MAR 19 1986

In Reply Refer To:
Docket: 50-267/85-17

Public Service Company of Colorado
ATTN: R. F. Walker, President
P. O. Box 840
Denver, Colorado 80201-0840

Gentlemen:

This provides additional information concerning our letter of October 31, 1985, which acknowledged your response (your letter serial P-85367 dated October 16, 1985) to our Notice of Violation and inspection report dated September 16, 1985.

Background. In your letter of October 16, 1985, you stated that you did not believe that single circulator trips should be considered reportable under the provisions of 10 CFR 50.72 and 10 CFR 50.73. You additionally stated that you believed RWP system actuations were not reportable. We forwarded your stated positions to the Office of Nuclear Reactor Regulation (NRR).

Interpretation. We have recently received a response from NRR. Based on this response, we are pleased to inform you that your contention that single circulator trips and RWP system actuations are not considered to be reportable has been sustained. A single circulator trip might, however, lead to a reportable condition, for example inadequate flow for the power level.

Action. As the result of this interpretation, Violation A for "Failure to Report" in our Notice of Violation dated September 16, 1985, is herewith withdrawn and considered for record purposes not to have been a violation.

Sincerely,

Original Signed By
Ramon E. Hall

E. H. Johnson, Director
Division of Reactor Safety
and Projects

cc:
J. W. Gahm, Manager, Nuclear
Production Division
Fort St. Vrain Nuclear Station
16805 WCR 19½
Platteville, Colorado 80651

8603280202 860319
PDR ADOCK 05000267
Q PDR

(cont. on next page)

RIV:RPB/A	RPB	EO	DRSP
JPJaudon/lk	for JEGagliardo	DAPowers	EHJohnson
3/20/86	3/20/86	3/20/86	3/20/86

IE01

L. Singleton, Manager, Quality
Assurance Division
(same address)

Colorado Radiation Control Program Director

bcc distrib. by RIV:

RPB	DRSP
Resident Inspector	R. D. Martin, RA
Section Chief (RPB/A)	RSB
Section Chief (RSB/ES)	R&SPB
MIS System	RSTS Operator
RIV File	

Public Service™

Public Service
Company of Colorado

16805 Weld County Road 19 1/2, Platteville, CO 80651

OCT 21 1985

October 16, 1985
Fort St. Vrain
Unit No. 1
P-85367

Regional Administrator
Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

Attn: Mr. E. H. Johnson

Docket No. 50-267

SUBJECT: I&E Inspection Report 85-17

REFERENCE: NRC Letter, Johnson to Lee, Dated 09/16/85 (G-85381)

Dear Mr. Johnson:

This letter is in response to the Notice of Violation (NOV) received as a result of an inspection conducted at Fort St. Vrain during the period of June 17, 1985, through August 16, 1985. The following response to the items contained in the Notice of Violation is hereby submitted:

1. Failure to Report

10CFR50.72, "Immediate Notification Requirements of Operating Nuclear Power Reactors, in paragraph (b) non-emergency events, (2) Four-hour reports, requires, in part, "... the licensee shall notify the NRC as soon as practical and in all cases within four hours of the occurrence of any of the following: ... (ii) Any event or condition that results in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)."

10CFR50.73, "Licensee Event Report System", also requires in part, a 30 day written report of Reactor Protection System actuations.

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The licensee's Technical Specification 2.9 states, "The plant protective system is the reactor protective circuitry and the circuitry oriented towards protecting various plant components from major damage. This system includes (1) scram, (2) loop shutdown, (3) circulator trip, and (4) rod withdraw prohibit."

Contrary to the above, the licensee has considered circulator trips as not reportable. The licensee's daily logs list 9 circulator trips in 1984 and 7 circulator trips to date in 1985. Of these 17 circulator trips, only the trip of August 11, 1985, was reported to the NRC.

This is a Severity Level IV violation (Supplement I.D) (50-267/85-17).

(1) The reason for the violation, if admitted:

Public Service Company of Colorado does not consider single circulator trip actuations to be reportable per 10CFR50.72 or 50.73. The reasons for this determination are related to the original design and definition of FSV's Plant Protective System (PPS). The PPS was originally designed to detect and initiate automatic action upon the onset of abnormal core parameters or abnormal equipment operation parameters. When the reporting requirements of 10CFR50.72 and 50.73 were initially proposed, the term Reactor Protection System (RPS) was not recognized for this plant. Through extensive FSAR design basis/accident analysis review and review of industry practice/precedent, it was determined that RPS was equivalent to the FSV Scram System. Reactor Protection System terminology is identified specifically in the Standard Technical Specification, Section 2.2.1, Reactor Trip System Instrumentation Setpoints. The FSV equivalence of this section is LSSS 3.3, which identifies scram, loop shutdown, and steam water dump actuations. FSAR accident analyses clearly rely only on scram and loop shutdown/steam water dump actuation to ultimately mitigate postulated accidents.

Although the PPS includes the reactor protective circuitry (scram) and engineered safety feature circuitry (steam water dump, and loop shutdown, FSAR Criterion 14), it also includes numerous other equipment protection functions. This is clearly stated in the FSV Technical Specifications, Section 2.9 (which was cited in the Notice of Violation):

> "The plant protective system is the reactor protective circuitry and the circuitry oriented towards protecting various plant components from major damage. This system includes (1) scram, (2) loop shutdown, (3) circulator trip, and (4) rod withdraw prohibit".

A single circulator trip is actuated by the equipment protection circuitry, as evidenced by the basis for Specification LCO 4.4.1.c which states in part:

> "All circulator shutdown inputs (except circulator speed high on water turbines) are equipment protection items which are tied to 'two loop trouble' through the loop shutdown system."

PSC contends that, provided it is clear that an actuation is a result of an identified source other than RPS or ESF actuation, then the actuation is not reportable. The following discussion demonstrates that a single circulator trip by itself does not cause a RPS or ESF actuation, nor is it caused by a RPS or ESF actuation.

Although a single circulator trip does provide an input to the loop shutdown logic, and therefore the reactor scram logic (indirectly through the two loop trouble logic), a single circulator trip, by itself, does not actuate any RPS or ESF system. A similarity in the LWR case would be a condition which resulted in one input in a "one out of two taken twice" logic for initiation of an RPS or ESF system. In the LWR case, this is not reportable per 10CFR50.72 or 50.73 since the RPS or ESF has not yet been actuated.

Figure 1 shows a simplified schematic for the inputs that can cause a circulator trip. Those inputs listed under equipment malfunctions are the only ones which result in a single circulator trip. Note that each parameter is associated with an abnormal condition for a single circulator. The ESF (loop shutdown) actuation of a circulator trip always causes both circulators in the loop to trip. This latter is a reportable event.

A similar situation in a BWR is where a reactor recirculation pump trips due to equipment protection circuitry, which is not reportable. However, a trip of both reactor coolant pumps due to actuation of the Anticipated Transient Without Scram (ATWS) protection circuitry is reportable.

RPS or ESF actuation can cause a simultaneous trip of both circulators in a given loop as part of loop shutdown. However, there is no case where RPS or ESF actuation can cause the trip of a single circulator in a loop.

The above reasoning highlights the basis for determining that single circulator trip actuations are not reportable in accordance with 10CFR50.72(b)(2)(ii) or 50.73(a)(iv). However, trip actuations and deficiencies associated with helium circulator operation are routinely reviewed and evaluated for reportability in accordance with other 50.72 and 50.73 criteria for other safety considerations. Circulator trip actuations and abnormal operations are also routinely reviewed and investigated for plant availability concerns. Although the circulators were designed to operate independently of one another, and the Technical Specifications only require that one circulator in each loop be operable during power operation, a single inoperable circulator would significantly limit plant operation.

PSC also contends that RWP actuations are not reportable per 10CFR50.72 or 50.73. The following discussion presents the basis for our determination.

As previously stated, FSAR and industry reviews originally determined that the RPS function as identified in 10CFR50.72 and 50.73 was equivalent to the FSV Scram circuitry. All the rod withdrawal accidents evaluated in Section 14.2 of the FSAR relied exclusively on the redundancy and reliability of the Scram system to initiate the required protective action. RWP parameters and setpoints were recognized as available, but were assumed to fail. The rod withdrawal accidents ultimately rely on the 140% power scram setpoint to terminate any credible increase in core reactivity, with no fuel failure or breach of the primary coolant boundary assumed. The combination of the reactor scram circuitry and the loop shutdown/steam water dump circuitry provide the necessary automatic reactor protective and engineered safety feature actions for all FSAR postulated accidents.

Although both systems were designed under the same basic design criteria as integral parts of the overall PPS, their basic functions distinguish their safety significance. The scram system actuates to deenergize the control rod brake power supplies, allowing for gravity driven insertion of all 37 control rod pairs. The Technical Specification definition of control rod operability is consistent with this function, as surveillance testing only considers free-fall insertion capability. The FSAR and original design specification also clearly portray this system as the reactor safety system.

As stated in the original PPS design specification, the basis for development of the RWP system was that:

> "During certain combinations of plant operating conditions, control rod withdrawal must be prohibited, but the conditions do not warrant plant shutdown (scram). The RWP system accomplishes this requirement."

This definition is consistent with the function of the RWP system, which deenergizes the power to the control rod drive motor. Control rod drive or withdrawal power is associated with reactor power level control, where brake power is associated with the basic safety function of reactor shutdown. One system limits reactor operation, whereas the other limits fuel damage during postulated accident conditions.

Although these systems obtain their inputs from common detector channels, the PPS design principle for Single Failure Criterion adequately prevents system interaction due to postulated faults or failures. These systems were installed under the same quality control specifications and are maintained through specific detailed surveillance testing.

This discussion provides the basis for determination that RWP actuations do not constitute ESF or RPS actuations in accordance with 10CFR50.72 and 50.73. Abnormal RWP actuations are, however, routinely evaluated for reportability in accordance with other criteria. Since they also present an operational limit, investigation and corrective action are highly desirable.

(2) The corrective steps which have been taken and the results achieved:

Until such time as PSC and the NRC can resolve this interpretation of 10CFR50.72 and 50.73 as it applies to single circulator trip and rod withdrawal prohibit actuations, all single circulator trip and rod withdrawal prohibit actuations not resulting from or a part of a preplanned sequence during testing or reactor operation, will be reported to the NRC.

(3) Corrective steps which will be taken to avoid further violations:

Upon resolution of this interpretation of 10CFR50.72 and 50.73, the following actions will be taken to eliminate any ambiguity in the future:

a) The procedure on reportable events will be revised to clarify that single circulator trip and Rod Withdrawal Prohibit actuations are not reportable.

b) Additional training will be provided to the operating staff on reportable events.

c) The FSAR will be revised to clearly define the RPS and its relationship to the PPS. Reporting requirements will be specified, and equipment protection features will be excluded.

(4) The date when full compliance will be achieved:

PSC believes that with regard to the reportability of single circulator trip, and Rod Withdrawal Prohibit actuations it has always been in full compliance with 10CFR50.72 and 50.73.

2. Violation of Limiting Condition for Operation (LCO)

LCO 4.2.7 of the Technical Specifications states, in part, "The PCRV shall not be pressurized to more than 100 psia unless: ...d) The Interspaces between the primary and secondary penetration closures are maintained at a pressure greater than primary system pressure with purified helium gas."

Contrary to the above, during the period from July 30, 1985, when PCRV pressure went above 100 psia, through August 10, 1985, no differential helium pressure was maintained in PCRV penetration interspaces B21 and B23.

This is a Severity Level IV violation (Supplement I.C.)

(1) The reason for the violation if admitted:

Valve PDV-11380, which regulates the differential pressure between the Loop 2 steam generator penetration interspaces and the cold reheat steam line, was in the closed position contrary to system operability requirements.

Operators in the Control Room had indication available to determine the differential pressure between the purified helium header and the Prestressed Concrete Reactor Vessel (PCRV), and they based their decision on compliance with LCO 4.2.7 on this indication (which was the correct determination for all but the Loop 2 steam generator pentration interspaces). However, they did not have Control Room indication of the differential pressure between the Loop 2 steam generator penetration interspaces and the PCRV cold reheat steam line. Therefore, they should have solicited this information from the Equipment Operator who obtains the information locally.

Equipment Operators log the Loop 2 steam generator interspace/cold reheat steam line differential pressure (PDI-11380) every eight hours, but they did not recognize that a minimum differential pressure of about five psid should have been expected when the PCRV is pressurized to greater than 100 psia. The value logged was zero psid.

The Equipment Operator Log Sheet did not adequately address the subject log entry as being associated with a Technical Specification LCO.

(2) The corrective steps which have been taken and the results achieved:

All personnel involved received a formal reprimand.

An operator adjusted the controller to about 25% open, and it was verified that PDV-11380 was operable with the controller in the automatic mode and that PDI-11380 was reading over 20 psid.

Pressure differential controller, PDC-11380, has been verified operable through functional testing.

(3) Corrective steps which will be taken to avoid further violations:

The procedure, OPOP-IBI, for raising the PCRV pressure over 100 psia will be revised to include reading PDI-11380 and assuring that the Loop 2 steam generator pentration interspaces are above cold reheat pressure.

The Turbine Equipment Operator log will be revised to designate that the steam generator penetration interspaces/cold reheat steam differential pressure is required to be in accordance with LCO 4.2.7 when PCRV pressure is above 100 psia.

The date when full compliance will be achieved:

Full compliance has been achieved, and the procedure and log revisions identified above will be completed by November 30, 1985.

Should you have any questions, please contact Mr. M.H. Holmes, (303) 571-8409.

Sincerely,

J. W. Gahm
Manager, Nuclear Production
Fort St. Vrain Nuclear
Generating Station

JWG:jlr

EQUIPMENT MALFUNCTIONS

GENERAL 2 OF 3 LOGIC

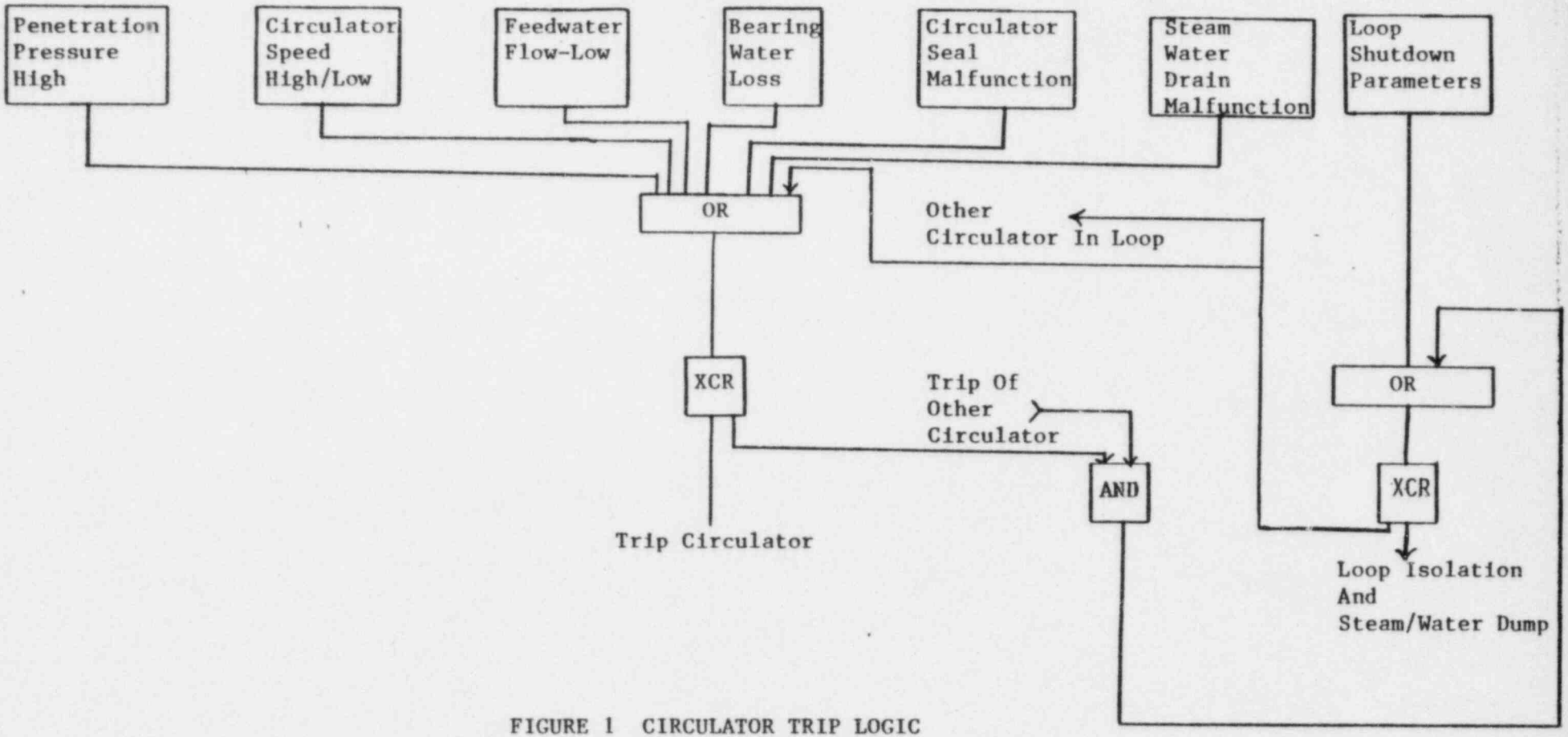

FIGURE 1 CIRCULATOR TRIP LOGIC