DISCUSSION OF
TMI LESSONS LEARNED
SHORT TERM REQUIREMENTS

This document provides only a portion of the Rochester Gas and Electric Corporation response to TMI Lessons Learned Short Term Requirements. Additional information is found in Rochester Gas and Electric letters from L. D. White, Jr. to Dennis L. Ziemann, USNRC, dated October 17, 1979 and November 19, 1979.

TABLE OF CONTENTS

TABLE OF CONTENTS
(Continued)

Section 2.1.1 - EMERGENCY POWER SUPPLY

Pressurizer Heaters

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.

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 LOCA.

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 1E 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)

RG&E Responses

The pressurizer heater power supply conforms to all the requirements specified. A new procedure, 0-8.1 "Restoration of Pressurizer Heaters to Maintain Natural Circulation at Hot Shutdown" has been incorporated. It includes criteria to prevent overloading of a diesel generator while providing for sufficient power to overcome pressurizer heat losses to maintain the reactor coolant system subcooled and to enhance natural circulation. Operator training in the use of this procedure has been completed. Existing emergency and operating procedures have been changed to refer to this procedure.

Section 2.1.1 - EMERGENCY POWER SUPPLY

Pressurizer Level and Relief Block Valves

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:

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 or the emergency power source when the offiste 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 or 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.

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

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.

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.

RG&E Responses

No additional response required.

Section 2.1.2 - PERFORMANCE TESTING FOR BWR AND PWR RELIEF AND SAFETY VALVES

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.

CLARIFICATION

1. Expected operating conditions can be determined through the use of analysis of accidents and anticipated operational occurrences 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.

RG&E Response

By letter dated December 17, 1979, Mr. William J. Cahill, Jr., Chairman of the EPRI Safety and Analysis Task Force, submitted to the NRC "Program Plan for the Performance Verification of PWR Safety/ Relief Valves and Systems".

The EPRI Program Plan provides for completion of the essential portions of the test program by July, 1981. RG&E will be participating in the EPRI program to provide program review and to supply plant specific data as required.

Section 2.1.3.a - DIRECT INDICATION OF POWER-OPERATED RELIEF VALVE AND SAFETY VALVE POSITION FOR PWRS AND BWRS

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.

CLARIFICATION

1. The basic requirement is to provide the operator with unambiguous 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 conjunction 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.

RG&E Response

RG&E has installed linear variable displacement transducers (LVDTs) with a control room position alarm on the pressurizer safety valves. All pressurizer relief and safety valves now have direct stem mounted position indication as well as discharge pipe thermocouple temperature indication. The present safety valve LVDTs are designed for a hostile environment (1000 psi and 300°F) and are constructed of radiation resistant materials (stainless steel and glass filled polymers). Since a similar LVDT will be available that is designed specifically for high radiation environments, the existing LVDTs will be replaced during the spring refueling outage. An alarm has been added to the existing relief valve stem mounted limit switch indication.

Section 2.1.3.b - INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING

SUBCOOLING METER

POSITION

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 NUREG-0578).

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 it is not to be used exclusive of other related plant parameters.

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 (sic) 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.

RG&E Responses

Two independent subcooling meters described in our November 19 letter have been installed. Procedure guidelines that are used by the operator in recognizing inadequate core cooling were submitted by the Westinghouse Owners' Group on October 30, 1979.

An existing procedure E-1.5, "Void Formation in the Reactor Coolant System" has been upgraded and modified as necessary to reflect the existing guidelines developed in a generic procedure produced by the Westinghouse Owners' Group. Information presently available from the plant computer and available to operating personnel has been included during this review. Formal training of licensed operators covering the content of this procedure prior to the recent upgrade and modification was held between June 11, 1979 and July 13, 1979. Additional guidelines have been recently provided by Westinghouse. These guidelines have been issued on a Plan of Day instruction and placed in the control room and will be reviewed by all operating personnel by January 1, 1980 except for those personnel on vacation who will review it as soon as they return to work.

Section 2.1.3.b - INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING

ADDITIONAL INSTRUMENTATION

POSITION

Licensees shall provide a description of any additional instrumentation or 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.

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 advanced 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.

RG&E Response

The Westinghouse Owners' Group, of which RG&E is a member, has performed analyses as required by Item 2.1.9 to study the effects of inadequate core cooling. These analyses were provided to the NRC "Bulletins and Orders Task Force" by a letter dated October 30, 1979. As part of the submittal made by the Owners' Group, an "Instruction to Restore Core Cooling during a Small LOCA" was included. This instruction provides the basis for procedure

changes and operator training required to recognize the existence of inadequate core cooling and restore core cooling based on existing instrumentation. RG&E has incorporated the considerations of this instruction into our procedures and Plan of Day instruction.

The October 30 submittal referenced above described the capabilities of the core exit thermocouples for determining the existence of inadequate core cooling conditions and their superiority in some instances to the loop RTDs for measuring true core conditions.

Other means of determining the approach to or existence of inadequate core cooling could be:

1. Incore detectors
2. Excore detectors
3. Reactor coolant pump motor currents
4. Steam generator pressure
5. Reactor vessel water level

A discussion of the possible use of these measurements is provided below.

The use of incore movable detectors to determine the existence of inadequate core cooling conditions appears doubtful. The detectors could be driven in to the top of the incore thimbles, which are located at the top of the core, following an accident. However, there may be a lack of sensitivity of the detectors to very low neutron levels and changes that would occur if the core is not covered. Gamma detectors could perhaps be employed, but they suffer from similar sensitivity problems, and from the fact that gamma levels in the fuel region change insignificantly between the covered and uncovered condition. As a result, it does not appear worthwhile to pursue incore movable detectors as a means of determining inadequate core cooling conditions.

Excore detectors have been discussed as a possibility in responding to core uncovery. The only detectors which would have the required sensitivity are the source range monitors since the intermediate and power range monitors are not sensitive enough to the low level changes resulting from vessel voiding. The use of the source range monitors will be investigated further as part of the ongoing study of inadequate core cooling being performed by the Westinghouse Owners' Group. However, their use is probably limited to those instances when significant voiding exists in the downcomer region, since water in the downcomer would effectively shield the detectors from the core region whether voids existed in the core or not.

Reactor coolant pump motor current, which could be indicative of core voiding, is appropriate if the pumps are running but a more reliable means of determining inadequate core cooling may be needed since a loss of off-site power or a pump trip shuts the pumps down.

Steam generator pressure indication, which already exists, is useful in the case where heat transfer from primary to secondary is interrupted due to loss of natural circulation. This, however, does not indicate the approach to inadequate core cooling, nor does it indicate the true condition of the core.

Reactor vessel water level determination is the most promising of the items discussed to provide additional capability of determining the approach to and the existence of inadequate core cooling. Several systems for determining water level are under review by the Westinghouse Owners' Group. A generic conceptual design of one system is given below:

Vessel Level System Description

After examining many different methods and principles for determining the water level in the reactor vessel, a basic delta pressure measurement from the bottom of the vessel to the top of the vessel appears to provide the most meaningful and reliable information to the operator. One of the reasons for choosing this system is that the sources of potential errors are better known for this system than for any other new or untested system.

Figure 1 shows a simplified sketch of the proposed vessel level instrumentation system. The bottom tap of the instrument would use a thimble of the incore movable detector system either at the seal table or in the thimble below the vessel. Use of the thimble as part of the water level indicator would not preclude use of the thimble for incore flux monitoring. For connection below the vessel the flux thimble guide tube would be tapped below the vessel and an instrument line connection made. The instrument line would have an isolation valve and would slope down to a hydraulic coupler connected to a sealed reference leg. For connection at the seal table, a special fitting would be utilized which would be connected to an isolation valve and sealed reference leg. The sealed reference leg would go to the differential pressure transmitter located at an elevation above the expected level of containment flooding. A similar seal leg would go to the top of the vessel and penetrate the head using the vent line or a special connection on a spare rod cluster control (RCC) mechanism penetration. Two trains of vessel level instrumentation could be provided from the single set of vessel taps.

The behavior of the signal generated by this level instrument under normal and accident conditions is being evaluated. The usefulness of this instrument to provide an unambiguous indication of inadequate core cooling is being evaluated as part of Item 2.1.9. The potential errors and accuracy of a final system configuration are being evaluated to assess its usefulness to provide information to the operator for proper operation of a vessel venting system and for normal water level control during periods when the primary system is open and a water level may exist in the vessel. The connection of the level system to the vessel head should be designed to be compatible with the head vent system. Operation of the vent system should not, if possible,

upset all indications of vessel level. If operation of the vent may have a potential impact on the level indication the problem can be avoided by some plants by using a separate instrument tap or by using more than one location. For other plants, such as Ginna, which have no spare RCC mechanism penetrations and have only a single existing head vent penetration, an evaluation will be made of the difficulties and additional radiation exposure required to install separate instrument taps on the reactor head against the potential for level indication accuracy degradation during and following venting. The need for vessel level indication following venting is also being evaluated and will be considered in the selection of instrument tap locations.

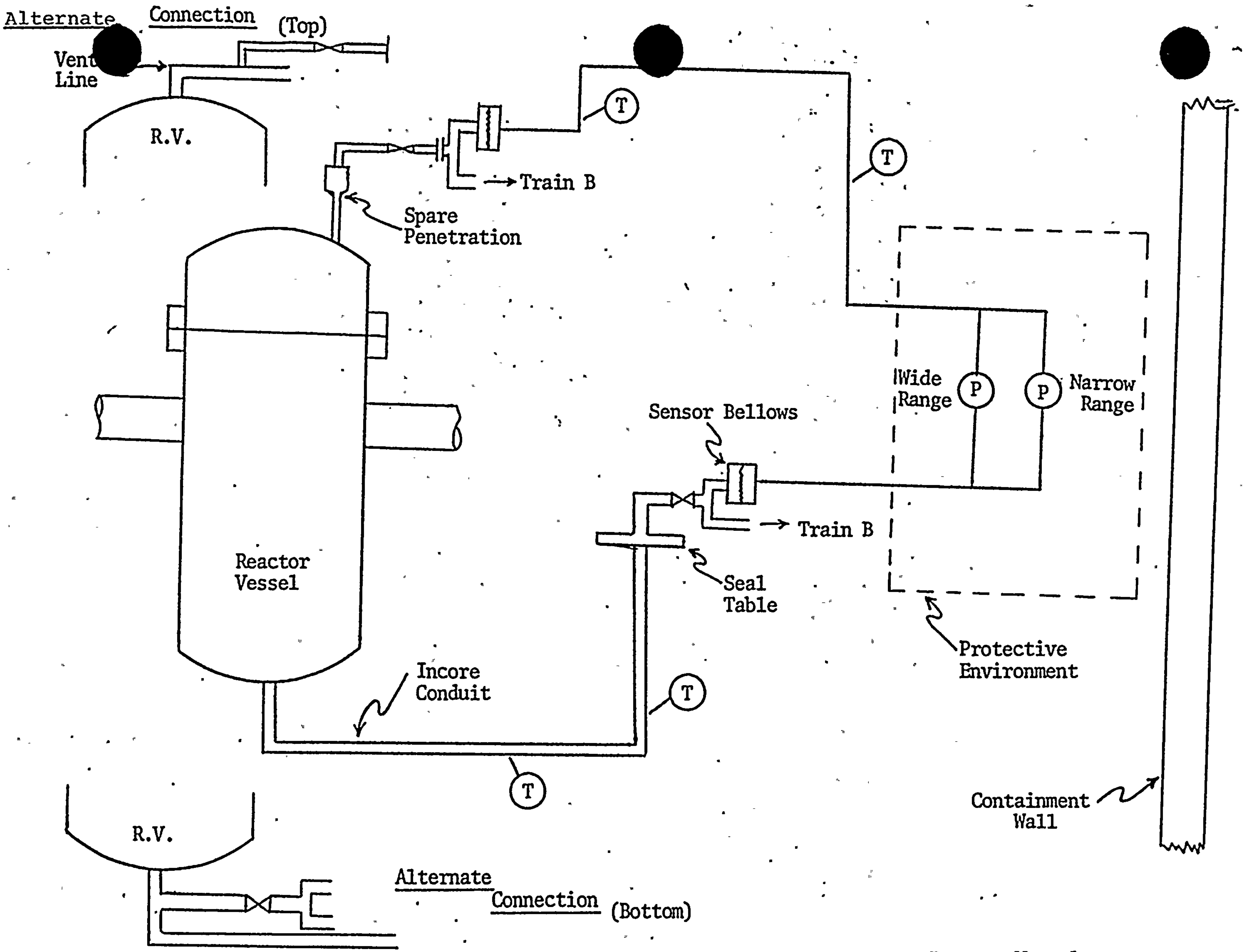


FIGURE 1

Reactor Vessel Level Instrumentation

Section 2.1.4 - CONTAINMENT ISOLATION

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 nonessential systems, shall identify each system determined to be essential, shall identify each system determined to be nonessential, 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 nonessential 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.

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.

RG&E Response

Our response of November 19, 1979 discussed the action we have taken to conform with the staff position. Equipment delivery times discussed in that response are unchanged.

Section 2.1.5.a - DEDICATED H_2 CONTROL PENETRATIONS

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.

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.

RG&E Response

Ginna has two hydrogen recombiners which are located inside containment. Therefore, dedicated penetrations are not required.

Section 2.1.5.c - CAPABILITY TO INSTALL HYDROGEN RECOMBINER AT EACH LIGHT WATER NUCLEAR POWER PLANT

POSITION

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.

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.

RG&E Response

Access to the hydrogen recombiner control panel has been evaluated in our shielding adequacy review. Access to the control panel under post-accident conditions for control of the recombiners is possible with acceptable personnel exposures. Details are given in 2.1.6.b.

Section 2.1.6.a - INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT LIKELY TO CONTAIN RADIOACTIVE MATERIALS FOR PWRs AND BWRs

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.

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.

RG&E Response

All practical leak reduction measures have been normal practice for systems that carry radioactive fluids outside containment. We will continue to assess possible improvements to these systems as identified by changes in the state of the art, improvements made at other plants and operating experience, to continually improve our systems leak rates. These measures were implemented following failed fuel experience a number of years ago. Despite good fuel performance in recent years, these leakage reduction measures have been continued.

A preventive maintenance program has been on going for mechanical equipment of these systems. A preventive maintenance program has been established for valves and piping on the systems shown on Table 2.1.6.a.1. The program will commence January 1, 1980 and will include regularly scheduled maintenance on maintainable valves and annual external inspection for leakage and/or deterioration. Drawings indicated in Table 2.1.6.a.1 were provided to the NRC for SEP in a letter dated September 12, 1978 to D. L. Ziemann. Systems which have not been included in the program include main steam, auxiliary feedwater, the turbine side, some of the auxiliary coolant, waste evaporator, boric acid evaporator, service water, drumming station, safety and relief valves and some of the reactor makeup water, waste disposal and component cooling water systems.

No release paths exemplified by the North Anna Unit 1 incident or similar release paths as identified in IE Circular 79-21 have been found.

Tests have been performed on the Residual Heat Removal System, Safety Injection System, and Containment Spray System with the system in operation to the extent practical to determine applicable leakage paths. No visible liquid leakage was detected. However, boron buildup was observed on some components indicating that a small amount of leakage was occurring. The affected components were cleaned and a followup inspection performed. Maintenance work requests have been submitted on components which have shown new boron accumulation, to reduce leakage to as low as practical. These components will also be given particular attention in our preventive maintenance program. The inspection was performed during December 1979.

A visual inspection of the radioactive Liquid Waste System, Chemical Volume Control System, and Primary Sampling System was made to determine if there was any system leakage. No visible liquid leakage was detected but boron buildup was observed. Similar action, as noted above, is being taken.

The radioactive Waste Gas System has been inspected to determine if there was any system leakage. The inspection was performed by Nuclear Consulting Services Inc. using a hydrogen detector method to determine a leak rate for the system. An ultrasonic leak detector was used to scan the system for audible leaks, to indicate any areas which would require maintenance, prior to proceeding with the hydrogen leak test. No areas were identified. A portable Gas Detection system (Sniffer) was then used to determine the actual leakage rates. The total leakage for the Waste Gas System was 6.91×10^{-2} cc/sec. We consider this leakage rate to be very small and it is as-low-as-practical. The inspection was performed during December 1979.

Inspections were made in response to IE Bulletin 79-17 regarding stress corrosion cracking at welds or base material in stagnant borated water systems. No indications were observed. Further inspections will be performed during our next annual refueling

outage as per our response to that bulletin. (See letters from L. D. White, Jr. to Mr. Boyce H. Grier, Office of Inspection and Enforcement dated August 24, 1979, October 24, 1979 and November 26, 1979).

As part of the In-Service Inspection Program requirement, containment isolation valve leak rate tests were performed during our 1979 refueling shutdown. The results of the leakage rate tests for the primary (see Table 2.1.6.a.2) and applicable secondary (see Table 2.1.6.a.3) isolation valves are attached. The primary isolation valves are those valves located inside containment with secondary isolation valves outside containment.

As required by Technical Specifications and Appendix J to 10 CFR 50, leak rate tests were performed during 1979 on the containment penetrations. The results of the leakage rate tests are shown on Table 2.1.6.a.4.

Table 2.1.6.a:

VALVE AND PIPING LEAKAGE PREVENTION MAINTENANCE PROGRAM

RESIDUAL HEAT REMOVAL (RHR)

PRINT 33013-436

Valves from A Loop Hot Leg through the A and B RHR pumps, A and B RHR Heat Exchangers and back to B Loop Cold Leg.

CONTAINMENT SPRAY (CS)

PRINT 33013-425
PRINT 33013-432

Valves from Refueling Water Storage Tank (RWST) through CS Pumps, to Containment Charcoal Filter Dousing System valves, and the Reactor Containment Ring Spray Header.

SAFETY INJECTION (SI)

PRINT 33013-425
PRINT 33013-432

Valves from RWST through the A,B,C Safety Injection Pumps, into the Containment to Loop A Hot Leg, Loop A Cold Leg, and fill line to Accumulator B. Also to Loop B Hot Leg, Loop B Cold Leg and fill line to Accumulator A.

CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)

PRINT 33013-433

Valves from Loop B Letdown line through Regenerative Heat Exchanger, non-regenerative Heat Exchanger, Demineralizers, Reactor Coolant Filter, Volume Control Tank, Charging Pumps, Seal Injection Filters, Reactor Coolant Pump Seals, Seal Water Heat Exchanger, and return to Volume Control Tank. Also the line to Loop B Hot Leg, Loop B Cold Leg, and Auxiliary Spray to Pressurizer by way of Regenerative Heat Exchanger.

CVCS

PRINT 33013-426

Valves on Boric Acid Batching Tank, Boric Acid Tanks A and B, Boric Acid Transfer Pumps A and B, and Blending System to Charging Pumps.

CVCS

PRINT 33013-426

Valves and piping both to and from the Hold Up Tanks and including Gas Stripper Feed Pumps.

WASTE DISPOSAL SYSTEM - WATER

PRINT 33013-431

Valves for Waste Water from Laundry and Hot Shower Tanks, Chemical Drain Tank, Reactor Coolant Drain Tank, to Hold Up Tanks and Waste Hold Up Tank. All drains to Waste Hold Up Tank and Recirculation Pump. Also the Containment and Auxiliary Building Sump.

WASTE DISPOSAL SYSTEM - GAS

PRINT 33013-430

Valves carrying gases from Holdup Tanks, Spent Resin Storage Tanks, Reactor Coolant Drain Tank, Volume Control Tank, Gas Stripper to and from Gas Compressors, Gas Decay Tanks, reuse and release lines to plant vent.

PRIMARY SAMPLE SYSTEM

Valves from Pressurizer Steam Space, Pressurizer Liquid Space, Loop B Hot Leg, Loop A Hot Leg, Failed Fuel System, RHR Loop, Mixed Bed Demineralizer Outlet and Inlet, and Volume Control Tank Gas Space to Nuclear Sample Room.

YEAR 1979	PRIMARY ISOLATION VALVE		Table 2.1.6.a.2		SHEET 1 of 7
PT-23 SERIES	NOMENCLATURE	PIV #	TEST ATMOSPHERE LEAKAGE	RESULTS	DATE
1	PRESSURIZER RELIEF TANK GAS ANAL.:	AOV 539	2.03 cc/min		2/16/79
2	H_2 SUPPLY TO PRESS. RELIEF TANK LINE:	CHECK VLV. 528	1.09 cc/min		2/20/79
3	MAKE UP H_2O TO PRESS. RELIEF TANK:	CHECK VLV. 529	1.09 cc/min		2/20/79
7	LETDOWN FROM REACTOR COOLANT SYSTEM:	AOV 371	1.09cc/min		4/22/79
8	RCS CHARGING LINE:	CHECK VLV. 370	1.09 cc/min		4/24/79
9A	"A" RCP SEAL WATER LINE:	CHECK VLV. 304 A	145.0 cc/min	Rate determined acceptable.	2/24/79
9B	"B" RCP SEAL WATER LINE:	CHECK VLV. 304 B	26.57 cc/min		2/22/79
10	ALTERNATE CHARGING LINE:	CHECK VLV. 383 B	1:09 cc/min		3/15/79

PT-23 SERIES	NOMENCLATURE	PIV #	TEST ATMOSPHERE LEAKAGE	RESULTS	DATE
11	RCP SEAL WATER RETURN & EXCESS LET.	MOV 313	0 cc/min		2/22/79
12A	STEAM SPACE SAMPLE:	AOV 966A	2.01 cc/min		2/16/79
12B	PRESSURIZER LIQUID SPACE SAMPLE:	AOV 966B	2.01 cc/min		2/16/79
12C	RCS SAMPLE LOOP "B":	AOV 966C	2.01 cc/min		2/16/79
13A	"A" STEAM GENERATOR SAMPLE:	AOV 5735	2.01 cc/min		2/16/79
13B	"B" STEAM GENERATOR SAMPLE:	AOV 5736	0 cc/min		2/16/79
14	CONTAINMENT AIR SAMPLE:	CHECK VLV. 1599	1.08 cc/min		2/13/79
15	CONTAINMENT AIR SAMPLE OUT:	AOV 1597	1.08 cc/min		2/13/79
16A	"A" STEAM GENERATOR BLOWDOWN:	AOV 5737	0 cc/min		2/26/79
16B	"B" STEAM GENERATOR BLOWDOWN:	AOV 5738	0 cc/min		2/26/79
17A	CONTAINMENT PRESSURE: PT-945	RV-1	1.09 cc/min		2/17/79
	PT-946	RV-2	1.09 cc/min		2/17/79
17B	CONTAINMENT PRESSURE PT-947	RV-1	1.1 cc/min		2/14/79
	SENSING TRANSMITTERS: [illegible]	[illegible]	1.1 cc/min		2/14/79

YEAR 19 — PRIMARY ISOLATION VALVE TEST — Table 2.1.6.a.2 —

PT-23 SERIL	NOMENCLATURE	PIV #	TEST ATMOSPHERE LEAKAGE	RESULTS	
17C	CONTAINMENT PRESSURE SENSING TRANS.:	RV-1	1.09cc/min		2/12/79
		RV-2	1.09cc/min		2/12/79
		RV-3	1.09cc/min		2/12/79
18A	"A" CONTAINMENT SPRAY HEADER:	CHECK VLV. 862A	2.01cc/min		2/21/79
		MANUAL VLV. 864A	2.01cc/min		2/21/79
18B	SPRAY HEADER:	CHECK VLV. 862A	2.01cc/min		2/21/79
		MANUAL VLV. 864B	2.01cc/min		2/21/79
19	SAFETY INJECTION SYSTEM:	CHECK VLV. 889A	2.02cc/min		2/21/79
		889B	2.02cc/min		2/21/79
		CHECK VLV. 870A	0cc/min		2/21/79
		& MANUAL VLV. 879	0cc/min		2/21/79
		CHECK VLV. 870B	0cc/min		2/21/79
		& MANUAL VLV. 879	0cc/min		2/21/79

PT-23 SERIL	NOMENCLAIURE	PIV #			
20	R.C.D.T. GAS HEADER:	AOV 1787	2.03 cc/min		2/20/79
		& CHECK VLV. 1713	2.03 cc/min		2/20/79
21	R.C.D.T. GAS ANALYZER:	AOV 1789	2.03 cc/min		2/16/79
22	R.C.D.T. DISCHARGE	AOV 1721	1.09 cc/min		2/23/79
23	SUMP "A" DISCHARGE	AOV 1728	1.09 cc/min		2/24/79
24	REACTOR SUPPORT COOLERS (INLET & OUTLET)	MOV 813	0 cc/min		2/19/79
		MOV 814	1.09 cc/min		2/19/79
26	AUX. COOLANT SYSTEM- "A" RCP	CHECK VLV. 750A	1.09 cc/min		2/16/79
27	AUX. COOLANT SYSTEM- "B" RCP	CHECK VLV. 750B	1.09 cc/min		2/17/79
28	AUX. COOLANT SYSTEM FROM- "A" RCP	MOV 759A	1.09 cc/min		2/16/79
29	AUX. COOLANT SYSTEM FROM- "B" RCP	MOV 759B	1.09 cc/min		2/18/79

ITEM 12 PRIMARY ISOLATION VALVE TEST Table 2.1.6.a.2 SHEET 5 OF 7

PT-23 SERIE	NOMENCLATURE	PIV #	TEST ATMOSPHERE LEAKAGE	RESULTS	
30	AUX. COOLANT SYSTEM EXCESS LETDOWN (SUPPLY & RETURN):	CHECK VLV. 743	1.09cc/min		2/17/79
		AOV 745	1.09 cc/min		2/17/79
32	INSTRUMENT AIR:	CHECK VLV. 5393	1.08 cc/min		3/21/79
33	SERVICE AIR:	MANUAL VLV. 7226	1.08 cc/min		3/17/79
34	DEPRESS. AT POWER:	AOV 7970	1.0 cc/min		2/20/79
35	PURGE SUPPLY:	AOV 5870	19.26 cc/min		11/15/79
36	PURGE EXHAUST:	AOV 5878	-56.17 cc/min	Negative leakage due to temp. variation during test	11/15/79
39	DEMINERALIZED WATER:	MANUAL VLV. 8419	19.1 cc/min		3/17/79
40	AUX. STEAM SUPPLY & CONDENSATE RETURN:	MANUAL VLV. 6151	1.08 cc/min		3/17/79
		MANUAL VLV. 6175	1.08 cc/min		3/17/79
42	LEAKAGE-TEST DEPRESS:	BLIND FLANGE	0 cc/min		2/12/79
43	LEAKAGE TEST SUPPLY:	BLIND FLANGE	0cc/min		2/12/79

25

YEAR 19 | PRIMARY ISOLATION VALVE TEST | Table 2.1.6.a.2 | SHEET 6 OF 7

PT-23 SERI.	NOMENCLATURE	PIV #	TEST ATMOSPHERE LEAKAGE	RESULTS	
44	LEAKAGE TEST DEPRESS.:	BLIND FLANGE	0 cc/min		2/12/79
45	LEAKAGE TEST INSTRUMENTATION:	TUBING CAP #1	0 cc/min		2/12/79
		#2	0 cc/min		2/12/79
		#3	0 cc/min		2/12/79
46	H_2 TO ACCUMALATOR:	AOV 846	1.09 cc/min		2/20/79
48	DEADWEIGHT TESTER:	MANUAL VLV. 549A	0 cc/min		2/12/79
49	CONT. FIRE SERVICE H_2O:	MANUAL VLV.	2.01 cc/min		2/24/79
50A	CONTAINMENT POST ACCIDENT AIR SAMPLE (CLEAN INT. BLDG.):	MANUAL VLV. #1	1.09 cc/min		2/12/79
		#2	1.09 cc/min		2/12/79
		#3	1.09 cc/min		2/12/79
50B	CONTAINMENT POST ACCIDENT AIR SAMPLE (CONTROLLED INT. BLDG.):	MANUAL VLV. 1563 #1	1.08 cc/min		2/14/79
		1566 #2	1.01 cc/min		2/14/79

YEAR 19 | PRIMARY ISOLATION VALVE TEST | Table 2.1.6.a.2 | SHEET 7 OF 7

PT-23 SERI.	NOMENCLATURE	PIV #	TEST ATMOSPHERE LEAKAGE	REMARKS	[illegible]
50C	CONT. POST ACCIDENT AIR SAMPLE (AUX. BLDG.):	MANUAL VLV. #1 1569	1.09 cc/min		2/17/79
		#2 1572	1.09 cc/min		2/17/79
51A	"A" HYDROGEN RECOMBINER:	SOLENOID VALVE IV3A	9.35 cc/min		2/13/79
		IV5A	1.08 cc/min		2/13/79
51B	"B" HYDROGEN RECOMBINER:	IV3B	1.1 cc/min		2/14/79
		IV5B	1.1 cc/min		2/14/79
51C	"A" & "B" HYDROGEN RECOMBINER OXYGEN MAKE-UP:	IV2A	1.0 cc/min		2/14/79
		IV2B	1.0 cc/min		2/14/79

YEAR 1979 SECONDARY ISOLATION VALVE TEST Table 2.1.6.a.3 SHEET 1 of 2

PT-23 SERIES	NOMENCLATURE	SIV#	TEST ATMOSPHERE LEAKAGE	RESULTS	
3	MAKE UP WATER TO PRESSURE RELIEF TANK	AOV 508	2.0 cc/min		2/20/79
14	CONT. AIR SAMPLE INLET:	AOV 1598	1.09 cc/min		4/20/79
20	RCDT GAS HEADER	AOV 1786	2.03 cc/min		2/20/79
22	R.C.D.T DISCHARGE	AOV 1003A	0.54 cc/min		2/23/79
		AOV 1003B	0.54 cc/min		2/23/79
23	SUMP "A" DISCHARGE:	AOV 1723	1.09 cc/min		2/24/79
32	INSTRUMENT AIR	AOV 5392	1.08 cc/min		3/21/79
34	DEPRESSURIZATION AT POWER	AOV 7971	1.0 cc/min		2/20/79
35	PURGE SUPPLY	AOV 5869	0 cc/min		11/15/79
36	PURGE EXHAUST:	AOV 5879	0 cc/min		11/15/79

YEAR 1979 SECONDARY ISOLATION VALVE TEST Table 2.1.6.a.3 SHEET 2 of 2

PT-23 SERIES	NOMENCLATURE	SIV#	TEST ATMOSPHERE LEAKAGE	RESULTS	
42	LEAK TEST DEPRESS:	MOV 7444	2.38 cc/min		2/12/79
43	LEAK TEST SUP. HEADER:	MOV 7443	2.38 cc/min		2/12/79
44	LEAK TEST DEPRESS.	MOV 7445	2.38 cc/min		2/12/79

Table 2.1.6.a.4

CONTAINMENT PENETRATIONS LEAK RATE TEST

TEST	TITLE	DATE	LEAKAGE	
22.1	Equipment Hatch (Door Seals)	11/9/79	Out 2.12	cc/min.
			In 0	"
22.2	Personnel Hatch (Door Seals)	11/16/79	Out 7.83	"
			In 0	"
22.3	Personnel Hatch (Between Doors)	5/10/79	148.95	"
22.4	Equipment Hatch (Between Doors)	5/11/79	-151.09	"
22.5	Personnel Hatch (Canopy)	2/19/79	0.00	"
22.6	Equipment Hatch (Canopy)	2/21/79	0.00	"
22.7	Equipment Hatch O-ring	2/21/79	0.00	"
22.8	Mech. Manifold A	11/15/79	1.20	"
22.9	Mech. Manifold B	2/20/79	2.08	"
22.10	Mech. Manifold C	2/20/79	0.00	"
22.11	Mech. Manifold E	2/20/79	1.13	"
22.12	Mech. Manifold F	2/20/79	0.00	"
22.13	Mech. Manifold G	11/15/79	3.11	"
22.14	Mech. Manifold H	11/15/79	5.65	"
22.15	Mech. Manifold I	5/11/79	-118.98	"
22.16	Mech. Manifold J	2/20/79	1.92	"
22.17	Mech. Manifold K	2/20/79	1.80	"
22.18	Electrical Manifold #1	2/19/79	212.99	"
22.19	Electrical Manifold #2	2/19/79	0.00	"
22.20	Electrical Manifold #3	2/22/79	32.22	"
22.21	Mechanical Manifold L	3/15/79	1.09	"
22.22	Fuel Transfer Flange	3/19/79	1.08	"

NOTE: Negative leakage due to temp. variations during test.

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

POSITION

With the assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4, 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.

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 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 occupany 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.

a. Areas Requiring Continuous Occupancy: $\leq$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.

RG&E Response

A radiation and design review has been completed to determine whether post-accident radiation fields unduly limit personnel access to areas necessary for mitigation of, or recovery from an accident; or unduly degrade the proper operation of safety equipment. This review was accomplished using the recommendations and guidelines of NUREG-0568 and subsequent clarifications as provided by the NRC. Appendix A presents details of the evaluation.

The review identified the most critical areas requiring personnel access following the onset of extreme accident conditions following a postulated major release of radioactivity into the Containment Building. Consideration was given to areas where predetermined post-accident functions would be performed (Nuclear Sample Room, Chemistry Laboratory, Count Room, Control Room and air sample penetrations) and to areas where personnel could be called upon to execute certain accident-mitigating or short-term recovery tasks (hydrogen recombiner panel, radwaste panel and building ventilation filters). Potential radiation exposures were determined for each task and included additional exposure due to accessing the areas where Tasks would be performed.

We have concluded that the actions which may be taken under the conservative post-accident conditions assumed in this analysis, can be accomplished without causing undue radiation exposures to personnel.

We have also identified certain areas which may require modifications to existing procedures, shielding, and/or relocation of equipment. These are discussed in detail in Section 4.0 of Appendix A and additionally in the RG&E response to Item 2.1.8.a. We are currently investigating both portable and permanent radiation shielding designs to reduce potential worker exposure resulting from post-accident sampling operations. Portable shadow shielding for post-accident sample collection, handling and analysis will be procured by January 1, 1981. Additional shielded sample containers for transport and disposal of samples will be made available at the earliest possible date prior to January 1, 1981.

Major modifications involving permanent shielding additions or piping relocations in relation to the primary sampling and waste gas systems are proposed in Section 4.0 of Appendix A. These are currently under review for seismic and other structural considerations. Relocation of radwaste panel indications and control functions is also proposed in Section 4.0 of Appendix A and is under current consideration.

Those major modifications which will be completed as of January 1, 1981 include the following:

	Location	Component	Modification
1.	Nuclear Sample Room (I.B. elev. 271')	Sample Coolers	Provide additional shielding around coolers.
2.	Access route by Nuclear Sample (I.B. elev. 271')	Primary sample lines	Provide additional shielding around sample lines; or possible line re-routing
3.	Access route into Auxiliary Building (I.B. elev. 271-278'; Aux. B. elev. 278')	RHR system sample line	Re-route sample line

The qualification of safety-related equipment has also been reviewed under the assumed accident condition guidance provided in NUREG-0578. The results of that review are presented in Section 5.2 of Appendix A.

Section 2.1.7.a - AUTOMATIC INITIATION OF THE AUXILIARY FEEDWATER SYSTEM (AFWS)

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 functions.

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 requirement.

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. <u>Other Considerations</u>

   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.

<u>RG&E Responses</u>

Conformance with the requirements of this position was documented by letter dated December 14, 1979 from L. D. White, Jr. to D. L. Ziemann, USNRC.

Section 2.1.7.b - AUXILIARY FEEDWATER FLOW INDICATION TO STEAM GENERATORS

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.

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 feed water flow channel may be backed up by a steam generator level channel.

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

RG&E Response

Conformance with the requirements of this position was documented by letter dated December 14, 1979 from L. D. White, Jr. to D. L. Ziemann, USNRC.

Section 2.1.8.a - IMPROVED POST-ACCIDENT SAMPLING CAPABILITY

POSITION

A design and operational review of the reactor coolant and containment atmosphere sampling systems shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) 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 safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (less than 2 hours) 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 performed promptly, i.e., the boron analysis within an hour and the chloride sample analysis within a shift.

DISCUSSION

The primary purpose of implementing Improved Post-Accident Sampling Capability is to improve efforts to assess and control the course of an accident by:

1. Providing information related to the extent of core damage that has occurred or may be occurring during an accident;

2. Determining the types and quantities of fission products released to the containment in the liquid and gas phase and which may be released to the environment;

3. Providing information on coolant chemistry (e.g., dissolved gas, boron and pH) and containment hydrogen.

The above information requires a capability to perform the following analyses:

1. Radiological and chemical analyses of pressurized and unpressurized reactor coolant liquid samples;

2. Radiological and hydrogen analyses of containment atmosphere (air) samples.

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_2) 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, 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 provisions 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 for 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:

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 here.

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 of the vicinity of the sampling station and analysis facilities shall be included in the evaluation.

4. High range portable survey instruments and personnel 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.

RG&E Response

A design and operational review of the reactor coolant and containment atmosphere sampling systems was performed to determine the improvements necessary for the prompt collection, handling and analysis of required post-accident samples without incurring ex-

cessive personnel radiation exposure. This review was performed following the guidelines of NUREG-0578 and subsequent clarification given in the October 30, 1979 letter from H. Denton to all operating nuclear power facilities.

We have identified certain procedural changes and equipment modifications which will be implemented in order to maintain personnel exposures within the accident dose criteria of 5 rem (GDC 19). Plant procedures for the handling and analysis of post-accident samples have been developed and will be implemented by January 1, 1980.

New procedures which have been provided are:

PC 23.1	Emergency Sampling of Primary Coolant
PC 23.2	Containment Atmosphere Sampling and Analysis During Containment Isolation

These procedures address the sample locations, radiological precautions (including use of shields), sample dilution requirements, means of handling the samples and necessary modifications to normal analytical procedures. Sample lines at a containment post-accident air sample penetration have been shortened to further reduce potential radiation exposure.

A variety of high range portable survey instruments and personnel dosimeters are available on site for rapid asssessment of personnel exposures.

Based upon the results of the shielding design review performed in response to Item 2.1.6.b, we have identified those plant areas where sampling system components need additional radiation shielding or relocation in order to reduce potential exposure to personnel under extreme post-accident conditions. Portable shadow shielding designs are now being investigated for use in sample collection, handling and analysis. Procurement of the portable shielding will be completed by January 1, 1981. Shielded sample containers are now available onsite and additional containers for transporting and disposing of the sample will be made available at the earliest possible date, prior to January 1, 1981. Alternative sampling methods for radiological and chemical analyses (boron, H_2, O_2) are currently being investigated. Continuous indication of hydrogen concentration in the containment atmosphere (0 to 10% hydrogen concentration range) will be provided in the Control Room by January 1, 1981 (see Response to Item 2.1.9).

Major sampling system modifications involving permanent shielding additions or piping relocation have been proposed in Section 4.0 of Appendix A to the Item 2.1.6.b response. These are currently under review for seismic and other structural considerations.

Therefore, we have concluded from our design and operational review that prompt collection and analysis of post-accident samples can be performed under the extreme conditions postulated

in NUREG-0578 without unacceptable exposure to personnel from radiation and airborne radioactivity. Certain procedural and equipment improvements will be implemented in a timely manner to maintain personnel exposures from emergency sampling operations within the dose guidelines of NUREG-0578.

Section 2.1.8.b - INCREASED RANGE OF RADIATION MONITORS

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 uCi/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) concentrations to a maximum of 10^5 uCi/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors shall overlap by a factor of ten.

2. Since iodine gaseous effluent monitors for the accident conditions 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 absorption on charcoal or other media, followed by on-site laboratory analysis.

3. In-containment radiation level monitors with a maximum range of 10^8 rad/hr [total or 10^7 rad/hr photon] 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.

DISCUSSION

The January 1, 1980 requirement, were specifically added by the Commission and were not included in NUREG-0578. The purpose of the interim January 1, 1980 requirement is to assure that licensees have methods of quantifying radioactivity releases should the existing effluent instrumentation go offscale.

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 serve only as a provisional fix with the more de-

tailed, 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.

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.

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 and 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 1.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.

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.

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
- SEISMIC - NO
- 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.

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.

RG&E Responses

2.1.8.b.1

Procedure PC-23.3 titled "Estimation of Noble Gas Release Rate From Plant Vent During Accident Conditions" to be used for estimating the rate of noble gas release through the plant vent if normal instrumentation is indicating above scale releases has been written. This procedure utilizes a hand-held instrument from which the curie content of a known volume in the vent line can be determined. The procedure has been reviewed by PORC and approved by the plant superintendent.

Eberline Model SPING-4 effluent samplers are planned for installation on release paths. The SPING-4 low and mid-range detectors overlap by a factor of 2.4 instead of the overlap of 10 suggested in the staff position. Conversations with members of the Environmental Evaluation Branch have confirmed that this is acceptable, however.

2.1.8.b.2

Procedure HP-11.2 Iodine in Air-Charcoal Cartridge Method is used to determine the concentration of iodine in plant effluents. A sample from the plant vent is collected on a SAI #CP 100 or equivalent charcoal cartridge by a Trapelo MAP 63 which is located adjacent to the vent. This unit has a continuous readout in the control room using a single channel Sodium Iodide crystal. The charocal cartridge is also counted in the laboratory on a Germanium - Lithium system after it has been purged of noble gases. Individual isotopes of radioiodines can be calculated as can other radionuclides from the prefilter. Samples will be transported from the sample location to the laboratory in a shielded container if high radiation doses are encountered.

Two counting laboratories are available on-site with independent power sources.

2.1.8.b.3

Two Victoreen model 875 high range radiation monitors have been ordered for installation in the containment. The monitors and associated equipment will be installed to meet the requirements of Table 2.1.8.b.3 and will have a range of 10^7 R/hr (photon only). In order to meet the required installation date of January 1, 1981, staff approval of this proposal is required by March 1, 1980. Approval after that date may mean installation will be delayed. Please notify us as early as possible of any potential problems.

Section 2.1.8.c - IMPROVED IN-PLANT IODINE INSTRUMENTATION UNDER ACCIDENT CONDITIONS

POSITION

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

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 at all areas where airborne iodine concentration data might be 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 Janaury 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.

RG&E Response

No additional response required.

Section 2.1.9 - TRANSIENT AND ACCIDENT ANALYSIS

POSITION

See NUREG-0578, page A-44.

DISCUSSION

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&P Task Force.

RG&E Response

Analyses of small break loss-of-coolant accidents, symptoms of inadequate core cooling and required actions to restore core cooling, and analysis of transient and accident scenarios including operator actions not previously analyzed are being performed on a generic basic by the Westinghouse Owners' Group, of which RG&E is a member. The small break analyses have been completed and were reported in WCAP-9600, which was submitted to the Bulletins and Orders Task Force by the Owners' Group on June 29, 1979. Incorporated in that report were procedure guidelines that were developed as a result of small break analyses. These guidelines have been reviewed and approved by the B&O Task Force and were presented to the Owners' Group utility representatives in a seminar held on October 16-19, 1979.

The work required to address the other two areas--inadequate core cooling and other transient and accident scenarios--has been performed in conjunction with schedules and requirements established by the Bulletins and Orders Task Force. Analysis related to the definition of inadequate core cooling and guidelines for recognizing the symptoms of inadequate core cooling based on existing plant instrumentation and for restoring core cooling following a small break LOCA were submitted on October 31, 1979. This analysis is a less detailed analysis than was originally proposed, and will be followed up with a more extensive and detailed analysis which will be available during the first quarter of 1980.

These guidelines were incorporated into specific plant procedures as applicable. The scope of revisions included altering the following existing emergency procedures:

1). E-1.1 Safety Injection System Actuation to Immediate Action and Diagnostics for Spurious Actuation of SI, LOCA, Loss of Secondary Coolant, and Steam Generator Tube Rupture.

2). E-1.2 Loss of Reactor Coolant

3). E-1.3 Steam Line Break to Loss of Secondary Coolant

4). E-1.4 Steam Generator Tube Rupture

5). E-1.5 Void Formation in Reactor Coolant System

Training of operators on these revised emergency procedures has been completed.

With respect to other transient and accidents contained in Chapter (14) of the FSAR, the Westinghouse Owners' Group will perform an evaluation of the actions which occur during an event by constructing sequence of event trees for each of the non-LOCA and LOCA transients. From these event trees a list of decision points for operator action will be prepared, along with a list of information available to the operator at each decision point. Criteria set for credible misoperation, and time available for operator decisions will be qualitatively assessed. The information developed will then be used to test Abnormal and Emergency Operating Procedures against the event sequences to determine if inadequacies exist in the procedures. The results of this study will be provided to the Bulletins and Orders Task Force by the end of the first quarter of 1980 as mutually agreed.

The Owners' Group has also provided test predictions analysis of the LOFT L3-1 nuclear small break experiment. This analysis was provided on December 15, 1979, in accordance with the schedule established mutually with the Bulletins and Orders Task Force.

Section 2.1.9 - CONTAINMENT PRESSURE INDICATION

POSITION

A continuous indication of containment pressure should 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.

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.

RG&E Response

RG&E will install containment pressure indication in conformance with the Staff position.

Section 2.1.9 - CONTAINMENT WATER LEVEL INDICATION

POSITION

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. A wide range instrument shall also be provided for PWRs and shall cover the range from the bottom of the containment to the elevation equivalent to a 600,000 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.

CLARIFICATION

1. The narrow range sump level instrument shall monitor the normal containment sump level vice 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 measurement 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.

RG&E Responses

No additional response required.

Section 2.1.9 - CONTAINMENT HYDROGEN INDICATION

POSITION

A continuous indication of hydrogen concentration in the containment atmosphere 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.

CLARIFICATION

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.

RG&E Response

RG&E intends to install containment hydrogen indication instrumentation in conformance with the Staff position.

Section 2.1.9 - REACTOR COOLANT SYSTEM VENTING

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

1. A description of the construction, 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.

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 noncondensible 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 plant's ability to maintain core cooling and containment integrity for events beyond the Design Basis.

B. BWR Design Considerations

1. Since the 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 noncondensible 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 noncondensible 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:

   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 noncondensible 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 from 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 noncondensible 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.

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 system 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 noncondensible 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.

RG&E Response

We are planning to establish a reactor coolant system high point vent system in accordance with the staff requirements by January 1, 1981. A system description and design details for the vent system (and particularly the reactor vessel head vent) are provided. In order to meet the installation date staff approval of our system design contained in the following system description is required by March 1, 1980. Approval of the design after March 1, will probably result in an inability to meet the January 1, 1981 installation date. This is because the current goal is to install the head vent during the spring refueling outage scheduled for March, 1980. If any potential problems are identified, please inform us as soon as possible, at least by February 1, 1980.

SYSTEM DESCRIPTION

REACTOR COOLANT SYSTEM
HIGH POINT VENT SYSTEM

1.0 INTRODUCTION

The basic functions of the R.E. Ginna reactor coolant system high point vent system are as follows:

- To remove noncondensable gases from the reactor vessel and head which may impair natural circulation or emergency core cooling.

- To remove a noncondensable or steam bubble in the pressurizer which prevents an orderly cold shutdown.

- To remove noncondensable gases from the steam generator U-tubes which may be impairing natural circulation.

2.0 GENERAL DESCRIPTION

2.1 REACTOR VESSEL HEAD VENT SYSTEM (RVHVS)

The reactor vessel head vent system is shown on Figure 1. The system as shown provides for venting the reactor vessel using only safety grade equipment. The active portion of the system consists of 4 one-inch solenoid operated globe valves connected to the normal vent pipe in two redundant flow paths. The isolation valves are powered by vital power supplies and are fail closed active valves in accordance with Regulatory Guide 1.48. The combination of safety-grade train assignments and valve failure modes will not prevent vessel head venting nor venting isolation with any single active failure.

The RVHVS connects to the existing 3/4 inch pipe used to manually vent the reactor during refilling operations. The remotely operated vent system branches off into 2 redundant flow paths whose piping and valves are supported by the seismic support platform or the lifting columns. The individual remotely operated vent paths are orificed to a diameter of 0.25" to allow two normal positive displacement charging pumps to maintain system inventory in the event of a break or valve failure downstream of the orifice. These orifices form the Safety Class 1 to 2 boundary of the system. The vent system piping discharges locally into the refueling cavity; a well ventilated, open area of the containment. Water released from the RVHVS resulting from leakage or saturated blowdown will drain to the A containment sump. System leakage will be detected by radiation and sump volume monitoring.

2.2 PRESSURIZER VENT SYSTEM

The power operated relief valves (PORV's) will be used to vent the top of the pressurizer. These valves function as a part of the automatic reactor coolant system pressure control system and can be manually controlled from the control room. The PORV's are air operated globe

valves which receive power from a vital bus. Each Ginna PORV is being replaced with a new valve which we expect to be seismically qualified and which will have enhanced availability under extreme environmental conditions. A seismically qualified nitrogen supply has been installed at Ginna to ensure the operability of the valve following a design earthquake.

2.3 STEAM GENERATOR U-TUBE VENTING

If the information available to the operator indicates a degradation of natural circulation, a reactor coolant pump(s) can be started for brief periods of time to sweep noncondensable gases from the U-tubes of the steam generators. The noncondensables can be swept into the reactor vessel head and pressurizer where the RVHVS or the PORV's can be opened if core cooling is being impaired by a noncondensable gas bubble.

3.0 SYSTEM DESIGN BASIS

3.1 REACTOR VESSEL HEAD VENT SYSTEM

The RVHVS is designed to vent in excess of one half the reactor coolant system in one hour from one of two available flow paths. The RVHVS is orificed to limit the blowdown from a break or inadvertently opened flow downstream of the orifice to within the capacity of two out of the three charging pumps at the Ginna plant. A venting period of at least 10 minutes is allowed before hydrogen concentrations reach 4 percent by volume in containment.

The system is completely safety grade and meets the single failure criteria for venting initiation and termination. The system is operated from the control room and has no automatic interlocks.

The RVHVS remains attached to the head during refueling and does not interfere with the manual venting process during vessel refill.

3.1.2 PRESSURIZER VENTING SYSTEM

The pressurizer venting system is designed to rapidly vent the pressurizer from one of two power operated relief valves.

The PORV's at the Ginna plant will be upgraded to improve their reliability after seismic events and during high temperature/humidity conditions. The PORV's will be replaced with the PORV's originally designated for the Sterling site. These valves will have a new high temperature diaphragm. In addition, the valves have seismically qualified N_2 supplies. The PORV's receive power from a vital bus. A failure modes and effects analysis was provided in a letter from L. D. White, Jr. to D. L. Ziemann, USNRC dated November 19, 1979.

4.0 ELECTRICAL POWER SUPPLIES

The power operated relief valves are powered from a vital bus. The PORV block valves are normally open fail-as-is motor operated valves. Each valve is powered by a separate vital bus. In the event that a previously opened PORV fails to close on demand, the corresponding block valve can be closed.

The reactor vessel head vent valves are powered by the vital busses. The head vent valves in series are powered from the same vital bus. This method of powering valves in conjunction with the fail closed design allows the RVHVS to meet the single failure criteria.

Each RVHVS flow path consists of two normally closed, normally de-energized valves. The two series valve arrangements eliminate the possibility of a spuriously opened flow path due to the spurious movement of one valve. As such, main control board power lockout of the valves is not necessary.

5.0 THERMAL INSULATION

The RVHVS piping is insulated up to the second isolation valves to minimize heat losses from the system.

6.0 EQUIPMENT PARAMETERS

6.1 REACTOR VESSEL HEAD VENT

Valves

- Solenoid Operated Globe Valves
  - Cv = 2.0
  - 1" Sch. 160S connections
  - Active valve per R.G. 1.48
  - Operating design pressure 2500 psig
  - Design temperature 680^{o}F
  - Design humidity 100%
  - Radiation environment - 10^8 rads γ post accident
  - Code compliance - ASME Sec. III, 1974, Class 2
  - Fail closed
  - Red/Green MCB status lights (Reed switches)

Piping

- Existing vent pipe -
  - 3/4" Sch. 80S
  - Code compliance - ANSI B31.1

- New piping
  - 3/4" and 1" Sch. 160S
  - Code compliance - ASME Sec. III, 1977, Class 1 and 2

Piping Supports and Support Structures

- Code compliance - ASME, Section III, 1977, Subsection NF for new supports and structures

6.2 PRESSURIZER VENT

Valves

- Power operated relief valves
  - Air operator
  - Cv=50
  - Design temperature - 680°F
  - Design pressure - 2485 psig
  - Design humidity - 100%
  - Fail closed
  - Code compliance - ASME, Secton III, Subsection NB
  - Radiation environment - 50R/hr Gamma

- PORV Block Valve
  - Motor operator
  - L/D = 13
  - Design temperature - 680°F
  - Design pressure - 2485 psig
  - Design humidity - 100%
  - Fail AS IS
  - Code compliance - ASME, Section VIII

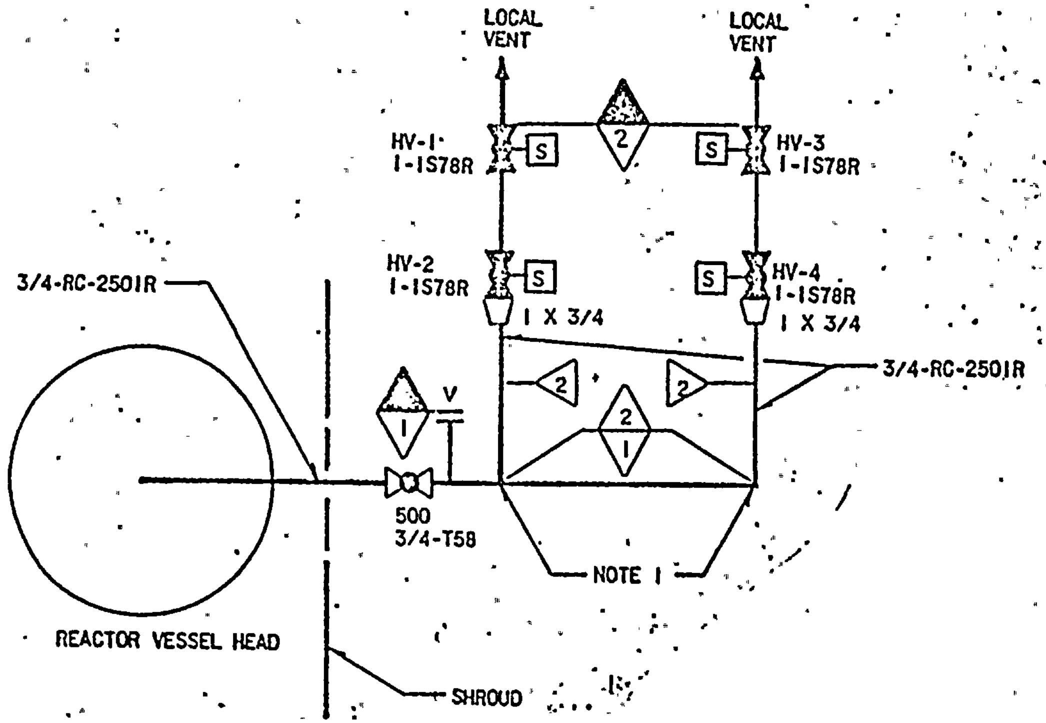
Piping

- Upstream of PORV
  - 3" and 4" Sch. 160S
  - Code Compliance - ANSI B31.1

- Downstream of PORV
  - 3" and 8" Sch. 40S
  - Non Nuclear Safety

Piping Supports and Support Structures

- PORV Upstream Piping is supported in accordance with ASME, Section III, Subsection NF



NOTES:

1. 1/4 INCH FLOW RESTRICTION ALLOWS TRANSITION FROM ANSI SAFETY CLASS 1 TO 2.

FIGURE 1

BY	DATE	BY	DATE	BY	DATE
PRELIMINARY INFO. ONLY		APPROVED FOR LAYOUT		APPROVED FOR CONSTRUCTION	

DRAWING STATUS

DFTM J ROBINSON 12-3-79
CHKR
DES ENG 12/5/79
MFG ENG
MTLS ENG
APP
APP
APP
DFTG SUPV

Westinghouse Electric Corporation
NUCLEAR ENERGY SYSTEMS PITTSBURGH PA

TITLE ROBERT E. GINNA NUCLEAR STATION
ROCHESTER GAS AND ELECTRIC
REACTOR VESSEL HEAD VENT SYSTEM

SCALE
DIMENSIONS IN INCHES
2654C03

Section 2.2.1.a - SHIFT SUPERVISOR RESPONSIBILITIES

POSITIONS

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 responsiblie 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.

CLARIFICATION

The attachment provides clarification to the above position.

Attachment

Section 2.2.1.A - SHIFT SUPERVISOR RESPONSIBILITY

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 Personnel, 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

RG&E Responses

POSITION 1:

The Vice President of Electric and Steam Production has issued, and plans to annually re-issue a directive to Ginna Station personnel which emphasizes the primary management responsibility of the Ginna Station Shift Supervisor for safe operation of the plant under all conditions and which clearly establishes his command duties. A copy has been sent to the USNRC Office of Inspection and Enforcement Region I Office.

POSITION 2:

Plant procedures have been reviewed to ascertain the extent of which duties, responsibilities and authority of the Ginna Station Shift Supervisor and Control Room Operators are defined. A definite line of command and clear delineation of command decision authority of the Shift Supervisor is provided in procedure A-52.1, Shift Organization and Responsibility in its section on Control Room Operations.

This is reinforced in the specific listing of duties in procedure A-201 Ginna Station Administrative and Engineering Staff Responsibilities. Regarding the additional particular emphasis required, the following information is provided:

a. Procedure A-52.1 Shift Organization and Responsibility has been revised to explicitly address the need for a broad perspective and avoidance of total involvement in any single operation.

b. Procedure A-52.1 also specifies that during accident conditions the Shift Supervisor shall remain in the Control Room until properly relieved. Those authorized to relieve the Shift Supervisor are specified.

c. The above procedure also provides clear indication of the control room command function of the Head Control Operator during temporary absence of the Shift Supervisor.

POSITION 3:

A management directive has been issued to the Shift Supervisors which emphasizes and reinforces their responsibility for safe operation under all conditions and their management function. It addresses the need for a broad perspective in operational conditions affecting plant safety and establishes their command duties. This directive will be reissued on an annual basis.

POSITION 4:

Duties of the Ginna Station Shift Supervisor have been reviewed by the Vice President, Electric and Steam Production in his

review of each Position Analysis for the job classifications in his department. The administrative duties performed by the Shift Supervisor have been recently reviewed and it was determined that all duties were related to safe operation of the plant.

Section 2.2.1.b - SHIFT TECHNICAL ADVISOR

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.

DISCUSSION

The NRC Lessons Learned Task Force has recommended the use of Shift Technical Advisors (STA) as a method of immediately improving the plant operation staff's capabilities for response to off-normal conditions and for evaluating operating experience.

In defining the characteristics of the STA, we have used the two essential functions to be provided by the STA. These are accident assessment and operating experience assement.

1. Accident Assessment

   The STA serving the accident assessment function must be dedicated to concern for the safety of the plant. The STA's duties will be to diagnose off-normal events and advise the shift supervisor. The duties of the STA should not include the manipulation of controls or supervision of operators. The STA must be available, in the control room, within 10 minutes of being summoned.

   The qualifications of the STA should include college level education in engineering and science subjects as well as training in reactor operations both normal and off-normal. Details regarding these qualifications are provided in paragraphs A.1, 2 and 3 of Enclosure 2 to our September 13, 1979 letter. In addition, the STA serving the accident assessment function must be cognizant of the evaluations performed as part of the operating experience assessment function.

2. Operating Experience Assessment

   The persons serving the operating experience assessment function must be dedicated to concern for the safety of the plant. Their function will be to evaluate plant operations from a safety point of view and should include such assignments as listed on pages A-50 and A-51 of NUREG-0578. Their qualifications are identical to those described previously under accident assessment and collectively this group should provide competence in all technical areas important to safety. It is desirable that this function be performed by onsite personnel.

CLARIFICATION

1. Due to the similarity in the requirements for dedication 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 judgments 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.

RG&E Response

The present shift operation organization, augmented by an additional licensed individual in the Control Room and further augmented by on-call Duty Engineers holding NRC Senior Reactor Operator Licenses and an expanded training program, represents a suitable alternative for accomplishing the objectives of the NRC Task Force position for a Shift Technical Advisor.

The Ginna Station Operation Department consists of five shifts with a minimum of five operators per shift, two of whom hold Senior Reactor Operator Licenses. The Shift Foremen have maintained their licenses since before January 1970 and have a broad knowledge of plant design and operation.

The present shift complements are normally comprised of a Shift Foreman (SRO), Head Control Room Operator (SRO), Control Room Operator (RO) and 3 Auxiliary operators (some with RO licenses). To meet the requirements of Safety Monitoring Function one additional licensed individual will be assigned to the control room when the reactor is critical and will be designated as Shift Technical Advisor. Thus the number of licensed individuals normally in the Control Room will be increased to 3. During the course of a reactor transient this additional licensed individual will be in the control room and will only be functioning in the Technical Advisory capacity to the Shift Supervisor. The Shift Technical Advisor will administratively report to the Technical Assistant - Operational Experience and Assessment Engineer.

The concept of an on-call Duty Engineer has been in existence since our commercial operation in 1970. The individuals on-call are plant staff engineers and possess an NRC Senior Reactor Operators License. We have purchased and assigned a 4 wheel drive vehicle with two way radio communication to the on-call duty engineer.

Administrative procedure A-201 "Ginna Station Administrative and Engineering Staff Responsibilities" has been revised to include the responsiblities of the Shift Technical Advisor and the responsibilities of the Technical Assistant - Operational Experience Assessment Engineer.

The Training Program to augment the technical education of the assigned Shift Technical Advisors for the safety monitoring function in the Control Room will be as follows.

Rochester Institute of Technology, College of Continuing Education will be providing the technical education required. This educational program will commence on January 8, 1980. Classes will be held in the morning and evening, two days per week, (forty-four weeks per year), with the same material presented during both classes of the same day to allow the necessary flexibility for shift personnel attendance. Classes will be held at the Brookwood (Ginna) Information and Science Center.

Table 2.2.1.b indicates the curriculum for the first three years. Besides providing the Shift Technical Advisor with the areas of knowledge suggested by the NRC, this program will allow participants to gain an AAS degree in an engineering discipline in approximately four years and a Bachelor degree in an engineering discipline in an additional two years.

Although we are not completing the program in 1 year, it is a major commitment to provide a quality program. We have 10 years of operating experience with a total of 164 man-years of licensed operator experience. This safe operation experience and an expanded training program provides a high level of confidence that the operating staff will perform safely and properly under accident conditions.

Table 2.2.1 b

Year I

First Quarter	Second Quarter	Third Quarter	Fourth Quarter
Technical Mathematics (CTAM 201) Elements of Electricity and Electronics (CTIL 201) Lab (CTIL 206)	Technical Mathematics (CTAM 202) Elements of Electricity and Electronics (CTIL 202) Lab (CTIL 207)	Introduction to Solution of Engineering Problems (CTEM 420) Elements of Electricity and Electronics (CTIL 203) Lab (CTIL 208)	Solution of Engineering Problems I (CTEM 421) Physics (CTCP 301) Lab (CTCP 306)

Year II

First Quarter	Second Quarter	Third Quarter	Fourth Quarter
Solution of Engineering Problems II (CTEM 422) Physics (CTCP 302) Lab (CTCP 307)	Materials Technology I (CTEM 414) Physics (CTCP 303) Lab (CTCP 308)	Materials Technology II (CTEM 415) Digital Systems (CTIE 321)	Thermodynamics and Heat Transfer (CTEM 441) Analog Systems (CTIE 322)

Year III

First Quarter	Second Quarter	Third Quarter	Fourth Quarter
Electromechanical Devices and Systems (CTIL 351) Mechanics of Fluids (CTEM 461)	Electromechanical Devices and Systems (CTIL 352) Reactor Physics*	Electromechanical Devices and Systems (CTIL 353) Reactor Thermodynamics*	Pneumatic and Hydraulic Systems (CTIL 303) Reactor Control*

*Accreditation not yet granted

Section 2.2.1.c - SHIFT AND RELIEF TURNOVER PROCEDURES

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 acceptance 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. Checklist 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 or test that by themselves could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate operational transients (what to check and criteria for acceptable status will be included on the checklist.)

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

CLARIFICATION

No clarification provided.

RG&E Responses

POSITION 1:

Procedure A-52.1 Shift Organization and Responsibility was revised so as to establish a new separate procedure for Shift Relief Turnover to formalize the transfer of operational information between shifts. It calls for review by the oncoming and offgoing Control Room Operators and the oncoming and offgoing Shift Supervisor as attested to by their signature.

a. It specifies review of a checklist of critical plant parameters which include alarm values and Plant Technical Specification limits, with direction to document any value not in the allowable range for the current operating mode.

b. It also requires documentation of checking the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents, by use of procedure O-6.2, Main Control Board Status. It includes the criteria for acceptance status. This procedure is completed each shift and requires documentation of any system not in its prescribed configuration.

c. It specifies review of systems and components that are in a degraded mode of operation permitted by Technical Specifications, as reported in accordance with procedures A-52.4, Control of Limiting Conditions for Operating Equipment, and A-52.5, Control of Limiting Conditions for System Specifications. The length of time in a degraded mode is identified in these reports.

<u>POSITION 2</u>:

Auxiliary Operators and Shift Health Physics Technicians maintain logs of the activities they perform for the purpose of informing their reliefs of activities performed or in progress and unusual conditions. New procedures have been written to provide a method for transferring information at shift change. Checklists have been provided to indicate equipment to be checked and criteria for acceptable status.

Section 2.2.2.a - CONTROL ROOM ACCESS

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 supervisor, 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 any 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.

CLARIFICATION

No clarification provided.

RG&E Response

An Administrative Procedure, A-52.1 Shift Organization and Responsibilities has been revised to formalize existing policies which allow the Shift Supervisor to restrict access to the control room during both normal operation and emergencies. This procedure establishes authority and responsibilities in the Control Room in the event of an emergency, and provides for a line of succession for the person in charge of the plant operations. The line of succession has been limited to those who possess a current NRC Senior Operator's License. The procedure provides that access by the NRC may be limited to the primary site inspector at the discretion of the Shift Supervisor.

Procedures have been established to designate individuals who should report to the Technical Support Center and the Operational Support Center in the event of a site emergency. These procedures include the lines of communication and authority of reporting personnel.

Section 2.2.2.b - ONSITE TECHNICAL SUPPORT CENTER

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 readily available to personnel in the TSC.

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. Assimulate 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 & 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.

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:

Plant Safety Systems Parameters for:

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

In-Plant Radiological Parameters 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.

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.

RG&E Response

Interim Technical Support Center

To meet the short term (interim) requirements for an on site technical support center (TSC) by January 1, 1980, the existing planalog room (present interim TSC) of the service building has been activated with the services necessary to meet the intent of the requirements.

Present modifications to the service building dictate that the TSC be relocated to a new room (new interim TSC) in the service building. The relocation will be completed in early 1980 without interruption to any service or capability requirement of the TSC. This TSC will be activated until January 1, 1981 at which time the permanent TSC will be activated.

The following tasks have been completed and fulfill the requirements of the interim TSC.

A. The present interim TSC has been established and a description of the facility is given in this response.

B. Plans and procedures for support and staffing of the TSC have been completed. Personnel have been assigned to man the center and their respective responsibilities designated.

The latest up-to-date drawings that are required to assess an incident are located in the TSC. QC document control procedures have been revised to insure that these drawings are kept current.

C. Communications between the TSC, NRC, emergency center, standby emergency center, onsite operational support center and control room have been established by adding two (2) company extensions to the TSC which can operate through the Rochester Telephone system. Additionally, two (2) Ontario (New York Telephone) phones have been added to the TSC. All four of these lines will be available on any of the four hand sets.

The NRC direct line to the control room has had an extension placed in the TSC.

A hard wired intercom system has been installed with the master unit located in the TSC and slave units located in the control room, emergency center, standby emergency center and the onsite operational support center. The master unit has the capability of calling and communicating with each slave station, but each slave unit can only call the TSC. Thus, a mode of communication directly under the TSC control is assured. One spare master unit will be located in the plant superintendent's office and a spare slave unit will be located in the TSC.

To further insure communications reliability, 6 portable radios are being purchased for the TSC which are of the same frequency as the base radios in the control room and emergency center.

D. A storage cabinet is located in the present interim TSC which contains masks with charcoal cannisters, DC battery lighting packs, and potassium iodide tablets. Action levels including evacuation to the control room have been established for the various protective responses.

A radiation monitor with local readout only is installed in the TSC. A radiation detector capable of local readout with alarm and remote readout has been ordered. Because of lead time involved in obtaining this instrument and the relatively

short period of time the present interim TSC will be in service, it is not intended to be installed there. However, the necessary cabling etc. has been installed to provide an installation in the new interim (early 1980) TSC. This monitor system will provide an indication of radiation levels in the TSC at the emergency center as well as perform the normal function of local alarm and indication in the TSC.

E. Access to vital plant parameters is provided for in the TSC by utilizing a plant computer data link. Software has been developed by which selected groups of parameters can be obtained in log sheet fashion. Printout of a single parameter or group of parameters is available.

Ninety percent of a control board video scan system is installed but will not be completed until the new interim TSC is placed in service.

Power for the present TSC is supplied by a normal nonvital bus source, however, upon loss of normal power, backup power is provided by an emergency lighting panel.

F. Procedure SC1.3E for manning the TSC has been approved. This procedure also directs personnel to the control room should the interim technical support center become unavailable.

Permanent Technical Support Center

The permanent technical support center (TSC) is intended to be completed by January 1, 1981. A preliminary review indicated that several thousand square feet of floor space might be needed to meet all the requirements of NUREG-0578. The TSC will occupy a one story addition to the present AVT building to the east of the turbine building and will be approximately 5000 square feet. The location clearly meets the requirement of "close to the control room".

The power supply for the TSC will be an uninterruptible power supply (UPS) backed up by a diesel generator. The system is planned to be identical to the present power supply now in operation for the Ginna Station security system.

A HVAC system will be installed for this addition that will provide a habitable environment at least equal to that of the control room. Ample shielding for occupants during an accident will be provided to limit exposures below those of GDC 19 considering major sources of radiation.

The accident assessment area will have computerized graphic display systems capable of reading out various vital plant parameters including radiation and meteorology data. The necessary dedicated communications will be provided in this area. In addition a radiation monitor will be provided.

A conference room, offices with telephones for at least RG&E and NRC, a working area, toilet facilities and a combination record, print, and reproduction room will also be provided.

Design of the facility is at a preliminary stage. It is our intent, however, to provide a permanent TSC which meets the staff requirements outlined in the clarifications above. More information will be available as the design progresses throughout the first quarter of 1980.

Section 2.2.2.c - ONSITE OPERATIONAL SUPPORT CENTER

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.

CLARIFICATION

No clarification provided.

RG&E Response

The auxiliary operator office and adjacent technical library have been designated as the Operational Support Center. This area is located at the east end of the turbine building one floor below the control room. It is equipped with telephones and a station intercom system which connects the Control Room, Technical Support Center and Emergency Survey Center. Procedure SC-1.3E for manning the center has been reviewed by PORC and approved by the plant superintendent as part of the emergency procedures series.

APPENDIX A