**Public Service Company of Colorado**

October 29, 1979
Fort St. Vrain
Unit No. 1
P-79249

Mr. Domeni. B. Vassallo, Acting Director
Division of Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Docket #50-267

Subject: Followup Actions Resulting From the NRC Reviews Regarding the Three Mile Island Unit 2 Accident

References:
1) NRC Letter D.B. Vassallo to J.K. Fuller Dated September 13, 1979
2) NRC Letter D.B. Vassallo to J.K. Fuller Dated July 25, 1979
3) PSC Letter P-79239, F.E. Swart to D.B. Vassallo Dated October 17, 1979

Gentlemen:

Enclosed are the PSC replies to the above referenced NRC letters concerning the NRC followup actions on the TMI-2 Lessons Learned Task Force Report, NUREG-0578.

It should be noted that a meeting was held on May 2, 1979 with the Division of Project Management, Advanced Reactors Branch, to discuss the Fort St. Vrain High Temperature Gas Cooled Reactor in light of the TMI-2 incident. The purpose of this meeting was to discuss the unique design and operating characteristics of the Fort St. Vrain HTGR and to provide other information as requested by the Staff to allow them to make a finding as to the susceptibility of the Fort St. Vrain HTGR to a TMI-2 type event.

As a result of this meeting, PSC was led to believe that the Staff had reached a conclusion and had issued an internal memo to the effect that it was not possible for the Fort St. Vrain reactor to experience an incident similar to that experienced at TMI-2, and that other incidents postulated for LWR's as a result of TMI-2 were not applicable to the Fort St. Vrain reactor design.

1281 013

7911010403

Since the meeting of May 2, 1979, PSC has not been included in the orders and bulletins that were issued to LWR owners, was not requested to respond to NUREG-0578, and was not a party to various NRC/Owner Group meetings. PSC and Fort St. Vrain were literally set aside from the NRC proceedings following the TMI-2 incident. Then on September 13, 1979, PSC was requested to commit within 30 days to various NRC positions based on the recommendations of the Lessons Learned Task Force with a rather short-term implementation schedule.

These NRC positions and the guidance and criteria given for their implementation are based on LWR technology, most of which is not applicable to gas-cooled technology. Needless to say, PSC considers this is unreasonably short notification, allowing very little time for evaluation and response to subject matter that LWR's have been developing since July, 1979. In this respect PSC does not feel it can meet the implementation schedules set forth.

In addition, PSC's review indicates that the recommendations set forth and the guidance given by the NRC were developed for LWR's without consideration of gas-cooled technology. On this basis many of the recommendations are clearly not applicable to Fort St. Vrain. Other recommendations must take into consideration the inherent safety features of an HTGR. Major differences occur during the development of an accident scenario, and there are significant differences in the consequences which result from accident scenarios in comparing gas-cooled reactors and water-cooled reactors.

Should you have questions regarding the enclosed PSC responses, please contact this office.

Very truly yours,

Frederic E. Swart

Frederic E. Swart
Nuclear Project Manager

FES/DWW:ler

Enclosures

1281 014

TABLE OF CONTENTS FOR PSC

REPLIES TO ENCLOSURE 6 OF

NRC LETTER VASSALLO TO FULLER

DATED SEPTEMBER 13, 1979

| Section | Title |
|---|---|
| Enclosure 3 to NRC Letter Vassallo to Fuller Dated Sept. 13, 1979 | Instrumentation to Monitor Containment During the Course of an Accident<br>1) Containment Pressure Monitoring<br>2) Containment Hydrogen Monitoring<br>3) Containment Water Level Monitoring |
| Enclosure 4 to NRC Letter Vassallo to Fuller Dated Sept. 13, 1979 | Installation of Remotely Operated High Point Vents in the Reactor Coolant System |
| 2.2.1a | Shift Supervisor's Responsibilities |
| 2.2.1b | Shift Technical Advisor |
| 2.2.1c | Shift and Relief Turnover Procedures |
| 2.2.2a | Control Room Access |
| 2.2.2b | Onsite Technical Support Center |
| 2.2.2c | Onsite Operational Support Center |

1281 016

Section 2.1.1 -- Emergency Power Supply Requirements for the Pressurizer Heaters, Power-Operated Relief Valves and Block Valves, and Pressurizer Level Indicators in PWRs.

NRC Position:

"Consistent with satisfying the requirements of General Design Criteria 10, 14, 15, 17, and 20 of Appendix A to 10 CFR Part 50 for the event of loss of offsite power, the following positions shall be implemented:

Pressurizer Heater Power Supply

1. The pressurizer heater power supply design shall provide the capability to supply, from either the offsite power source or the emergency power source (when offsite power is not available), a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation at hot standby conditions. The required heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability.

2. Procedures and training shall be established to make the operator aware of when and how the required pressurizer heaters shall be connected to the emergency buses. If required, the procedures shall identify under what conditions selected emergency loads can be shed from the emergency power source to provide sufficient capacity for the connection of the pressurizer heaters.

3. The time required to accomplish the connection of the preselected pressurizer heater to the emergency buses shall be consistent with the timely initiation and maintenance of natural circulation conditions.

4. Pressurizer heater motive and control power interfaces with the emergency buses shall be accomplished through devices that have been qualified in accordance with safety-grade requirements.

Power Supply for Pressurizer Relief and Block Valves and Pressurizer Level Indicators

1. Motive and control components of the power-operated relief valves (PORVs) shall be capable of being supplied from either the offsite power source of the emergency power source then the offsite power is not available.

2. Motive and control components associated with the PORV block valves shall be capable of being supplied from either the offsite power source of the emergency power source when the offsite power is not available.

3. Motive and control power connections to the emergency buses for the PORVs and their associated block valves shall be through devices that have been qualified in accordance with safety-grade requirements.

1281 017

4. The pressurizer level indication instrument channels shall be powered from the vital instrument buses. These buses shall have the capability of being supplied from either the offsite power source or the emergency power source when offsite power is not availabe."

PSC Reply to Specific NRC Concerns:

Redundant emergency power for 1) pressurizer heaters and 2) control and motive power systems for power operated relief valves and associated block valves is not required at Fort St. Vrain (FSV) for the following reasons:

A. A pressurizer and power operated relief valves are not incorporated into the FSV primary system design, as pressurizers are not utilized to establish and maintain natural circulation at hot standby conditions, and a "feed and bleed" mode of reactor coolant system operation is not used at FSV for decay heat removal.

B. As no pressurizer is incorporated into the FSV design, it follows that the requirement for powering the pressurizer level indication instrument channels from the vital instrument buses is not applicable.

Additional PSC Reply to General NRC Concerns:

The NRC has indicated concern that unavailability of the pressurizer heaters in PWR designs could necessitate the operation of high pressure emergency core systems to maintain reactor coolant pressure during operating transients. The frequency with which the high pressure emergency core cooling system operates (or in the general case, the frequency with which any safety system operates) may then exceed the previously understood and accepted design basis.

A. At FSV most safety systems are normally functioning during routine plant operations and are not challenged to function from a standby condition. These normally operating systems are designed and intended to operate for the life of the plant with only routine maintenance and repair.

B. The FSV safety systems and equipment that are not in service during normal plant operation, such as the diesel driven firewater pump and the standy diesel generators, are tested on a periodic basis per Technical Specification surveillance requirements. Based on actual plant experience to date, these standby safety systems and equipment have not experienced an excessive usage or a unanticipated number of operating cycles.

## Section 2.1.2 -- Performance Testing for BWR and PWR Relief and Safety Valves

NRC Position:

"Pressurized water reactor and boiling water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents. The licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The single failures applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test pressures shall be the highest predicted by conventional safety analysis procedures. Reactor coolant system relief and safety valve qualification shall include qualification of associated control circuitry piping and supports as well as the valves themselves."

PSC Reply:

The Prestressed Concrete Reactor Vessel (PCRV) safety valves at Fort St. Vrain (FSV) provide the ultimate protection against the primary coolant system pressure exceeding the PCRV reference pressure of 845 psig. Two safety valves are provided, either of which has adequate capacity to prevent the pressure in the PCRV from exceeding the PCRV reference pressure under the design basis accident conditions. The design basis accident for the PCRV safety valves, described in the FSV Final Safety Analysis Report (FSAR), Section 6.8, assumes the complete offset rupture of a steam generator subheader at the feedwater inlet end, which is the only credible means of substantially increasing the primary coolant pressure.

For the assumed accident to increase primary coolant pressure to the point where the safety valves are required to operate, it is necessary to postulate all of the following additional failures:

1. The primary coolant moisture monitors fail to identify, shut off the feedwater to, and dump the failed steam generator, scram the reactor and

2. the plant protective system fails to shut off the feedwater to and dump a preselected steam generator, scram the reactor on high primary coolant pressure, and

3. the operator fails to manually scram the reactor and dump and shut off feedwater to the failed steam generator.

For the assumed accident, the only two fluids present in the PCRV are helium and steam as the result of the steam generator subheader rupture. No liquid is present in the primary system. The PCRV relief valves are therefore required to operate passing 100% helium, 100% steam or a mixture of the two gases. Conditions of two-phase flow through the relief valves are not present at Fort St. Vrain.

The PCRV relief valves are Target Rock Model 69D-000 valves utilizing three stages of control. We are informed by the manufacturer that these valves are very similar to the safety valves used on BWR's.

The PCRV relief valves are not required to operate under any normal plant transient or accident condition except as previously described. Under these conditions, operation of the valves is not expected to occur during the life of the plant.

The PCRV safety valve tank, the piping leading to this tank from the penetration, and the piping and components internal to the tank conform to the requirements of Specification 11-T-1 (seismic design). The seismic analysis performed on this piping and components resulted in no loss of function for a maximum horizontal and vertical force of 7.21g and 3.40g respectfully. In addition, the PCRV safety valve tank is a secondary containment housing the piping from the PCRV, isolation valves, rupture discs and safey relief valves. In the event of leakage in these devices, the leakage will be contained within the secondary containment.

The PCRV safety valve discharge piping has also been seismically analyzed for the combination of thermal, weight and pressure stresses, and an operating base earthquake (OBE). The analysis also considered a discharge involving weight and pressure stresses and a design basis earthquake. All of the pipe hangers were analyzed for the worst case condition which is a combination of thermal, weight and pressure stresses, and a DBE earthquake. For all of these conditions, the seismic analysis resulted in no loss of function to the piping and supports.

As previously discussed, the duty of the relief valves is minimal because it passes only helium, steam, or a mixture of the two. No liquid water is passed through the valves. Operation of this type of valve as a steam relief valve has been proven at numerous installations over a number of years.

Thus, as has been discussed above, the PCRV safety valves, piping and supports have been adequately qualified to perform their intended function under expected operating conditions for design basis transients and accidents. Therefore, further qualification testing is not necessary of the safety valve and piping installation at Fort St. Vrain.

1281 020

Section 2.1.3a -- Direct Indication of Power-Operated Relief Valve and Safety Valve Position for PWRs and BWRs

NRC Position:

"Reactor system relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe."

PSC Reply:

The reactor containment system of the FSV reactor does not utilize power-operated safety or relief valves. Unlike PWR's or BWR's, operation of the safety valves utilized as ultimate protection for the PCRV is not expected during system operational transients.

Normal operating pressure in the primary system of the FSV reactor is about 700 psi at 100% reactor power. The pressure protection for the PCRV is designed to relieve at about 845 psig, the design working pressure of the PCRV. The coolant used is a compressable gas, and pressure changes experienced during system transients are relatively small. No high capacity make-up systems are installed, nor would the installed primary coolant make-up systems need to be used during a small loss of coolant accident insofar as the reactor is capable of being adequately cooled with atmospheric pressure in the primary system at decay heat levels. Reactor trip signals are generated as the result of low primary system pressure to minimize the heat which must be removed during a low primary coolant pressure condition.

The PCRV relief valves are preceded in the system piping by leak-tight metallic rupture disks. The interspace between the rupture disks and the relief valves is continually monitored for pressure. An increase in pressure to 5 psig indicates a leak in the rupture disk and indicates the need for repair or replacement. Figure 2.1.3a is included for clarification. Refer to FSV FSAR Section 6.8.2 for additional information.

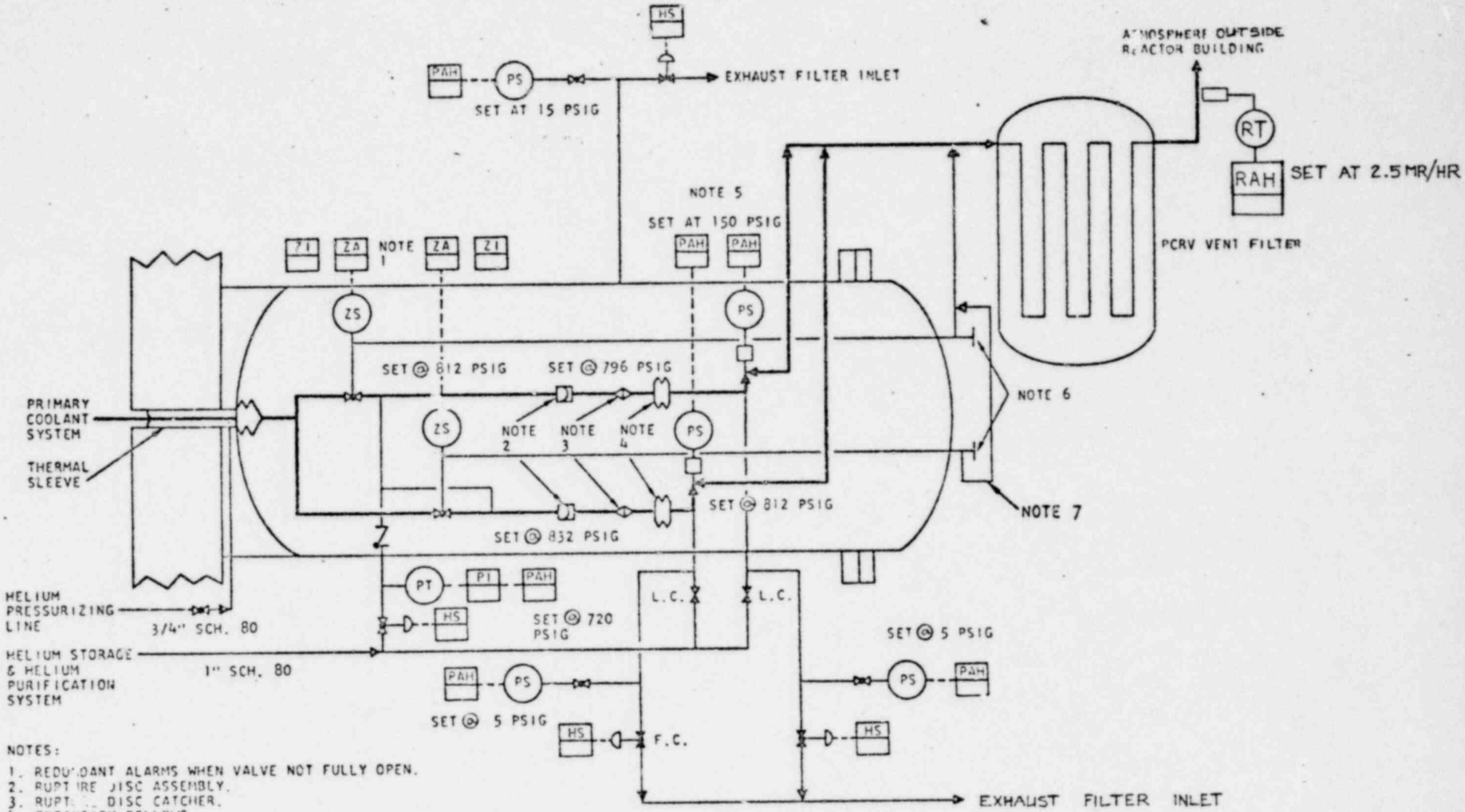
The only identified mechanism for increasing reactor containment pressure to the relieving pressure of the rupture disk/relief valves would be an unchecked moisture leak into the PCRV. This condition would require the failure of the moisture monitors to detect and isolate the leaking steam generator and trip the reactor, the failure of the reactor to trip on high primary system pressure, and the failure of the operator to take corrective action. (Reference FSAR Section 6.8.1.)

In the event the rupture disk/safety valves do function, the interspace pressure alarm will sound in the control room, along with a high radiation alarm on the relief valve vent pipe to the atmosphere. These alarms will indicate to the operator that the valve has opened. The primary coolant pressure indicators in the control room will also tell the Operator an "over-pressure" condition exists.

Reset presure of the relief valves is approximately 680 psig. Reactor coolant pressure is indicated in the control room by three pressure indicators. In the event a relief valve does not close, reactor pressure will continue to drop. Operator response to this indication would be to manually isolate the relief valve.

In the event the relief valve was not isolated and the reactor coolant system depressurizes to atmospheric pressure, the reactor can still be adequately cooled at previously stated. (Reference FSAR Section 14.4.3).

PSC therefore considers that primary coolant pressure indication is the most critical parameter during any loss of coolant condition on a gas-cooled reactor and that FSV PCRV relief valve flow/position instrumentation, while indirect, adequately indicates the full range of possible relief valve conditions to operating personnel.




NOTES:

1. REDUNDANT ALARMS WHEN VALVE NOT FULLY OPEN.
2. RUPTURE DISC ASSEMBLY.
3. RUPTURE DISC CATCHER.
4. EXPANSION BELLOWS.
5. DETECT SAFETY VALVE BELLOWS LEAKAGE.
6. HANDWHEELS LOCKED OPEN.
7. VALVE STEM PACKING GLAND VENT LINES

FIG. 2.1.3a --Process flow diagram PCRV safety valve installation

1281 023

Section 2.1.3b -- Instrumentation for Detection of Inadequate Core Cooling in PWRs and BWRs

NRC Position:

1. "Licensees shall develop procedures to be used by the operator to recognize inadequate core cooling with currently available instrumentation. The licensee shall provide a description of the existing instrumentation for the operators to use to recognize these conditions. A detailed description of the analyses needed to form the basis for operator training and procedure development shall be provided pursuant to another short-term requirement, "Analysis of Off-Normal Conditions, Including Natural Circulation" (see Section 2.1.9 of this appendix).

   In addition, each PWR shall install a primary coolant saturation meter to provide on-line indication of coolant saturation condition. Operator instruction as to use of this meter shall include consideration that is not to be used exclusive of other related plant parameters.

2. Licensees shall provide a description of any additional instrumentation of controls (primary or backup) proposed for the plant to supplement those devices cited in the preceding section giving an unambiguous, easy-to-interpret indication of inadequate core cooling. A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided."

PSC Reply:

Sufficient information is presently available at Fort St. Vrain to detect inadequate core cooling based upon the following:

1. The primary coolant utilized at FSV is single-phase helium gas, eliminating the possibility of core voiding due to a change in the coolant phase. Thus, primary coolant saturation meters or level indicators are not applicable to FSV.

2. Helium circulator shaft speed indication is available in the control room for each of the four helium circulators.

3. Pressure differential instrumentation is provided on each of the four helium circulators to indicate average helium coolant flow through the core in the control room.

4. Core region outlet thermocouples are installed in the outlet of each of the 37 fuel regions at FSV to provide region temperature indication and alarm in the control room. Inadequate primary system flow would result in an increase in region temperatures which would be indicated by thermocouple response.

5. The ratio of core thermal power to core helium flow is measured, indicated and recorded in the control room.

6. Three pressure transducers provide reactor coolant pressure input to control room indication and the plant protective system. One indication is of primary coolant differential pressure across the core, which would be used by the operator to access adequacy of core cooling.

Procedures such as the "Safe Shutdown Under Highly Degraded Conditions" procedure presently exist at FSV to define operator action to be taken under slowly degrading plant conditions. Under extreme accident conditions, such as the design basis accidents, emergency procedures are utilized. For accident conditions that are less severe than design basis accidents, operator corrective actions are based upon assessments of plant conditions utilizing the formal emergency procedures as "guides" rather than strictly following a formal procedure that is responsive to a specific hypothesized incident. This operator action philosophy is appropriate as the FSV gas-cooled reactor has inherent safety characteristics (such as the ability of the core to sustain a total loss of coolant flow for a period of 30 minutes with no fuel or primary system component damage) that allow time for operator analysis of the plant abnormal condition prior to instituting corrective action.

PSC believes that operator assessment of corrective measures to be taken under accident conditions, rather than operator actions based upon following established procedures, is consistent with the plant design philosophy and is in the best interest of the health and safety of the public.

1281 025

Section 2.1.4 -- Containment Isolation Provision for PWRs and BWRs

NRC Position:

1. "All containment isolation system designs shall comply with the recommendations of SRP 6.2.4; i.e., that there be diversity in the parameters sensed for the initiation of containment isolation.

2. All plants shall give careful reconsideration to the definition of essential and non-essential systems, shall identify each system determined to be essential, shall identify each system determined to be non-essential, shall describe the basis for selection of each essential system, shall modify their containment isolation designs accordingly, and shall report the results of the re-evaluation to the NRC.

3. All non-essential systems shall be automatically isolated by the containment isolation signal.

4. The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action."

PSC Reply:

Diverse containment isolation is not directly applicable to the FSV plant because as designed and licensed, the FSV plant does not utilize a reactor containment similar to PWR and BWR reactor plants. The FSV primary coolant system is completely contained within the Prestressed Concrete Reactor Vessel (PCRV) with the vessel's steel liner, steam generator tubing and PCRV penetrations and primary closures constituting the primary containment. Secondary closures on the PCRV penetrations and the PCRV concrete structure constitute the secondary containment. There is normally no radioactive primary coolant contained in piping external to the PCRV except for very small bore primary coolant sample lines used to draw samples for radiological and chemical analyses. To further confine and process any accidental radioactive releases, the PCRV and reactor plant associated systems are located in a "reactor building". The reactor building is a vented tertiary confinement containing a continuously operating ventilation system, including high efficiency particulate air filters (HEPAs) and charcoal adsorbers that process any accidental radioactive releases in the ventilation system's exhaust stack.

Isolation valves are provided in all piping that passes through PCRV penetrations in compliance with Design Criterion 53 in Appendix C of the FSV FSAR, which states "penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus. Automatic operation of these isolation valves is initiated by the detection of radiation in the external system, in the reactor building, or by detection of higher-than-expected flow rates as appropriate to the individual system design.

No transfer of potentially contaminated fluids is made automatically from one system to another or from the PCRV containment to systems outside the PCRV. All such transfers require specific operator action.

Transfers of potentially contaminated fluids from the reactor building for disposal outside the reactor building are made automatically. However, all contaminated fluid discharge lines are continuously monitored for radioactivity above acceptable levels, and discharges are automatically isolated and the fluid contained for processing upon receiving a high activity alarm.

Containment isolation as referred to in the NRC position stated above is not applicable to the "containment" utilized at Fort St. Vrain.

PSC has reviewed the containment philosophy implemented at Fort St. Vrain and has concluded that no modifications to systems are required.

Section 2.1.5a -- Dedicated Penetrations for External Recombiners or Post-Accident Purge Systems

NRC Position:

"Plants using external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere should provide containment isolation systems for external recombiner or purge systems that are dedicated to that service only, that meet the redundancy and single failure requirements of General Design Criteria 54 and 56 of Appendix A to 10 CFR Part 50, and that are sized to satisfy the flow requirements of the recombiner or purge system."

Section 2.1.5c -- Capability to Install Hydrogen Recombiner at Each Light Water Nuclear Power Plant

NRC Position (Minority View):

1. "All licensees of light water reactor plants shall have the capability to obtain and install recombiners in their plants within a few days following an accident if containment access is impaired and if such a system is needed for long-term post-accident combustible gas control.

2. The procedures and bases upon which the recombiners would be used on all plants should be the subject of a review by the licensees in considering shielding requirements and personnel exposure limitations as demonstrated to be necessary in the case of TMI-2."

PSC Reply:

The need for hydrogen recombiners and post-accident combustible gas control of the containment atmosphere of the Fort St. Vrain reactor is precluded since the basic design of the FSV gas-cooled reactor is substantially different from PWR's and BWR's.

The Fort St. Vrain gas-cooled reactor incorporates a ceramic core (graphite) cooled by an inert gas, helium. Under normal and most abnormal or accident conditions, no chemical reaction between reactor core or primary system materials and the inert primary coolant helium gas is expected, and therefore no hydrogen or other combustible gases are generated.

In the event moisture is accidently injected into the primary coolant as the result of a steam generator tube leak, reference FSAR Section 14.5.3, or due to an upset in the helium circulator bearing water system, the moisture and core graphite will under o a chemical reaction at normal reactor operating temperatures resulting in the production of hydrogen and carbon monoxide. Because of this reaction, provisions are made to continuously monitor the primary coolant for moisture and to automatically shut down the reactor in the event the dew point temperature of the primary coolant reaches 67°F ($\approx$ 500 ppm $H_2O$ at 700 psia). Restrictions on reactor operating temperatures as a function of primary system moisture concentration are incorporated into the plant Technical Specifications and operating procedures.

1281 028

Because hydrogen and carbon monoxide are <u>normally</u> expected to be present in the primary coolant, provisions have been incorporated into the plant design to continuously remove and process these gases.

It is therefore Public Service Company of Colorado's position that no additional provisions or processing equipment is required at the Fort St. Vrain facility to handle the combustible gases generated in the primary system as the result of the chemical reaction of the core graphite and moisture that may on occasion enter this system.

Primary coolant containing trace amounts of the combustible gases hydrogen and carbon monoxide would not accumulate inside the reactor confinement building even if the primary coolant were to be accidentally released from the PCRV. In the highly unlikely event a PCRV rupture disk/relief valve assembly should be called upon to function, the primary coolant passing through the assembly would be <u>vented to the atmosphere</u>, not into an enclosed containment structure.

In the event primary coolant should leak into the reactor confinement building, the leak would be readily detected by the installed area radiation monitors and building ventilation system exhaust stack radiation monitors. Operator action to identify and correct the leak would be initiated.

There would be no concern for an accumulation of combustible gases within the reactor confinement building insofar as the primary coolant would contain these gases in only trace amounts, and the continuing operation of the reactor building ventilation system would further reduce the combustible gas concentration and expel it to the atmosphere.

It is therefore Public Service Company of Colorado's position that no additional provisions are required for monitoring and controlling the reactor building atmosphere to detect and control the concentration of combustible gases during normal and/or post-accident conditions.

1281 029

Section 2.1.6a -- Integrity of Systems Outside Containment Likely to Contain Radioactive Materials (Engineered Safety Systems and Auxiliary Systems)

NRC Position:

Applicants and licensees shall immediately implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

1. Immediate Leak Reduction

   a. Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

   b. Measure actual leakage rates with system in operation and report them to the NRC.

2. Continuing Leak Reduction

   Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at a frequency not to exceed refueling cycle intervals.

PSC Reply:

The entire primary coolant system for the Fort St. Vrain HTGR is contained within the prestressed concrete reactor vessel (PCRV). The only system that processes primary coolant is the helium purification system. All helium purification system equipment items containing significant activity, except the hydrogen removal and regeneration equipment, are enclosed in PCRV top-head penetrations and wells. Piping from equipment enclosed in PCRV wells to areas outside the PCRV have remote manual isolation valves outside the wells. There is no need for personnel access to the PCRV wells following an accident.

The regeneration and hydrogen removal equipment located outside the PCRV contain activity that could be released through leakage or failure, but such releases would be swept away and filtered by the reactor building ventilation system. The consequences of such leakage are less significant than other postulated releases, all of which are within 10 CFR 20 limitations.

In the event of an accident involving permanent loss of forced circulation cooling, the primary coolant loop is depressurized by transferring helium to storage via the helium purification system and the primary coolant circuit is isolated. Due to the inherent characteristics of an HTGR, there is ample time to complete this depressurization before the primary coolant gas temperature or the gasborne fission product activity increase to levels affecting helium purification system performance.

Primary coolant sampling lines are the only system external to the containment that would handle liquids or gases containing large radioactive inventories after an accident. The sampling lines, only one of which is in operation at any given time, are constructed of welded 1/4" OD x 1/16" ID stainless steel tubing and are provided with automatic isolation valves actuated by area radiation monitors. Moreover, the lines are located within the Reactor Building which provides a third confinement boundary preventing direct release of radioactive materials to the environment.

The radioactive gas waste system collects, filters and monitors waste gases generated in the reactor plant and limits their discharge to rates conforming to 10 CFR 20 requirements. The consequences of total release of the maximum radioactive inventory present in the gas waste system have been evaluated in the FSAR (Section 14.6.2) and shown to be within 10 CFR 20 limitations.

The radioactive liquid waste system permits storage and identification of liquid wastes so as to permit disposal of the wastes in accordance with 10 CFR 20 limitations. Leakage from this system would be contained within the reactor building.

Each of the above systems is an active, operating system during normal plant operation. As such, any leakage would be detected by area radiation monitoring equipment and corrected. Additional periodic testing and preventive maintenance is not considered necessary for the Fort St. Vrain HTGR.

Section 2.1.6.b -- Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used In Post-Accident Operations

NRC Position:

"With the assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radioiodine and 100% of the core noble gas inventory are contained in the primary coolant), each licensee shall perform a radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility."

PSC Reply:

PSC will perform the radiation and shielding design review requested by Section 2.1.6b. PSC has established a tentative date of January 1, 1980 for submission of the results of the review to the NRC.

1281 032

Section 2.1.7.a -- Automatic Initiation of the Auxiliary Feedwater System for PWRs.

NRC Position:

"Consistent with satisfying the requirements of General Design Criterion 20 of Appendix A to 10 CFR Part 50 with respect to the timely initiation of the auxiliary feedwater system, the following requirements shall be implemented in the short term:

1. The design shall provide for the automatic initiation of the auxiliary feedwater system.

2. The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

3. Testability of the initiating signals and circuits shall be a feature of the design.

4. The initiating signals and circuits shall be powered from the emergency buses.

5. Manual capability to initiate the auxiliary feedwater system from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function.

6. The a-c motor-driven pumps and valves in the auxiliary feedwater system shall be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

7. The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFWS from the control room.

In the long term, the automatic initiation signals and circuits shall be upgraded in accordance with safety-grade requirements.

PSC Reply:

The most important difference between the HTGR and a PWR in regard to the need for automatic initiation of auxiliary feedwater is in the length of time that the reactor core can tolerate a loss of forced circulation cooling without damage. For a PWR, this time is very short, however, for the Fort St. Vrain HTGR if forced circulation is restored within five hours, there is no significant fuel damage (i.e. fuel particle coating failures), and there is no affect on the health and safety of the public. Therefore, manual actuation of the emergency cooling water systems is acceptable.

In addition, the FSV "TGR provides several back-up sources of cooling water for each of the two ondary coolant loops:

a) The emergency feedwater header automatically supplies water directly to the steam generators on loss of pressure in the main dwater line. This header also supplies feedwater to the helium circulator water turbine drives, the circulator bearing water surge tanks, the circulator bearings and the main steam desuperheaters.

b) In the event of total failure of all three boiler feedwater pumps, an emergency condensate line is provided to supply water to the steam generators and helium circulator water turbine drives. One or more of the four condensate pumps (2-powered from essential buses, 2-powered from non-essential buses) normally supply water to this line. Use of this source of cooling water is initiated by remote manual actuation of the isolation valves following steam generator depressurization.

c) Two auxiliary boiler feedwater pumps (powered from essential buses) can also supply water to the emergency condensate line via remote manual actuation of isolation valves.

d) In the event that all of the above pumps are inoperable, two fire water pumps (one diesel engine driven, one motor driven from essential buses) can be connected to the emergency condensate line via a remote manual actuated isolation valve or to the emergency feedwater header via a removable spool piece.

Each of these backup cooling methods is capable of removing stored and residual heat following a reactor scram from full power. Also, with the exception of the firewater system, all of the above systems are operating systems, not standby systems, thus their availability is, to a large extent, assured at all times.

The existing FSV backup secondary cooling systems meet the operability, testability and initiation time requirements appropriate for an HTGR.

1281 034

Section 2.1.7b -- Auxiliary Feedwater Flow Indication to Steam Generators For PWRs

NRC Position:

"Consistent with satisfying the requirements set forth in GDC 13 to provide the capability in the control room to ascertain the actual performance of the AFWS when it is called to perform its intended function, the following requirements shall be implemented:

1. Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

2. The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements of the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

PSC Reply:

Instrumentation is provided in the Fort St. Vrain Control Room to indicate steam generator cooling water flow, regardless of its source, under all normal and abnormal modes of operation.

This instrumentation includes both safety-grade and nonsafety-grade flow detectors, transmitters, controllers, monitors, indicators and recorders, including those listed in Table 2.1.7b-1.

In addition to indicating and recording steam generator cooling water flow in the Control Room, the listed instrumentation includes a low flow alarm (set at 22% of full flow) in the Control Room, remote indication of steam generator cooling water flow at the Remote Shutdown Panel in the 480V Switchgear Room and local indication of emergency feedwater flow.

Safety-grade instrumentation powered by essential busses is provided for all system control and reactor safety functions. Although some of the indicating and recording instrumentation is not classified safety grade, it is powered by an instrument bus that is fed by transformers connected to the 480V essential busses and is therefore considered highly reliable.

## TABLE 2.1.7b-1

## FEEDWATER FLOW INSTRUMENTATION

### I. Loop 1 Feedwater Flow

| Type | Number | Location* | Safety Grade |
|---|---|---|---|
| Flow Element | FE 2205 | Local | Yes |
| Flow Transmitter, High Rg. | FT 2205 | Local | Yes |
| Flow Monitors | FM 2205-1,2,7 | I35 (AEER) | Yes |
| Flow Controller | FC 2205 | I05 (CR) | Yes |
| Flow Monitors | FM 2205-4,5,6 | I35 (AEER) | No |
| Flow Indicator | FI 2205 | I49 (480V) | No |
| Flow Indicator | FI 2205-1 | I05 (CR) | No |
| Flow Indicator | FI 2205-2 | I05 (CR) | No |
| Flow Recorder | FR 2205 | I05 (CR) | No |
| Flow Switch Low | FSL 2205-1 | I70 (AEER) | No |
| Flow Alarm Low | FAL 2205 | I05 (CR) | No |
| Valve Position Indicator | ZI 2205 | I05 (CR) | Yes |
| Flow Transmitter, Low Range | FT 2207 | I55 (TB, MEZ) | Yes |
| Flow Monitor | FM 2207 | I05 (CR) | No |
| Flow Controller | FC 2207 | I35A (AEER) | No |
| Flow Transmitter (to PPS) | FT 2209 | I55 (TB, MEZ) | Yes |
| Flow Monitor | FM 2209-1 | I39 (AEER) | Yes |
| Flow Transmitter (to PPS) | FT 2211 | I55 (TB, MEZ) | Yes |
| Flow Monitor | FM 2211-1 | I40 (AEER) | Yes |
| Flow Transmitter (to PPS) | FT 2213 | I55 (TB, MEZ) | Yes |
| Flow Monitor | FM 2213-1 | I43 (AEER) | Yes |

### II. Loop 1 Emergency Condensate to S/G Reheater Section

| Type | Number | Location* | Safety Grade |
|---|---|---|---|
| Flow Element | FE 2293 | Local | Yes |
| Flow Transmitter | FT 2239 | I128 (RB, EL4759') | Yes |
| Flow Monitor | FM 2239 | I35B (AEER) | Yes |
| Flow Controller | FC 2239 | I05 (CR) | Yes |
| Flow Recorder | FR 2239 | I05 (CR) | Yes |

### III. Loop 2 Feedwater Flow

| Type | Number | Location* | Safety Grade |
|---|---|---|---|
| Flow Element | FE 2206 | Local | Yes |
| Flow Transmitter, High Rg. | FT 2206 | I54 (TB, MEZ) | Yes |
| Flow Monitors | FM 2206-1,2,7 | I36A (AEER) | Yes |
| Flow Controller | FC 2206 | I05 (CR) | Yes |
| Flow Monitors | FM 2206-4,5,6, | I36A (AEER) | No |
| Flow Indicator | FI 2206 | I49 (480V) | No |
| Flow Indicator | FI 2206-1 | I05 (CR) | No |
| Flow Indicator | FI 2206-2 | I05 (CR) | No |
| Flow Recorder | FR 2206 | I05 (CR) | No |
| Flow Switch Low | FSL 2206-1 | I70 (AEER) | No |
| Flow Alarm Low | FAL 2206 | I05 (CR) | No |
| Valve Position Indicator | ZI 2206 | I05 (CR) | Yes |

Table 2.1.7b-1 (Continued)

III. Loop 2 Feedwater Flow (Continued)

| Type | Number | Location* | Safety Grade |
|---|---|---|---|
| Flow Transmitter, 'ow Rg. | FT 2208 | I54 (TB, MEZ) | Yes |
| Flow Monitor | FM 2208 | I36A (AEER) | No |
| Flow Controller | FC 2208 | I05 (CR) | No |
| Flow Transmitter (to PPS) | FT 2210 | Local | Yes |
| Flow Monitor | FM 2210-1 | I39 (AEER) | Yes |
| Flow Transmitter (to PPS) | FT 2212 | I54 (TB, MEZ) | Yes |
| Flow Monitor | FM 2212-1 | I40 (AEER) | Yes |
| Flow Transmitter (to PPS) | FT 2214 | I54 (TB, MEZ) | Yes |
| Flow Monitor | FM 2214-1 | I43 (AEER) | Yes |

IV. Loop 2 Emergency Condensate to S/G Reheater Sections

| Type | Number | Location* | Safety Grade |
|---|---|---|---|
| Flow Element | FE 2240 | Local | Yes |
| Flow Transmitter | FT 2240 | I139 (RB, EL4759') | Yes |
| Flow Monitor | FM 2240 | I36B (AEER) | Yes |
| Flow Controller | FC 2240 | I05 (CR) | Yes |
| Flow Recorder | FR 2240 | I05 (CR) | Yes |

V. Emergency Feedwater

| Type | Number | Location* | Safety Grade |
|---|---|---|---|
| Flow Element | FE 2297 | Local | Yes |
| Flow Transmitter | FT 2297 | I71 (TB, Grade) | Yes |
| Flow Monitor | FM 2297 | I45 (AEER) | No |
| Flow Indicating Switches High | FISH 2297,8,9 | Local | Yes |
| Flow Indicator | FI 2297 | I02 (CR) | No |

IV. Steam Generator Module Feedwater Flow

| Type | Number | Location* | Safety Grade |
|---|---|---|---|
| Flow Elements | FE 2222-1 - 12 | Local | Yes |
| Flow Transmitters | FT 2222-1 - 12 | I125, I131, I134, I140 (RB, EL4799') | Yes |
| Flow Monitors | FM 2222-1 - 12 | I35B, I36B (AEER) | No |
| Flow Recorder, Multipoint | FR 2222 | I13 (CR) | No |

* Location by equipment rack number and physical location. Physical locations given as:

CR -- Control Room
AEER -- Auxiliary Electrical Equipment Room
480V -- 480V Switchgear Room
TB, MEZ -- Turbine Building, Mezzanine Level
TB, Grade -- Turbine Building, Grade Level
RB, EL4759' -- Reactor Building, Elevation 4759'

## Section 2.1.8a -- Improved Post-Accident Sampling Capability

NRC Position:

"A design and operational review of the reactor coolant and containment atmosphere sampling systems shall be performed to determine that capability of personnel to promptly obtain (less th[illegible] hours) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18 3/4 Rems to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safety obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operation review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly (less than 2 hours) quantify certain radioisotopes that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and non-volatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly; i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift."

PSC Reply:

PSC will perform the design review of post-accident sampling capability requested by Section 2.1.8a. PSC has established a tentative date of January 1, 1980 for submission of the results of the review to the NRC.

Because of the short time span available until January 1, 1980, PSC can make no commitment as to the date for implementation of revised procedures, the description of any proposed modifications or implementation of any proposed modification.

Following the design review to be completed by January 1, 1980, commitment dates for procedure revisions, description of any proposed modification and implementation of any proposed modification will be submitted.

Section 2.1.8b -- Increased Range of Radiation Monitors

NRC Position:

"The requirements associated with this recommendation should be considered as advanced implementation of certain requirements to be included in a revision to Regulatory Guide 1.97, "Instrumentation to Follow the Course of an Accident," which has already been initiated, and in other Regulatory Guides, which will be promulgated in the near-term.

1. Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions; multiple monitors are considered to be necessary to cover the ranges of interest.

   a. Noble gas effluent monitors with an upper range capacity of $10^5$ μCi/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

   b. Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (ALARA) to a maximum of $10^5$ μCi/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors should overlap by a factor of ten.

2. Since iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by adsorption on charcoal or other media, followed by onsite laboratory analysis.

3. In-containment radiation level monitors with a maximum range of $10^8$ rad/hr shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be designed and qualified to function in an accident environment."

PSC Reply:

As indicated in the PSC reply to Section 2.1.8c, a noble gas effluent monitor providing a continuous display, recording and alarm in the control room exists at FSV. The instrument range, however, is limited to 1 x $10^6$ cpm.

The FSV radio chemistry laboratory contains gamma spectroscope equipment which can be used for the analysis of effluent samples.

PSC will further evaluate noble gas effluent and in-containment radiation level monitoring and provide the NRC with a response by January 1, 1980.

Section 2.1.8c -- Improved In-Plant Iodine Instrumentation

NRC Position:

"Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident."

PSC Reply:

Equipment exists at the FSV facility to discriminate noble gases from iodine gases in the reactor building ventilation exhaust stack. The vent stack airborne iodine concentration is continuously displayed, alarmed and recorded in the control room. Two control room alarm functions are provided; the first being a trouble alarm on the iodine detector to indicate loss of background signal, loss of power, or an increased level of detected radiation above background but below the instrument setpoints, and the second being the high radiation alarm. The instrument range, however, is limited to $1 \times 10^6$ cpm.

For accident situations, instrumentation is provided that is capable of measuring iodine levels up to $1 \times 10^4$ mr/hr, but this instrumentation does not discriminate between iodine and noble gas.

PSC will review the equipment and associated training and procedures for determining airborne iodine concentration at FSV during accident conditions and will provide the results of the review to the NRC. The results of this review are tentatively scheduled for submission to the NRC by January 1, 1980.

1281 040

Section 2.1.9 -- Analysis of Design and Off-Normal Transients and Accidents

NRC Position:

"Analyses, procedures and training addressing the following are required:

1. Small break loss-of-coolant accidents;

2. Inadequate core cooling; and

3. Transients and accidents.

Some analysis requirements for small breaks have already been specified by the Bulletins and Orders Task Force. These should be completed. In addition, pretest calculations of some of the Loss of Fluid Test (LOFT) small break tests (scheduled to start in September, 1979) shall be performed as means to verify the analyses performed in support of the small break emergency procedures and in support of an eventual long-term verification of compliance with Appendix K of 10 CFR Part 50.

In the analysis of inadequate core cooling, the following conditions shall be analyzed using realistic (best-estimate) methods:

1. Low reactor coolant system inventory (two examples will be required - LOCA with forced flow, LOCA without forced flow).

2. Loss of natural circulation (due to loss of heat sink).

These calculations shall include the period of time during which inadequate core cooling is approached as well as the period of time during which inadequate core cooling exists. The calculations shall be carried out in real time far enough that all important phenomena and instrument indications are included. Each case should then be repeated taking credit for correct operator action. These additional cases will provide the basis for developing appropriate emergency procedures. These calculations should also provide the analytical basis for the design of any additional instrumentation needed to provide operators with an unambiguous indication of vessel water level and core cooling adequacy (see Section 2.1.3.b in this appendix).

The analyses of transients and accidents shall include the design basis events specified in Section 15 of each FSAR. The analyses shall include a single active failure for each system called upon to function for a particular event. Consequential failures shall also be considered. Failures of the operators to perform required control manipulations shall be given consideration for permutations of the analyses. Operator actions that could cause the complete loss of function of a safety system shall also be considered. At present, these analyses need not address passive failures or multiple system failures in the short term. In the recent analysis of small break LOCA's, complete loss of auxiliary feedwater was considered. The complete loss of auxiliary feedwater may be added to the failures being considered in the analysis of transients and accidents it if is concluded that more is needed in operator training beyond the short-term actions to upgrade auxiliary feedwater system reliability. Similarly, in the long term, multiple failures and passive failures may be considered depending in part on staff review of the results of the short-term analyses.

1281 041

The transient and accident analyses shall include event tree analyses, which are supplemented by computer calculations for those cases in which the system response to operator actions is unclear or these calculations could be used to provide important quantitative information not available from an event tree. For example, failure to initiate high-pressure injection could lead to core uncovery for some transients, and a computer calculation could provide information on the amount of time available for corrective action. Reactor simulators may provide some information in defining the event trees and would be useful in studying the information available to the operators. The transient and accident analyses are to be performed for the purpose of identifying appropriate and inappropriate operator actions relating to important safety considerations such as natural circulation, prevention of core uncovery, and prevention of more serious accidents.

The information derived from the preceding analyses shall be included in the plant emergency procedures and operator training. It is expected that analyses performed by the NSSS vendors will be put in the form of emergency procedure guidelines and that the changes in the procedures will be implemented by each licensee or applicant.

In addition to the analyses performed by the reactor vendors, analyses of selected transients should be performed by the NRC Office of Research, using the best available computer codes, to provide the basis for comparisons with the analytical methods being used by the reactor vendors. These comparisons together with comparison to data, including LOFT small break test data, will constitute the short-term verification effort to assure the adequacy of the analytical methods being used to generate emergency procedures."

PSC Response:

During the licensing review of the Fort St. Vrain reactor, careful evaluation was made of those events that could result in loss of primary system coolant and the resultant ability to provide continued cooling of the reactor core. The results of these analyses is contained in FSAR Section XIV, Parts 14.4, 14.7, 14.8 and 14.11. Also included is the analyses of the facility's ability to cope with and to provide core cooling in the event "Permanent Loss of Forced Circulation" is experienced.

In summary, the evaluation of the NRC Staff at the time of the licensing review identified the loss of primary coolant through the rupture of a 2" diameter line in the helium purification system regeneration piping. In the case of the Fort St. Vrain reactor, this is considered a "small break loss-of-coolant accident". The only other line failure that could result in the direct loss of primary coolant would be the rupture of a primary coolant sample line. This line has an inner diameter of approximately 1/8" and its short-term affect on reactor coolant inventory would be neglible.

A massive failure in the primary system with sudden release of its contents has been considered and analyzed. The results of this analysis is contained in FSAR Section XIV, Part 14.11, Design Basis Accident No.2 "Rapid Depressurization/Blowdown".

In summary, the analysis indicated that the reactor core could be adequately cooled with the primary system at "atmospheric" pressure to remove decay and residual heat.

From the standpoint of inadequate core cooling, at the time of the licensing review, an analysis of core cooling using the PCRV Liner Cooling System to remove decay and residual heat was made. The results of this analysis is contained in FSAR Section XIV, Part 14.10, Design Basis Accident No. 1, "Permanent Loss of Forced Circulation (LOFC)". This analysis also assumed the loss of one of the two PCRV Cooling Water Loops.

In summary, the results of this analysis indicated that the reactor core decay and residual heat could be removed, not without experiencing fuel particle damage, by the liner cooling system and that the health and safety of the plant operating staff and the public would not be jeopardized.

In conclusion, Public Service Company of Colorado considers the evaluations, analyses and conclusions documented in the Fort St. Vrain FSAR, Section XIV to meet the intent of the analysis requested by the Staff for small break loss-of-coolant accidents, inadequate core cooling, and transients and analysis.

The results of these analyses and the required operator action to cope with these accidents is incorporated in the facilities Operating and Emergency Procedures and are included as a part of Operator Training.

1281 043

Enclosure 3 to NRC Letter Vassallo to Fuller Dated September 13, 1979 -- Instrumentation to Monitor Containment Conditions During the Course of an Accident

NRC Position:

"Consistent with satisfying the requirements set forth in General Design Criterion 13 to provide the capability in the control room to ascertain containment conditions during the course of an accident, the following requirements shall be implemented:

1. A continuous indication of containment pressure shall be provided in the control room. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and minus five psig for all containments.

2. A continuous indication of hydrogen concentration in the containment atmosphere shall be provided in the control room. Measurement shall be provided in the control room. Measurement capability shall be provided over the range of 0 to 10% hydrogen concentration under both positive and negative ambient pressure.

3. A continuous indication of containment water level shall be provided in the control room for all plants. A narrow range instrument shall be provided for PWRs and cover the range from the bottom to the top of the containment sump. Also for PWRs, a wide range instrument shall be provided and cover the range from the bottom of the containment to the elevation equivalent to a 500,000 gallon capacity. For BWRs, a wide range instrument shall be provided and cover the range from the bottom to 5 feet above the normal water level of the suppression pool.

The containment pressure, hydrogen concentration and wide range containment water level measurements shall meet the design and qualification provisions of Regulatory Guide 1.97, including qualification, redundancy, and testability. The narrow range containment water level measurement instrumentation shall be qualified to meet the requirements of Regulatory Guide 1.89 and shall be capable of being periodically tested."

PSC Reply:

1. Containment Pressure Monitoring

The primary containment at Fort St. Vrain has three pressure monitors which indicate, record and alarm high and low pressure in the control room. The pressure range of these instruments is 0-1000 psi. There is no need for pressure instrumentation above this range due to PCRV pressure protection by two parallel relief valves set at stepped relief pressures of 796 psig and 812 psig. As an additional monitor, a recent modification installed two additional pressure transmitters and a digital control room indicator. One transmitter has a 0-100 psia range and the other has a 0-1000 psig range.

The reactor confinement building has six over-pressure monitors that alarm in the control room. The pressure range of these instruments is 0-30"$H_2O$ positive. The reactor builidng is normally maintained at -1/4"$H_2O$ via a pressure differential indicator/controller with a -2"$H_2O$ to 0"$H_2O$ range. If the reactor building reaches a positive pressure of 3"$H_2O$, the reactor building louvers will open and vent the building. The reactor building ov -pressure indicators would sense, indicate and alarm this condition.

Based on the above, no PCRV primary coolant or reactor confinement building pressure indication problems will occur at Fort St. Vrain. Existing containment pressure instrumentation is adequate.

2. Containment Hydrogen Monitoring

All reactor coolant (helium) at Fort St. Vrain, including hydrogen gas and other entrained gases, remains in the PCRV not only for cooling the reactor core but for processing and purification. Therefore, hydrogen gas inside the primary containment at Fort St. Vrain will not be detrimental to the cooling capability or safe shutdown capability of the reactor.

In the unlikely event that hydrogen gas enters the reactor confinement building at Fort St. Vrain, the gas would be vented off by the reactor building ventilation system through exhaust filters and would not be retained in the building. In addition, discharge of the primary system relief valves is through the PCRV relief filter and not to the reactor confinement building.

Based on the above, continuous control room indication of primary coolant and reactor confinement building hydrogen levels at Fort St. Vrain is not considered necessary.

3. Containment Water Level Monitoring

The nuclear reactor at Fort St. Vrain is cooled by gas and not water. Shutdown of the reactor is accomplished by control rod insertion. Emergency shutdown in the event of rod failure is accomplished by pressurized shutdown hoppers that drop boron balls into the reactor. Water is not used for shutdown or emergency core spray of the reactor in a HTGR. Venting of reactor cooling quench water and/or primary reactor coolant water to the containment sump is not applicable for Fort St. Vrain. Therefore, continuous indication in the control room of containment sump water level is not necessary at Fort St. Vrain.

1281 045

Enclosure 4 to NRC Letter Vassallo to Fuller Dated September 13, 1979 -- Installation of Remotely Operated High Point Vents in the Reactor Coolant System

NRC Position:

"Each applicant and licensee shall install reactor coolant system and reactor vessel head high point vents remotely operated from the control room. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents shall conform to the requirement of Appendix A to 10 CFR Part 50 General Design Criteria. In particular, these vents shall be safety grade, and shall satisfy the single failure criterion and the requirements of IEEE-279 in order to ensure a low probability of inadvertent actuation.

Each applicant and licensee shall provide the following information concerning the design and operation of these high point vents:

1. A description of the contruction, location, size, and power supply for the vents along with results of analyses of loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should be demonstrated to be acceptable in accordance with the acceptance criteria of 10 CFR 50.46.

2. Analyses demonstrating that the direct venting of noncondensable gases with perhaps high hydrogen concentrations does not result in violation of combustible gas concentration limits in containment as described in 10 CFR Part 50.44, Regulatory Guide 1.7 (Rev. 1), and Standard Review Plan Section 6.2.5.

3. Procedural guidelines for the operators' use of the vents. The information available to the operator for initiating or terminating vent usage shall be discussed."

PSC Reply:

The nuclear reactor at Fort St. Vrain is cooled by helium gas. Providing reactor high point vents on a HTGR is not necessary. Therefore, the recommended installation of reactor coolant system and reactor vessel head high point vents remotely operated from the control room is not applicable to Fort St. Vrain.

Section 2.2.1a -- Shift Supervisor's Responsibilities

NRC Position:

1. "The highest level of corporate management of each licensee shall issue and periodically reissue a management directive that emphasizes the primary management responsibility of the shift supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

2. Plant procedures shall be reviewed to assure that the duties, responsibilities, and authority of the shift supervisor and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the shift supervisor in the control room relative to other plant management personnel. Particular emphasis shall be placed on the following:

   a. The responsibility and authority of the shift supervisor shall be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The idea shall be reinforced that the shift supervisor should not become totally involved in any single operation in times of emergency when multiple operations are required in the control room.

   b. The shift supervisor, until properly relieved, shall remain in the control room at all times during accident situations to direct the activities of control room operators. Persons authorized to relieve the shift supervisor shall be specified.

   c. If the shift supervisor is temporarily absent from the control room during routine operations, a lead control room operator shall be designated to assume the control room command function. These temporary duties, responsibilities, and authority shall be clearly specified.

3. Training programs for shift supervisors shall emphasize and reinforce the responsibility for safe operation and the management function the shift supervisor is to provide for assuring safety.

4. The administrative duties of the shift supervisor shall be reviewed by the senior officer of each utility responsible for plant operations. Administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant shall be delegated to other operations personnel not on duty in the control room."

PSC Reply:

By January 1, 1980, the Vice President Production will issue a management directive that emphasizes the primary responsibilities of the Shift Supervisor for safe operation of the plant and that shall clearly establish the Shift Supervisors authority and responsibilities. This management directive will be subject to annual review and will be revised or updated as necessary as a result of the annual review.

Existing Administrative Procedures and Policies shall be reviewed and revised as necessary to address control room operational activities during accident conditions, lines of authority and succession, temporary relief and temporary absences. These procedures will be issued prior to January 1, 1980.

Section 2.2.1b -- Shift Technical Advisor

NRC Position:

"Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor may serve more than one unit at a multi-unit site if qualified to perform the advisor function for the various units.

The shift technical advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The shift technical advisor shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the shift technical advisors that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience."

PSC Reply:

With respect to the Task Force position for Shift Technical Advisors, PSC proposes to meet the January, 1980 requirements as follows:

1. Three engineers will be assigned the duties of site Technical Advisors and will be placed on call to respond to accident conditions at the plant.

2. The use of three engineers will ensure necessary response to the plant site, will permit on-call coverage on a rotating basis, and will ensure adequate coverage for vacations, sickness, and routine absences from the site.

3. The Engineers will be assigned to the Technical Services Department under the supervision of the Technical Services Supervisor who reports directly to the Manager of Nuclear Production. This organizational assignment will provide necessary independence from plant operations and will permit a reporting authority to a high level of management (See Figure 2.2.1b for overall organizational diagram).

4. Within the Technical Services Department, the engineers will be given the responsibilities of the Shift Technical Advisor with additional responsibilites for operational and plant maintenance engineering functions which will keep them abreast of daily plant conditions, proposed modifications, modifications in progress, operational problems, procedural and licensing changes, degrading trends, equipment and system problems, and overall plant status.

5. Although the position of Shift Technical Advisor does not require an operating license, the engineers will be required to attend initial licensing classes and will be subjected to training and internal testing to ensure their knowledge and comprehension of plant operations. The engineers will be subject to continuing training such as the operator requalification program to ensure they maintain a high level of proficiency concerning plant operations, emergency procedures, etc.

With the exception of having these personnel on shift and the recommended 10-minute response time, we believe the above proposed methods meet the requirements for the position of Shift Technical Advisor.

PSC does not consider it necessary to provide shift coverage or to meet the 10-minute response time which are criteria developed based on water reactor technology. At Fort St. Vrain accident conditions develop very slowly in comparison with water reactors, thereby providing substantially greater response times.

Fort St. Vrain has in the past experienced loss of feedwater and temporary interruption of forced reactor cooling and has recovered with no damage to fuel and/or other primary system components. Since the reactor utilizes helium as a coolant and a fully ceramic fuel and refractory-type core, it is not possible for a TMI-2 accident scenario to develop at Fort St. Vrain.

During the FSV licensing review, recovery of forced circulation cooling of the reactor following a 30-minute interruption in cooling from 100% reactor power was considered and analyzed and was found to result in no damage to the fuel or other primary system components.

An evaluation and analysis of a "permanent loss of reactor forced circulation cooling" was also made during the licensing review and is documented in the FSV FSAR. In summary, the Fort St. Vrain reactor can experience a permanent loss of forced circulation cooling without impairing the health and safety of the Plant Operating Staff or the public.

Under loss of forced circulation cooling conditions (see FSAR DBA-1, Section 14.10), we have up to five (5) hours to restore forced cooling. Even with a permanent loss of forced cooling, the maximum off-site dose over a six (6) month duration are orders of magnitude less than 10 CFR 100 limits.

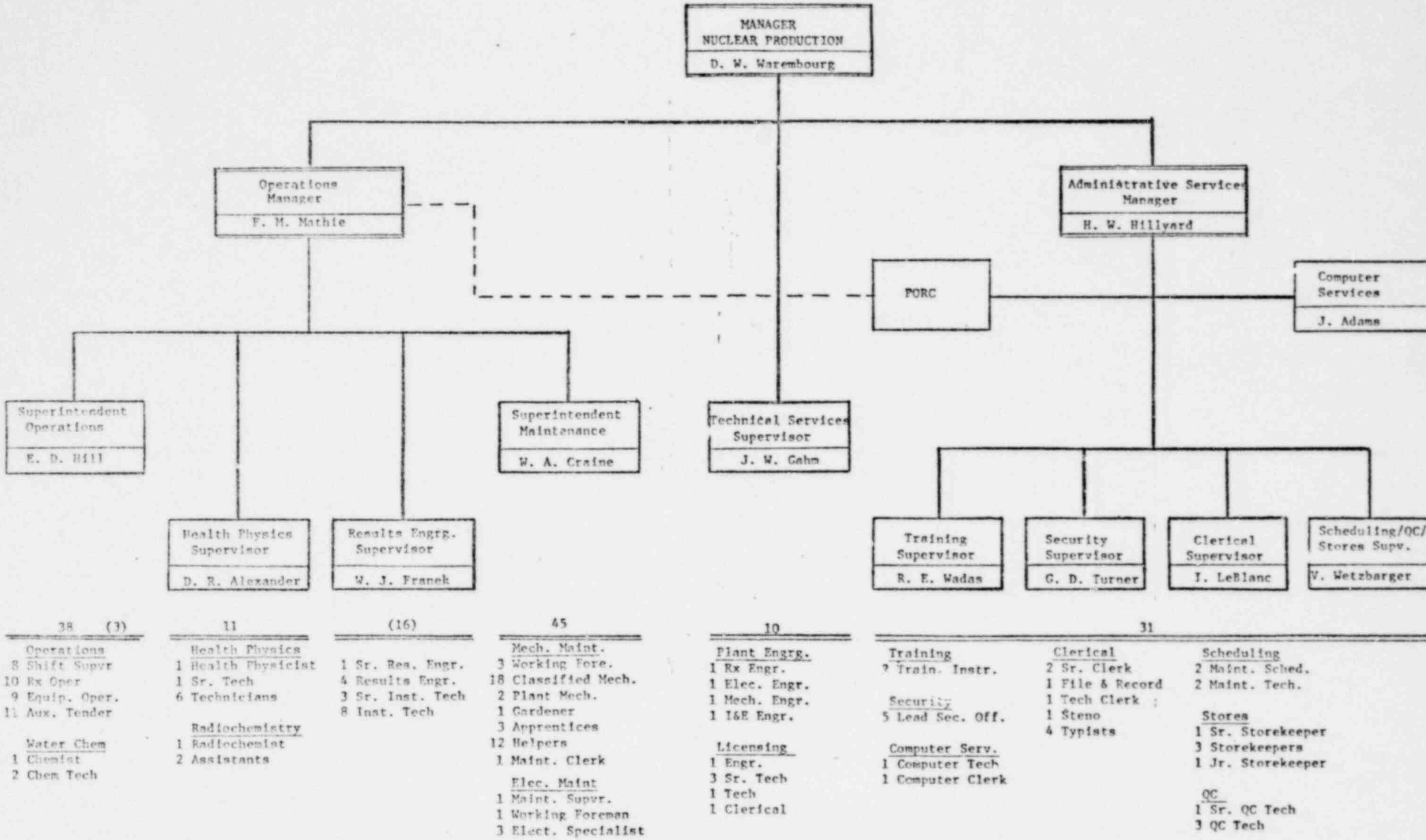
Under the maximum depressurization accident (DBA-2, see FSAR Section 14.11), a total release of the primary coolant results in off-site doses that are about one-half of 10 CFR 100 limits.

It is not possible to experience disassociation of the reactor coolant into explosive mixtures, and no change will occur in the graphite core structure even at high temperatures.

Given these inherent safety characteristics and long reactor system response times, the role of the Shift Technical Advisor is minimized and most certainly the need for shift coverage and immediate response is not warranted. PSC considers a response time of two (2) hours for the Technical Advisor to be more than adequate to respond to conditions that may develop at Fort St. Vrain.

Our proposal for utilizing on-call Technical Advisors would provide for a response of less than two hours under the most adverse conditions. This response, along with the response of the remainder of our technical staff, considering the nature and characteristics of Fort St. Vrain, will ensure more than adequate assessment of accident situations that may develop.

We would anticipate having the three (3) Technical Advisors on-call January 1, 1980. Recognizing that engineers with HTGR background may be difficult to obtain, these positions may have to be filled on a temporary basis utilizing contract personnel until a permanent staff can be developed and trained.




1281 051

Public Service Company of Colorado
Fort St. Vrain Station
Organization Chart

POOR ORIGINAL

Figure 2.2.1b

Section 2.2.1c -- Shift and Relief Turnover Procedures

NRC Position:

"The licensee shall review and revise as necessary the plant procedure for shift and relief turnover to assure the following:

1. A checklist shall be provided for the oncoming and offgoing control room operators and the oncoming shift supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist:

   a. Assurance that critical plant parameters are within allowable limits (parameters and allowable limits shall be listed on the checklist).

   b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console (what to check and criteria for acceptable status shall be included on the checklist);

   c. Identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode shall be compared with the Technical Specifications action statement (this shall be recorded as a separate entry on the checklist).

2. Checklists or logs shall be provided for completion by the offgoing and oncoming auxiliary operators and technicians. Such checklists or logs shall include any equipment under maintenance of test that by themselves could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate an operational transients (what to check and criteria for acceptable status shall be included on the checklist); and

3. A system shall be established to evaluate the effectiveness of the shift and relief turnover procedure (for example, periodic independent verification of system alignments)."

PSC Reply:

Existing Administrative Procedures for Fort St. Vrain provide for shift turnover. These administrative procedures will be reviewed and revised as necessary in light of the Lessons Learned Task Force guidelines. These procedures may require extensive review and evaluation in developing and preparing new checklists and logs. It is anticipated that such reviews and evaluations can be completed to affect an implementation date of March 1, 1980.

Section 2.2.2a -- Control Room Access

NRC Position:

"The licensee shall make provisions for limiting access to the control room to those individuals responsible for the direct operation of the nuclear power plant (e.g., operations supervisor, shift superviosr, and control room operators), to technical advisors who may be requested or required to support the operation, and to predesignated NRC personnel. Provisions shall include the following:

1. Develop and implement an administrative procedure that establishes the authority and responsibility of the person in charge of the control room to limit access.

2. Develop and implement procedures that establish a clear line of authority and responsibility in the control room in the event of an emergency. The line of succession for the person in charge of the control room shall be established and limited to persons possessing a current senior reactor operator's license. The plan shall clearly define the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside of the control room."

PSC Reply:

An Administrative Procedure will be written to formalize existing policies which allow the Shift Supervisor, the Superintendent of Operations or plant management to restrict access to the control room during both normal and emergency operations.

Existing Emergency Procedures establish the emergency director as the Shift Supervisor and provide for the succession of the person in charge of plant operations to possess a current Senior Operators License. FSV Emergency Procedures are currently under review and will be revised to explicitly incorporate control room access control and will clearly establish the lines of authority and responsibility in emergency situations.

Procedures will be revised by January 1, 1980.

Section 2.2.2b -- Onsite Technical Support Center

NRC Position:

"Each operating nuclear power plant shall maintain an onsite technical support center separate from and in close proximity to the control room that has the capability to display and transmit plant status to those individuals who are knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident. The center shall be habitable to the same degree as the control room for postulated accident conditions. The licensee shall revise his emergency plans as necessary to incorporate the role and location of the technical support center.

Records that pertain to the as-built conditions and layout of structures, systems and components shall be stored and filed at the site and accessible to the technical support center under emergency conditions. Examples of such records include system descriptions, general arrangement drawings, piping and instrument diagrams, piping system isometrics, electrical schematics, wire and cable lists, and single line electrical diagrams. It is not the intent that all records described in ANSI N45.2.9-1974 be stored and filed at the site and accessible to the technical support center under emergency conditions; however, as stated in that standard, storage systems shall provide for accurate retrieval of all pertinent information without undue delay."

PSC Reply:

1. Interim On-Site Technical Support Center

The existing Emergency Procedures establish the equivalent of the interim On-Site Technical Support Center, although by title it is called the On-Site Command Post.

The interim On-Site Command Post at Fort St. Vrain is located adjacent to the control room and, by the use of controlled access, permits ready access by specifically authorized personnel to the control room to monitor instruments and plant status during an emergency.

The On-Site Command Post will house approximately eight (8) technical support personnel and will be equipped with commercial telephones to permit primary communications with the Control Room as well as other emergency centers. In addition, the center will have back-up radio communications (battery operated). (See Figure 2.2.2b for overall emergency center plans and communications for various emergency centers.) Emergency Procedures will be revised to designate those personnel assigned to the Technical Support Center and to define the responsibilities of those personnel during accident conditions.

If for some reason the primary area designated for the On-Site Command Post cannot be utilized, alternate locations will be designated.

The interim Technical Support Center (On-Site Command Post) will be equipped with essential drawings and procedures (i.e., P&I diagrams, FSAR, one-line electrical schematics, Technical Specifications and Emergency Procedures). The location of the Center provides reasonable access to all drawings and procedures, provides reasonable control room access, and provides communication for off-site support.

From the information gained at our Regional Meeting in Las Vegas, the above will fulfill the requirements of the interim Technical Support Center required by January 1, 1980.

2. Final on-Site Technical Support Center

We are presently reviewing the requirements for the final On-Site Technical Support Center (required by January 1, 1981), and we will be addressing these requirements in future correspondence.

1281 055

EMERGENCY RESPONSE CO TROL CENTER

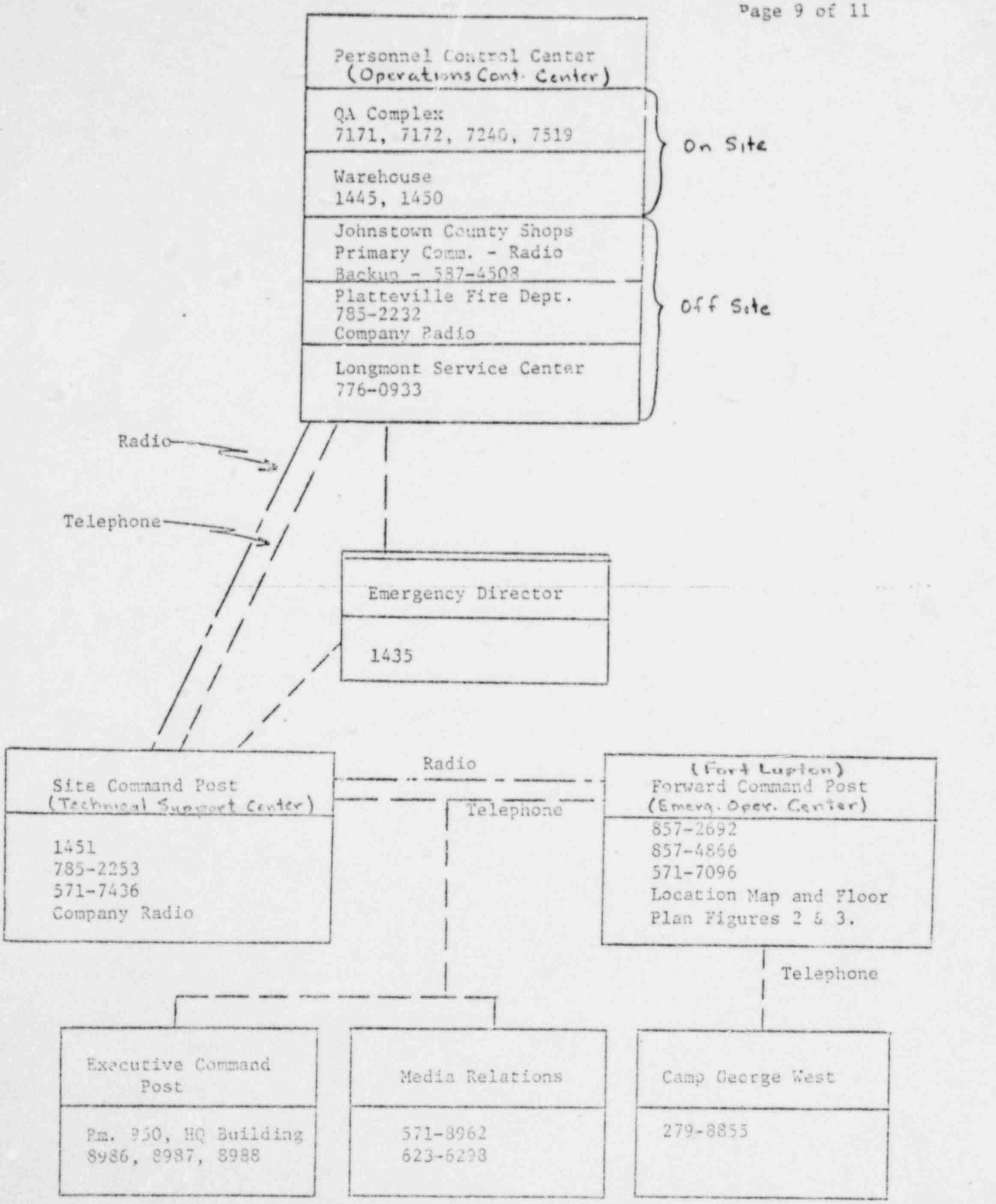



Figure 2.2.2b

1281 056

Section 2.2.2c -- Onsite Operation Support Center

NRC Position:

"An area to be designated as the onsite operational support center shall be established. It shall be separate from the control room and shall be the place to which the operations support personnel will report in an emergency situation. Communications with the control room shall be provided. The emergency plan shall be revised to reflect the existence of the center and to establish the methods and lines of communication and management."

PSC Reply:

Upon identification of an operational situation existing in the facility that may require immediate response from operational and support personnel, the plant emergency siren is sounded and, in accordance with present emergency procedures, all plant personnel, operational and support, go immediately to their "emergency stations". In general, these "emergency stations" are those locations from which they normally work. In-plant communications exist between the "emergency stations" and the Control Room so that in the event help is required, it is immediately available through the in-plant communications system. All plant supervisory personnel report to the Technical Support Center to take direction from the Shift Supervisors in responding to the event. The supervisors are charged with the responsibility of establishing and maintaining communications with the "emergency stations".

In the event plant ambient radiological conditions would require plant evacuation, the equivalent of an on-site operational Support Center has been established per the Fort St. Vrain Emergency Procedures, although by title it is called the Personnel Control Center. The Emergency Procedures also make provisions for alternate on-site Personnel Control Centers to accommodate direction of releases. Three alternate off-site locations have also been designated in the event the site should have to be evacuated.

Communications with the control room as well as other emergency centers can be maintained by commercial telephone. The Personnel Control Center Emergency Kit is also equipped with a portable radio for back-up communications.

We believe we are in full compliance wtih the requirements for an On-Site Operational Support Center.

## RESPONSE TO ENCLOSURE 7

The following response is keyed to the paragraph numbers of Enclosure 7 to NRC letter Vassallo to Fuller dated September 13, 1979:

1. Upgrade Emergency Plans

PSC has been working with the State of Colorado for some time now to upgrade the Radiological Emergency Response Plan. To date, the plan has been updated and reviewed by the IRAP and RAP Committees with the exception of the communications annex. The State is developing the communications annex and is procuring necessary communications equipment with a target date for completion in early December, 1979.

The Emergency Plan is presently scheduled for review by the NRC review team in April, 1980. PSC has, however, objected to this late review (see PSC letter P-79205 attached) on the basis that guidelines which have been published are not entirely applicable to Fort St. Vrain and that compliance to guidelines issued for water reactors are not warranted for Fort St. Vrain. To date, we have received no response to the attached letter.

With reference to the uniform action guide levels, PSC has just received NUREG-0610, and we are not in agreement with the proposed action guide levels. PSC intends to comment on NUREG-0610 by the requested comment date of December 1, 1979.

Given the position set forth by letter P-79205 attached, the December 1, 1979 comment date for NUREG-0610 and the scheduled review of April, 1980 for Fort St. Vrain Emergency Plans, it is not possible to comply with the implementation schedule set forth by Enclosure 8. It is anticipated that compliance could be achieved by mid-1980 only if the problem areas identified by letter P-79205 are resolved in a timely fashion.

2. Radiation Monitors

This subject is addressed in the responses to NRC positions 2.1.8a, 2.1.8b and 2.1.8c in this letter. The response of our radiation monitoring system was also addressed some time ago by PSC letter P-79130 which has been attached for your information.

3. Emergency Operations Center

As indicated in P-79205, PSC has made plans and arrangements to utilize a facility in the City of Fort Lupton (approximately 10 miles from the site) for the Emergency Operations Center. At the regional NRC meetings, the licensees in attendence were informed that the Emergency Operations Center should be within two (2) miles of the site. PSC is of the opinion that the Fort Lupton facility is more than adequate to meet the requirements of the Emergency Operations Center, but we need a timely resolution to our position as expressed in P-79205 before proceeding further.

Our plans for the Technical Support Center have been outlined in the response to NRC position 2.2.2b in this letter.

4. Environmental Monitoring

Again, PSC has set forth its position on environmental monitoring in PSC letter P-79205. Until specific guidelines are established for Fort St. Vrain, we cannot proceed further.

5. State/Local Plans

As indicated under Item 1 above, PSC has updated the Radiological Emergency Response Plan, and we are working with the State of Colorado to ensure appropriate emergency actions. As indicated in letter P-79205, PSC is not in agreement with the extension of the EPZ to ten (10) miles although our present plans include provisions for evacuation of the affected segment out to eight (8) miles. Until we can resolve the position set forth by letter P-79205 PSC cannot proceed further.

6. Test Exercises

The present draft of the Radiological Emergency Response Plan includes provisions for review of plans and test exercises that meet or exceed the NRC guidelines.

1281 059