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October 19, 1979

BECo Ltr. #79-206

License No. DPR-35
Docket No. 50-293

G. CARL ANDOBNINI
SUPERINTENDENT
NUCLEAR OPERATIONS DEPARTMENT

Mr. Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Follow-Up to Reviews Regarding the
Three Mile Island Unit 2 Accident

Reference: Division of Operator Reactors Letter dated 9/13/79,
Follow-Up Actions Resulting from the NRC Staff Reviews
Regarding Three Mile Island Unit 2 Accident.

Dear Mr. Eisenhut:

You have requested that all Operating Reactor Licensees implement the actions contained in NUREG-0578, as modified and/or supplemented by items (a) through (f) of the above reference, within the schedule constraints as specified in enclosure (6) to the above reference.

The Boston Edison Company has reviewed all relevant material on the subject and has concluded that the issue of NUREG-0578 implementation can best be handled within the framework of the currently existing General Electric Boiling Water Reactor Owners Group, which was created specifically to address the issues raised by the Three Mile Island accident of significance to boiling water reactors.

Well defined acceptance criteria for many of the recommendations of NUREG-0578 are needed in order to ensure timely implementation. The recent clarification meetings and discussions have been of benefit, but others may be necessary to develop adequate acceptance criteria. These acceptance criteria, when fully developed, may impact implementation schedules due to hardware availability as well as affecting the ability to optimize utilization of our scheduled refueling outage for such implementation.

We are attempting in our planning to complete category A items before the beginning of the next Pilgrim Station operating cycle. Each NUREG-0578 position and our response is addressed in the attachment to this letter. Our planning to meet this tight time schedule is underway. We will advise you if it is determined that any of the planned changes required by NUREG-0578 positions cannot be achieved on this schedule.

In addition, the referenced letter requests that we meet the requirements of Enclosure 7 in accordance with the implementation schedules shown in Enclosure 8. As stated above, we will advise you if we determine that we are unable to meet any of the implementation schedules shown in Enclosure 8.

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ADD:
S/1
R SWAIDER
W MILLS
J BRARD
F SKOPEC
N ANDERSON

BOSTON EDISON COMPANY

Mr. Darrell G. Eisenhut, Acting Director
October 19, 1979
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We trust this letter is responsive to your requirements; however, should you desire additional information or clarification please feel free to contact us.

Very truly yours,

Harold Redgrave
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Attachment
JV/gs

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ATTACHMENT

Responses to NUREG-0578 Positions and Emergency Preparedness

1205 077

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Emergency Power Supply Requirements for the Pressurizer Heaters, Power-Operated Relief Valves and Block Valves, and Pressurizer Level Indicators in PWR's (Section 2.1.1)

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 the when and how the required pressurizer heaters shall be connected to the emergency buses. If required, the procedures shall identify under what conditions selected emergency loads can be shed from the emergency power source to provide sufficient capacity for the connection of the pressurizer heaters.
3. The time required to accomplish the connection of the preselected pressurizer heater to the emergency buses shall be consistent with the timely initiation and maintenance of natural circulation conditions.
4. Pressurizer heater motive and control power interfaces with the emergency buses shall be accomplished through devices that have been qualified in accordance with safety-grade requirements.

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

1. Motive and control components of the power-operated relief valves (PORV's) shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.
2. Motive and control components associated with the PORV block valves shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.
3. Motive and control power connections to the emergency buses for the PORV's 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. These buses shall have the capability of being supplied from either the offsite power source or the emergency power source when offsite power is not available.

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Discussion

As discussed in NEDO-247081 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. 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.

Response

For the reasons stated above, there is no need for action in response to position 2.1.1 for Pilgrim Nuclear Power Station, which is a G.E. BWR.

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Performance Testing for BWR and PWR Relief and Safety Valves (Section 2.1.2)

Position

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

Discussion

The BWR design basis includes no transients or accidents in which two-phase flow or subcooled liquid flow at high pressure is calculated or expected. 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, 50 cases have occurred; in 3 such cases, the safety and relief valves passed two-phase flow. Tables 1 and 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 valve 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).

B. Valve Usage

1. Dual-Function S/RV Plants. The 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 valves 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 shutdown using familiar and uncomplicated procedures.
2. Plants with relief (and/or S/RV) and Safety valves. Steam in the relief functions is discharged to the containment suppression pool and the discussion of (a) applies. The safety valve set-point is sufficiently higher than the relief set-point that the safety valves are almost never required

to operate (Table 2 documents the three cases in which safety valves have ever lifted 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.

C. Two-Phase Flow

Expected operating conditions and transients do not include two-phase flow through S/RV's, safety, or relief valves. However in 3 incidents, 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 3. In these events, Electromatic relief valves and direct acting safety valves water actuated, discharged a steam/water mixture and reclosed, indicating that the flow media did not cause a stuck-open valve condition. Construction of other BWR direct acting S/RV's is equivalent to the designs used in these early plants. These events did not lead to any concern over core uncover. 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.

1. 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 is indicated by field service.
2. Satisfactory operation of Dresser safety valves is indicated by field service.

D. Field Experience

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

Although a stuck open S/RV is ordinarily of no safety concern, programs are underway to reduce the frequency of each event. From Table 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 it is seen that experience with Dresser safety valves has always been good.

E. 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 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;

4. In the three events wherein BWR S/RV's did pass two-phase flow the valves reclosed;
5. Spring safety and Electromatic relief 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 core uncover, and the valve need not pass two-phase flow;
6. Dual-function S/RV's are frequently called on to operate and 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. There is no concern for core uncover, and the valve need not pass two-phase flow. Improvement programs are reducing the frequency of such events.

Response

Based on the above discussions, concerns regarding safety/relief valve performance have been addressed and no additional testing is required provided that the following criteria are met:

1. A procedure shall exist for responding to a stuck open relief, S/RV, or safety valve.
2. The procedure shall address prevention of inadvertent overfilling of the reactor vessel.
3. A control grade system, actuated by reactor vessel high water level, shall be provided to prevent overfilling of the vessel.

The Boston Edison Company intends to comply with the above criteria.

TABLE 1
S/RV BLOWDOWNS IN BWR OPERATION

YEAR	3-STAGE TARGET ROCK			2-STAGE TARGET ROCK		CROSBY-OKANO-DIKKERS		TOTAL S/RV BLOWDOWNS	TOTAL S/RVs IN SERVICE	TOTAL BLOWDOWNS DIVIDED BY TOTAL VALVES IN SERVICE
	TOTAL BLOWDOWNS	STUCK OPEN FOLLOWING DEMAND	# OF VALVES IN SERVICE	TOTAL BLOWDOWNS	# OF VALVES IN SERVICE	TOTAL BLOWDOWNS	# OF VALVES IN SERVICE			
1971	2	2	14					2	4	0.5
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 for information on this equipment.

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TABLE 2
SAFETY AND ELECTROMATIC RELIEF VALVE
BLOWDOWNS IN BWR OPERATION

A. Dresser Safety Valves.

Only one event has ever occurred with partially stuck open valves - the Dresden 2 event described in Table 3. The lifting levers which jammed 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 3). The total number of valves in service is 76⁽¹⁾.

POOR ORIGINAL

B. Dresser Electromatic Relief Valves.

There have been two occurrences of a stuck open Electromatic relief valve, one of which followed a demand. These events occurred at the same plant in April 1973 and March 1977. The number of valves in service is 37.⁽¹⁾

⁽¹⁾ Some BWRs are in the process of replacing Dresser Safety valves and Electromatic relief valves with Target Rock S/RVs.

TABLE 3

BWR EVENTS IN WHICH TWO-PHASE FLOW OR
LIQUID PASSED THROUGH SAFETY/RELIEF VALVES

POOR ORIGINAL

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. As a result, the containment was pressurized to an estimated 20 psig and an estimated temperature of approximately 300°F. Damage within the drywell was generally limited to over-heating of most of the flux monitoring instrumentation cables and water impingement on insulation. 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 2 - 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). Pressurization of the containment

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TABLE 3 (cont'd)

POOR ORIGINAL

continued as high as 20 psig. Inspections showed that the high humidity and temperature in the drywell following the release to the containment damaged LPRM cables, which required replacement. Other results of the discharge from the safety valve included damage to an electromatic relief valve controller, damage to insulation near the safety valve, scoured paint on the drywell walls, and a damaged ventilation duct. There was never any concern for 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 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 concern for maintaining adequate water supply to the reactor core, and there was no question of adequate core cooling.

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NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Direct Indication of Power-Operated Relief Valve and Safety Valve Position for PWR's and BWR's (Section 2.1.3.a)

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.

Discussion

Boston Edison offers the following generic discussion. BWR safety and relief valves are arranged in three ways in the various operating reactors:

1. Discharges piped to the containment suppression pool;
2. Discharges manifolded and piped to the 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. Relief Valves Individually Piped to the Suppression Pool

The safety/relief valves at Pilgrim Station Unit #1 are individually piped to the suppression pool. In the case of a stuck-open valve, the containment pressure will not increase because of the submerged discharge. There is benefit in direct indication, because the operator would be aware of which S/RV valve had actuated.

B. Spring Safety Valve Discharge Directly to Containment or Drywell

All spring safety valves are configured this way. Because these valves are large (642,100 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 ECCS operation. Because the valves discharge steam from the main steam lines into the containment, the effect of a stuck-open valve is identical to a small steam line break. Because the operator has no capability of attempting to reseal 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 50 events of stuck-open relief valves of similar capacity in operating BWR's (see "Discussion and Response to NUREG-0578 Position 2.1.2"). High reactor water level trips preclude water in the steam lines, and operators are sensitive to the undesirability of overfilling the reactor vessel. Thus, there is no need for special precautions due to the possibility of two-phase flow in the valves. Even if the valves were to reseal, the operator's action would be no different than for a small steam line break.

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(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.

Response

1. For reasons stated above in paragraph B, "Spring Safety Valve Discharge Directly to Containment or Drywell" no further action is necessary for the Pilgrim Nuclear Power Station.
2. Safety/Relief valves at PNPS discharge to the Torus. Acoustic monitoring devices will be installed on each Safety/Relief valve discharge to insure that operations personnel have a positive indication of flow in the discharge line.
3. The Boston Edison Company will provide acoustic monitoring which meets the following criteria:
 - a. There will be at least one sensing device per discharge line;
 - b. Sensing devices may be either inside or outside the drywell;
 - c. 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;
 - d. All components will be seismically qualified;
 - e. The system will be powered by one division of emergency power;
 - f. With sensing devices inside the drywell, non-class IE electrical penetrations may be used if insufficient IE penetrations are available.

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Instrumentation for Detection of Inadequate Core Cooling in PWR's and BWR's (Section 2.1.3.b)

Position

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

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

2. Licensees shall provide a description of any additional instrumentation 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.

Discussion

Boston Edison believes that additional hardware to identify inadequate core cooling on BWR's is not necessary. Licensee procedures will identify the diverse methods of determining inadequate core cooling, using existing instrumentation. The results of analysis being performed in response to Section 2.1.9 will be factored into procedures, as required, after the analysis is complete.

Response

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 Order 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 further 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. Boston Edison believes that a subcooling meter, as required by Enclosure 6 of the NUREG-0578 implementation letter of September 13, 1979, is not necessary since the BWR normally operates at saturated conditions.

1205 089

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Containment Isolation Provisions for PWR's and BWR's (Section 2.1.4)

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

Discussion

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

Response

1. Diversity of parameters sensed for the initiation of containment isolation shall be provided in accordance with SRP-6.2.4.
2. Careful reconsideration shall be given to the definition of essential and non-essential systems, each system shall be identified as essential or non-essential and the bases for the selection of each essential system shall be provided.
3. All systems not identified as essential will be reviewed. If automatic isolation is not provided, justification for not isolating them will be prepared.
4. Isolation control systems and administrative controls will be reviewed as appropriate, such that no isolation valve will open when the isolation logic is reset.

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Dedicated Penetrations for External Recombiners or Post-Accident Purge Systems (Section 2.1.5.a)

Position

Plants using external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere should provide penetrations for external recombining 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 recombining or purge system.

Discussion

Dedicated penetrations for the use of external hydrogen recombiners have been determined to be a necessary modification.

Response

Penetrations will be provided for purge system or external hydrogen control use. These penetrations will be single failure proof, dedicated to hydrogen control functions and engineered to include the installation and operational requirements of standard hydrogen recombiners.

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Capability to Install Hydrogen Recombiner at Each Light Water Nuclear Power Plant (Section 2.1.5.c)

Position (minority View)

1. All licensees of light water reactor plants shall have the capability to obtain and install recombiners in their plants within a few days following an accident if containment access is impaired and if such a system is needed for long-term post-accident combustible gas control.
2. The procedures and bases upon which the recombiners would be used on all plants should be the subject of a review by the licensees in considering shielding requirements and personnel exposure limitations as demonstrated to be necessary in the case of TMI-2.

Discussion

As discussed at the NRC Region I meeting on TMI in King of Prussia, PA. on September 24, 1979, this position requires reactor sites with installed hydrogen recombiners to develop procedures for using these recombiners.

Response

Since the Pilgrim Nuclear Power Station does not have installed hydrogen recombiners, this position is not applicable.

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Integrity of Systems Outside Containment Likely to Contain Radioactive Materials (Engineered Safety Systems and Auxiliary Systems) for PWR's and BWR's (Section 2.1.6.a)

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.

Discussion

Several of the engineered safety features and auxiliary systems located outside reactor containment will or may have to function during a serious transient or accident with large radioactive inventories in the fluids they process. The leakage from these systems, when operated, must be minimized or eliminated to prevent the release of significant amounts of radioactive material to the environment.

Response

1. System leakage will be determined by periodic surveillance of those systems outside of primary containment which could contain highly radioactive material. The status of any leakage will be evaluated and corrective action taken to keep system leakage as low as practical.
2. All plant systems will be evaluated and those systems outside the primary containment that could contain highly radioactive material will be included in the surveillance program.

1205 093

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Design Review of Plant Shielding of Spaces for Post-Accident Operations (Section 2.1.6.b)

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.

DISCUSSION:

After an accident in which significant core damage occurs, the radiation source terms may approximate those of Regulatory Guides 1.3 and 1.4. Large radiation fields, resulting from large radiation sources being contained in systems not designed for such activity, may make it difficult to effectively perform accident recovery operations. Vital areas such as control rooms, radwaste panels, emergency power supplies, and instrument rack areas may fall within the radiation fields of such systems.

RESPONSE:

Boston Edison is reviewing the accessibility requirements following any credible accident releasing 100% Noble gas, 50% halogen, and 1% solids to the coolant. Locations requiring access will be defined and procedures to perform the work will be prepared. The final goal will be to establish the plant capability to remain in a safe cooldown/shutdown condition following an accident which would result in the above-mentioned releases.

1205 094

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Automatic Initiation of the Auxiliary Feedwater System for PWR's
(Section 2.1.7.a)

Position

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

1. The design shall provide for the automatic initiation of the auxiliary feedwater system.
2. The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
3. Testability of the initiating signals and circuits shall be a feature of the design.
4. The initiating signals and circuits shall be powered from the emergency buses.
5. Manual capability to initiate the auxiliary feedwater system from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function.
6. The a-c motor-driven pumps and valves in the auxiliary feedwater system shall be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
7. The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFWS from the control room.

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

Discussion

As discussed at the NRC Region I meeting on TMI in King of Prussia, PA., on September 24, 1979, this position is only applicable to PWR's.

Response

Since the Pilgrim Nuclear Power Station is a Boiling Water Reactor, this position is not applicable.

1205 095

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Auxiliary Feedwater Flow Indication to Steam Generators for PWR's
(Section 2.1.7.b)

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 APWS 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.

Discussion

As discussed at the NRC Region I meeting on TMI in King of Prussia, PA., on September 24, 1979, this position is only applicable to PWR's.

Response

Since the Pilgrim Nuclear Power Station is a BWR this position is not applicable.

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Improved Post-Accident Sampling Capability (Section 2.1.8.a)

POSITION:

A design and operational review 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 to excess of 3 and 18 3/4 Rems to the whole body 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 completed promptly; i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift.

DISCUSSION:

Timely information from reactor coolant and containment air samples can be important to reactor operators for their assessment of system conditions and can influence subsequent actions to maintain the facility in a safe condition. Following an accident, significant amounts of fission products may be present in the reactor coolant and containment air, creating abnormally high radiation levels throughout the facility. These high radiation levels may delay the obtaining of information from samples because people taking and analyzing the samples would be exposed to high levels of radiation. In addition, the abnormally high background radiation, high sample radiation, and high levels of airborne contamination may render in-plant radiological spectrum analysis equipment inoperable during and after an accident.

1205 097

RESPONSE:

Reactor coolant and containment atmosphere sampling systems will be reviewed to determine the capability to promptly obtain 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. The review will consider procedural and/or hardware modifications to achieve the desired results and will be completed by 1/1/80. Modifications, if required, will be identified and a plan to complete them will be transmitted to the NRC.

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Increased Range of Radiation Monitors (Section 2.1.8.b)

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", to 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 a minimum of 10^{-7} uCi/cc (Xe-133) 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 condition are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by absorption on charcoal or other media, followed by onsite laboratory analysis.
3. In-containment radiation level monitors with a maximum range of 10^8 rad/hr shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be designed and qualified to function in an accident environment.

Discussion

Boston Edison Company recognizes and concurs with the position as modified and discussed during the NRC Region I meeting on September 24, 1979.

Response

The requirements of position 2.1.8.b items 1,2 and 3 will be implemented.

1205 099

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Improved-In-Plant Iodine Instrumentation (Section 2.1.8.c)

Position

Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration throughout the plant under accident conditions.

Discussion

Boston Edison Company recognizes the concurs with the position.

Response

1. The requirements of position 2.1.8.c will be implemented.
2. Procedures will be developed to accurately determine in-plant iodine concentrations.

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Analysis of Design and Off-Normal Transients and Accidents
(Section 2.1.9)

POSITION

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

1. Small break loss-of-coolant accidents;
2. Inadequate core cooling; and
3. Transients and accidents.

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

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

1. Low reactor coolant system inventory (two examples will be required - LOCA with forced flow, LOCA without forced flow).
2. Loss of natural circulation (due to loss of heat sink).

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

The analyses of transients and accidents shall include the design basis events specified in Section 15 of each FSAR. The analyses shall include a single active failure for each system called upon to function for a particular event. Consequential failures shall also be considered. Failures of the operators to perform required control manipulations shall be given consideration for permutations of the analyses. Operator actions that could cause the complete loss of function of a safety system shall also be considered. At present, these analyses need not address passive failures or multiple system failures in the short term. In the recent analysis of small break LOCAs, complete loss of auxiliary feedwater was considered. The complete loss of auxiliary feedwater may be added to the failures

being considered in the analysis of transients and accidents if it is concluded that more is needed in operator training beyond the short-term actions to upgrade auxiliary feedwater system reliability. Similarly, in the long term, multiple failures and passive failures may be considered depending in part on staff review of the results of the short-term analyses.

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

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

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

DISCUSSION

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

RESPONSE

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

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Instrumentation to Monitor Containment Conditions During the Course of an Accident (Section 2.1.9, "New 1, 2 & 3")

Position

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

1. A continuous indication of containment pressure shall be provided in the control room. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and minus five psig for all containments.
2. A continuous indication of hydrogen concentration in the containment atmosphere shall be provided in the control room. Measurement 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 PWR's and cover the range from the bottom to the top of the containment sump. Also for PWR's, a wide range instrument shall be provided and cover the range from the bottom of the containment to the elevation equivalent to a 5000,000 gallon capacity. For BWR's, 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 Boston Edison Company agrees with the BWR Owner's Group position and ACRS recommendations for additional instrumentation for the following parameters:

- a. Containment water level monitoring
- b. Containment pressure monitoring
- c. Containment hydrogen monitoring

Response

1. BECo will provide containment pressure and water level systems which will be designed and installed to meet engineered safety systems criteria.

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2. The existing hydrogen and oxygen monitoring system will be reviewed and upgraded if necessary so that it has a redundant capability to monitor hydrogen and oxygen in both the drywell and suppression chamber.
3. The lowest suppression pool water level monitored will be at or below the elevation of the lowest ECCS pump suction penetrations.

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Installation of Remotely Operated High Point Vents in the Reactor Coolant System (Section 2.1.9 "New 4")

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 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 Standard Review Plan Section 6.2.5.
3. Procedural guidelines for the operator's use of the vents. The information available to the operator for initiating or terminating vent usage shall be discussed.

Discussion

Pilgrim Nuclear Power Station Unit #1 is provided with four (4) power operated safety grade relief valves (RV-203-3A, 3B, 3C & 3D). A complete description of these valves is provided in the PNPS #1 FSAR. The location of these valves (on main steam lines A, B, C & D) are such that accumulation of noncondensable gases above this point will not affect natural circulation cooling of the reactor core.

Although the ADS valves fully satisfy the intent of the requirement, other means also provide protection against accumulation of non-condensables. The Pilgrim Nuclear Power Station Unit #1 contains a small (2") continuous vent pipe upstream of the RPV head vent valves connecting directly to the main steam piping. This continuous vent line contains a normally open globe valve and during normal operations conveys non-condensables mixed with the steam to the main turbine.

During reactor vessel isolation, non-condensables are carried to the RCIC turbine via the continuous vent line to the RCIC steam line. Additionally, non-condensables can also be vented to the suppression pool by intermittent use of the automatic depressurization system (ADS). During LOCA conditions, non-condensables are carried to the suppression pool via continuous vent to main steam line to HPCI and to RCIC turbine. If HPCI is inoperable, ADS operation will serve to vent non-condensables.

In addition, there is a Reactor Head Vent valve (remote manual from the control room) available for venting non-condensables gas to the drywell.

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, and;
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 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 Section 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 noncondensable gases with perhaps high hydrogen concentrations does not result in violation of combustible gas concentration limits in containment."

Boston Edison concludes that adequate reactor coolant system venting is provided by the existing plant design as detailed above.

Response

1. Boston Edison believes that adequate reactor coolant system venting is provided by the existing plant design.
2. Plant procedures will be reviewed and revised as necessary to govern the operator's use of the relief valves for venting the reactor pressure vessel.
3. No new 10 CFR 50.46 conformance calculations 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 is not applicable to the Pilgrim Nuclear Power Station.
5. In response to a request from the October 11, 1979, topical meeting, the effect of non-condensables in HPCI/RCIC turbine steam will be addressed.

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Shift Supervisor's Responsibilities (Section 2.2.1.a)

POSITION:

1. The highest level of corporate management of each licensee shall issue and periodically reissue a management directive that emphasizes the primary management responsibility of the shift supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.
2. Plant procedures shall be reviewed to assure that the duties, responsibilities, and authority of the shift supervisor and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the shift supervisor in the control room relative to other plant management personnel. Particular emphasis shall be placed on the following:
 - a. The responsibility and authority of the shift supervisor shall be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The idea shall be reinforced that the shift supervisor should not become totally involved in any single operation in times of emergency when multiple operations are required in the control room.
 - b. The shift supervisor, until properly relieved, shall remain in the control room at all times during accident situations to direct the activities of control room operators. Persons authorized to relieve the shift supervisor shall be specified.
 - c. If the shift supervisor is temporarily absent from the control room during routine operations, a lead control room operator shall be designated to assume the control room command function. These temporary duties, responsibilities, and authority shall be clearly specified.
3. Training programs for shift supervisors shall emphasize and reinforce the responsibility for safe operation and the management function the shift supervisor is to provide for assuring safety.
4. The administrative duties of the shift supervisor shall be reviewed by the senior officer of each utility responsible for plant operations. Administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant shall be delegated to other operations personnel not on duty in the control room.

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DISCUSSION:

The Boston Edison Company offers the BWR owners' group discussion as follows:

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 above, the entire sentence will be interpreted as follows: The shift supervisor, 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.

RESPONSE:

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

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Shift Technical Advisor (Section 2.2.1.b)

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:

In developing the recommendation for the Shift Technical Advisor, the Lessons Learned Task Force concentrated on the two functions that needed to be provided; namely, an accident assessment function and an operating experience assessment function. The proper performance of these functions requires the provision of certain characteristics as described in enclosure (2) to Mr. D. G. Eisenhut's letter of September 13, 1979, to utilities of all Operating Nuclear Power Plants.

RESPONSE:

The Boston Edison Company is adopting the concept of the Shift Technical Advisor as presented in Enclosure 2 to Mr. D.G. Eisenhut's letter as follows:

1. Senior Reactor Operators will be selected for upgrading to the position of Shift Superintendent who will perform the accident assessment function. These individuals would be relieved of all administrative duties. The Shift Superintendent will receive additional specific training in the response and analysis of the plant for transients and accidents as well as core physics, thermal hydraulics, system response and management skills.
2. The Boston Edison Company will provide two Senior Reactor Operators on shift at all times, one of which will be in the Control Room. Boston Edison Company is evaluating the possibility of placing an administrative assistant on either the day watch or on all shifts as deemed necessary.
3. Boston Edison Company will develop a systematic method for multi-disciplinary review of operating experience to improve both reliability and safety. This group will perform the operational review assessment function.
4. 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.

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Shift and Relief Turnover Procedures (Section 2.2.1.c)

POSITION

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

1. A checklist shall be provided for the oncoming and offgoing control room operators and the oncoming shift supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist:
 - a. Assurance that critical plant parameters are within allowable limits (parameters and allowable limits shall be listed on the checklist).
 - b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console (what to check and criteria for acceptable status shall be included on the checklist);
 - c. Identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode shall be compared with the Technical Specifications action statement (this shall be recorded as a separate entry on the checklist).
2. Checklists or logs shall be provided for completion by the offgoing and oncoming auxiliary operators and technicians. Such checklists or logs shall include any equipment under maintenance of test that by themselves could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate an operational transients (what to check and criteria for acceptable status shall be included on the checklist); and
3. A system shall be established to evaluate the effectiveness of the shift and relief turnover procedure (for example, periodic independent verification of system alignments).

DISCUSSION

The Boston Edison Company agrees with the BWR owners group position in that knowledge of plant status, especially for those systems required to mitigate the consequences of an accident, should be transferred in a systematic manner from one shift to the next. To be most effective as a means of information transfer, the information to be provided will be that which can be summarized on a single list on a single piece of paper. The information provided by the list will be reviewed by the Shift Superintendent and the Nuclear Plant Operators to insure that an adequate transfer of information occurs between cognizant individuals on each shift.

RESPONSE

1. A checklist will be devised to insure 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 developed under item 1 above will be reviewed by personnel other than the Shift Superintendent and Nuclear Plant Operators to provide an independent view of the effectiveness of the shift turnover procedure.
3. The checklist will be kept in the Control Room at all times.

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Control Room Access (Section 2.2.2.a)

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

DISCUSSION:

The Boston Edison Company agrees with the BWR owners' group position in that we believe 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.

RESPONSE:

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

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: On-site Technical Support Center (Section 2.2.2.b)

POSITION:

Each operating nuclear power plant shall maintain an on-site 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.

A complete set of as-built drawings and other records, as described in ANSI N45.2.9-1974, shall be properly stored and filed at the site and accessible to the technical support center under emergency conditions. These documents shall include, but not be limited to, general arrangement drawings, P&ID's piping system isometrics, electrical schematics, and photographs of components installed without layout specifications (e.g., field-run piping and instrument tubing).

DISCUSSION:

The Boston Edison Company is in general agreement with the NRC position stated above and as modified by the October 12, 1979, topical meeting on the Technical Support Center.

RESPONSE:

1. A location will be designated in the emergency plan. This may be a temporary location
2. Communications link will be established with the control room. These may be temporary.
3. The staffing and activation criteria will be specified in the emergency plan.
4. The TCS will have access to the records (systems, description, arrangement drawings, etc.) in accordance with the revised NUREG-0578 position.
5. The implementation criteria for the Category B schedule Technical Support Center will be issued later.

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Onsite Operational Support Center (Section 2.2.2.c)

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.

Discussion

The Boston Edison Company agrees with the BWR owners group position in that it may be more appropriate that more than one location be designated in the emergency 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.

Response

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

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Near Term Requirements for Improving Emergency Preparedness

Position

While the emergency plans of all power reactor licensees have been reviewed in the past for conformance to the general provisions of Appendix E to 10 CFR Part 50, the most recent guidance on emergency planning, primarily that given in Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants", has not yet been fully implemented by most reactor licenses. Further, there are some additional areas where improvements in emergency planning have been highlighted as particularly significant by the TMI-2 accident.

We plan to undertake an intensive effort over about the next year to improve licensee preparedness at all operating power reactors and those reactors scheduled for an operating license decision within the next year. This effort will be closely coordinated with a similar effort by the Office of State Programs to improve State and local response plans through the concurrence process and the efforts of the Office of Inspection and Enforcement to verify proper implementation of licensee emergency preparedness activities. Further, the Commission has initiated a rulemaking procedure, now scheduled for completion in January 1980, in the area of Emergency Planning and Preparedness. Additional requirements are to be expected when this rulemaking is completed and some modifications to the emergency preparedness requirements contained in this letter may be necessary.

Our near term requirements in this effort are as follows:

1. Upgrade licensee emergency plans to satisfy Regulatory Guide 1.101, with special attention to the development of uniform action level criteria based on plant parameters.
2. Assure the implementation of the related recommendations of the Lessons Learned Task Force involving instrumentation to follow the course of an accident and relate the information provided by this instrumentation to the emergency plan action levels. This will include instrumentation for post-accident sampling, high range radioactivity monitors, and improved in-plant radioiodine instrumentation. The implementation of the Lessons Learned Task Force's recommendations on instrumentation for detection of inadequate core cooling will also be factored into the emergency plan action level criteria.
3. Determine that an emergency operations center for Federal, State and local personnel has been established with suitable communications to the plant, and that upgrading of the facility in accordance with the Lessons Learned Task Force's recommendation for an in-plant technical support center is underway.
4. Assure that improved licensee offsite monitoring capabilities (including additional thermoluminescent dosimeters or the equivalent) have been provided for all sites.

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5. Assess the relationship of State/local plans to the licensee's and Federal plans so as to assure the capability to take appropriate emergency actions. Assure that this capability will be extended to a distance of ten miles. This item will be performed in conjunction with the Office of State Programs and the Office of Inspection and Enforcement.
6. Require test exercises of approved emergency plans (Federal, State, local and licensees), review plans for such exercises, and participate in a limited number of joint exercises. Tests of licensee plans will be required to be conducted as soon as practical for all facilities and before reactor startup for new licensees. Exercises of State plans will be performed in conjunction with the concurrence reviews of the Office of State Programs. As a preliminary planning bases, assume that joint test exercises involving Federal, State, local and licensees will be conducted at the rate of about ten per year, which would result in all sites being exercised once each five years. Revised planning guidance may result from the ongoing rulemaking.

Response

1. The Pilgrim Unit #1 emergency plan will be based on and will satisfy Regulatory Guide 1.101. Special attention will be given to the development of uniform action level criteria based on plant parameters.
2. Commitments in this area have been addressed in the Pilgrim Unit #1 responses to NUREG-0578. The implementation of the lessons learned recommendations on instrumentation for detection of inadequate core cooling will also be factored into the emergency plan action level criteria.
3. An Emergency Operations Center will be established for Federal, State and local personnel with suitable communications between the plant and the Emergency Operations Center. As indicated in the responses an inplant technical support center will be provided.
4. The existing offsite monitoring program will be augmented by the addition of thermoluminescent dosimeters. In addition we intend to install and operate a pressurized ion chamber. Combined operation of this device will depend on our evaluation of its contribution to the offsite monitoring program.
5. Boston Edison is currently cooperating with the Commonwealth of Massachusetts in its development of an emergency action plan out to a radius of 10 miles from Pilgrim Station. It is our understanding that the Commonwealth of Massachusetts will obtain NRC concurrence with the plan prior to 1/1/81.
6. The applicants will participate in test exercises of approved Emergency Plans (Federal, State, local, licensees). We will participate in reviews of plans for such exercises, and participate in joint exercises. We will participate in exercises of State plans to be performed in conjunction with the concurrence reviews of the Offsite of State Programs.

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