Public Service Company of Colorado

December 12, 1979
Fort St. Vrain
Unit No. 1
P-79299

Mr. Steven A. Varga
Acting Assistant Director
for Light Water Reactors
Division of Project Management
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Docket # 50-267

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

Reference: 1) PSC Letter P-79249, F. E. Swart to D. B. Vassallo dated October 29, 1979

2) Denton to All Operating Nuclear Power Plants, Dated October 30, 1979

Gentlemen:

In our correspondence to you dated October 29, 1979, Reference 1, PSC responded to Mr. Vassallo's letter of September 13, 1979 to Mr. Fuller, which contained Mr. Eisenhut's letter to All Operating Nuclear Power Plants. Since submitting our response of October 29, 1979, PSC has had a number of conversations with members of Staff concerning the contents of our reply.

At a meeting attended by PSC, GAC and members of the NRC Staff on November 20, 1979, at the Fort St. Vrain site, PSC first became aware of Mr. Denton's October 30, 1979 letter, Reference 2. PSC had not received a copy of the letter through official channels at that time and to date still has not been sent a copy of the letter officially. We were handed a copy by a member of the Staff on a "for your information basis" and since have been held accountable for responding to the letter by the NRC.

PSC has, since being handed the October 30, 1979 letter, had numerous discussions over the phone with members of the Staff concerning our initial responses to the September 13, 1979 letter and the clarification and revised requirements contained in the October 30, 1979 letter.

D002 S/1 ADD: C ANDERSON C NELSON G CWALINA G IMBRO

7912270 233

The attached contains PSC clarified, and, in some cases, revised responses to the September 13, 1979 letter, based upon the clarifications contained in the October 30, 1979 letter and our discussions with the Staff.

It should be specifically noted that PSC has not committed to meet all the requirements and dates contained in the NRC's September 13 and October 30, 1979 letters. The exceptions have been taken based upon the failure of the NRC to include PSC and the Fort St. Vrain facility in owners group meetings, to include PSC on the mailing list for the orders and bulletins that were issued to LWR owners, and because of the unique physical and safety features utilized in the Fort St. Vrain High Temperature Gas-Cooled Reactor design.

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/MLP:pa

Attachments

TABLE OF CONTENTS FOR PSC

REPLIES TO NRC LETTER

"DISCUSSION OF LESSONS LEARNED

SHORT TERM REQUIREMENTS" DATED

OCTOBER 30, 1979

1647 154

1647 155

Section 2.1.1 - Emergency Power Supply Requirements

Pressurizer Heaters

NRC CLARIFICATION:

1. "In order not to compromise independence between the sources of emergency power and still provide redundant capability to provide emergency power to the pressurizer heaters, each redundant heater or group of heaters should have access to only one Class 1E division power supply.

2. The number of heaters required to have access to each emergency power source is that number required to maintain natural circulation in the hot standby condition.

3. The power sources need not necessarily have the capacity to provide power to the heaters concurrent with the loads required for LOOA.

4. Any change-over of the heaters from normal offsite power to emergency onsite power is to be accomplished manually in the control room.

5. In establishing procedures to manually reload the pressurizer heaters onto the emergency power sources, careful consideration must be given to:

   a. Which ESF loads may be appropriately shed for a given situation.

   b. Reset of the Safety Injection Actuation Signal to permit the operation of the heaters.

   c. Instrumentation and criteria for operator use to prevent overloading a diesel generator.

6. The Class IE interfaces for main power and control power are to be protected by safety-grade circuit breakers. (See also Reg. Guide 1.75)

7. Being non-Class IE loads, the pressurizer heaters must be automatically shed from the emergency power sources upon the occurrence of a safety injection actuation signal. (See item 5.b. above)"

Pressurizer Level and Relief Block Valves

NRC CLARIFICATION:

1. "While the prevalent consideration from TMI Lessons Learned is being able to close the PORV/block valves, the design should retain, to the extent practical, the capability to open these valves.

2. The motive and control power for the block valve should be supplied from an emergency power bus different from that which supplies the PORV.

1647 156

3. Any changeover of the PORV and block valve motive and control power from the normal offsite power to the emergency onsite power is to be accomplished manually in the control room.

4. For those designs where instrument air is needed for operation, the electrical power supply requirement should be capable of being manually connected to the emergency power sources."

PSC REPLY:

The PSC reply to Section 2.1.1 in the Reference 1 letter remains applicable to the NRC clarifications provided above.

1647 157

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

NRC CLARIFICATION:

1. "Expected operating conditions can be determined through the use of analysis of accidents and anticipated operational occurences referenced in Regulatory Guide 1.70.

2. This testing is intended to demonstrate valve operability under various flow conditions, that is, the ability of the valve to open and shut under the various flow conditions should be demonstrated.

3. Not all valves on all plants are required to be tested. The valve testing may be conducted on a prototypical basis.

4. The effect of piping on valve operability should be included in the test conditions. Not every piping configuration is required to be tested, but the configurations that are tested should produce the appropriate feedback effects as seen by the relief or safety valve.

5. Test data should include data that would permit an evaluation of discharge piping and supports if those components are not tested directly.

6. A description of the test program and the schedule for testing should be submitted by January 1, 1980.

7. Testing shall be complete by July 1, 1981."

PSC REPLY:

The PSC response to Section 2.1.2 in the Reference 1 letter remains applicable to the NRC clarification provided above.

Regarding structural qualification of the FSV PCRV relief value discharge piping, its piping has been analyzed by PSC considering thermal, pressure and seismic loadings and was found to meet the FSV FSAR Class I requirements. Fluid reaction loads were found to be minimal (less than 250 pounds) and were not further analyzed.

1647 158

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

NRC CLARIFICATION:

1. "The basic requirement is to provide the operator with unambigious indication of valve position (open or closed) so that appropriate operator actions can be taken.

2. The valve position should be indicated in the control room. An alarm should be provided in conjuction with this indication.

3. The valve position indication may be safety grade. If the position indication is not safety grade, a reliable single channel direct indication powered from a vital instrument bus may be provided if backup methods of determining valve position are available and are discussed in the emergency procedures as an aid to operator diagnosis and action.

4. The valve position indication should be seismically qualified consistent with the component or system to which it is attached. If the seismic qualification requirements cannot be met feasibly by January 1, 1980, a justification should be provided for less than seismic qualification and a schedule should be submitted for upgrade to the required seismic qualification.

5. The position indication should be qualified for its appropriate environment (any transient or accident which would cause the relief or safety valve to lift). If the environmental qualification program for this position indication will not be completed by January 1, 1980, a proposed schedule for completion of the environmental qualification program should be provided."

PSC REPLY:

The PSC reply to Seciton 2.1.3.a in the Reference 1 letter remains applicable to the NRC clarifications provided above.

1647 159

Section 2.1.3.b - Instrumentation for Detection of Inadequate Core Cooling

Subcooling Meter:

NRC CLARIFICATION:

1. "The analysis and procedures addressed in paragraph one above will be reviewed and should be submitted to the NRC "Bulletins and Orders Task Force" for review.

2. The purpose of the subcooling meter is to provide a continuous indication of margin to saturated conditions. This is an important diagnostic tool for the reactor operators.

3. Redundant safety grade temperature input from each hot leg (or use of multiple core exit in T/C's) are required.

4. Redundant safety grade system pressure measures should be provided.

5. Continuous display of the primary coolant saturation conditions should be provided.

6. Each PWR should have: (A) Safety grade calculational devices and display (minimum of two meters) or (B) A highly reliable single channel environmentally qualified, and testable system plus a backup procedure for use of steam tables. If the plant computer is to be used, its availability must be documented.

7. In the long term, the instrumentation qualifications must be required to be upgraded to meet the requirements of Regulatory Guide 1.97 (Instrumentation for Light Water Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident) which is under development.

8. In all cases, appropriate steps (electrical, isolation, etc.) must be taken to assure that the addition of the subcooling meter does not adversely impact the reactor protection or engineered safety features systems.

9. The attachment provides a definition of information required on the subcooling meter."

Additional Instrumentation:

NRC CLARIFICATION:

1. "Design of new instrumentation should provide an unambiguous indication of inadequate core cooling. This may require new measurements to or a synthesis of existing measurements which meet safety-grade criteria.

2. The evaluation is to include reactor water level indication.

3. A commitment to provide the necessary analysis and to study advantages of various instruments to monitor water level and core cooling is required in the response to the September 13, 1979 letter.

4. The indication of inadequate core cooling must be unambiguous, in that, it should have the following properties:

   a) it must indicate the existence of inadequate core cooling caused by various phenomena (i.e., high void fraction pumped flow as well as stagnant boil off).

   b) it must not erroneously indicate inadequate core cooling because of the presence of an unrelated phenomenon.

5. The indication must give advance warning of the approach of inadequate core cooling.

6. The indication must cover the full range from normal operation to complete core uncovering. For example, if water level is chosen as the unambiguous indication, then the range of the instrument (or instruments) must cover the full range from normal water level to the bottom of the core."

PSC REPLY:

The PSC reply to Section 2.1.3.b in the Reference 1 letter remains applicable to the NRC clarification provided above.

1647 161

Section 2.1.4 - Containment Isolation

NRC CLARIFICATION:

1. "Provide diverse containment isolation signals that satisfy safety-grade requirements.

2. Identify essential and non-essential systems and provide results to NRC.

3. Non-essential systems should be automatically isolated by containment isolation signals.

4. Resetting of containment isolation signals shall not result in the automatic loss of containment isolation."

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 radioactive gaseous 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 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 whichever is appropriate to the individual system design. Resetting the isolation signal will not result in automatic reopening of containment isolation valves. Deliberate operator action is required to return all the valve hand switches to the closed position before any of the valves can be reopened.

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.

1647 162

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 relative to potential leakage down the Helium Circulator shaft is provided, when the circulator is operating, by the buffer helium purged double labyrinth shaft seal, buffer helium seal, and bearing water seal.

When the Helium Circulator is shut down, there is a static mechanical shutdown seal. As a back-up, bearing water can be used to prevent helium leakage. Lastly, all connections to the Helium Circulator contain positive isolation valves and are designed for the pressure and temperature associated with primary coolant. Modifications to the Helium Circulator auxiliary systems are being planned which will greatly enhance the operational flexibility of those systems and will provide for separation of the individual loops to minimize operational upsets. The target date for implementation of the modifications is January 31, 1981.

In light of the differences between what is traditionally considered containment for a PWR/BWR and what constitutes containment of the HTGR, the concept of essential/non-essential systems does not apply to the Fort St. Vrain HTGR.

Section 2.1.5.a - Dedicated H_2 Control Penetrations

NRC CLARIFICATION:

1. "This requirement is only applicable to those plants whose licensing basis includes requirements for external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere.

2. An acceptable alternative to the dedicated penetration is a combined design that is single-failure proof for containment isolation purposes and single-failure proof for operation of the recombiner or purge system.

3. The dedicated penetration or the combined single-failure proof alternative should be sized such that the flow requirements for the use of the recombiner or purge system are satisfied.

4. Components necessitated by this requirement should be safety grade.

5. A description of required design changes and a schedule for accomplishing these changes should be provided by January 1, 1980. Design changes should be completed by January 1, 1981."

Section 2.1.5.c - Capability to Install Hydrogen Recombiner at Each LWR Plant

NRC CLARIFICATION:

1. "This requirement applies only to those plants that included Hydrogen Recombiners as a design basis for licensing.

2. The shielding and associated personnel exposure limitations associated with recombiner use should be evaluated as part of licensee response to requirement 2.1.6.B, "Design Review for Plant Shielding."

3. Each licensee should review and upgrade, as necessary, those criteria and procedures dealing with recombiner use. Action taken on this requirement should be submitted by January 1, 1980."

PSC REPLY:

The PSC replies to Sections 2.1.5.a and 2.1.5.c in the Reference 1 letter remain applicable to the NRC clarifications provided above.

Hydrogen can be generated in the FSV PCRV, together with carbon monoxide, as a consequence of the steam-graphite reaction during postulated water ingress accidents. Since the primary coolant (helium) is an inert gas, the helium-water-gas (i.e., hydrogen and carbon monoxide in a 1:1 ratio) mixture is not combustible.

The maximum amount of water reacting with graphite, in the event of water ingress, does not exceed 919 pounds (FSV FSAR, Amendment 26 to Chapter 14, p. 14.5-8, Case 6). Thus, a binary mixture between the generated water-gas and the helium contained in the PCRV has only 5.25% of water-gas at the most.

1647 164

If the binary mixture (5.25% water gas, 94.75% helium) is released from the PCRV through the safety relief valves, it does not exhaust to the confinement building, but flows through relief train exhaust ducts to the vent stack. Even if the mixture were to someway enter the confinement building, it would form a non-flammable ternary mixture of water-gas-helium-air. By the addition of air, the water gas fraction is reduced and equilibrium ternary mixtures formed between air and the PCRV effluent pose no flammability hazard.

1647 165

Section 2.1.6.a. - Integrity of Systems Outside Containment Likely to Contain Radioactive Materials for PWRs and BWRs

NRC CLARIFICATION:

"Licensees shall, by January 1, 1980, provide a summary description of their program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident. Examples of such systems are given on page A-26 of NUREG-0578. Other examples include the Reactor Core Isolation Cooling and Reactor Water Cleanup (Letdown function) Systems for BWRs. Include a list of systems which are excluded from this program. Testing of gaseous systems should include helium leak detection or equivalent testing methods. Consider in your program to reduce leakage potential release paths due to design and operator deficiencies as discussed in our letter to you regarding North Anna and Related Incidents dated October 17, 1979."

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 for 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 detected 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 at the exclusion area boundary.

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 highly radioactive gases 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.

1647 166

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 limited by Tech. Specs. 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.

Within the reactor building all areas containing equipment handling radioactive fluids outside the reactor vessel are continuously monitored by area, equipment and process radiation monitors. Most of these monitors are included in the Fort St. Vrain Technical Specifications which require periodic testing and calibration to assure operability and proper performance. The remainder of the monitors are on a scheduled Prevention Maintenance Program. The radioactive systems are in continuous use in normal operation (i.e. the regeneration system is operated frequently-- at least four to five times monthly at a regeneration pressure of 100 psia; the gaseous radwaste system is in continuous operation receiving gaseous waste from the helium purification system and continuous return from sample lines; and the gas waste release system is in use on an average of 8 to 10 times monthly). On the basis of the frequency of operation of the systems, the radiation monitors which are all alarmed in the control room provide continuous assessment of the integrity of these systems.

In addition, area radiation surveys are made on a routine basis (daily) to ensure as-low-as-practical levels are maintained.

Helium leakage has been reduced to as low a level as practical as a result of extensive helium leakage surveys and corrective actions performed during the construction and startup phases. The total helium inventory is determined and leakage calculated on a daily basis. Unanticipated departures from established leakage rates are investigated and corrected on an as-needed basis.

1647 167

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 CLARIFICATION:

"Any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident is designated as a vital area. In order to assure that personnel can perform necessary post-accident operations in the vital areas, we are providing the following guidance to be used by licensees to evaluate the adequacy of radiation protection to the operators:

1. Source Term

   The minimum radioactive source term should be equivalent to the source terms recommended in Regulatory Guides 1.3, 1.4, 1.7 and Standard Review Plan 15.6.5. with appropriate decay times based on plant design.

   a. Liquid Containing Systems: 100% of the core equilibrium noble gas inventory, 50% of the core equilibrium halogen inventory and 1% of all others are assumed to be mixed in the reactor coolant and liquids injected by HPCI and LPCI.

   b. Gas Containing Systems: 100% of the core equilibrium noble gas inventory and 25% of the core equilibrium halogen activity are assumed to be mixed in the containment atmosphere. For gas containing lines connected to the primary system (e.g., BWR steam lines) the concentration of radioactivity shall be determined assuming the activity is contained in the gas space in the primary coolant system.

2. Dose Rate Criteria

   The dose rate for personnel in a vital area should be such that the guidelines of GDC 19 should not be exceeded during the course of the accident. GDC 19 limits the dose to an operator to 5 Rem whole body or its equivalent to any part of the body. When determining the dose to an operator, care must be taken to determine the necessary occupancy time in a specific area. For example, areas requiring continuous occupancy will require much lower dose rates than areas where minimal occupancy is required. Therefore, allowable dose rates will be based upon expected occupancy, as well as the radioactive source terms and shielding. However, in order to provide a general design objective, we are providing the following dose rate criteria with alternatives to be documented on a case-by-case basis. The recommended dose rates are average rates in the area. Local hot spots may exceed the dose rate guidelines provided occupancy is not required at the location of the hot spot. These doses are design objectives and are not to be used to limit access in the event of an accident.

1647 168

a. Areas Requiring Continuous Occupancy: 15mr/hr. These areas will require full time occupancy during the course of the accident. The Control Room and onsite technical support center are areas where continuous occupancy will be required. The dose rate for these areas is based on the control room occupancy factors contained in SRP 6.4.

b. Areas Requiring Infrequent Access: GDC 19. These areas may require access on a regular basis, but not continuous occupancy. Shielding should be provided to allow access at a frequency and duration estimated by the licensee. The plant Radiochemical/Chemical Analysis Laboratory, radwaste panel, motor control center, instrumentation locations, and reactor coolant and containment gas sample stations are examples where occupancy may be needed often but not continuously."

PSC REPLY:

PSC will perform the radiation protection design reviews required by Section 2.1.6.b, utilizing the source terms recommended in Regulatory Guides 1.3, 1.4, and 1.7, and will submit the results of the review to the NRC by January 1, 1980. Where doses received are in excess of GDC 19 guidelines, PSC will take those steps necessary to permit post-accident operations in vital areas. Any required modifications will be completed by January 1, 1981.

1647 169

Section 2.1.7.a - Automatic Initiation of the Auxiliary Feedwater Systems (AFWS)

NRC CLARIFICATION:

"Control Grade (Short-Term)

1. Provide automatic/manual initiation of AFWS.

2. Testability of the initiating signals and circuits is required.

3. Initiating signals and circuits shall be powered from the emergency buses.

4. Necessary pumps and valves shall be included in the automatic sequence of the loads to the emergency buses. Verify that the addition of these loads does not compromise the emergency diesel generating capacity.

5. Failure in the automatic circuits shall not result in the loss of manual capability to initiate the AFWS from the control room.

6. Other Considerations

   a. For those designs where instrument air is needed for operation, the electric power supply requirement should be capable of being manually connected to emergency power sources."

PSC REPLY:

The PSC reply to Section 2.1.7.a in the Reference 1 letter remains applicable to the NRC clarifications provided above.

1647 170

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

NRC CLARIFICATION:

A. "Control Grade (Short-Term)

1. Auxiliary feedwater flow indication to each steam generator shall satisfy the single failure criterion.

2. Testability of the auxiliary feedwater flow indication channels shall be a feature of the design.

3. Auxiliary feedwater flow instrument channels shall be powered from the vital instrument buses.

B. Safety/Grade (Long-Term)

1. Auxiliary feedwater flow indication to each steam generator shall satisfy safety-grade requirements.

C. Other

1. For the Short-Term, the flow indication channels should by themselves satisfy the single failure criterion for each steam generator. As a fall-back position, one auxiliary feedwater flow channel may be backed up by a steam generator level channel.

2. Each auxiliary feedwater channel should provide an indication of feed flow with an accuracy on the order of ±10%."

PSC REPLY:

Feedwater flow is measured (1) in each feedwater header downstream of the back-up supplies, (2) in the emergency feedwater header, (3) in each emergency condensate header supplying the steam generator reheater sections and (4) at each of the twelve steam generator modules. All steam generator cooling water, regardless of its source, must flow through one or more of the flow elements. Each of the measurements is indicated in the Control Room and some of them are recorded there as well.

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.

The design of this instrumentation provides redundancy to satisfy the single failure criterion, testability, and accuracy within ± 10%.

1647 171

Safety-grade instrumentation powered by essential buses is presently provided for all necessary system control and reactor safety functions and some of the indicating/recording functions. The indicating and recording instrumentation not presently classified safety grade is powered by instrument buses that are fed by redundant transformers connected to the 480V essential buses and is, therefore, considered to be equally as reliable as safety grade instrumentation.

Nevertheless, PSC will provide fully-safety grade auxiliary feedwater flow indication in the Control Room by January 1, 1981.

1647 172

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

1647 173

Table 2.1.7b-1 (Continued)

III. Loop 2 Feedwater Flow (Continued)

Type	Number	Location*	Safety Grade
Flow Transmitter, Low 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'

1647 174

Section 2.1.8.a - Improved Post-Accident Sampling Capability

NRC CLARIFICATION:

"The licensee shall have the capability to promptly obtain (in less than 1 hour) pressurized and unpressurized reactor coolant samples and a containment atmosphere (air) sample.

The licensee shall establish a plan for an onsite radiological and chemical analysis facility with the capability to provide, within 1 hour of obtaining the sample, quantification of the following:

1. certain isotopes that are indicators of the degree of core damage (i.e., noble gases, iodines and cesiums and non-volatile isotopes),

2. hydrogen levels in the containment atmosphere in the range 0 to 10 volume percent,

3. dissolved gases (i.e., H_2, O) and boron concentration of liquids.

Or have in-line monitoring capabilities to perform the above analysis. Plant procedures for the handling and analysis of samples, minor plant modifications for taking samples and a design review and procedural modifications (if necessary) shall be completed by January 1, 1980. Major plant modifications shall be completed by January 1, 1981.

During the review of the post accident sampling capability consideration should be given to the following items:

1. Provisions shall be made to permit containment atmosphere sampling under both positive and negative containment pressure.

2. The licensee shall consider provision for purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, and for passive flow restrictions to limit reactor coolant loss or containment air leak from a rupture of the sample line.

3. If changes or modifications to the existing sampling system are required, the seismic design and quality group classification of sampling lines and components shall conform to the classification of the system to which each sampling line is connected. Components and piping downstream of the second isolation valve can be designed to quality Group D and nonseismic Category I requirements.

The licensee's radiological sample analysis capability should include provisions to:

1647 175

a. Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source terms given in Lessons Learned Item 2.1.6.b. Where necessary, ability to dilute samples to provide capability for measurement and reduction of personnel exposure, should be provided. Sensitivity of onsite analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1 μCi/gm to the upper levels indicated.

b. Restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of ventilation system design which will control the presence of airborne radioactivity.

c. Maintain plant procedures which identify the analysis required, measurement techniques and provisions for reducing background levels. The licensees chemical analysis capability shall consider the presence of the radiological source term indicated for the radiological analysis.

In performing the review of sampling and analysis capability, consideration shall be given to personnel occupational exposure. Procedural changes and/or plant modifications must assure that it shall be possible to obtain and analyze a sample while incurring a radiation dose to any individual that is as low as reasonably achievable and not in excess of GDC 19. In assuring that these limits are met, the following criteria will be used by the staff.

1. For shielding calculations, source terms shall be as given in Lessons Learned Item 2.1.6.b.

2. Access to the sample station and the radiological and chemical analysis facilities shall be through areas which are accessible in post accident situations and which are provided with sufficient shielding to assure that the radiation dose criteria are met.

3. Operations in the sample station, handling of highly radioactive samples from the sample station to the analysis facilities, and handling while working with the samples in the analysis facilities shall be such that the radiation dose criteria are met. This may involve sufficient shielding of personnel from the samples and/or the dilution of samples for analysis. If the existing facilities do not satisfy these criteria, then additional design features, e.g., additional shielding, remote handling, etc. shall be provided. The radioactive sample lines in the sample station, the samples themselves in the analysis facilities, and other radioactive lines in the vicinity of the sampling station and analysis facilities shall be included in the evaluation.

1647 176

4. High range portable survey instruments and pesonnel dosimeters should be provided to permit rapid assessment of high exposure rates and accumulated personnel exposure.

The licensee shall demonstrate their capability to obtain and analyze a sample containing the isotopes discussed above according to the criteria given in this section."

PSC REPLY:

In the Reference 1 letter to the NRC, the first paragraph in the PSC reply to the NRC Position on Seciton 2.1.4 indicated that the Fort St. Vrain plant does not utilize a reactor containment similar to PWR and BWR 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. Thus, the NRC requirement for obtaining primary coolant and "containment" atmosphere samples is applicable solely to obtaining a primary coolant sample from within the PCRV at Fort St. Vrain.

In this light, PSC commits to 1) performing a design review of the existing primary coolant sampling system, 2) performing necessary required interim minor modifications for primary coolant sample collection, in accordance with the requirements of Section 2.1.8.a by January 1, 1980.

The NRC clarification indicates the need for promptly obtaining (less than one hour) reactor coolant samples and containment samples to assess indication of possible fuel damage in accident situations.

At Fort St. Vrain, we have a continuous on-line sampler (RT 9301) that monitors primary coolant activity and provides a continuous indication of fuel degradation. This radiation monitor is supplemented by a continuous on-line analyzer monitoring core moisture and CO content. These on-line monitoring systems (which read out in the control room), provide continuous indication of possible fuel degradation.

It should also be noted as indicated in our October 29 response, that there are no accident conditions postulated for Fort St. Vrain that would result in fuel damage within one (1) hour after an accident. Fuel damage in our ceramic/refractory type core occurs slowly and over a period of hours. The response time of one (1) hour referenced, while applicable to water reactors, is totally inapplicable to Fort St. Vrain. As indicated in our October 29, 1979 response, we can lose forced cooling for periods up to five (5) hours without fuel damage.

In addition to the continuous monitoring systems mentioned above, we do have the ability to sample both the primary coolant and the reactor building atmosphere and the provision for analyzing such samples to determine the isotopic content for those accident conditions postulated in the FSAR. Likewise, samples can also be taken for analysis of CO, CO_2, H_2O content (hydrogen and boron concentration as recommended for water reactors are not primary indications of fuel damage at Fort St. Vrain).

1647 177

On-site analysis of samples, however, is predicated on the activity level in the reactor building as our radio-chem lab is presently located in this building (i.e. background level in the radio-chem lab may be too high for meaningful analysis). Options do exist, however, for obtaining off-site analysis of the samples on a contract basis. Procedures are in place for obtaining and analyzing grab samples that are within accident activity levels postulated in the FSAR.

Provisions do not exist for obtaining, handling and analyzing highly radioactive samples as postulated in the NRC position statement. Provisions for handling and analyzing such samples will be evaluated along with the high level radiation monitors required by January 1, 1981. (It should be noted that within the accident scenarios for Fort St. Vrain as presented in the FSAR, we do not have any accident situations that would result in the stack monitors going off scale nor do we have release rate concentrations that approach 10^5 ci/cc.)

With reference to the limitations of analysis capability imposed by having the radio-chem lab located in the reactor building, we are presently planning to relocate the radio-chem lab outside the reactor building. Based on present scheduling, it is anticipated that the radio-chem will be relocated by January 1, 1981.

1647 178

Section 2.1.8.b - Increased Range of Radiation Monitors

NRC CLARIFICATION:

1. "Radiological Noble Gas Effluent Monitors

   A. January 1, 1980 Requirements

   Until final implementation in January 1, 1981, all operating reactors must provide, by January 1, 1980, an interim method for quantifying high level releases which meets the requirements of Table 2.1.8.b.1. This method is to service only as a provisional fix with the more detailed, exact methods to follow. Methods are to be developed to quantify release rates of up to 10,000 Ci/sec for noble gases from all potential release points, (e.g., auxiliary building, radwaste building, fuel handling building, reactor building, waste gas decay tank releases, main condenser air ejector, BWR main condenser vacuum pump exhaust, PWR steam safety valves and atmosphere steam dump valves and BWR turbine buildings) and any other areas that communicate directly with systems which may contain primary coolant or containment gases, (e.g., letdown and emergency core cooling systems and external recombiners). Measurements/analysis capabilities of the effluents at the final release point (e.g., stack) should be such that measurements of individual sources which contribute to a common release point may not be necessary. For assessing radioiodine and particulate releases, special procedures must be developed for the removal and analysis of the radioiodine/particulate sampling media (i.e., charcoal canister/filter paper). Existing sampling locations are expected to be adequate; however, special procedures for retrieval and analysis of the sampling media under accident conditions (e.g., high air and surface contamination and direct radiation levels) are needed.

   It is intended that the monitoring capabilities called for in the interim can be accomplished with existing instrumentation or readily available instrumentation. For noble gases, modifications to existing monitoring systems, such as the use of portable high range survey instruments, set in shielded collimators so that they "see" small sections of sampling lines is an acceptable method for meeting the intent of this requirement. Conversion of the measured dose rate (mR/hr) into concentration (μCi/cc) can be performed using standard volume source calculations. A method must be developed with sufficient accuracy to quantify the iodine releases in the presence of high background radiation from noble gases collected on charcoal filters. Seismically qualified equipment and equipment meeting IEEE-279 is not required.

   The licensee shall provide the following information on his methods to quantify gaseous releases of radioactivity from the plant during an accident.

1647 179

1. Noble Gas Effluents

a. System/Method description including:

i) Instrumentation to be used including range or sensitivity, energy dependence, and calibration frequency and technique,

ii) Monitoring/sampling locations, including methods to assure representative measurements and background radiation correction,

iii) A description of method to be employed to facilitate access to radiation readings. For January 1, 1980, control room read-out is preferred: however, if impractical, in-situ readings by an individual with verbal communication with the Control Room is acceptable based on (iv) below.

iv) Capability to obtain radiation readings at least every 15 minutes during an accident.

v) Source of power to be used. If normal AC power is used, an alternate back-up power supply should be provided. If DC power is used, the source should be capable of providing continuous readout for 7 consecutive days.

b. Procedures for conducting all aspects of the measurement/analysis including:

i) Procedures for minimizing occupational exposures

ii) Calculational methods for converting instrument readings to release rates based on exhaust air flow and taking into consideration radionuclide spectrum distribution as function of time after shutdown.

iii) Procedures for dissemination of information.

iv) Procedures for calibration.

B. January 1, 1981 Requirements

By January 1, 1981, the licensee shall provide high range noble gas effluent monitors for each release path. The noble gas effluent monitor should meet the requirements of Table 2.1.8.b.2. The licensee shall also provide the information given in Sections 1.A.1.a.i, 1.A.1.a.ii, 1.A.1.b.ii, 1.A.1.b.iii, and 1.A.1.b.iv above for the noble gas effluent monitors.

1647 180

2. Radioiodine and Particulate Effluents

A. For January 1, 1980 the licensee should provide the following:

1. System/Method description including:

a) Instrumentation to be used for analysis of the sampling media with discussion on methods used to correct for potentially interfering background levels of radioactivity.

b) Monitoring/sampling location.

c) Method to be used for retrieval of handling of sampling media to minimize occupational exposure.

d) Method to be used for data analysis of individual radionuclides in the presence of high levels of radioactive noble gases.

e) If normal AC power is used for sample collection and analysis equipment, an alternate back-up power supply should be provided. If DC power is used, the source should be capable of providing continuous read-out for 7 consecutive days.

2. Procedures for conducting all aspects of the measurement analysis including:

a) Minimizing occupational exposure

b) Calculational methods for determining release rates

c) Procedures for dissemination of information

d) Calibration frequency and technique

B. For January 1, 1981, the licensee should have the capability to continuously sample and provide onsite analysis of the sampling media. The licensee should also provide the information required in 2.A above.

3. Containment Radiation Monitors

Provide by January 1, 1981, two radiation monitor systems in containment which are documented to meet the requirements of Table 2.1.8.b.2. It is possible that future regulatory requirements for emergency planning interfaces may necessitate identification of different types of radionuclides in the containment air, e.g., noble gases (indication of core damage) and non-volatiles (indication of core melt). Consequently, consideration should be given to the possible installation or future conversion of these monitors to perform this function."

1647 181

TABLE 2.1.8.b.1

INTERIM PROCEDURES FOR QUANTIFYING HIGH LEVEL

ACCIDENTAL RADIOACTIVITY RELEASES

Licensees are to implement procedures for estimating noble gas and radioiodine release rates if the existing effluent instrumentation goes off scale.

Examples of major elements of a highly radioactive effluent release special procedures (noble gas).

- Preselected location to measure radiation from the exhaust air, e.g., exhaust duct or sample line.
- Provide shielding to minimize background interference.
- Use of an installed monitor (preferable) or dedicated portable monitor (acceptable) to measure the radiation.
- Predetermined calculational method to convert the radiation level to radioactive effluent release rate.

1647 182

TABLE 2.1.8.b.2

HIGH RANGE EFFLUENT MONITOR

- NOBLE GASES ONLY
- RANGE: (Overlap With Normal Effluent Instrument Range)
  - UNDILUTED CONTAINMENT EXHAUST 10^{+5} μCi/CC
  - DILUTED (>10: 1) CONTAINMENT EXHAUST 10^{+4} μCi/CC
  - MARK I BWR REACTOR BUILDING EXHAUST 10^{+4} μCi/CC
  - PWR SECONDARY CONTAINMENT EXHAUST 10^{+4} μCi/CC
  - BUILDINGS WITH SYSTEMS CONTAINING PRIMARY COOLANT OR GASES 10^{+3} μCi/CC
  - OTHER BUILDINGS (E.G., RADWASTE) 10^{+2} μCi/CC
- NOT REDUNDANT - 1 PER NORMAL RELEASE POINT
- POWER - VITAL INSTRUMENT BUS
- SPECIFICATIONS - PER R.G. 1.97 AND ANSI N320-1979
- DISPLAY*: CONTINUOUS AND RECORDING WITH READOUTS IN THE TECHNICAL SUPPORT CENTER (TSC) AND EMERGENCY OPERATIONS CENTER (EOC)
- QUALIFICATIONS - NO

*Although not a present requirement, it is likely that this information may have to be transmitted to the NRC. Consequently, consideration should be given to this possible future requirement when designing the display interfaces.

1647 183

TABLE 2.1.8.b.3

HIGH RANGE CONTAINMENT RADIATION MONITOR

- RADIATION: TOTAL RADIATION (ALTERNATE: PHOTON ONLY)
- RANGE:
  - UP TO 10^8 RAD/HR (TOTAL RADIATION)
  - ALTERNATE: 10^7 R/HR (PHOTON RADIATION ONLY)
  - SENSITIVE DOWN TO 60 KEV PHOTONS*
- REDUNDANT: TWO PHYSICALLY SEPARATED UNITS
- SEISMIC: PER R.G. 1.97
- POWER: VITAL INSTRUMENT BUS
- SPECIFICATIONS: PER R.G. 1.97 REV. 2 AND ANSI N320-1978
- DISPLAY: CONTINUOUS AND RECORDING
- CALIBRATION: LABORATORY CALIBRATION ACCEPTABLE

*Monitors must not provide misleading information to the operators assuming delayed core damage when the 80 KEV photon Xe-133 is the major noble gas present.

1647 184

PSC REPLY:

In our October 29, 1979 response to 2.1.8.b, we neglected to indicate that actions that are indicated by our evaluation of the noble gas effluent monitors would be implemented by January 1, 1981. Our response is modified accordingly.

As indicated in our response to 2.1.8.a, we do not have any postulated accident condition that will result in our present radiation monitors going off scale. We are, however, evaluating the use of temporary portable high range monitoring equipment similar to that indicated in your October 30, 1979 letter and will make every effort to make provisions to install a portable high range monitor by January 1, 1980 along with necessary interim procedures for determining high level releases should our existing monitors go off scale.

For radioiodine effluent analysis, the existing charcoal filters in the plant iodine radiation stack monitors are routinely removed and analyzed using a computer based gamma analysis system. Procedures are in effect for their analysis. These methods and procedures are applicable for accident situations, and therefore, meet the intent of NRC's position in the October 30, 1979 letter.

The primary radioiodine effluent monitor is on the essential bus which provides an essential power supply off the emergency diesel generators upon loss of normal power.

1647 185

Section 2.1.8.c - Improved In-Plant Iodine Instrumentation Under Accident Conditions

NRC CLARIFICATION:

"Use of Portable Versus Stationary Monitoring Equipment

Effective monitoring of increasing iodine levels in the buildings under accident conditions must include the use of portable instruments for the following reasons:

a. The physical size of the auxiliary/fuel handling building precludes locating stationary monitoring instrumentation all areas where airborne iodine concentration data might required.

b. Unanticipated isolated "hot spots" may occur in locations where no stationary monitoring instrumentation is located.

c. Unexpectedly high background radiation levels near stationary monitoring instrumentation after an accident may interfere with filter radiation readings.

d. The time required to retrieve samples after an accident may result in high personnel exposures if these filters are located in high dose rate areas.

Iodine Filters and Measurement Techniques

A. The following are short-term recommendations and shall be implemented by the licensee by January 1, 1980. The licensee shall have the capability to accurately detect the presence of iodine in the region of interest following an accident. This can be accomplished by using a portable or cart-mounted iodine sampler with attached single channel analyzer (SCA). The SCA window should be calibrated to the 365 keV of ^{131}I. A representative air sample shall be taken and then counted for ^{131}I using the SCA. This will give an initial conservative estimate of presence of iodine and can be used to determine if respiratory protection is required. Care must be taken to assure that the counting system is not saturated as a result of too much activity collected on the sampling cartridge.

B. By January 1, 1981:

The licensee shall have the capability to remove the sampling cartridge to a low background, low contamination area for further analysis. This area should be ventilated with clean air containing no airborne radionuclides which may contribute to inaccuracies in analyzing the sample. Here, the sample should first be purged of any entrapped noble gases using nitrogen gas or clean air free of noble gases. The licensee shall have the capability to measure accurately the iodine concentrations present on these samples and effluent charcoal samples under accident conditions."

1647 186

PSC REPLY:

PSC will identify the areas at Fort St. Vrain which require continuous occupancy to mitigate the consequences of an accident. Acceptable portable or cart mounted iodine samplers with attached single channel analyzers are not on the market today to the best of our knowledge.

To meet the requirements of this Section, PSC proposes to take air samples, utilizing charcoal filter adsorbers, from those locations that require continuous habitability during accident conditions. Sampling will be analyzed utilizing a multi-channel analyzer, which will be located external to the reactor building to assure analytical capability in a timely fashion under accident conditions.

These procedures will be in effect by January 1, 1980. We will continue to evaluate the development of portable iodine monitors and will purchase such equipment if and when reliable equipment becomes available.

By January 1, 1981, PSC will have the capability to remove an iodine sampling cartridge to a permanent, low background, low contamination area where accident condition iodine concentrations can be accurately measured.

1647 187

Section 2.1.9 - Transient and Accident Analysis

NRC DISCUSSION Clarification Not Provided):

"The scope of the required transient and accident analysis is discussed in NUREG-0578. The schedule for these analyses is included in NUREG-0578 and is reproduced in the Implementation Schedule attachment to this letter. The Bulletins and Orders Task Force has been implementing these required analyses on that schedule. The analysis of the small break loss of coolant accident has been submitted by each of the owners groups. These analyses are presently under review by the B&O Task Force. The scope and schedule for the analysis of inadequate core cooling have been discussed and agreed upon in meetings between the owners groups and the B&O Task Force, and are documented in the minutes to those meetings.

The analysis of transients and accidents for the purpose of upgrading emergency procedures is due in early 1980 and the detailed scope and schedule of this analysis is the subject of continuing discussions between the owners groups and the B&O Task Force."

PSC REPLY:

PSC has been routinely completing additional system and transient analysis over the past few years as a result of the plant startup testing program. As a result of this reanalysis, various plant hardware, system and procedural modifications have been made. Reanalyses have been made of the following systems, individual component or procedures with the indicated result:

1. Reanalysis of reactor cooldown following DBDA utilizing one helium circulator driven by boiler feedwater and operating at 3,000 rpm.

   Results: With helium circulator speed reduced from 10,500 rpm to 8,000 rpm higher fuel temperatures are to be expected. These temperatures are still projected to be less than the 2900°F safety limit established for the fuel. Plant procedures were updated to reflect the reduction in helium circulator speed, and the pressure setpoint for the water pressure regulators ahead of the pelton wheel speed values were revised to limit helium circulator speed when on water drive to approximately 9,000 rpm.

2. As the result of testing completed using simulated firewater conditions and reanalysis of the low temperature adsorber, it was determined that increased helium circulator speed was required when being driven by firewater to facilitate adequate decay heat removal.

   Results: To achieve higher helium circulator speeds when operating on the pelton wheel drives being supplied with firewater, two Firewater Booster Pumps were installed. Procedures and Technical Specifications were updated to reflect the addition of these pumps.

3. A reanalysis of the capacity of the Helium Purification System Low Temperature Adsorber (LTA) indicated the need to revise the Emergency Procedures and physical plant to prevent overloading the LTA during depressurization of the primary coolant system as called for in the LOFC scenario.

   Results: As the result of this reanalysis, start of primary system depressurization was moved from 5 hours to 2 hours following onset of LOFC from 100% power, and the depressurization piping and valves were modified to assure system depressurization in 9 hours following onset of LOFC. Plant procedures were modified to reflect the new requirements and plant system configuration.

   Depressurization capability was confirmed by test.

4. The Firewater cooldown scenario presented in the FSAR assumed instantaneous start of reactor cooling using firewater on the steam generator and driving one helium circulator. Because reactor cooling using firewater is initiated manually, some delay in start of cooling is expected.

   A reanalysis of the Firewater cooldown assuming a one and one-half hour delay confirmed the adequacy of the systems involved. In conjunction with the above analysis, a study was initiated to identify the major manual valves that may have to be operated in the event of a seismic disturbance that would disrupt core cooling utilizing normal plant cooling. This Manual Valve Study has been integrated into the Emergency Procedures.

5. Following the events at Browns Ferry, PSC undertook an analysis of the plant from the standpoint of determining methods to sustain reactor forced circulation cooling under continuously deteriorating plant conditions.

   Results: This study resulted in the generation of additional operational guides for the operators to utilize to maintain forced circulation cooling of the core under conditions not previously considered during the preparation of Plant Operating and Emergency Procedures.

   PSC is presently considering two additional studies of plant operating situations not previously analyzed:

   1. Conditions or common mode failures in the Helium Purification System that could preclude primary system depressurization, and

   2. Helium circulator buffer seal upset resulting in moisture injection into the primary coolant system prior to onset of LOFC.

1647 189

The schedule for completion of these studies has not yet been determined.

In addition, PSC is evaluating the possible use of fault tree and event tree analysis to evaluate the scenarios of the design basis accident events as analyzed in the FSAR.

The schedule for completion of the additional studies listed above and our decision whether to evaluate present design basis accident scenarios using fault tree analysis will be forwarded to the NRC at the earliest possible date.

1647 190

Miscellaneous Items:

Containment Pressure Indication

NRC CLARIFICATION:

1. "The containment pressure indication shall meet the design provisions of Regulatory Guide 1.97 including qualification, redundancy, and testability.

2. The containment pressure monitor shall be installed by January 1, 1981."

Containment Water Level Indication

NRC CLARIFICAITON:

1. "The narrow range sump level instrument shall monitor the normal containment sump level vice (sic) the containment emergency sump level.

2. The wide range containment water level instruments shall meet the requirements of the proposed revision to Regulatory Guide 1.97 (Instrumentation for Light-Water Cooled Nuclear Power Plant to Assess Plant Conditions During and Following an Accident).

3. The narrow range containment water level instruments shall meet the requirements of Regulatory Guide 1.89 (Qualification of Class IE Equipment of Nuclear Power Plants).

4. The equivalent capacity of the wide range PWR level instrument has been changed from 500,000 gallons to 600,000 gallons to ensure consistency with the proposed revision to Regulatory Guide 1.97. It should be noted that this capability is based on recent plant designs. For older plants with smaller water capacities, licensees may propose deviations from this requirement based on the available water supply capability at their plant.

5. The containment water level indication shall be installed by January 1, 1981."

Containment Hydrogen Indication

NRC CLARIFICAITON:

1. "The containment hydrogen indication shall meet the design provisions of Regulatory Guide 1.97 including qualification, redundancy, and testability.

2. The containment hydrogen indication shall be installed by January 1, 1981."

PSC REPLY:

1647 191

In the Reference 1 letter, PSC's reply to Enclosure 3 of NRC letter Vassallo to Fuller dated September 13, 1979 titled "Instrumentation to Monitor Containment Conditions During the Course of an Accident" addressed 1) containment pressure monitoring, 2) containment hydrogen monitoring and 3) containment water level monitoring. The PSC replies remain applicable to the NRC clarifications provided above.

Miscellaneous Item:

Reactor Coolant System Venting

NRC CLARIFICATION:

A. "General

1. The two important safety functions enhanced by this venting capability are core cooling and containment integrity. For events within the present design basis for nuclear power plants, the capability to vent non-condensible gases will provide additional assurance of meeting the requirements of 10CFR50.46 (LOCA criteria) and 10CFR50.44 (containment criteria for hydrogen generation). For events beyond the present design basis, this venting capability will substantially increase the plant's ability to deal with large quantities of non-condensible gas without the loss of core cooling or containment integrity.

2. Procedures addressing the use of the RCS vents are required by January 1, 1981. The procedures should define the conditions under which the vents should be used as well as the conditions under which the vents should not be used. The procedures should be based on the following criteria: (1) assurance that the plant can meet the requirements of 10CFR50.46 and 10CFR50.44 for Design Basis Accidents; and (2) a substantial increase in the plants ability to maintain core cooling and containment integrity for events beyond the Design Basis.

B. BWR Design Considerations

1. Since BWR owners group has suggested that the Present BWR designs inherent capability of venting, this question relates to the capability of existing systems. The ability of these systems to vent the RCS of non-condensible gas must be demonstrated. In addition, the ability of these systems to meet the same requirements as the PWR vent systems must be documented. Since there are important differences among BWR's, each licensee should address the specific design features of his plant.

2. In addition to reactor coolant system venting, each BWR licensee should address the ability to vent other systems such as the isolation condenser, which may be required to maintain adequate core cooling. If the production of a large amount of non-condensible gas would cause the loss of function of such a system, remote venting of that system is required. The qualifications of such a venting system should be the same as that required for PWR venting systems.

C. PWR Vent Design Considerations

1. The locations for PWR vents are as follows:

1647 192

a. Each PWR licensee should provide the capability to vent the reactor vessel head.

b. The reactor vessel head vent should be capable of venting non-condensible gas from the reactor vessel hot legs (to the elevation of the top of the outlet nozzle) and cold legs (through head jets and other leakage paths). Additional venting capability is required for those portions of each hot leg which cannot be vented through the reactor vessel head vent. The NRC recognizes that it is impractical to vent each of the many thousands of tubes in a U-tube steam generator. However, we believe that a procedure can be developed which assures that sufficient liquid or steam can enter the U-tube region so that decay heat can be effectively removed for the reactor coolant system. Such a procedure is required by January 1981.

c. Venting of the pressurizer is required to assure its availability for system pressure and volume control. These are important considerations especially during natural circulation.

2. The size of the reactor coolant vents is not a critical issue. The desired venting capability can be achieved with vents in a fairly large range of sizes. The criteria for sizing a vent can be developed in several ways. One approach, which we consider reasonable, is to specify a volume of non-condensible gas to be vented and a venting time i.e., a vent capable of venting a gas volume of 1/2 the RCS in one hour. Other criteria and engineering approaches should be considered if desired.

3. Where practical, the RCS vents should be kept smaller than the size corresponding to the definition of a LOCA (10CFR50 Appendix A). This will minimize the challenges to the ECCS since the inadvertent opening of a vent smaller than the LOCA definition would not require ECCS actuation although it may result in leakage beyond Technical Specification Limits. On PWRs the use of new or existing valves which are larger than the LOCA definition will require the addition of a block valve which can be closed remotely to terminate the LOCA resulting from the inadvertent opening of the vent.

4. An indication of valve position should be provided in the control room.

5. Each vent should be remotely operable from the control room.

6. Each vent should be seismically qualified.

1647 193

7. The requirements for a safety grade system is the same as the safety grade requirement on other Short Term Lessons Learned items, that is, it should have the same qualifications as were accepted for the reactor protection system when the plant was licensed. The exception to this requirement is that we do not require redundant valves at each venting location. Each vent must have its power supplied from an emergency bus. A degree of redundancy should be provided by powering different vents from different emergency buses.

8. For systems where a block valve is required, the block valve should have the same qualifications as the vent.

9. Since the RCS vent system will be part of the reactor coolant systems boundary, efforts should be made to minimize the probability of an inadvertent actuation of the system. Removing power from the vents is one step in the direction. Other steps are also encouraged.

10. Since the generation of large quantities of non-condensible gas could be associated with substantial core damage, venting to atmosphere is unacceptable because of the associated released radioactivity. Venting into containment is the only presently available alternative. Within containment, those areas which provide good mixing with containment air are preferred. In addition, areas which provide for maximum cooling of the vented gas are preferred. Therefore, the selection of a location for venting should take advantage of existing ventilation and heat removal systems.

11. The inadvertent opening of an RCS vent must be addressed. For vents smaller than the LOCA definition, leakage detection must be sufficient to identify the leakage. For vents larger than the LOCA definition, an analysis is required to demonstrate compliance with 10CFR50.46."

PSC REPLY:

In the Reference 1 letter, PSC's reply to Enclosure 4 of NRC letter Vassallo to Fuller, dated September 13, 1979, titled "Installation of Remotely Operated High Point Vents in the Reactor Coolant System" addressed reactor coolant system venting. The PSC reply remains applicable to the NRC clarification provided above.

1647 194

Section 2.2.1.a - Shift Supervisor Responsibilities

NRC CLARIFICATION:

"NUREG-0578 POSITION (POSITION NO.)	CLARIFICATION
Highest Level of Corporate Management (1.)	V.P. For Operations
Periodically Reissue (1.)	Annual Reinforcement of Company Policy
Management Direction (1.)	Formal Documentation of Shift Presonnel, All Plant Management, Copy to IE Region
Properly Defined (2.0)	Defined in Writing in a Plant Procedure
Until Properly Relieved (2.B)	Formal Transfer of Authority, Valid SRO License, Recorded in Plant Log
Temporarily Absent (2.C)	Any Absence
Control Room Defined (2.C)	Includes Shift Supervisor Office Adjacent to the Control Room
Designated (2.C)	In Administrative Procedures
Clearly Specified	Defined in Administrative Procedures
SRO Training	Specified in ANS 3.1 (Draft) Section 5.2.1.8
Administrative Duties (4.)	Not Affecting Plant Safety
Administrative Duties Reviewed (4.)	On Same Interval as Reinforcement: i.e., Annual by V. P. for Operations."

PSC REPLY:

The PSC response to Section 2.1.2 in the Reference 1 letter remains applicable to the NRC Clarification provided above.

1647 195

Section 2.2.1.b - Shift Technical Advisor

NRC CLARIFICATION:

1. "Due to the similarity in the requirements for dedicatio to safety, training and onsite location and the desire that the accident assessment function be performed by someone whose normal duties involve review of operating experiences, our preferred position is that the same people perform the accident and operating experience assessment functions. The performance of these two functions may be split if it can be demonstrated the persons assigned the accident assessment role are aware, on a current basis, of the work being done by those reviewing operating experience.

2. To provide assurance that the STA will be dedicated to concern for the safety of the plant, our position has been that STA's must have a clear measure of independence from duties associated with the commercial operation of the plant. This would minimize possible distractions from safety judgements by the demands of commercial operations. We have determined that, while desirable, independence from the operations staff of the plant is not necessary to provide this assurance. It is necessary, however, to clearly emphasize the dedication to safety associated with the STA position both in the STA job description and in the personnel filling this position. It is not acceptable to assign a person, who is normally the immediate supervisor of the shift supervisor to STA duties as defined herein.

3. It is our position that the STA should be available within 10 minutes of being summoned and therefore should be onsite. The onsite STA may be in a duty status for periods of time longer than one shift, and therefore asleep at some times, if the ten minute availability is assured. It is preferable to locate those doing the operating experience assessment onsite. The desired exposure to the operating plant and contact with the STA (if these functions are to be split) may be able to be accomplished by a group, normally stationed offsite, with frequent onsite presence. We do not intend, at this time, to specify or advocate a minimum time onsite.

4. The implementation schedule for the STA requirements is to have the STA on duty by January 1, 1980, and to have STAs, who have all completed training requirements, on duty by January 1, 1981. While minimum training requirements have not been specified for January 1, 1980, the STAs on duty by that time should enhance the accident and operating experience assessment function at the plant."

1647 196

PSC REPLY:

In our October 29, 1979 letter, we neglected to mention in our reference to training in Item 5, the schedule for completion of training. As stated in our response, we may have to utilize contract personnel on a temporary basis until a permanent staff can be developed. Contract personnel that are to be initially utilized to fill the position of Technical Advisor have been on-site for several years and are thoroughly familiar with the HTGR and the operating aspects of Fort St. Vrain, and only very minimal training will be required for these personnel. As the permanent staff is obtained, they will be developed and trained within a period of one (1) year following the date of their employemnt. Likewise, any new personnel that may be required in the future due to attrition etc., will be trained within one year following their employment date.

In our October 29, 1979 letter, we provided our justification for a 2-hour response time for the technical advisor. Although we still feel justified in the 2-hour response time, we are willing, based on various conversations with the Staff, to commit to a 1- hour response time.

Section 2.2.1.c - Shift and Relief Turnover Procedures

NRC CLARIFICATION:

"No clarification provided."

PSC REPLY:

Based on conversations with the NRC staff subsequent to our October 29, 1979 response, we have re-evaluated our position on shift and relief turn over procedures and our response is modified as follows:

Startup of the reactor is accomplished by a series of Overall Plant Operating Procedures (OPOP's). These procedures provide detailed startup instructions including valve line-ups, instrumentation checks and verification of safety system parameters and technical specification requirements. These OPOP's are utilized for every plant startup and are written to assure proper valve line-up and operating status of various systems.

Changes in system status that occur subsequent to completion of the OPOP's are controlled by various methods such as the use of "system status tags", "system clearance tags", temporary modification systems (jumper control and instrument set point control), all of which are logged to reflect system status.

On a shift basis a surveillance test (PMO-30) is conducted to ensure compliance with Tech. Spec. LCO's and to ensure systems meet operational and safety requirements. Any discrepancies such as LCO's, etc., noted in this test are logged and reviewed during shift turnover.

Tech. Spec. surveillance test logs are maintained to ensure compliance with safety system requirements. These logs are reviewed and updated on a weekly basis.

All critical plant safety parameters are logged on a detail log sheet. Once per day (usually the midnight shift) the Shift Supervisor is required to review the log against tolerances provided. Discrepancies, if any, resulting from this review are placed on the Plant Trouble Report System for evaluation and corrective action.

Each of the various operating levels (i.e. Shift Supervisor, reactor operator, equipment operator and auxiliary tender), maintain a narrative log which is utilized to document unusual circumstances, clearances issued, plant trouble reports issued and other unusual plant status conditions or problems. Each of the oncoming shift personnel must review the narrative log for his job position and must sign the log indicating that he has read and understood the entries in the narrative log. In addition, oncoming shift personnel receive a verbal briefing by off-going shift personnel during the shift turnover process, at which in any abnormal conditions or problems are discussed.

These procedures, in addition to the fact that systems utilized to respond to plant emergencies and accidents at Fort St. Vrain are systems that are in normal operation, provide assurance that system status is properly documented and that information is passed on from shift to shift. We feel our existing practices adequately meet the intent of NRC's position statement on this subject.

As indicated in our October 29, 1979 response, we do not have a detailed formal procedure that describes the shift and relief turnover activities. We will, however, issue necessary operation orders and/or administrative procedures to describe the shift and relief turnover responsibilities. These orders/procedures will be issued and in effect by January 1, 1980.

1647 199

Section 2.2.2.a - Control Room Access

NRC CLARIFICATION:

"No clarification provided."

PSC REPLY:

The PSC reply to Section 2.2.2a in the Reference 1 letter remains applicable.

1647 200

Section 2.2.2.b - Onsite Technical Support Center

NRC CLARIFICATION:

1. "By January 1, 1980, each licensee should meet items A-G that follow. Each licensee is encouraged to provide additional upgrading of the TSC (items 2-10) as soon as practical, but no later than January 1, 1981.

   A. Establish a TSC and provide a complete description,

   B. Provide plans and procedures for engineering/management support and staffing of the TSC,

   C. Install dedicated communications between the TSC and the control room, near site emergency operations center, and the NRC,

   D. Provide monitoring (either portable or permanent) for both direct radiation and airborne radioactive contaminants. The monitors should provide warning if the radiation levels in the support center are reaching potentially dangerous levels. The licensee should designate action levels to define when protective measures should be taken (such as using breathing apparatus and potassium iodide tablets, or evacuation to the control room),

   E. Assimilate or ensure access to Technical Data, including the licensee's best effort to have direct display of plant parameters, necessary for assessment in the TSC,

   F. Develop procedures for performing this accident assessment function from the control room should the TSC become uninhabitable, and

   G. Submit to the NRC a longer range plan for upgrading the TSC to meet all requirements.

2. Location

   It is recommended that the TSC be located in close proximity to the control room to ease communications and access to technical information during an emergency. The center should be located onsite, i.e., within the plant security boundary. The greater the distance from the CR, the more sophisticated and complete should be the communications and availability of technical information. Consideration should be given to providing key TSC personnel with a means for gaining access to the control room.

3. Physical Size and Staffing

   The TSC should be large enough to house 25 persons, necessary engineering data and information displays (TV monitors, recorders, etc.). Each licensee should specify staffing levels and disciplines reporting to the TSC for emergencies of varying severity.

1647 201

4. Activation

The center should be activated in accordance with the "Alert" level as defined in the NRC document "Draft Emergency Action Level Guidelines, NUREG-0610" dated September, 1979, and currently out for public comment. Instrumentation in the TSC should be capable of providing displays of vital plant parameters from the time the accident began (t = 0 defined as either reactor or turbine trip). The Shift Technical Advisor should be consulted on the "Notification of Unusual Event" however, the activation of the TSC is discretionary for that class of event.

5. Instrumentation

The instrumentation to be located in the TSC need not meet safety-grade requirements but should be qualitatively comparable (as regards accuracy and reliability) to that in the control room. The TSC should have the capability to access and display plant parameters independent from actions in the control room. Careful consideration should be given to the design of the interface of the TSC instrumentation to assure that addition of the TSC will not result in any degradation of the control room or other plant functions.

6. Instrumentation Power Supply

The power supply to the TSC instrumentation need not meet safety-grade requirements, but should be reliable and of a quality compatible with the TSC instrumentation requirements. To insure continuity of information at the TSC, the power supply provided should be continuous once the TSC is activated. Consideration should be given to avoid loss of stored data (e.g., plant computer) due to momentary loss of power or switching transients. If the power supply is provided from a plant safety-related power source, careful attention should be given to assure that the capability and reliability of the safety-related power source is not degraded as a result of this modification.

7. Technical Data

Each licensee should establish the technical data requirements for the TSC, keeping in mind the accident assessment function that has been established for those persons reporting to the TSC during an emergency. As a minimum, data (historical in addition to current status) should be available to permit the assessment of:

1647 202

Plant Safety Systems Parameters for:

Reactor Coolant System
Secondary System (PWRs)
ECCS Systems
Feedwater & Makeup Systems
Containment

In-Plant Radiological Paramenters for:

Reactor Coolant System
Containment
Effluent Treatment
Release Paths

Offsite Radiological

Meteorology
Offsite Radiation Levels

8. Data Transmission

In addition to providing a data transmission link between the TSC and the control room, each licensee should review current technology as regards transmission of those parameters identified for TSC display.

Although there is not a requirement at the present time, each licensee should investigate the capability to transmit plant data offsite to the Emergency Operations Center, the NRC, the reactor vendor, etc.

9. Structural Integrity

A. The TSC need not be designed to seismic Category I requirements. The center should be well built in accordance with sound engineering practice with due consideration to the effects of natural phenomena that may occur at the site.

B. Since the center need not be designed to the same stringent requirements as the Control Room, each licensee should prepare a backup plan for responding to an emergency from the control room.

10. Habitability

The licensee should provide protection for the technical support center personnel from radiological hazards including direct radiation and airborne contaminants as per General Design Criterion 19 and SRP 6.4.

A. Licensee should assure that personnel inside the technical support center (TSC) will not receive doses in excess of those specified in GDC 19 and SRP 6.4 (i.e., 5 Rem whole body and 30 Rem to the thyroid for the duration of the accident). Major sources of radiation should be considered.

1647 203

B. Permanent monitoring systems should be provided to continuously indicate radiation dose rates and airborne radioactivity concentrations inside the TSC. The monitoring systems should include local alarms to warn personnel of adverse conditions. Procedures must be provided which will specify appropriate protective actions to be taken in the event that high dose rates or airborne radioactive concentrations exist.

C. Permanent ventilation systems which include particulate and charcoal filters should be provided. The ventilation systems need not be qualified as ESF systems. The design and testing guidance of Regulatory Guide 1.52 should be followed except that the systems do not have to be redundant, seismic, instrumented in the control room or automatically activated. In addition, the HEPA filters need not be tested as specified in Regulatory Guide 1.52 and the HEPA's do not have to meet the QA requirements of Appendix B to 10 CFR 50. However, spare parts should be readily available and procedures in place for replacing failed components during an accident. The systems should be designed to operate from the emergency power supply.

D. Dose reduction measures such as breathing apparatus and potassium iodide tablets cannot be used as a design basis for the TSC in lieu of ventilation systems with charcoal filters. However, potassium iodide and breathing apparatus should be available."

PSC REPLY:

In our response of October 29, 1979, it was apparently not clear that PSC does intend to provide a permanent or final on-site technical support center by January 1, 1981.

1647 204

Section 2.2.2.c - Onsite Operational Support Center

NRC CLARIFICATION:

"No clarification provided."

PSC REPLY:

The PSC reply to Section 2.2.2.c in the Reference 1 letter remains applicable.

1647 205