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 letter from L. D. White, Jr. to Dennis L. Ziemann, USNRC, dated October 17, 1979.

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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 nave 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 at the Ginna Nuclear Plant conforms to all requirements set forth above. Procedures to manually load the heaters onto emergency power sources will be completed by January 1, 1980. They will include criteria to prevent overloading a diesel generator. Consideration will be given to the necessity to shed ESF loads and the reset of the safety injection signal.

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.

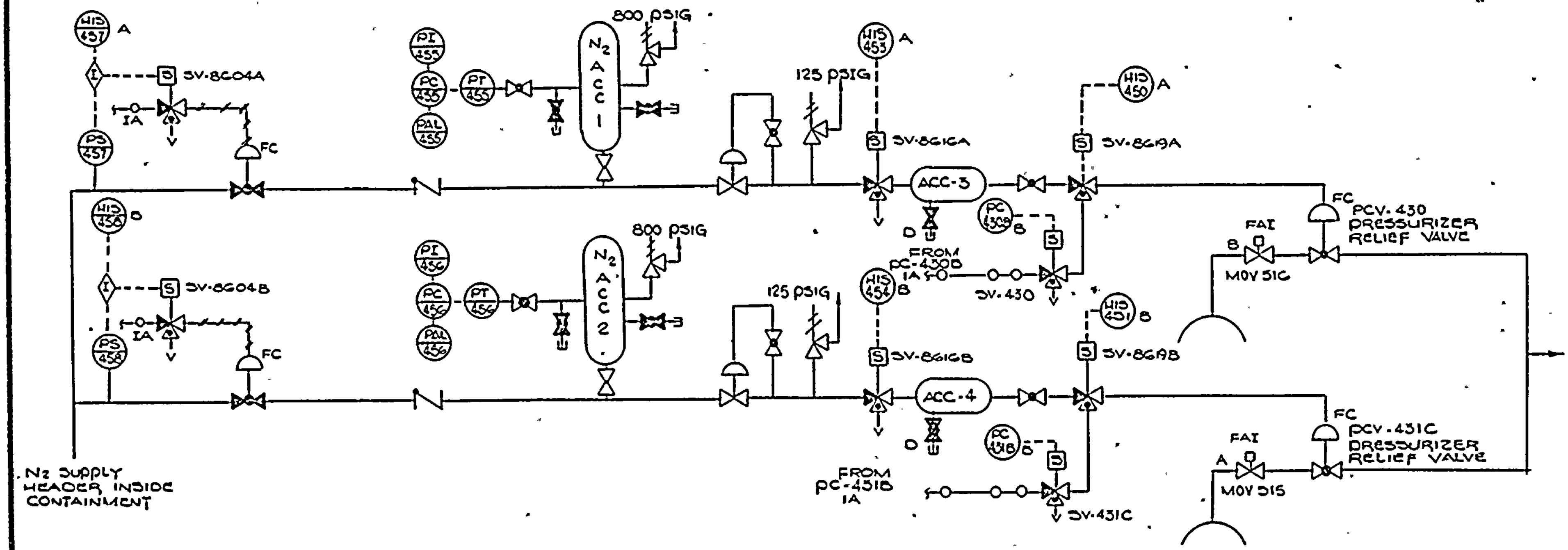
2. The motive and control power for the block valve should be supplied from an emergency power bus different from that which supplies the PORV.

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

The attached Failure Mode and Effects Analysis systematically reviews the consequences of all single failures in the pressurizer PORV/block valve system, including the motive and control power systems. The system is shown to be capable of performing its safety function (closure) after any single failure. It is also shown that pressurizer pressure relief (using PORVs) may be accomplished after any single failure.



FC - FAIL CLOSE
FAI - FAIL AS IS
V - VENT
A - A-TRAIN POWER
B - B-TRAIN POWER
IA - INSTRUMENT AIR

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		11-16-79	11-16-79	11/16/79	[illegible]
NUMBER	REVISION	DRAWN BY	CHECKED	RESP. ENG.	ENG. MANG'R.

FACILITY GINNA STA. — ROCHESTER GAS & ELECTRIC CORP. ROCHESTER, NEW YORK

SCALE NONE

PRESSURIZER PORV/BLOCK VALVE FAILURE MODE AND EFFECT ANALYSIS PIPING, INSTRUMENTATION AND CONTROL DIAGRAM

JOB NO.	DRAWING NO	REV.
EWR-2601	21489-299	0

1. COLD SHUTDOWN
2. POWER OPS

FAILURE MODE AND EFFECTS ANALYSIS

PRESSURIZER CONTROL SYSTEM

COMPONENT IDENTIFICATION	PLANT OPERATING MODE	FUNCTION	FAILURE MODE	FAILURE MECHANICS	EFFECT ON SYSTEM	METHOD OF FAILURE DETECTION	REMARKS
1. A-TRAIN POWER (A.C. & D.C.)	1 OR 2	PROVIDE MOTIVE & CONTROL POWER TO THE A-ASSOCIATED VALVES: MOV-515, SV-8619A, SV-8616A & SV-8604A	LOSS OF POWER	SHORTED BUSS, LOSS OF SUPPLY POWER	INABILITY TO CLOSE VALVE MOV-515	ALARM IN MAIN CONTROL ROOM	ISOLATION CAPABILITY STILL PROVIDED BY ASSOCIATED PORV, PCV-431C
2. B-TRAIN POWER (A.C. & D.C.)	1 OR 2	PROVIDE MOTIVE & CONTROL POWER TO THE B-ASSOCIATED VALVES: MOV-516, PC-430, PC-431C, SV-8619B, SV-8616B & SV-8604B	LOSS OF POWER	SHORTED BUSS, LOSS OF SUPPLY POWER	INABILITY TO CLOSE VALVE MOV-516; PCV-430, PCV-431C BOTH FAIL CLOSE, INABILITY TO REOPEN	ALARM IN MAIN CONTROL ROOM	ISOLATION OBTAINED FROM PORV'S
3. PLANT INSTRUMENT AIR SYSTEM	1 OR 2	INSTRUMENT AIR SUPPLY TO AIR OPERATED VALVES	COMPLETE OR PARTIAL LOSS OF PRESSURE	ANY DESIGN BASIS EVENT - NON QUALIFIED SYSTEM	PCV-430, PCV-431C BOTH FAIL CLOSE	ALARM IN MAIN CONTROL ROOM	ISOLATION OBTAINED
4. BLOCK VALVES (MOV-515, MOV-516)	1 OR 2	TO PROVIDE SECONDARY MEANS OF ISOLATING PRESSURIZER PORV LINES UPON FAILURE OF PORV TO CLOSE AS NEEDED	a) FAIL CLOSE b) FAIL OPEN	a) MECHANICAL FAILURE, MOTOR OPERATOR ELECTRICAL FAILURE b) MECHANICAL FAILURE	NONE INABILITY TO CLOSE VALVE	VALVE STEAM MOUNTED LIMIT SWITCH & PERIODIC TESTING	ISOLATION CAPABILITY STILL PROVIDED BY ASSOCIATED PORV'S
5. PRESSURIZER POWER OPERATED RELIEF VALVES (PCV-430, PCV-431C)	1 OR 2	TO PROVIDE MITIGATION IN THE EVENT THE RCS IS OVER-PRESSURIZED	a) FAIL CLOSE b) FAIL OPEN	a) MECHANICAL FAILURE b) MECHANICAL FAILURE	a) NONE b) LOSS OF FUNCTION OF ONE POWER RELIEF VALVE	PERIODIC TESTING ALARM IN MAIN CONTROL ROOM	b) ASSOCIATED BLOCK VALVE WOULD BE CLOSED TO OBTAIN ISOLATION
6. SOLENOID OPERATED VALVES (SV-431C, SV-430)	1 OR 2	TO PROVIDE CONTROL AIR SIGNAL TO OPEN PRESSURIZER POWER RELIEF VALVES WHEN PRESSURIZER CONTROL SYSTEM SENDS "OPEN" SIGNAL	a) FAIL VENT b) FAIL OPEN	a) MECHANICAL FAILURE ELECTRICAL FAILURE b) MECHANICAL FAILURE	PORV CLOSES, ISOLATION OBTAINED SPURIOUS OPENING OF RELIEF VALVE	PERIODIC TESTING ALARM IN MAIN CONTROL ROOM	ASSOCIATED BLOCK VALVE WOULD BE CLOSED TO OBTAIN ISOLATION
7. SOLENOID OPERATED VALVES (SV-8619A, SV-8619B)	1. 2.	TO PROVIDE N_2 TO OPEN PCV-430, PCV-431C WHEN "OPEN" SIGNAL RECEIVED FROM OVER-PRESSURE PROTECTION SYSTEM TO PROVIDE CONTROL AIR SIGNAL FROM PRESSURIZER CONTROL SYSTEM, PC-430B & PC-431C	a) FAIL VENT b) FAIL OPEN c) FAIL VENT d) FAIL OPEN	a) MECHANICAL FAILURE ELECTRICAL FAILURE b) MECHANICAL FAILURE c) MECHANICAL FAILURE ELECTRICAL FAILURE d) MECHANICAL FAILURE	a) LOSS OF FUNCTION OF ONE POWER OPERATED RELIEF VALVE ISOLATED b) PORV OPENS c) NONE d) NONE	PERIODIC TESTING PERIODIC TESTING	a) REDUNDANT RELIEF VALVE AVAILABLE b) ASSOCIATED BLOCK VALVE WOULD BE CLOSED TO OBTAIN ISOLATION d) VENTING WOULD OCCUR THROUGH VALVE SV-8616A
8. SOLENOID OPERATED VALVES (SV-8616A, SV8616B)	1. 2.	TO ALLOW CHARGING OF SECONDARY ACCUMULATORS* TO VENT THE SECONDARY ACCUMULATORS TO ASSURE PREVENTION OF SPURIOUS ACTUATION	a) FAIL VENT b) FAIL OPEN c) FAIL VENT d) FAIL OPEN	a) MECHANICAL FAILURE ELECTRICAL FAILURE b) MECHANICAL FAILURE c) MECHANICAL FAILURE ELECTRICAL FAILURE d) MECHANICAL FAILURE	a) LOSS OF FUNCTION OF ONE POWER OPERATED RELIEF VALVE ISOLATED b) NONE c) NONE d) NONE	PERIODIC TESTING PERIODIC TESTING	REDUNDANT RELIEF VALVE AVAILABLE IF NEEDED TO RELIEVE PRESSURE, OTHERWISE IT IS ISOLATED

*ACCUMULATORS 1 & 2 CAN EACH CYCLE A PORV 150 TIMES

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

RG&E is a member of an Owners group formed by utilities owning and operating Westinghouse reactors. The Westinghouse Owners Group is working in conjunction with other PWR owners and the Electric Power Research Institute (EPRI) to develop a program for qualification of relief and safety valves under expected operating conditions. We will follow the program and schedule developed and carried out by EPRI. Although the program is not yet complete, the intent is to comply with the NRC clarifications.

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

The power operated relief valves have direct stem position indication functionally conforming with the Staff position. Functional indication of safety valve position is provided in accordance with the following position.

The thermocouple located in the discharge pipe of each (of two) pressurizer safety valve provides unambiguous indication of valve movement or significant seat leakage. Opening of a safety valve will cause a rapid elevation in discharge temperature which is alarmed and annunciated in the Control Room. The thermocouple response time is short with respect to operator capability to observe, evaluate and take action. Since the only other valves which can normally discharge steam into this system are the PORV's, which have direct indication of stem position, any possible ambiguity is removed by checking the PORV position indication.

When the operator has determined that a pressurizer safety valve has opened, he must initiate a course of action based on the assumption that the safety valve will not reseat. It should be noted that this differs in principle with action taken in response to an open PORV, where initial operator action is to either close the valve itself or to close the associated block valve. The operator's subsequent actions following the opening of the safety valve must be based on primary system temperature and pressure and not on the safety valve position. In this way the operator is not dependent on safety valve closure for safe recovery of the plant.

Reseating of a safety valve after opening will result in a rapid initial drop in discharge temperature although not to normal levels. This indication may be used with the primary pressure and temperature and pressurizer relief tank level, temperature and pressure to provide a basis for the decision to terminate actions taken to mitigate the effects of an open safety valve. However, it should be noted that such action should never be taken on safety valve position indication alone, since unlike PORV's, the safety valves do not have backup block valves, and incorrect position indication could lead the operator to terminate mitigating actions prematurely unless primary system parameters are properly considered.

It is our position that the above technical review provides a totally adequate and sound basis for safe plant operation until at least the scheduled refueling outage of March 1980. Little additional safety margin is provided by direct safety valve position indication. Nevertheless, RG&E proposes to install direct valve stem position indication during the refueling outage to augment the instrumentation described above. The RG&E design will utilize linear variable differential transformers (LVDTs) to provide continuous valve stem position indication, from fully closed to fully open.

The environmental qualification of electrical components has been reviewed with respect to accidents associated with safety and PORV lifting. The materials and components are suitable for such environments.

We believe this response to be technically sound and totally responsive to the Staff concerns in this matter.

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 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

RG&E is installing two redundant channels of subcooling margin monitoring. Each channel will be composed of 1) existing RCS temperature and pressure measurements, 2) a dedicated, fully qualified, analog, saturation temperature calculator and alarm, and 3) an analog display showing subcooling margin. The principle components for this system are on order, with projected delivery dates in mid December. Delays in procurement which might be caused by component testing, manufacturing difficulties or shipping problems may result in an extended schedule, however, we are expediting the delivery to the extent possible. Based on the projected delivery schedule RG&E will install this equipment prior to January 1, 1980. The equipment data requested by the Staff is attached.

Procedure guidelines that are used by the operator in recognizing inadequate core cooling will be submitted by the Westinghouse Owners Group for review by the Bulletins and Orders Task Force.

INFORMATION REQUIRED ON THE SUBCOOLING METER

Display	
Information Displayed (T-Tsat, Tsat, Press, etc.)	$T_{SAT} - T_{HOT}$
Display Type (Analog, Digital, CRT)	Analog
Continuous or on Demand	Continuous
Single or Redundant Display	Single display/channel
Location of Display	Control Room
Alarms (include setpoints)	one per channel/ $T_{SAT} - T_{HOT} < 50°F$
Overall uncertainty (°F, PSI)	2.5°F
Range of Display	0-100°F
Qualifications (seismic, environmental, IEEE323)	None (Display non-1E)
Calculator	
Type (process computer, dedicated digital or analog calc.)	Analog
If process computer is used specify availability. (% of (time)	-
Single or redundant calculators	one calculator/channel
Selection Logic (highest T., lowest press)	None
Qualifications (seismic, environmental, IEEE323)	Seismic & Environmental, IEEE-323 & 344
Calculational Technique (Steam Tables, Functional Fit, ranges)	function to fit to saturation curve
Input	
Temperature (RTD's or T/C's)	RTD-200 ohm Pt.
Temperature (number of sensors and locations)	one sensor/channel RCS hot leg
Range of temperature sensors	500-700°F

Uncertainty* of temperature sensors (°F at 1)	see note 1
Qualifications (seismic, environmental, IEEE323)	see note 2
Pressure (specify instrument used)	Foxboro 611GM
Pressure (number of sensors and locations)	one/channel, Pressurizer
Range of Pressure sensors	1700-2500 psig
Uncertainty* of pressure sensors (PSI at 1)	± 2 psi
Qualifications (seismic, environmental, IEEE323)	Qualified for 60 psig and 286°F note 3
Backup Capability	
Availability of Temp & Press	Core exit TC and pressurizer pressure indication in the control room
Availability of Steam Tables etc.	available in control room
Training of operators	complete
Procedures	complete

*Uncertainties must address conditions of forced flow and natural circulation

Note 1: Accuracy of RTD's

Temperature (°F)	Accuracy (±°F)
32	.011
525	.055
625	.065

Note 2: RTD Qualifications (Reference: Rosemount Engineering Co. Drawing 176JA)

1. RTD (except lead wires) capable of exposure of -30 to 650°F. Lead wires: -30 to 200°F
2. Vibration: 10G peak from 20 to 2000 HZ along and through mutually perpendicular axis for 15 minutes.
3. Radiation: 200R/hour

Note 3: Qualification information is given in RG&E letter from L. D. White, Jr. to A. Schwencer, USNRC, dated February 24, 1978.

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

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 have been performed on a generic basis by the Westinghouse Owners Group, of which RG&E is a member.

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

The existing containment isolation system at the Ginna Nuclear Plant conforms with the diversity requirements of the Staff position. See Westinghouse drawing 882D612 sheet 6 submitted in a letter to D. L. Ziemann on November 22, 1978. This drawing also shows which sensors provide containment isolation and containment ventilation isolation. The system automatically isolates all nonessential systems not already isolated. Essential and non-essential systems are identified on Table 2.1.4.

The effect of resetting containment isolation and containment ventilation isolation was discussed in detail in our responses to item 9 of IE Bulletin 79-06A dated April 28, 1979 and June 22, 1979 and in our letters to the NRC dated January 2, 1979, February 16, 1979 and March 30, 1979. As identified in those letters, there are certain valves which could reopen upon reset of the containment isolation or containment ventilation isolation if their controllers were set in the open position.

The reopening of valves is currently precluded by several means. First, the operator is directed to place all valve position controllers in the closed position so that no valve will open on initiation of the reset. The reset of containment ventilation isolation can be actuated only through use of a key switch. The key is under the control of the shift foreman. Therefore, no single operator error can result in improper use of this reset function. The reset for containment isolation, originally a reset button, has been replaced with a key switch. To further reduce the likelihood of inadvertent reopening of valves, a system modification has been designed to provide for individual resetting of all isolation valves to eliminate any possibility of an inadvertent opening. Equipment for this modification was recently ordered, following the necessary phases of preliminary engineering, bid requests, and bid evaluations and every effort is being made to expedite delivery. The vendor estimates twenty-two weeks for delivery. It should be noted that in order not to degrade the Class 1E system of which it will be a part, this equipment (including over 150 relays) must be fully qualified seismically and environmentally. The equipment being procured is housed in four large cabinets which will be located in the relay room. Installation also involves wiring between the relay room and the control room.

Table 2.1.4

Essential and Nonessential System Containment Penetrations

PENT. NO.	IDENTIFICATION/DESCRIPTION	ESSENTIAL VS. NONESSENTIAL
29	Fuel transfer tube	Nonessential
100	charging line to "B" loop	Nonessential
101	SI Pump 1B discharge	Essential
102	Alternate charging to "A" cold leg	Nonessential
105	Containment Spray Pump 1A	Essential
106	"A" Reactor Coolant Pump (RCP) seal water inlet	Essential
107	Sump A discharge to Waste Holdup Tank	Nonessential
108	RCP seal water out and excess letdown to VCT	Nonessential
109	Containment Spray Pump 1B	Essential
110	"B" RCP seal water inlet	Essential
110	SI test line	Nonessential
111	RHR to "B" cold leg Essential	
112	letdown to Non-regen. Heat exchanger	Nonessential
113	SI Pump 1A discharge	Essential
120	Nitrogen to Accumulators	Nonessential
120	Pressurizer Relief Tank (PRT) to Gas Analyzer (GA)	Nonessential
121	Nitrogen to PRT	Essential
121	Reactor Makeup water to PRT	Nonessential
121	Cont. Press. transmitter PT-945	Essential
121	Cont. Press. transmitter PT-946	Essential
123	Reactor Coolant Drain Tank (RCDT) to GA	Nonessential
124	Excess letdown supply and return to heat exchanger	Nonessential

Table 2.1.4 (continued)

PENT. NO.	IDENTIFICATION/DESCRIPTION	ESSENTIAL VS. NONESSENTIAL
124	Post Accident air sample "C" fan	Nonessential
125	Component Cooling Water (CCW) from 1B RCP	Essential
126	CCW from 1A RCP	Essential
127	CCW to 1A RCP	Essential
128	CCW to 1B RCP	Essential
129	RCDT & PRT to Vent Header	Nonessential
130	CCW to reactor support cooling	Nonessential
131	CCW to reactor support cooling	Nonessential
132	Depressurization at power	Nonessential
140	RHR pump suction from "A" Hot leg	Essential
141	RHR-#1 pump suction from Sump B	Essential
142	RHR-#2 pump suction from Sump B	Essential
143	RCDT pump suction	Nonessential
201	Reactor Compart. cooling Unit A & B	Essential
202	Hydrogen recombiner pilot & main "B"	Nonessential
203	Contain. Press. transmitter PT-947 & 948	Essential
203	Post accident air sample to "B" fan	Nonessential
204	Purge Supply Duct	Nonessential
205	Hot leg loop sample	Nonessential
206	Przr. liquid space sample	Nonessential
206	"A" S/G sample	Nonessential
207	Przr. Steam space sample	Nonessential
207	"B" S/G sample	Nonessential
209	Reactor Compart. cooling Unit A & B	Essential
210	Oxygen makeup to A & B recombiners	Nonessential
300	Purge Exhaust Duct	Nonessential

Table 2.1.4 (continued)

PENT. NO.	IDENTIFICATION/DESCRIPTION	ESSENTIAL VS. NONESSENTIAL
301	Aux. steam supply to containment	Nonessential
303	Aux. steam condensate return	Nonessential
304	Hydrogen recombiner pilot and main to "A"	Nonessential
305	Radiation Monitors R-11, R-12 & R-10A Auto Inlet Isol.	Nonessential
305	R-11, R-12 & R-10A Outlet	Nonessential
305	Post Accident air sample (containment)	Nonessential
308	Service Water to "A" fan cooler	Essential
309	leakage test depressurization	Nonessential
310	Service Air to Contain.	Nonessential
310	Instrument Air to Contain.	Nonessential
311	Service Water from "B" fan cooler	Essential
312	Service Water to "D" fan cooler	Essential
313	leakage test depressurization	Nonessential
315	Service Water from "C" fan cooler	Essential
316	Service Water to "B" fan cooler	Essential
317	leakage test supply	Nonessential
318	Dead weight tester	Nonessential
319	Service Water from "A" fan cooler	Essential
320	Service water to "C" fan cooler	Essential
321	A S/G Blowdown	Nonessential
322	B S/G Blowdown	Nonessential
323	Service Water from "D" fan cooler	Essential
324	Demineralized water to Containment	Nonessential
332	Cont. Press. Trans. PT-944, 949 & 950	Essential
332	Leakage test instrumentation lines	Nonessential

Table 2.1.4 (continued)

PENT. NO.	IDENTIFICATION/DESCRIPTION	ESSENTIAL VS. NONESSENTIAL
401	Main steam from A S/G	Nonessential*
402	Main steam from B S/G	Nonessential*
403	Feedwater line to A S/G	Essential, used for Auxiliary Feedwater
404	Feedwater line to B S/G	Essential, used for Auxiliary Feedwater
1000	Personnel Hatch	Nonessential
2000	Equipment Hatch	Nonessential

*Signals which cause main steam line isolation are shown on Westinghouse drawing 882D612 sheet 6.

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 Station 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

The presently available procedures which govern the use of hydrogen recombiners have been reviewed and found to be adequate. In view of the fact that our recombiners are inside containment, no further review is required to consider shielding requirements and personnel exposure limitations resulting from the recombiners. Access to the control panel will be considered in response to Section 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

The schedule for completion of the NRC staff requirements is given in our response of October 17, 1979 and is in agreement with staff requirements.

We will consider in our program for reducing leakage the NRC letter to RG&E dated October 17, 1979 regarding the North Anna and related incidents.

We are reviewing and discussing with consultants a satisfactory method for determining the leakage rate for the waste gas system. We intend to comply as required by January 1, 1980.

We will provide to the NRC by January 1, 1980, a summary description of our program to reduce leakage from systems outside containment that contain highly radioactive fluids during a serious transient or accident.

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 Plant 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: ≤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 shielding design review will be completed by January 1, 1980 which identifies the location of vital areas and equipment in which personnel occupancy may be limited or safety equipment unduly degraded by radiation fields during post-accident conditions. The review will use the source term and dose rate criteria of this position. The design review will determine the types of corrective actions needed for the affected vital areas.

We expect to implement plant shielding modifications and procedure changes which may be required by January 1, 1981 unless major modifications which are affected by Systematic Evaluation Program (SEP) topics are identified. There may be a selected number of modifications which we will recommend including in the integrated assessment of the SEP. This would be in those cases where the modification could potentially interact with reviews being conducted under SEP such as topics III-4.A, III-4.B, III-4.C, III-4.D (missiles), III-5.B (pipe break outside containment), III-6 (seismic), and VI-8 (control room habitability). Recommendations for inclusion in SEP will be reached on a case-by-case basis and will be presented to the NRC for concurrence.

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. Other Considerations

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

RG&E Responses

We meet the requirements of this position. Please see our response of October 17, 1979.

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

A.

   1. The auxiliary feedwater flow indication to each steam generator does not, by itself, satisfy the single failure criterion. However, as allowed by clarification C.1, redundant indication is provided by steam generator level.

2. The auxiliary feedwater flow indication channels are testable.

3. Auxiliary feedwater flow instrument channels are powered by vital instrument buses.

B.

1. Long term auxiliary feedwater flow indication to each steam generator which meets safety grade requirements will be installed by January 1, 1981.

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 samples 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 or 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.

<u>RG&E Response</u>

We are performing an operational and design review of the reactor coolant and containment atmosphere sampling systems to determine the improvements necessary for prompt collection, handling and

analysis of required post-accident samples without incurring excessive personnel exposure. Sampling procedure changes and minor sample collection modifications will be completed by January 1, 1980. The ongoing design review will consider those items identified in the Commission's October 30, 1979 clarification letter and will implement by January 1, 1981 the necessary major plant modifications with the possible exception of selected modifications that may be affected by SEP review. These may be incorporated into the Systematic Evaluation Program. (see response to 2.1.6.b)

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.b1

Noble gas effluent monitor with an upper range capacity of 10^5 $\mu Ci/cc$ (Xe-133) is being procured to be installed on the plant vent by January 1, 1981. In order to meet this date staff approval of our design is required by March 1, 1980. The additional information required by the staff position concerning our interim methods for quantifying high level releases will be submitted by January 1, 1980.

2.1.8.b.2

RG&E currently has the capability to monitor iodine gaseous releases. This equipment is in place and used for routine analysis. Prior to January 1, 1980 we will provide the information required by paragraph 2.A. of the clarifications above.

2.1.8.b.3

Two high range containment radiation monitors have been ordered for installation in the containment prior to January 1, 1981. In order to meet this date staff approval of our design is required by March 1, 1980. These monitors will meet the requirements of Table 2.1.8.b.3 and will have a range of 10^7 R/hr (photon only).

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

We have mobile instrumentation located in various areas throughout the plant to monitor airborne iodine concentrations. Portable air samplers are available in the Health Physics office and at the Emergency Survey Center for use in collecting iodine samples. Both charcoal and silver zeolite are available as iodine collectors.

We have the capability to accurately detect the presence of iodine using mobile air monitors which have a single channel analyzer calibrated to the I 131 energy. We also have a low background, low contamination counting facility where a sample can be purged of noble gases to assure accurate iodine measurements. Procedures are in use and Health Physics technicians are trained to use the GeLi detector in isotopic analysis. This procedure is used routinely to determine MPC hours of exposure.

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 were reported to the NRC in WCAP-9600, submitted by the Westinghouse Owners group on June 29, 1979. The sensitivity to reactor coolant pump trip was addressed in WCAP-9584 which was submitted on August 31, 1979. NRC approval of the procedure guidelines contained in these two reports, as amended, was transmitted by letter dated November 5, 1979 from Mr. D. F. Ross, Jr.

Analyses of inadequate core cooling was submitted by the Owners group on October 30, 1979.

As identified in a letter from Cordell Reed, Chairman, Westinghouse Owners Group, to Mr. D. F. Ross, Jr., dated October 29, 1979, we expect to submit the analysis of transients and accidents by January 1, 1980.

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

RG&E will install containment water level indication in conformance with the Staff position, except that the depth may not be greater than that equivalent to 500,000 gallons. The maximum capacity of the refueling water storage tank (338,000 gal.) plus the reactor coolant system (approximately 50,000 gal.) plus the accumulators (less than 10,000 gal. each) is considerably less than 500,000 gallons.

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 systems boundary, efforts should be made to minimize the probability of an inadvertent actuation of the system. Removing power from the vents is one step in the direction. Other steps are also encouraged.

10. Since the generation of large quantities of 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.

<u>RG&E Response</u>

We are planning to install reactor coolant system and reactor vessel head high point vents in accordance with the staff's requirements. We intend to submit the design details by January 1, 1980; and have the venting system installed by January 1, 1981 as required. In order to meet the installation date staff approval of our design will be required by March 1, 1980.

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

We will comply with the staff position and clarifications. See our response of October 17, 1979.

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

See our response of October 17, 1979. Additional information concerning our training program will be provided by January 1, 1980.

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

We will comply with the staff position as described in our response of October 17, 1979.

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

We will comply with the staff position as described in our response of October 17, 1979.

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

1. The following tasks will be accomplished by January 1, 1980. Those items describing design of the facility apply to the interim TSC. Our intent is to provide the equipment as described, although changes in design may be required as the design is finalized.

   A. An interim TSC will be established and a description provided.

   B. Plans and procedures for support and staffing of the TSC will be complete. The TSC will be manned by designated personnel.

   C. Communications between the TSC and the control room, site emergency operations center and the NRC will be established. A tie to the TSC with the existing direct line from the control room to the NRC will be installed. A hardwired

[illegible]

intercom system with a master unit in the TSC and slave units in the control room, emergency center, alternate emergency center and the operations center will be installed. The TSC will also have phone communications and portable radios.

D. A radiation monitor will be provided in the TSC. In addition portable airborne monitors are already available in the plant. Guidelines will be established as an aid for qualified personnel to decide when protective measures should be taken.

E. To meet the short term requirements plant data will be made available in the TSC by means of a data link terminal. A video system capable of scanning the control board is being investigated.

F. Procedures to use the control room as a backup TSC will be developed.

G. Plans for our permanent TSC will be provided.

Design details given in items 2 through 10 apply to the permanent TSC. Our intent is to provide a facility with the features described below although changes may be made as the design progresses. As noted in 1.G above, a description of our longer range plans will be submitted by January 1, 1980.

2. The TSC will be on site with a means for gaining access to the control room.

3. Procedures are being prepared to direct staff to the TSC for emergencies of varying severity. The design of the permanent TSC is not complete but it will be capable of housing approximately 25 people and the equipment necessary to assess the emergency situation.

4. Guidelines for activation of the TSC are being prepared which are based upon specific plant conditions. The guidelines are not inconsistent with the "Alert" level as defined in NUREG-0610. Vital plant parameters for the period from the beginning of the event until activation of the TSC will be recoverable in the TSC.

5. Instrumentation to be located in the TSC will be of good quality and will be capable of displaying information independent from actions in the control room. The control room and other plant functions will not be degraded by the TSC.

6. An uninterruptable power supply will be provided for TSC instrumentation which is independent from existing emergency power supplies.

7. Technical data requirements for the TSC will be established which permit assessment of the plant safety systems and in-plant and offsite radiological conditions listed above.

8. The technology best suited to meet the requirements for the TSC will be implemented, however, RG&E always tries to remain flexible to meet ever changing requirements.

9. The TSC will be designed and built in accordance with sound engineering practice.

10. A. We will provide appropriate radiological protection for the technical support center personnel so that dose limitations specified in GDC 19 and SRP 6.4 will not be exceeded. We expect to fulfill this objective utilizing a combination of an installed HVAC system, radiation shielding and administrative dose control measures.

B. Permanent monitoring systems and procedures will be provided to meet this requirement.

C. A ventilation system which meets this requirement will be installed.

D. Breathing apparatus and potassium iodide will be available but it will not be used as a design basis for the TSC.

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

We will comply with the staff position. See our response of October 17, 1979.