None	Reactor Building Exhaust Filters	5.5.3
4.6.1	Auxiliary Electric System	5.6.1
None	Station Battery	5.6.2
4.7.1	Fuel Handling in the Reactor	None
4.7.2	Fuel Handling Machine	5.7.1
4.7.3	Fuel Storage Facility	5.7.2
4.7.4	Spent Fuel Shipping Container	None
4.8	Deleted	-
4.9.1	Fuel Loading and Initial Rise to Power	None
4.9.2	Plant Protection System Dew Point Moisture Monitor Tests During Phase 2	None
4.10.1	Room Isolation Dampers, Three Room Control Complex	5.10.1
4.10.2	Halon Fire Suppression System, Three Room Control Complex	5.10.2
4.10.3	Smoke Detectors and Alarms for Three Room Control Complex and Congested Cable Areas	5.10.3
4.10.4	Fire Barrier Penetration Seals	5.10.4
None	Breathing Air System	5.10.5
4.10.5	Fixed Water Spray System	5.10.6
4.10.6	Carbon Dioxide Fire Suppression System, Emergency Diesel Generator Rooms	5.10.7
4.10.7	Fire Hose Stations	5.10.8
4.10.8	Yard Fire Hydrants and Hydrant Hose Houses	5.10.9

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUL 5 1984

MEMORANDUM FOR: John T. Collins, Regional Administrator
Region IV

Darrell G. Eisenhut, Director
Division of Licensing

FROM: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

SUBJECT: FORT ST. VRAIN (FSV)

The recent event of the failure of several control rods to insert (June 23, 1984) has prompted me to request that Region IV and the Division of Licensing conduct an assessment of the overall conduct of operations at FSV. I want to be satisfied prior to continued operation of FSV that they have analyzed the control rod failure to insert event and have submitted a report for our review and approval prior to FSV leaving their refueling mode.

I understand that there may also be some questions dealing with their failure to initially report that certain control rods did not insert and that Region IV will be looking into this matter for possible enforcement action.

I consider the June 23, 1984 incident, the overall conduct of operations situation, the construction of Building 10, the continued water ingress problem, and the maintenance and housekeeping situation to be of significance; therefore, these subjects should be the focus of our immediate assessment of licensee performance. We should do a thorough overall assessment of the licensing conditions at FSV to determine that plant safety has not diminished. To this end, we will contact the licensee to arrange for a site meeting among NRR, its consultants, the licensee and Region IV. Jim Miller of my staff will be in touch with Mr. Richard Ireland of your staff to obtain your help and participation in this effort. In addition NRR is assigning individuals from the Divisions of Safety Technology and Human Factors to assist in this review. We also will be contacting other Regions for help in specific areas so that a different perspective can be obtained. We will be initiating this assessment beginning July 9, 1984.

We have enclosed a tentative outline of the areas to be reviewed and have developed some suggested staffing levels.

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/enclosure:
W. Dircks
D. Eisenhut
G. Lainas
J. Miller
[redacted]

Fort ST. Vrain Licensing Assessment July 9-11, 1984

I. Items To Be Reviewed:

A. Overall Conduct - of Operation (Management and Licensing)

(i) Staffing (G. Holahan, GL Plumlee, Region Rep.)

B. Review of Building 10 Situation

(Region Rep., Wagner)

C. Water ingress (ATWS - June 22-23, 1984)

(Miller, LANL)

D. Maintenance (Housekeeping)

(John Jankovich)

E. Technical Specifications (Procedures)

(T. King, D. Brinkman, D. Ireland)

APPENDIX B

Summary of Recent History and Region IV
Initiatives Related to Fort St. Vrain

As discussed in the Assessment Report, a series of events culminating with the failure of six rod pairs to automatically scram, caused the Director of the Office of Nuclear Reactor Regulation to order an assessment of plant operations. To complete the picture, this appendix discusses some recent history involving FSV and the actions of Region IV.

SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE (SALP)

On January 17, 1984, the staff of Region IV, assisted by staff from NRR, met with PSC staff to discuss the results of the SALP Board assessment of PSC performance during the period September 1, 1982, through September 30, 1983. The SALP Board found that PSC's performance in four functional areas: plant operations; licensing activities; design, design changes, and modifications; and management controls were in performance category 3. Although this level of performance is minimally satisfactory, it is indicative of weaknesses that require increased management attention and oversight to ensure that further degradation in performance would not occur. The area of plant operations, it was noted, was also in performance category 3 during the previous SALP period. As noted in the SALP report, the principal reason for this was the large number of LER's and violations that could be attributed to personnel errors and/or failures to adhere to station procedures.

The licensee responded to the SALP Board report on January 24, 1984, and outlined steps that were being taken to improve performance in this area. In the final SALP report (issued as NRC report 50-267/83-30 and containing: (1) SALP Board report dated January 9, 1984; (2) licensee response to SALP Board report; and (3) NRC response to PSC response), the Regional Administrator noted that the steps were being taken to improve performance and agreed that, if they were implemented as described, the level of performance in this area would improve.

Based on Region IV's early assessment of licensee performance in the area of plant operations, we have determined that some improvements have been made, however, a dramatic turnaround had not occurred and the findings in the previous SALP report continued to apply in principle.

We have noted that the staff audit found deficiencies and had concerns that were similar to those noted in the SALP report, and which supported the SALP Board findings.

COMMISSIONER'S VISIT

On May 21, 1984, an NRC Commissioner, accompanied by his technical assistant, the resident inspector, and a Region IV representative, toured the Fort St. Vrain plant. Following the tour, the Commissioner met with PSC management and expressed concern about housekeeping, maintenance, and operator discipline. In order to more fully assess these concerns, Region IV personnel conducted a special inspection on June 4, 1984. The results of this inspection were reported in Inspection Report 50-267/84-16 dated June 22, 1984. In addition to a Notice of Violation for failing to have a shift turnover procedure in effect, the report stated that the licensee should devote more attention to housekeeping, reduce the delay in completing maintenance activities, and maintain a more professional atmosphere in the control room.

REGIONAL ADMINISTRATOR'S MEETING WITH PSC MANAGEMENT

On June 25, 1984, the Regional Administrator, along with the Director, Division of Reactor Safety and Projects, met with executives from PSC to discuss the findings of special inspection and other concerns that had recently developed within the NRC concerning FSV.

The topics discussed at this meeting were: (1) the NRC review of PSC's compliance to Branch Technical Position 9.5-1 (fire protection); (2) the licensing requirements for building 10; (3) NRC notification of the control rod failure to automatically insert; (4) documentation of disciplinary actions taken on identified instances of procedural noncompliance; (5) equipment clearance tags in the plant that were several years old; (6) the scenario for the radiological emergency exercise; and (7) a controller in the liquid nitrogen system that had been installed in a temporary fashion in 1980 and still had not been made permanent.

The licensee responded to this meeting and addressed each of the concerns in a letter dated July 6, 1984 (PSC letter No. P-84194, O. R. Lee, PSC, to J. T. Collins, NRC). This letter is available in the Public Document Room.

NOTIFICATION TO THE NRC OF THE FAILURE OF CONTROL RODS TO AUTOMATICALLY INSERT

Following the above management meeting, a special review of the events surrounding the failure of 6 control rods to automatically insert and the notification made by PSC, was conducted by the Deputy Regional Administrator and the Acting Chief, Special Projects and Engineering Section of RIV on July 5, 1984. This was supplemented by a visit during the week of July 9, 1984, by the RIV lead license examiner. These reviews determined that the FSV operating personnel followed plant operating and emergency procedures during the event. With regard to the notification required by 10 CFR 50.72, Region IV determined through discussions with several shift supervisors that they were knowledgeable of the reporting regulations (revised as of January 1, 1984). The shift supervisors noted that these regulations are decidedly vague on the reportability of the control rod failure to automatically insert in light of the subsequent successful insertion of the rods. Although the control rod failure to automatically insert was reported early in the next morning, the failure to include this information in the initial report to the NRC operations center can be viewed in retrospect as lacking good judgement.

016

Mr. R. F. Walker, President
Public Service Company of Colorado
P. O. Box 840
Denver, Colorado 80201

DISTRIBUTION:
Docket File
TColburn
ECase
JTCollins, Rgn IV
FMiraglia
ACRS (10)
PDR
JRMiller
TColburn
PCheck, Rgn IV
RIreland, Rgn IV
PWagner, Rgn IV
DSells

Dear Mr. Walker:

In early July I directed my staff, with assistance from Region IV, to perform an audit of Fort St. Vrain operations including problem areas associated with the June 23, 1984 event regarding the failure of a number of control rods to insert.

A preliminary report (copy attached) has been developed by the staff which documents the results of our assessment. The report contains findings that the staff believes should be implemented before and after station restart. These findings are contained in the Executive Summary and at the end of each section in the body of the report. The report is preliminary in that various options to solve staff's findings are available to the licensee and need to be discussed prior to final resolution.

Public Service Company of Colorado should review and evaluate the report and determine what followup actions are appropriate. We intend to schedule a meeting to discuss your proposed actions to resolve these findings and will be in contact with you to schedule such a meeting.

Sincerely,

Original Signed by
H. R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc: See next page

8410240115 841015
PDR ADOCK 05000267
S PDR

50-267

*See previous concurrence page

ORB#3:DL*	ORB#3:DL*	ORB#3:DL*	AD:OR:DL	D:DIR	NRR
PKreutzer	TColburn:dd	JRMiller	GCLainas	DGEisenhut	HRDenton
9/18/84	9/19/84	9/19/84	10/11/84	10/15/84	10/15/84

OK'd by RIV on 10/12 10/12

Mr. R. F. Walker, President
Public Service Company of Colorado
P. O. Box 840
Denver, Colorado 80201

DISTRIBUTION:
Docket File
ECase
JTCollins, Rgn IV
DGEisenhut
FMiraglia
JRMiller
TColburn
PCheck Rgn IV
Johnson Rgn IV
RIreland Rgn IV
PWagner Rgn IV
DSells

Dear Mr. Walker:

In early July I directed my staff, with assistance from Region IV, to perform an audit of Fort St. Vrain operations covering the problem areas that have been identified including the failure of a number of control rods to insert (June 23, 1984).

The staff has completed its report, which is enclosed. The report contains findings that the staff believes should be implemented before and after station restart. These findings are contained in the Executive Summary and at the end of each section in the body of the report. I concur with the conclusions and findings contained in the staff's report.

I recommend that Public Service Company of Colorado review the report and that we schedule a meeting to discuss your comittments to resolve these findings. The Region IV Regional Administrator will contact you to arrange a meeting.

Sincerely,

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
As stated

ORB#3:DL PKreutzer 9/ /84	ORB#3:DL TColburn 9/ /84		
ORB#3:DL JRMiller/dn 9/ /84	RGN:IV JTCollins 9/ /84	AD:OR:DL GCLainas 9/ /84	D:DIR DGEisenhut 9/ /84

See Previous Concurrence
See Previous Concurrence

DISTRIBUTION:
Docket File
ECase
JTCollins, Rgn IV
DGEisenhut
FMiraglia
JRMiller
TColburn
PCheck Rgn IV
Johnson Rgn IV
RIreland Rgn IV
PWagner Rgn IV
DSells

Mr. R. F. Walker, President
Public Service Company of Colorado
P. O. Bcx 840
Denver, Colorado 80201

Dear Mr. Walker:

In early July I directed my staff, with assistance from Region IV, to perform an audit of Fort St. Vrain operations covering the problem areas that have been identified including the failure of a number of control rods to insert (June 23, 1984).

The staff has completed its report, which is enclosed. The report contains findings that the staff believes should be implemented before and after station restart. These findings are contained in the Executive Summary and at the end of each section in the body of the report. I concur with the conclusions and findings contained in the staff's report.

I recommend that Public Service Company of Colorado review the report and that we schedule a meeting to discuss your comittments to resolve these findings Mr. John T. Collins will contact you to arange a meeting.

Sincerely,

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
As stated

See Brown

ORB#3:DL	ORB#3:DL		
PKreutzer	TColburn		
9/ /84	9/ /84		
ORB#3:DL	RGN:IV	AD:OR:DL	D:DIR
JRMiller/dn	JTCollins	GCLainas	DGEisenhut
9/27/84	9/26/84	9/26/84	9/ /84

DISTRIBUTION:

Docket File	TColburn	
Case	Check Rgn IV	
JTCollins Rgn IV		Johnson Rgn IV
DGEisenhut	RIreland Rgn IV	
FMiraglia	PWagner Rgn IV	
JRMiller	DSells	

Mr. R. F. Walker, President
Public Service Company of Colorado
P. O. Box 840
Denver, Colorado 80201

Dear Mr. Walker:

In early July I directed my staff, with assistance from Region IV, to perform an audit of Fort St. Vrain operations covering the problem areas that have been identified including the failure of a number of control rods to insert (June 23, 1984).

The staff has completed its report, which is enclosed. The report contains recommendations that the staff believes should be implemented before and after station restart. These recommendations are contained in the Executive Summary and at the end of each section in the body of the report. I concur with the recommendations contained in the staff's report.

I recommend that Public Service Company of Colorado review the report and that we schedule a meeting to discuss your comittments to resolve these findings Mr. John T. Collins will contact you to arange a meeting.

Sincerely,

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
As stated

See previous concurrence

ORB#3:DL	ORB#3:DL	ORB#3:DL	RGN:IV	AD:OR:DL	D:DIR
PKreutzer	TColburn	JRMiller:dd	JTCollins	GCLainas	DGEisenhut
9/ /84	9/ /84	9/19/84	9/ /84	9/ /84	9/ /84

DISTRIBUTION:
Docket File
Case
JTCollins Rgn IV
DGEisenhut
FMiraglia
JRMiller
GHolahan
TColburn
Check Rgn IV
Johnson Rgn IV
RIreland Rgn IV
PWagner Rgn IV
DSells

Mr. R. F. Walker, President
Public Service Company of Colorado
P. O. Box 840
Denver, Colorado 80201

Dear Mr. Walker:

In early July I directed my staff, with assistance from Region IV, to perform an audit of Fort St. Vrain operations covering the problem areas that have been identified since the failure of a number of control rods to insert (June 23, 1984).

The staff has completed its report, which is enclosed. The report contains recommendations that the staff believes should be implemented before and after station restart. These recommendations are contained in the Executive Summary and at the end of each section in the body of the report. I concur with the recommendations contained in the staff's report.

I recommend that Public Service Company of Colorado review the report and that we schedule a meeting to answer any questions that you may have. The Fort St. Vrain project manager, Mr. Phil Wagner, should be contacted to arrange this meeting.

Sincerely,

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
As stated

ORB#3:DL
DSells
9/18/84

ORB#3:DL	ORB#3:DL	ORB#3:DL	RGN:IV	AD:OR:DL	D:DIR
PKreutzer	TColburn	JRMiller:dd	JTCollins	GCLainas	DGEisenhut
9/18/84	9/19/84	9/19/84	9/ /84	9/ /84	9/ /84
	TBC				