TO: PAT MARRIOTT

PAT, PLEASE MAIL TO THE OWNERS GROUP

VERMONT YANKEE NUCLEAR POWER CORPORATION

SEVENTY SEVEN GROVE STREET

RUTLAND, VERMONT 05701

October 17, 1979

Mr. Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, Maryland 20014

Subject: BWR Owners Group Positions on NUREG-0578

Enclosure: GE BWR Owners Group NUREG-0578 Positions Submittal Schedule

Dear Mr. Eisenhut:

As previously discussed with you and members of your organization, the General Electric Boiling Water Reactor Owners Group has developed a number of detailed technical positions for NUREG-0578 recommendations as they apply to Boiling Water Reactors. The purpose of this letter is to advise you of the status of preparation of these positions and to provide you with a schedule for submittal of those documents, as indicated in the enclosure to this letter. The headings of the various columns in the enclosure are self-explanatory and do not require discussion.

The positions being submitted to you represent a consensus of the operating plant members of our Owners Group and it is anticipated that member companies will be utilizing these standards being mailed to you as part of their commitment to implement the recommendations of NUREG-0578. In order to streamline processing, typing and reproduction, the General Electric Company has acted as the focal point for submittal of the documents on our behalf. Consequently the positions as indicated in the enclosure will be mailed directly to your office from General Electric Company. Submittal of these positions by the Owners Group does not constitute a commitment by or for any individual utility. These commitments will be made in each individual utility implementation letter.

For those items requiring a January 1, 1980, implementation date it is necessary that we have benefit of any comments from your organization on those Owners Group positions which in your judgement require modification. Because of this short time frame it is necessary that your comments be provided to us within two weeks from date of receipt of these position documents. If we do not hear from your organization within that time period we will assume that the positions provided are acceptable and will proceed with implementation as indicated by those positions.

We believe the positions being taken by our Owners Group are responsive to the intent of NUREG-0578 for Boiling Water Reactors and trust you will concur with

dupe of 7910230474

our position on these matters. If you have any questions or concerns on the positions being forwarded to you, please call the working chairman of the NUREG Implementation Task Force within our Owners Group, Mr. Joe Bynum of the Tennessee Valley Authority (615 755-3635). If you are unable to contact Mr. Bynum please advise me of your questions or concerns and I will insure that a representative of our Owners Group contacts you with specific responses to those concerns.

Sincerely yours,

Thomas D. Keenan, Chairman
General Electric Boiling Water
Reactor Owners Group

TDK/cms

GE BWR OWNERS GROUP NUREG-0578 POSITIONS SUBMITTAL SCHEDULE

NUREG Item	Subject	NRC Deferral	Submitted to NRC on 10/17	N/A to BWRs	To be Provided to NRC by 11/15	Individu Compan Positio
2.1.1	Emergency Power Supply		X			
2.1.2	Relief and Safety Valve Test		X			
2.1.3.a	Direct Indication of Valve Position		X			
2.1.3.b	Instrumentation for Inadequate Core Cooling		X			
2.1.4	Containment Isolation		X			
2.1.5.a	Dedicated H_2 Control Penetrations					X
2.1.5.b	Inerting BWR Containments	X				
2.1.5.c	Recombiner Procedures					X
2.1.6.a	Systems Integrity for High Radioactivity				X	
2.1.6.b	Plant Shielding Review				X	
2.1.7.a	Auto Initiation of Auxiliary Feedwater			X		
2.1.7.b	Auxiliary Feedwater Flow Indication			X		
2.1.8.a	Post-Accident Sampling				X	
2.1.8.b	High Range Radiation Monitors		X			
2.1.8.c	Improved Iodine Instrumentation		X			
2.1.9	Transient and Accident Analysis		X			
2.2.1.a	Shift Supervisor Responsibilities		X			
2.2.1.b	Shift Technical Advisor		X			
2.2.1.c	Shift Turnover Procedures		X			
2.2.2.a	Control Room Access		X			
2.2.2.b	On Site Technical Support Center		X			
2.2.2.c	On Site Operational Support Center		X			
2.2.3	Revised Limiting Conditions for Operation	X				
	Containment Pressure		X			
	Containment Water Level		X			
	Containment H_2 Concentration		X			
	High Point Venting		X			

NUREG-0578 AND IMPLEMENTATION LETTER REQUIREMENTS

BWR OWNERS' GROUP IMPLEMENTATION CRITERIA

Prepared by:

BWR OWNERS' NUREG-0578 IMPLEMENTATION SUBGROUP

October, 1979

FOREWORD

This report contains generic BWR implementation criteria for the short-term requirements of the USNRC Lessons Learned Task Force (Ref. 1), as modified by the enclosures to the implementation letter to operating plants (Ref. 2), and by the NRC interpretation presented at the Inspection and Enforcement Regional Meetings (Ref. 3). These criteria also include interpretations drawing from formal and informal discussions with the NRC staff, notably from a September 20, 1979, meeting to discuss generic LWR issues relative to NUREG-0578 implementation, and from Topical Meetings on selected issues on October 10-12, 1979.

These criteria were developed by the BWR Owners' Group NUREG-0578 Implementation Subgroup for use by individual utilities in preparing their own NUREG-0578 implementation commitments as required by Reference 2. This document does not constitute or imply a commitment by any individual utility to the criteria: such commitments will be made by each utility individually, and there will necessarily be plant-specific differences.

Each implementation criterion is prefaced by a statement of the NRC position and a discussion of the BWR Owners' Group position. For compactness, the NRC positions from the main text of NUREG-0578 are cited: the Owners' Group has, however, used the more detailed positions in Appendix A of NUREG-0578, plus References 2 and 3, for definitive guidance.

The following NUREG-0578 requirements are not part of this submittal for the reasons stated:

- 2.1.5A: Criteria will be plant-unique
- 2.1.5B: Deferred for additional study by USNRC
- 2.1.5C: Criteria will be plant-unique
- 2.1.7A: Applies only to PWRs
- 2.1.7B: Applies only to PWRs
- 2.2.3 : Deferred for additional study by USNRC

Requirement 2.1.1, although intended to apply strictly to PWRs, has been examined for its applicability to BWRs and is included in this submittal.

FOREWORD (p.2)

References:

1. USNRC, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," NUREG-0578, July, 1979.

2. Letter, D. G. Eisenhut to All Operating Nuclear Power Plants, "Followup Actions Resulting From the NRC Staff Reviews Regarding the Three Mile Island Unit 2 Accident," September 13, 1979.

3. USNRC, "Regional Meetings - TMI Short-Term Implementation Action," handout material at Inspection and Enforcement Regional Meetings, September 24-28, 1979.

NUREG-0578 Requirement 2.1.1: "Emergency Power Supply Requirements For the Pressurizer Heaters, Power-Operated Relief Valves, and Pressurizer Level Indicators in PWRs"

Provide redundant emergency power for the minimum number of pressurizer heaters required to maintain natural circulation conditions in the event of loss of offsite power. Also provide emergency power to the control and motive power systems for the power-operated relief valves and associated block valves and to the pressurizer level indication instrument channels.

Discussion:

As discussed in NEDO-24708, natural circulation in the BWR is strong and inherent in all off-normal modes of operation, independent of any powered system, as long as sufficient inventory is maintained. This is because even in normal operation the BWR is essentially an augmented natural circulation machine. Because the BWR operates in all modes with both liquid and steam in the reactor pressure vessel, saturation conditions are always maintained irrespective of system pressure (the BWR does not have a pressurizer). Thus there is no need for emergency power to maintain natural circulation or to keep the system pressurized.

The power-operated relief valves in BWR's are already powered by emergency power. They have no block valves.

The reactor vessel level indication instrument channels for safety system activation and control are already powered by emergency power.

BWR Owners' Group Implementation Criteria:

For the reasons stated above, there is no need for action in response to Recommendation 2.1.1 for any General Electric BWR.

NUREG-0578 Requirement 2.1.2: "Performance Testing for BWR and PWR Relief and Safety Valves"

Commit to provide performance verification by full scale prototypical testing for all relief and safety valves. Test conditions shall include two-phase slug flow and subcooled liquid flow calculated to occur for design-basis transients and accidents.

Discussion:

The BWR design basis includes no transients or accidents in which two-phase flow or subcooled liquid flow at high pressure through relief, safety/relief, or safety valves is calculated or expected*. The BWR therefore satisfies the intent of the requirement in the strictest sense. The need for performance verification, however, has been studied in a broader sense. The remainder of this discussion is intended to demonstrate that performance verification in the field fully satisfies the broad intent of the requirement.

In determining the need for special testing of BWR safety and relief valves it is essential to consider the service duty to which the primary system relief and safety valves of the BWR are exposed, and the consequences of maloperation of these valves. Relief valves are routinely used to mitigate the effects of system transients. A stuck-open valve is not an event of great significance in a BWR: in 300 reactor years of experience, 54 cases have occurred. Tables 2.1.2-1 and 2.1.2-2 summarize the experience to date. This experience, as will be explained, clearly shows that there is no need for an extensive testing program for BWR safety and relief valves.

A. BWR Safety and Relief Valves

Table 2.1-3 of NEDO-24708 shows the complement of safety and relief valves for all domestic operating BWR's. Most BWR's have relief valves or dual-function safety/relief valves (S/RV), the discharges of which are piped to the suppression pool. Spring safety valves discharge directly to the drywell (or the containment in a dry containment), except for Humboldt Bay, in which the safety valves discharge to the suppression pool.

B. Valve Usage

(1) Relief valves and dual-function S/RV's in BWR/2-6. The relief valves and dual-function S/RV's are designed to routinely mitigate the effect of system transients. Their discharges are piped to the containment suppression pool. This massive heat sink prevents significant containment heatup. Complication of a system transient by a stuck-open valve has essentially no effect on reactor vessel water level measurement or on forced or natural circulation capability. The flow through the valve is saturated steam. If the valve cannot be closed by operator action the plant can be shut down using familiar and uncomplicated procedures.

*Liquid flow is expected as an alternate shutdown mode in some units. This flow, however, is controlled and occurs at low pressure.

2.1.2 (p. 2)

B. (2) Spring safety valves in BWR/2-4. The safety valve set-point is sufficiently higher than the relief valve set-point that the safety valves are almost never required to operate (Table 2.1.2-3 documents the three cases in which safety valves have ever listed in BWR operation). Should a safety valve inadvertently lift, which has never happened in BWR operation, the effect is the same as a small steam line break inside containment. Even in this remote event, the flow through the valves will be saturated steam at all times.

(3) Dresden 1. For pressurization events, such as a turbine trip, the two relief valves, which are located downstream of the main steam isolation valves (MSIV), are sized to relieve pressure directly to the main condenser without requiring safety valve action. In the event of MSIV closure, reactor scram is initiated from MSIV position switches, which also initiate the redundant isolation condensers. Even one isolation condenser will limit reactor pressure to well below the safety valve set-point. The results of a stuck-open safety valve would be as described in (2) above.

(4) Big Rock Point. An isolation condenser is provided, containing redundant cooling loops either one of which (when automatically actuated at 1450 psig) keeps reactor pressure below the spring safety valve set-point. The results of a stuck-open safety valve would be as described in (2) above.

(5) Humboldt Bay. All spring safety valves are piped to the containment suppression pool. Therefore, the results of a stuck-open valve would be as described in (1) above.

C. Two-Phase Flow.

Expected operating conditions and transients do not include two-phase flow through S/RV's, safety, or relief valves. However, on three occasions, circumstances combined to cause high pressure water to flow down the steamlines and a steam/water mixture to flow through the valves. A summary of these events is given in Table 2.1.2-3. In these events, Electromatic relief valves and direct acting safety valves were actuated, discharged a steam/water mixture and reclosed, indicating that the flow media did not cause a stuck-open valve condition. These events did not lead to any concern over adequate core cooling. However, following these events, high water level trips were added to all new BWR's and retrofitted to most of the BWR's in operation.

D. Valve Qualification.

(1) Crosby, Dikkers, Okano and two-stage Target Rock S/RV's are tested for the expected saturated steam flow conditions. This includes life-cycle testing of 300 actuations as well as environmental qualifications including seismic, thermal, mechanical and radiation effects.

(2) Three-stage Target Rock S/RV's were subjected to restricted flow steam tests to qualify the set-point and valve opening time delay. Solenoid valves (used during power actuation) are qualified by autoclave test for the LOCA environment. Satisfactory valve operation has been demonstrated by field service.

(3) Dresser Electromatic relief valve solenoids were qualified by autoclave test for the LOCA environment. Satisfactory valve operation has been demonstrated by field service.

(4) Satisfactory operation of Dresser safety valves has been demonstrated by field service.

E. Field Experience

Since 1971 there have been 50 events in BWR plant operation wherein S/RV's have stuck open (Table 2.1.2-1). In each of these cases the reactor was depressurized, the stuck valve was repaired or replaced, and the plant was placed back into service.

Although a stuck-open S/RV is of no significant safety concern in the BWR, programs are underway to reduce the frequency of such events. From Table 2.1.2-1 it is seen that the total number of S/RV blowdowns has steadily decreased since the mid-70's. The improvement in the number of S/RV blowdowns as a factor of number of S/RV's in service has been even more dramatic.

From Table 2.1.2-2 it is seen that experience with spring safety valves and Electromatic relief valves has always been good: there have been only four blowdowns.

F. Summary

(1) BWR S/RV's are routinely tested for the only expected mode of operation (saturated steam), both by in-place functional tests and by frequent usage in mitigating plant transients;

(2) There is no design-basis transient or accident which requires safety, relief, or dual function S/RV's to pass two-phase or liquid flow at high pressure;

(3) Inadvertent passage of two-phase flow is not likely where high pressure feedwater and injection systems are tripped by high vessel water level.

2.1.2 (p. 4)

F. (4) In the three events wherein BWR S/RV's did pass two-phase flow, the valves reclosed;

(5) Spring safety valves are almost never required to open; in the even less likely event that one should stick open, the effect is identical to that of a small steam line break. There is no concern for adequate core cooling.

(6) Electromatic relief valves and dual-function S/RV's are frequently called on to operate, and dual-function S/RV's occasionally stick open. The consequences of a stuck-open valve are minimal and reactor shutdown is uncomplicated, as proven by numerous field occurrences. In some BWR's the procedures for responding to a stuck-open relief valve includes the opening of additional relief valves. Improvement programs are reducing the frequency of such events.

BWR Owners' Group Implementation Criteria:

It is concluded that concerns regarding safety/relief valve performance have been addressed and no special performance testing is required provided that the following criteria are met:

(1) A procedure shall exist for responding to a stuck-open relief, safety/relief, or safety valve.

(2) The procedures shall address prevention of inadvertent overfilling of the reactor vessel.

(3) Control grade systems, actuated by reactor vessel high water level, shall be provided to prevent the feedwater and high pressure injection systems from overfilling the vessel (or steam drum, on plants so equipped).

(4) S/RV performance in "Alternate shutdown" mode will be specifically addressed, and a justification of why a test is not needed will be provided.

TABLE 2.1.2-1

S/RV BLOWDOWNS IN BWR OPERATION

YEAR	3-STAGE TARGET ROCK			2-STAGE TARGET ROCK		CROSBY-OKANO-DIKKERS				
	TOTAL BLOWDOWNS	STUCK OPEN FOLLOWING DEMAND	# OF VALVES IN SERVICE	TOTAL BLOWDOWNS	# OF VALVES IN SERVICE	TOTAL BLOWDOWNS	# OF VALVES IN SERVICE	TOTAL S/RV BLOW-DOWNS	TOTAL S/RVs IN SERVICE	TOTAL BLOWDOWNS DIVIDED BY TOTAL VALVES IN SERVICE
1971	2	2	14					2	14	0.15
1972	1	1	23					1	23	0.04
1973	1	1	56					1	56	0.02
1974	10	1	108					10	108	0.09
1975	7	0	127					7	127	0.06
1976	11	1	149					11	149	0.07
1977	9	4	157					9	157	0.06
1978	5	3	157	0	11	0	35	5	203	0.02
1979 to Sept.	4	1	132	0	36	0	52	4	220	0.02

NOTE: The above table does not include Dresser Safety Valves (unpiped discharge) or "Electromatic" relief valves. See Table 2.1.2-2 for information on this equipment.

TABLE 2.1.2-2

SAFETY AND ELECTROMATIC RELIEF VALVE

BLOWDOWNS IN BWR OPERATION

A. Safety Valves.

Only one event has ever occured with partially stuck open valves - the Dresden 2 event described in Table 2.1.2-3. The lifting levers which cocked the valves partially open were subsequently removed from safety valves at all plants and there have been no further occurrences. There have only been three occurrences in which safety valves have ever lifted during operation (see Table 2.1.2-3). The total number of valves in service is 76[1].

B. Electromatic Relief Valves.

There have been three occurrences of a stuck open Electromatic relief valve, two of which followed a demand. The number of valves in service is 37.[1]

[1] Some BWR's are in the process of replacing safety valves and Electromatic relief valves with Target Rock S/RV's.

TABLE 2.1.2-3

EVENTS IN WHICH TWO-PHASE FLOW OR

LIQUID PASSED THROUGH BWR SAFETY OR RELIEF VALVES

DRESDEN 2 - JUNE 5, 1970

During the course of the initial test program on Dresden 2 with the unit operating at 75% power, a spurious signal in the reactor pressure control system occurred. This spurious signal resulted in simultaneous opening of the control and the turbine bypass valves with resultant turbine trip, reactor scram, and main steamline isolation.

In response to the initial and expected water level drop, the operator switched to manual control of the feedwater system and began filling the reactor vessel at the maximum rate. Water level misinterpretation led to reactor water overflowing into the main steam lines. A pressure surge resulted in the main steam lines when relief valves were cycled. This momentarily opened one of the safety valves, resulting in a discharge directly to the containment (unpiped discharge). The fluid impinged upon the lifting levers of two other safety valves causing these safety valves to cock slightly open. The water-steam mixture from the two safety valves pressurized the primary containment to an estimated 20 psig and an estimated temperature of approximately 300°F. At no time during the event was there difficulty maintaining adequate water supply to the reactor core, and there was no question of adequate core cooling.

DRESDEN 3 - DECEMBER 8, 1971

Unit 3 was operating about 98% power on December 8, 1971, when the plant was shut down due to a reactor low water level scram. The scram resulted from a condensate/condensate booster pump trip and the subsequent trip of two reactor feed pumps on low suction pressure. Following the scram, the standby feed pump started. The vessel was overfilled and the steam lines flooded. Due to a pressure surge in the main steam lines, a safety valve lifted causing discharge directly to the containment (unpiped discharge). The containment was pressurized to approximately 20 psig. At no time during the event was there difficulty maintaining adequate water supply to the reactor core, and there was no question of adequate core cooling.

KRB (GERMANY) - JANUARY 13, 1977

The unit was operating at 100% power when a bus on two of its 200 KV lines opened. The plant was scrammed and isolated. Manual feedwater

TABLE 2.1.2-3 (cont'd)

control was initiated which resulted in flooding of the steam lines. Safety valves opened and discharged water, steam and two-phase media. The valves discharged directly to the containment (unpiped discharge). The safety valves opened and reclosed several times. Because of the unique piping arrangement (which is not present in any US-BWR), reaction forces of the discharging valves caused or contributed to a pipe rupture in two of the fourteen flanged nozzles by which the valves are connected to a U-shaped header. At no time during the event was there difficulty maintaining adequate water supply to the reactor core, and there was no question of adequate core cooling.

NUREG-0578 Requirement 2.1.3A: "Direct Indication of Power-Operated Relief Valve and Safety Valve Position for PWR's and BWR's"

Provide in the control room either a reliable, direct position indication for the valves or a reliable flow indication device downstream of the valves.

Discussion:

BWR safety and relief valves are arranged in three ways in the various operating reactors:

1) Valve discharges piped to the containment suppression pool;

2) Valve discharges manifolded and piped to suppression pool;

3) Discharging directly to the drywell free volume, in pressure suppression containments, or to the containment free volume in dry containments.

The configuration of the valve discharge, and the operator's ability to diagnose and act on stuck-open valve events, will determine what information is to be provided in the control room. The environment experienced by the installed instrumentation during a stuck-open valve event will determine the proper qualification requirements.

A. Valve Discharges Individually Piped to the Suppression Pool.

All dual-function safety/relief valves and most relief valves are configured this way. Given a stuck-open valve, the containment pressure will not increase because of the submerged discharge. There is benefit in direct indication, not only because the operator would be given an early warning of S/RV discharge, but because he can attempt to reseat a stuck-open valve from the control room. Most such valves have no external stem, which precludes direct position indication.

B. Valve Discharges Manifolded and Piped to the Suppression Pool.

Relief valves in one domestic BWR are configured this way. The system response and operator action considerations are identical to those for valves piped individually to the suppression pool. A pressure switch in each discharge manifold will give a positive indication that one of the two or three valves on each manifold is open or closed. The existing temperature sensors in the discharge of each valve will then be sufficient to determine exactly which valve has actuated, so that the operator can attempt to reseat it.

C. Valves Discharge Directly to Containment or Drywell.

All spring safety valves are configured this way. Because these valves are large (200,000 to 940,000 lbm/hr capacity per valve) compared to the containment free volume, a stuck-open valve will cause a rapid rise in containment pressure, causing almost immediate scram and ECCS operation in most units and an unambiguous indication of loss of coolant in all. Because the valves discharge steam from the steam drum or main steam lines into the containment, the effect of a stuck-open valve is identical to a small main steam line break. Because the operator has no capability of attempting to re-seat a stuck-open safety valve from the control room, his actions would be identical to those for a main steam line break (that is, whether the "LOCA" is due to a stuck-open valve or due to a pipe break is of no interest in operator action). Spring safety valves almost never open in BWR's, but even if one were to open and remain open, two-phase flow would not be expected, as shown in 54 events of stuck-open relief valves of similar capacity in operating BWR's (see Owners' Group position on NUREG-0578 Requirement 2.1.2). High reactor water level trips preclude water in the steam lines in most units, and operators are sensitive to the undesirability of overfilling the reactor vessel or steam drum in all units. Thus there is no need for special precautions due to the possibility of two-phase flow in the valves. Even if the valve were to reseat, the operator's action would be no different than for a small steam line break (maintain reactor water level, depressurize the reactor, cool the suppression pool). For all of these reasons, the existing high drywell or containment pressure instrumentation provides all the information the operator can use in analyzing and acting on a stuck-open spring safety valve. Existing instrumentation is therefore a sufficiently "reliable flow indication device" for spring safety valves.

BWR Owners' Group Implementation Criteria:

A. Valve Discharges Individually Piped to the Suppression Pool.

The Owners' Group considers two types of monitoring to be acceptable methods of positive valve indication: pressure switches in the valve discharge lines and acoustic monitors. A suitable pressure switch system is outlined in the Appendix, in response to an NRC request in the September 24, 1979, Region I meeting.

Either type of system will be designed to the following broad requirements:

(1) There will be at least one sensing device per discharge line;

(2) Sensing devices may be either inside or outside the drywell;

(3) Sensing devices and other components need not be qualified for a LOCA (pipe break) environment, but only for the environment expected during S/RV discharge to the suppression pool;

(4) All components will be seismically qualified;

(5) The system will be powered by one division of emergency power;

(6) With sensing devices inside the drywell, non-class-IE electrical penetrations may be used if insufficient IE penetrations are available.

B. <u>Valve Discharges Manifolded and Piped to the Suppression Pool:</u>

A single pressure switch in each manifold, otherwise conforming to "A" criteria above, will be provided.

C. <u>Valves Discharge Directly to Containment or Drywell.</u>

No further action is necessary based on the discussion above.

APPENDIX to OWNERS' GROUP POSITION 2.1.3A

USE OF DISCHARGE PIPING PRESSURE SWITCHES

FOR

S/R VALVE POSITION MONITORING

Main stem position is not accessible on the manufacturer's designs of safety/relief valves in BWR service. General Electric has evaluated several concepts including magnetic or proximity switches, acoustic devices, temperature, and pressure switches.

The use of pressure switches on the discharge lines has been selected as the most simple, direct and proven technique for monitoring valve position. The Safety/Relief valve discharge is piped to the torus, discharging below the water line. Pressure near the valve discharge can be straightforwardly calculated and tested; it is in the range of 250 psig when the reactor is at rated pressure. This pressure is sufficiently high that a positive and unambiguous signal is available with ample margin for tolerances in set point calibration. When the valve re-closes, pressure returns to normal in a fraction of a second. Thus a pressure signal does not have the slow response time which characterizes temperature monitoring.

Test data are available confirming the transient and steady-state response of S/RV discharge line pressure. These data were obtained during extensive in-plant measurements of suppression-pool loading resulting from safety relief valve actuations. These test data confirm the analytical basis for selection of set points.

Pressure switches are available in industry which are suitable for this service. Similar devices are used routinely for the protection of plant and equipment. Plant personnel will be familiar with the calibration, testing, and maintenance of these devices. No development testing is required to prove a satisfactory device, other than qualification tests which would be required for any device.

With the use of pressure switches, no device is mounted on or near the safety/relief valve. The technique will work for all types of piped BWR safety/relief valves in service. It will have no effect on valve performance. The pressure switches may be located at some distance from the safety/relief valve where they will not be subjected to severe temperature or vibration conditions. Where suitable piping penetrations are available it is possible to locate the switches outside the drywell.

The pressure switches will be qualified for a 212°F, 100% humidity environment. This is adequate for the intended service even if the pressure switches are inside the drywell because actuation of the S/R valves, inadvertent or planned, will not cause these environmental

conditions to be exceeded. In the event of a small pipe break the safety/relief valves in the ADS system would be initiated early in the transient, before degradation of the switches could have occurred. In the event of a large pipe break the safety/relief valves are not required to operate. No failure mode has been identified that would result in an erroneous indication that the valve was open.

The signals from pressure switches may be interfaced with indicating lights, control room annunciators, an event counter, or the process computer. Any one or all of these functions may be implemented. Each safety/relief valve can be monitored independently of the other valves.

NUREG-0578 Requirement 2.1.3B: "Instrumentation for Detection of Inadequate Core Cooling in PWR's and BWR's."

Perform analyses and implement procedures and training for prompt recognition of low reactor coolant level and inadequate core cooling using existing reactor instrumentation (flow, temperature, power, etc.) or short-term modifications of existing instruments. Describe further measures and provide supporting analyses that will yield more direct indication of low reactor coolant level and inadequate core cooling such as reactor vessel water level instrumentation.

Discussion:

Additional hardware to identify inadequate core cooling on BWR's has not been determined to be necessary at this time. Licensees' procedures will identify the diverse methods of determining inadequate core cooling, using existing instrumentation. The results of analysis being performed in response to 2.1.9 will be factored into procedures as required, after the analysis is complete.

Because the BWR operates in all modes with both liquid and steam in the reactor pressure vessel, saturation conditions are always maintained irrespective of system pressure. This there is no need for a subcooling meter in the BWR.

BWR Owners' Group Implementation Criteria:

1) Analyses and operator guidelines for the detection and mitigation of inadequate core cooling are currently being developed per Requirement 2.1.9 and questions from the Bulletins and Orders Task Force. These studies include an evaluation of currently installed reactor vessel water level instrumentation, and the possible use of other instrumentation, to detect inadequate core cooling. The need for futher measures, if any, will be addressed after these analyses and operator guidelines are complete. Implementation of emergency procedures and retraining will be done on a schedule consistent with those established with the Bulletins and Orders Task Force.

2) A subcooling meter, as required by Enclosure 6 of the NUREG-0578 implementation letter of September 13, 1979, will not be provided.

NUREG-0578 Requirment 2.1.4: "Containment Isolation Provisions for PWR's and BWR's."

Provide containment isolation on diverse signals in conformance with Section 6.2.4 of the Standard Review Plan, review isolation provisions for non-essential systems and revise as necessary, and modify containment isolation designs as necessary to eliminate the potential for inadvertent reopening upon reset of the isolation signal.

Discussion:

There is diversity in the parameters sensed for the initiation of BWR containment isolation. Following an isolation, deliberate operator action is required to open valves in most cases.

BWR Owners' Group Implementation Criteria:

1) Diversity of parameters sensed for the initiation of containment isolation shall be provided in accordance with SRP 6.2.4.

2) A review shall be made of all systems penetrating primary containment to identify all essential systems. The basis of such classification shall be documented and supplied to the NRC.

3) All systems not identified as essential will be reviewed. If automatic isolation is not provided, justification for not isolating will be presented to the NRC.

4) Licensees will review and modify isolation control systems and administrative controls, as appropriate, such that no isolation valve will open when the isolation logic is reset. Those plants that have valves that will automatically open when the isolation logic is reset, will change the isolation logic to prevent the valves from opening when reset. Administrative controls to prevent valves from reopening will be implemented by 1/1/80; logic modifications will be implemented by 1/1/81.

NUREG-0578 Requirement 2.1.8b: "Increased Range of Radiation Monitors"

Provide high range radiation monitors for noble gases in plant effluent lines and a high-range radiation monitor in the containment. Provide instrumentation for monitoring effluent release lines capable of measuring and identifying radioiodine and particulate radioactive effluents under accident conditions.

Discussion:

The Owners' Group recognizes and concurs with the position as modified in the NRC regional meetings the week of September 24, 1979.

BWR Owners' Group Implementation Criteria:

1) The Owners will implement the requirements of position 2.1.8b, items 1, 2, and 3, consistent with commercial availability of equipment.

2) Procedures will be developed to estimate noble gas and radioiodine release rates if the existing effluent instrumentation goes off scale.

NUREG-0578 Requirement 2.1.8c: "Improved In-Plant Iodine Instrumentation."

Provide instrumentation for accurately determining in-plant airborne radioiodine concentrations to minimize the need for unnecessary use of respiratory protection equipment.

Discussion:

The Owners' Group recognizes and concurs with the position.

BWR Owners' Group Implementation Criteria:

1) The Owners will implement the requirements of position 2.1.8c.

2) Procedures will be developed to accurately determine in-plant iodine concentrations.

NUREG-0578 Requirement 2.1.9: "Analysis of Design and Off-Normal Transients and Accidents"

a. Provide the analysis, emergency procedures, and training to substantially improve operator performance during a small break loss-of-coolant accident.

b. Provide the analysis, emergency procedures, and training needed to assure that the reactor operator can recognize and respond to conditions of inadequate core cooling.

c. Provide the analysis, emergency procedures, and training to substantially improve operator performance during transients and accidents, including events that are caused or worsened by inappropriate operator actions.

Discussion:

The specific requirements and schedules are being developed in a continuing series of meetings between the utility owners' groups and the NRC Bulletins and Orders Task Force.

BWR Owners' Group Implementation Criteria:

The implementation of emergency procedures and retraining will be done on a schedule consistent with those established with the Bulletins and Orders Task Force.

NUREG-0578 Requirement 2.2.1.a: "Shift Supervisor's Responsibilities"

Review plant administrative and management procedures. Revise as necessary to assure that reactor operations command and control responsibilities and authority are properly defined. Corporate management shall revise and promptly issue an operations policy directive that emphasizes the duties, responsibilities, and authority and lines of command of the control room operators, the shift technical advisor, and the person responsible for reactor operations command in the control room (i.e., the senior reactor operator).

Discussion:

The Owners' Group agrees with the intent of the staff's position. However, in order to remove any ambiguity from the meaning of the term "accident situation" in item 2.b of the staff's position in Appendix A of NUREG-0578,* the entire sentence will be interpreted as follows: The shift supervisor (or equivalent, such as the supervising control operator in some plants), until properly relieved, shall remain in the control room at all times whenever a site or general emergency has been declared to direct the activities of control room operators.

BWR Owners' Group Implementation Criteria:

The staff's position will be implemented as stated and subject to the interpretation of item 2.b as discussed above.

*2.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.

NUREG-0578 Requirement 2.2.1b: "Shift Technical Advisor"

Provide on shift at each nuclear power plant a qualified person (the shift technical advisor) with a bachelor's degree or equivalent in a science or engineering discipline and with specific training in the plant response to off-normal events and in accident analysis of the plant.

Shift technical advisors shall serve in an advisory capacity to shift supervisors. The licensee shall assign normal duties to the shift technical advisor that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

Discussion:

Implementation of the Shift Technical Advisor (STA) as proposed by the Task Force would place a graduate engineer independent and detached from plant operations in the control room at or shortly following the occurrence of an accident or abnormal transient. Because the STA would not be in the direct operational chain of command and, in fact, would not need to be licensed, he could neither manipulate nor direct licensed operators to manipulate the controls of the reactor plant. He would be empowered to advise operations but not responsible to operations for his advice.

The shift supervisor is correctly charged with the responsibility for safe operation of the plant at all times. During the early phase of an accident, he discharges this responsibility by coordinating and directing the response of the control room staff. The actions of the operators are procedural, being governed by their training and emergency procedures, and during this phase the entire control room staff including the shift supervisor is completely occupied with responding to the accident. Plant operating experience indicates that there is a period of time following initiation of any accident or transient wherein the shift supervisor has sufficient time to analyze, diagnose, and respond to the condition of the plant but does not have sufficient time to carefully consider an independent assessment of the accident, resolve any conflicts between his and the independent assessment and, on the basis of such assessment, decide to alter the procedural actions of the operators. Dialogue regarding such an assessment or time spent resolving such conflicts can only distract and delay the shift supervisor and consequently degrade the response of the control room staff to the accident.

Even though the roles of shift supervisor and STA can be carefully delineated by procedure and training, industrial and military experience indicates that a direct-line organization wherein authority and responsibility are interdependent is required to effectively operate in a crisis environment. The proposed STA is empowered to advise operations but not responsible to operations for his advice. His authority and responsibility are not interdependent. A potential for conflict and confusion exists which cannot be completely eliminated by procedure or training because procedure and training can address only those event sequences which have been postulated in advance. One important lesson learned from the experience at Three Mile Island and at other facilities is that not all event sequences can be postulated in advance. Therefore, an alternative which avoids this potential for conflict and confusion but improves the functions intended by the proposed STA is recommended.

Two functions are intended to be improved by the proposed STA: (1) accident assessment and (2) operating experience assessment. In order to improve the accident assessment function while avoiding the degradation in accident response which accompanies the proposed STA, the course of an accident is considered in three sequential phases: immediate, intermediate, and recovery.

The immediate phase extends from the point at which an abnormal condition affecting plant safety can be detected in the control room until the point at which the shift supervisor has sufficient time to carefully consider an independent assessment and, on the basis of such assessment, decide to alter the procedural actions of the operators. The intermediate phase extends from the end of the immediate phase until the point at which the Technical Support Center (TSC) is manned and ready. The recovery phase extends from the end of the intermediate phase until the point at which recovery is complete.

For the immediate phase, the accident assessment function can be improved only by upgraded training to enchance the operators' abilities to recognize, diagnose, and respond to accident conditions. During this phase the operators' actions are governed by training and emergency procedures, and by definition there is insufficient time for the careful consideration of an independent assessment which would be required before such an assessment could become the basis for altering the procedural actions of the operators.

For the intermediate phase, the accident assessment function can be improved by either of two alternative means. An operator can be educated in science and engineering in order that he might provide an assessment which could be considered and acted upon by the shift supervisor. Alternatively, a graduate engineer or equivalent can be trained in plant operations and made available to the shift supervisor on call in order that he might provide such an assessment. In either case, the shift supervisor must have sufficient time to carefully consider the assessment and, based on such assessment, decide to alter the procedural actions of the operators.

2.2.1b: "Shift Technical Advisor" (p.3)

For the recovery phase, the accident assessment function can be improved by manning the TSC. The collective engineering resource within the TSC will be able to develop a detailed independent assessment of plant conditions and provide appropriate procedures with which to recover from the accident.

The operating experience assessment function can best be provided by a team which reviews operating experience at the plant and at plants of like design. Varying team membership as appropriate to the operating experience being assessed assures accomplishment of this function by the best qualified individuals.

BWR Owners' Group Implementation Criteria:

The two functions intended to be improved by the proposed STA will be improved as follows:

1. Accident Assessment

    a. Immediate Phase

    An operator or supervisor in the direct operational chain of command on each shift (normally in charge in the control room) will receive additional specific training in the response and analysis of the plant for transients and accidents. This training will be coordinated with the schedule for preparation and review of analysis and guidelines under the NRC Bulletins and Orders Task Force.

    All operators and supervisors will receive additional training appropriate to their responsibilities in the response of the plant to transients and accidents. This longer term training and qualification criteria will be provided by the Institute of Nuclear Power Operations.

    b. Intermediate Phase (Alternatives)

    An operator or supervisor in the direct operational chain of command on each shift will receive substantial additional education in basic engineering and science sufficient to aid him in assessing unusual situations not explicitly covered in the current operator training.

    - OR -

    A graduate engineer or equivalent trained in the response and analysis of the plant for transients and accidents and in plant design and layout, including the capabilities of instrumentation and controls in the control room, will be available to the individual in charge in the control room on call. He may be stationed on or off site as appropriate to plant location, communication capabilities, operator training and education, extent and detail of emergency procedures, etc.

2.2.1b: "Shift Technical Advisor" (p.4)

c. Recovery Phase

Individuals knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident will be available on call to staff the On-Site Technical Support Center.

2. Operating Experience Assessment

Where it does not already exist, a team will be designated by the licensee to assess the operating experience at his plant or plants and at plants of like design. Team membership may vary as appropriate to the operating experience being assessed but will include experience in systems engineering and familiarity with or routine access to persons experienced in the principles of human engineering or human factors.

NUREG-0578 Requirement 2.2.1.c: "Shift and Relief Turnover Procedures"

Review and revise plant procedures as necessary to assure that a shift turnover checklist is provided and required to be completed and signed by the on-coming and off-going individuals responsible for command of operations in the control room. Supplementary checklists and shift logs should be developed for the entire operations organization, including instrument technicians, auxiliary operators, and maintenance personnel.

Discussion:

The Owners' Group agrees that knowledge of plant status, especially for those systems requird to mitigate the consequences of an accident, should be transferred in a systematic manner from one shift to the next. The Group is also convinced that to be most effective as a means of information transfer in the course of a shift or relief turnover, the information must be limited to that which can be summarized on a single list on a single piece of paper. Furthermore, the information provided by the list should be reviewed not only by the shift supervisor and control room operators, but by other plant personnel (auxiliary operators, technicians, etc.) as appropriate, thus eliminating the need for separate checklists, as apparently required in the staff's position.

BWR Owners' Group Implementation Criteria:

1) A checklist will be devised to ensure that control room status of systems that are required to mitigate the consequences of an accident are monitored on a shift turnover basis. This list will include system lineups and alarms located in the main control room. Systems and components in a degraded condition will be identified as required by plant status.

2) The checklist will be kept in the control room at all times.

3) The checklist will be reviewed by personnel other than the shift supervisor and control room operators as appropriate.

NUREG-0578 Requirement 2.2.2.a: "Control Room Access"

Review plant emergency procedures, and revise as necessary, to assure that access to the control room under normal and accident conditions is limited to those persons necessary to the safe command and control of operations.

Discussion:

The Owners' Group agrees that it is necessary to limit access to the control room and to establish a clear line of authority and responsibility in the control room in the event of an emergency.

BWR Owners' Group Implementation Criteria:

Procedures will be developed and implemented which will meet the intent of the staff's position.

NUREG-0578 Requirement 2.2.2.b: "Onsite Technical Support Center"

A separate technical support center shall be provided for use by plant management, technical, and engineering support personnel. In an emergency, this center shall be used for assessment of plant status and potential offsite impact in support of the control room command and control function. The center should also be used in conjunction with implementation of onsite and offsite emergency plans, including communications with an offsite emergency response center. Provide at the offsite technical support center the as-built drawings of general plant arrangements and piping, instrumentation, and electrical systems. Photographs of as-built system layouts and locations may be an acceptable method of satisfying some of these needs.

Discussion:

The Owners' Group agrees that it is important to have a technical support center (TSC) designated where "individuals who are knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident" can go to consistent with the intent to limit access to the control room. Furthermore, it is agreed that it is appropriate that the emergency plans will designate the role and location of the technical support center. There is, however, one area in particular which needs further discussion.

The requirement that the TSC be onsite and in close proximity to the control room is not necessarily the best choice under all circumstances for meeting the intent of the position. The location of the TSC should be dictated by its accessibility to the engineering and management personnel who will occupy it, rather than by its physical proximity to the control room. For example, multi-unit sites which share engineering and management personnel, or so-called outdoor sites which have administrative buildings detached from the plant, may designate locations which may not be judged as in close proximity to the control room, but make sense from a personnel access viewpoint. Furthermore, "close proximity" would only seem to be required as a means of supplementing the transmittal of plant status from the control room to the TSC, and in that sense then becomes inconsistent with the desire to limit access to the control room during emergencies. Thus, the requirement for close proximity could be eliminated on the basis that the plant status must be monitored from the TSC.

The Owners' Group also agrees that monitoring equipment may vary from plant to plant, and that there is no single best way in which to monitor plant status in the TSC. There was agreement that TV monitors which could read and transmit information from the control room panels to the TSC would meet the requirement to display and transmit plant status. It was also agreed that the TSC should have two-way communications links with the control room, other onsite telephones, the offsite Emergency Operations Center, and the NRC. It was further agreed that the existing direct link between the NRC and the control room would be switched over to the TSC upon its activation in accordance with the intent to limit access to the control room. Finally, it was agreed that the staffing and activation criteria for the TSC would be specified in the emergency plan.

BWR Owners' Group Implementation Criteria:

Phase I (by Jan. 1, 1980):

1) A location will be designated in the emergency plan. This may be a temporary location.

2) Communications links will be established with the control room, the on-site Operational Support Center, the off-site Emergency Operations Center, and the NRC. These may be temporary.

3) The staffing and activation criteria will be specified in the emergency plan.

4) The TSC will have access to the records (system descriptions, arrangement drawings, etc.) in accordance with the revised NUREG-0578 position.

Phase II

The implementation criteria of Phase II will be issued after further discussions between the Owners' Group and the NRC staff.

NUREG-0578 Implementation Letter Requirements Relative to Containment Level, Pressure, and Hydrogen Monitoring

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

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

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

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

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

Discussion:

The Owners' Group concurs with the ACRS recommendations for additional instrumentation for the following parameters:

1) Containment water level monitoring

2) Containment pressure monitoring

3) Containment hydrogen monitoring

For practical reasons, it is not desirable to monitor suppression pool water level all the way to the bottom of the suppression pool. This is because an instrument tap at the very bottom could become obstructed by sludge and small debris. The Owners' Group believes that water level monitoring down to the elevation of the lowest ECCS pump suction is more practical and fully satisfies the intent of the requirement.

Containment Monitoring (p

It is the Owners' Group's current interpretation that the hydrogen monitoring requirement is associated with ECCS performance and core degradation, rather than with containment atmospheric control.

Owners' Group Implementation Criteria:

1) The Owners' Group intends to implement containment pressure, water level, and hydrogen monitoring which will be designed and installed to meet Engineered Safety System criteria.

2) The lowest suppression pool water level monitored will be at or below the elevation of the lowest ECCS pump suction.

NUREG-0578 Implementation Letter Requirement Relative to Remotely Operated High Point Vents.

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 applicant 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 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.

Discussion:

Domestic BWRs are provided with a number of power operated safety grade relief valves which can be manually operated from the control room to vent the reactor pressure vessel. The point of connection of the vent lines from the vessel to these valves is such that accumulation of gases above that point in the vessel will not affect natural accumulation of gases of the reactor core.

These power operated relief valves satisfy the intent of the NRC position. Information regarding the design, qualification, power source, etc., of these valves has been provided to the individual plant Safety Analyses Reports.

The Owners' position is that the requirement of single failure criteria for prevention of inadvertent actuation of these valves, and the requirement (stated in the October 11 topical meeting) that power be removed

Discussion (Cont)

during normal operation, are not applicable to BWRs. These valves serve an important function in mitigating the effects of transients and in many plants provide ASME code overpressure protection. Therefore, the addition of a second "block" valve to the vent lines could result in a less safe design and in some cases a violation of the code. Also, inadvertent opening of relief valve in a BWR is a design basis event and is a controllable transient (this is discussed in our position on NUREG-0578, item 2.1.2).

In addition to the power-operated relief valves, operating BWRs include various other means of high-point venting. Information on which plants are equipped with which features has been provided in individual plant Safety Analysis Reports, and may be summarized by individual licensees in their NUREG-0578 implementation letters. Among these are:

1) Normally closed reactor vessel head vent valves, operable from the control room, which discharge to the drywell;

2) Normally open reactor head vent line, which discharges to a main steam line;

3) Main steam-driven Reactor Core Isolation Cooling (RCIC) System turbines, operable from the control room, which exhaust to the suppression pool;

4) Main steam-driven High Pressure Coolant Injection (HPCI) System turbines, operable from the control room, which exhaust to the suppression pool;

5) Isolation condenser primary side vent valves, operable from the control room, which discharge to containment or a main steam line.

Although the power-operated relief valves fully satisfy the intent of the requirement, these other means also provide protection against the accumulation of noncondensibles in the reactor pressure vessel.

In the October 11, 1979, topical meeting on this subject, three procedural questions were raised:

1) Where to vent to (suppression pool vs. containment);

2) When to vent;

3) When not to vent.

Under most circumstances, there would be no choice as to where to vent to or when to vent, since the relief valves (as part of the Automatic Depressurization System), HPCI, and RCIC will function automatically in their designed modes to ensure adequate core cooling, and these will provide

High Point Vents (p.3)

Discussion (cont)

continuous venting to the suppression pool. The current assessment is that it would not be desirable to interfere with emergency core cooling functions in order to prevent venting, but the matter will be studied further.

The result of a break in the safety/relief valve discharge line, or any of the other systems enumerated above, would be the same as a small steam line break. A complete steam line break is part of the plants' design basis, and smaller-size breaks have been shown to be of lesser severity. A number of reactor system blowdowns due to stuck-open relief valves (also equivalent to a small steam line break) have confirmed this in practice (see Owners' Group position on Requirement 2.1.2). Thus no new analyses to show conformance with 10 CFR 50.46 are required.

Because the relief valves, HPCI, and RCIC will vent the reactor continuously, and because containment hydrogen calculations in normal safety analysis calculations assume continuous venting, no special analyses are required to demonstrate "that the direct venting of noncondensible gases with perhaps high hydrogen concentrations does not result in violation of combustible gas concentration limits in containment."

BWR Owners' Group Implementation Criteria:

1) The Owners' Group believes that adequate reactor coolant system venting is provided by the existing plant design.

2) Plant procedures will be provided to govern the operator's use of the relief valves for venting the reactor pressure vessel.

3) No new 10 CFR 50.46 conformance calulations or containment combustible gas concentration calculations are required, since systems in the plant's original design and covered by the original design bases are used;

4) In response to a request from the October 11, 1979, topical meeting, the use of isolation condenser tube side vents will be considered;

5) In response to a request from the October 11, 1979, topical meeting, the effect of noncondensibles in HPCI/RCIC turbine steam will be addressed.

NUREG-0578 Requirement 2.2.2.c: "Onsite Operational Support Center"

Each operating nuclear power plant should establish and maintain a separate onsite operational support center outside the control room, In the event of an emergency, shift support personnel (e.g., auxiliary operators and technicians) other than those required and allowed in the control room shall report to this center for further orders and assignment.

Discussion:

The Owners' Group agrees with the position as stated, with the clarification that there may be plant unique situations where it may be more appropriate that more than one location be designated in the emeegency plan. As long as these locations are known and the "methods and lines of communication and management" are specified in the emergency plan, the intent of the position will have been met.

BWR Owners' Group Implementation Criteria

The staff's position will be implemented as stated and subject to the clarification on location stated above.