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October 17, 1979

Mr. Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
US Nuclear Regulatory Commission
Washington, DC 20555

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

Reference: Division of Operating Reactors Letter dated September 13, 1979, 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 enc' ure (6) to the above reference. The enclosure to this letter ntains our position on each request specified in your letter of S er 13, 1979.

Many of the NUREG 0578 issues are generic to all boiling water reactor facilities, and have been subjected to a review by the 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. We have participated in the evaluation of these issues through the Owners Group. Our response to your letter is a result of both the joint study associated with the Owners Group, and internal Philadelphia Electric Company studies that were of a plant specific nature. Our response is consistent with the general recommendations of the BWR Owners Group.

Complete implementation of NUREG Item 2.1.3a (Direct Indication of Valve Position), and NUREG Item 2.1.4 (Diverse Containment Isolation) will require plant outages to effect major modifications. Short term measures have been implemented

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addressing the concerns expressed by these NUREG items, and will provide adequate protection until the long term modifications can be implemented during a scheduled outage in 1980.

Well defined acceptance criteria for many of the recommendations of NUREG 0578 are needed in order to ensure timely implementation. These acceptance criteria, when fully developed as a result of discussions between the BWR Owners Group and the Nuclear Regulatory Commission, may impact implementation schedules. Additionally, hardware availability may affect the ability to optimize utilization of scheduled refueling outages for such implemenentation. Thus, we believe a degree of flexibility may be necessary in the implementation schedules.

We trust this letter is responsive to your requirements. If additional clarification of our position is necessary, please advise us.

Very truly yours,

Enclosure

PHILADELPHIA ELECTRIC COMPANY
COMMITMENTS TO NUCLEAR REGULATORY COMMISSION REQUIREMENTS

Reference: 1) NRC letter dated September 13, 1979; D. G. Eisenhut to All Operating Nuclear Power Plants; titled: "Follow up Actions Resulting from the NRC Staff Reviews Regarding the Three Mile Island Unit 2 Accident."

2) NUREG-0578, titled: "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations"

1. <u>NRC Requirement - Emergency Power Supply (NUREG - 0578, Section 2.1.1)</u>

This section describes the emergency power supply requirements for the pressurized heaters, power operated relief and block valves, and pressurizer level indicators in PWRs

RESPONSE

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. 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 powered by emergency power. They have no block valves.

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

For the reasons stated above, there is no need for action in response to Recommendation 2.1.1 for the Peach Bottom facility.

2. <u>NRC Requirement - Relief and Safety Valve Testing NUREG 0578, Section 2.1.2</u>

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.

RESPONSE

The issue of relief and safety valve testing is generic to all BWR facilities. The General Electric BWR Owners Group has developed a position on this subject and we concur with the conclusion that additional safety/relief valve qualification testing per recommendation 2.1.2 is not required. The consequences of a stuck open valve are minimal and reactor shutdown is uncomplicated, as proven by numerous field occurrences. Control systems to prevent over filling of the reactor vessel presently exist at PBAPS on the feedwater and emergency cooling water injection systems.

3. NRC Requirement - Direct Indication of Valve Position (NUREG 0578, Section 2.1.3A)

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

RESPONSE

Valve position instrumentation based on acoustic monitors, and developed under a research program financed by the Department of Energy is presently installed on all safety/relief valves on both units. This system provides reliable valve indication in the interim period until a permanent indicating system is installed.

The BWR Owners Group has been working with the General Electric Company in the selection of a reliable technique for monitoring valve position. We have participated in the evaluation of this issue through the Owners Group, and concur with their conclusions that the use of pressure switches on the discharge lines provides a single, direct, and proven technique for this function. The results of this study will be presented by the Owners Group to the Commission within the next several weeks. We will install an improved position monitoring system for all safety/relief valves in accordance with the criteria described below by January 1, 1981.

The two coded spring loaded safety valves, on each Peach Bottom primary coolant system, discharge directly to the containment. A stuck-open valve will therefore cause a rapid rise in containment pressure, causing almost immediate scram and ECCS operation and an unambiguous indication of loss of coolant. 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 main steam line break. The operator has no capability of attempting to re-seat a stuck-open safety valve from the control room and his actions would be identical to those for a main steam line 1203 255
break. For 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. Therefore, it is our conclusion that direct position indication are not required on the coded safety valves.

Since the specific criteria applicable to "safety grade" system are not clearly delineated by the NRC in NUREG 0578 or elsewhere, the following clarifies our understanding of what constitutes an acceptable "safety grade" system in this application:

a) Failure criteria - a single channel system will be provided. There will be no redundancy in sensing, signal conditioning, power supply, or indication.

b) Quality Assurance - a reliable system will be provided by procuring components of high quality. Although activities within PECO will be subject to the requirements of the PECO quality assurance program, the individual components may be procured from a vendor without a full quality assurance program. Acceptance testing will be performed to assure high quality of these components.

c) Power supply - safeguard power will be provided.

d) Qualification - Components located inside containment and in the reactor building will be tested to assure operability in the environment in which they will be installed. Components located in the control room will be seismically qualified.

e) A single alarm will be provided to indicate whenever any valve is open. The alarm circuit may not be safety-related.

f) Calibration and functional testing of the position instrumentation will be performed once per refueling cycle.

Completion: Permanent system satisfying the criteria identified above will be installed by January 1, 1981, unless precluded by material or equipment unavailability. A temporary system presently provides reliable valve position indication.

4. <u>NRC Requirement - Instrumentation for Inadequate Core Cooling (NUREG 0578, Section 2.1.3B)</u>

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.

RESPONSE

a. In response to IE Bulletin 79-08, April 14, 1979, a comprehensive review of Peach Bottom operating procedures has been completed. The review addressed the need to provide additional information and instructions to the operators for responding to plant transients, and the need to identify other plant parameter indications in evaluating plant conditions. The review included the area of instructions provided the operators to recognize inadequate core cooling using existing instrumentation. The category of operating procedures reviewed included:

   1. Plant emergency operating procedures
   2. Plant operational transient procedures
   3. Specific engineered safeguard system operating procedures.

   Completion: Neccessary revisions to operating procedures related to this NUREG requirement will be completed by January 1, 1980.

b. General Electric Company, under the direction of the Boiling Water Reactor Owners Group, is presently performing an analysis of transients and accidents identified in NUREG 0578, Section 2.1.9 and by the Bulletins and Orders Task Force as requested by the Bulletin and Order letter, dated October 1, 1979, General Electric will include consideration of the adequacy of existing instrumentation and established operating procedures as part of this study, and will recommend, if necessary, improvements in the design of plant instrumentation, and in the plant emergency procedures related to the detection and monitoring of inadequate core cooling conditions. General Electric expects to complete this study by late November, 1979.

   (1) Philadelphia Electric Company will incorporate the operational information provided by the above mentioned study, as appropriate, into the plant operating procedures and training programs.

   Completion: 60 days following NRC Acceptance of the General Electric study.

   (2) Philadelphia Electric Company will review all recommendations, and inform the Commission of those improvements it plans to incorporate into the design of plant instrumentation.

   Completion: Installation of new instrumentation - January 1, 1981, except as restrained by material or equipment unavailability.

c. Philadelphia Electric Company will incorporate the revisions to operating procedures resulting from the analysis described above into the operator training program.

Completion: 60 days following procedural revisions.

5. NRC Requirement - Diverse Containment Isolation (NUREG 0578, Section 2.1.4)

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.

RESPONSE

a. The containment isolation system design will be reviewed to ensure compliance with SRP 6.2.4 regarding diversity in the parameters sensed for the initiation of containment isolation. All systems penetrating primary containment will be evaluated and classified as "essential" or "non essential" . Systems determined to be non essential will be provided with automatic isolation, or justification for an alternate approach will be submitted to the NRC. Modifications will be performed as required

Completion: Design revew: January 1, 1980
Modifications: Next scheduled refueling outage, unless precluded by material or equipment unavailability.

b. The control systems for automatic isolation valves will be modified, as necessary, such that resetting the isolation signal will not result in the automatic reopening of these valves. Procedural controls regarding the reset of containment isolation have been revised and re-emphasized to prohibit automatic re-opening of isolation valves following reset.

Completion: Modifications - first scheduled outage in 1980, unless precluded by material or equipment unavailability.

6. NRC Requirement - Dedicated H2 Control Penetrations (NUREG 0578, Section 2.1.5.A)

For plants that have external recombiners or purge systems, provide dedicated penetrations and isolation systems that meet the redundancy and single failure requirements of the Commission regulations. Modify design as necessary so that these systems are not connected to, or are branch lines of, the large containment purge penetrations.

RESPONSE

The Containment Atmospheric Dilution (CAD) purge system at Peach Bottom is used for post-accident combustible gas control. The design of the containment penetrations associated with CAD system operation will be reviewed to verify that the isolation provisions for interconnected lines remain single failure proof during CAD operation, and that the purge lines have been properly sized.

Completion: Design review - January 1, 1980
Modifications if required - January 1, 1981, unless precluded by material or equipment unavailability.

7. NRC Requirements - Inerting BWR Containments (NUREG 0578, Section 2.1.5B)

This section describes the inerting requirements for Mark I and Mark II BWR containments.

RESPONSE

The Containment Atmospheric Dilution system provides containment inerting capabilities during accident conditions.

8. NRC Requirements - Combustible Gas Control Recombiner (NUREG 0578, Section 2.1.5C)

This section describes the requirements for the capability to install a hydrogen recombiner.

RESPONSE

In accordance with the NRC statement in the letter of September 13, 1979, no action is required for the Peach Bottom Station at this time.

9. NRC Requirements - System Integrity for High Radioactivity (NUREG 0578, Section 2.1.6A)

Perform leakage rate tests on systems outside containment that process primary coolant and could contain high level radioactive materials. Develop and implement a periodic testing program and preventive maintenance programs.

RESPONSE

Systems outside containment that would be expected to contain highly radioactive fluids during a serious transient or accident will be identified by January 1, 1980. A routine leak inspection program for these systems will be established to identify

abnormal sources of leakage. The inspection will consist of an external visual inspection, or radioactive airborne area sampling, for leakage identification while the system is pressurized. Most the systems in the program will initially be inspected quarterly. The CRD scram discharge volume will be pressurized and inspected for leaks once per operating cycle during the primary coolant system hydro test. The leakage identification program will include methods for identifying equipment to be repaired, and the frequency of inspection will be adjusted, if necessary, to adequately monitor abnormal leakage. The leakage identification program will not include pressure - decay type integrated leak rate tests to determine external lekage from the subject systems. These systems have boundaries that under normal circumstances, seperate them from the torus compartment or the reactor vessel, and because those bundaries are not required to be leak-tight, an integrated leak test of those systems is not indicative of the leakage paths of concern. The quarterly leakage identification program will be initiated during the first quarter of 1980.

:. <u>NRC Requirement</u> - <u>Plant Shielding Review (NUREG 0578, Section 2.1.6B)</u>

Perform a design review of the shielding of systems processing primary coolant outside of containment. Determine any areas or equipment that are vital for post-accident occupancy or operation and assure that access and performance will not be unduly impaired due to radiation from these systems.

<u>RESPONSE</u>

A radiation and shielding design review of vital areas and equipment will be completed in order to determine what corrective action is required, if any, to provide for adequate access to this equipment and to provide protection for this equipment during accident conditions.

Philadelphia Electric Company response to Bulletin 79-01, which addresses Environmental Qualification of Class 1E Equipment, uses Reg. Guide 1.3 for the assumption of post-accident release of radioactive materials. The response uses a total dose rate for 180 days. As indicated in our letter dated October 5, 1979, our response to Bulletin 79-01 will be completed by January 31, 1980. This response is an essential part of addressing this recommendation. The recent guidance on using new sourve terms rather than those stated in Reg. Guide 1.3 will be examined to determine the impact on equipment.

If corrective action is required to provide adequate access to and protection for vital equipment, modifications will be expedited.

Completion: Design review: January 1, 1980
Modifications: January 1, 1981, unless precluded by material or equipment unavailability.

11. NRC Requirements - Auxiliary Feedwater System (NUREG 0578, Section 2.1.7)

This section describes the auxiliary feedwater system reliability for PWRs.

RESPONSE

As described in section 6 of the Peach Bottom Final Safety Analysis Report the Emergency Core Cooling Systems are automatically initiated upon loss of the normal feedwater delivery systems. Another safety related system, Reactor Core Isolation Cooling System, is also started automatically to backup normal feedwater capabilities. Flow indication is provided in the control room for these systems.

12. NRC Requirement - Post Accident Sampling (NUREG 0578, Section 2.1.8A)

Review and upgrade the capability to obtain samples from the reactor coolant system and containment atmosphere under high radioactivity conditions. Provide the capability for chemical and spectrum analysis of high-level samples on site.

RESPONSE

The radioactive gas and liquid sampling capabilities during accident conditions will be reviewed, and modifications implemented as necessary to assure that sampling can be performed in a timely manner without incurring excessive radiation exposures. Additionally, the radiological spectrum and chemical analysis capabilities will be reviewed to assure that the appropriate analyses can be performed in a timely manner. The criteria, presented in NUREG 0578, Section 2.1.8a will be utilized as far as practical in the above evaluations. Deviations from the stated criteria, if any, will be reviewed with the Commission following completion of this study.
Completion: Design review: January 1, 1980
Modifications: January 1, 1981 unless precluded by material or equipment unavailability.

13. NRC Requirement - High Range Effluent Monitor (NUREG 0578, Section 2.1.8B)

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.

RESPONSE

a. A method for estimating noble gas release rates from the off-gas stack and reactor building exhaust vents will be developed for concentrations above the range of existing installed instrumentation. High level radioiodine is addressed in 2.1.8b, position 'c' below.
Completion: January 1, 1980 unless precluded by material or equipment unavailability

b. High range noble gas effluent monitors will be installed to monitor effluents from the off gas stack and reactor building exhaust vents.
Completion: January 1, 1981 unless precluded by material or equipment unavailability.

c. An evaluation will be performed to determine (1) accessibility of the particulate filters and iodine cartridges in the off gas stack monitors and the reactor building exhaust vent monitors during accident conditions, and (2) the capabilities of performing a laboratory analysis during accident conditions. The evaluation will be performed as part of our review for section 2.1.8a. Modifications, or alternate provisions will be implemented as required.
Completion: Evaluation: January 1, 1980
Modifications: January 1, 1981 unless precluded by material or equipment unavailability

d. Two in-containment high range radiation level monitors will be installed.
Completion: January 1, 1981 unless precluded by material or equipment unavailability

14. NRC Requirement - Improved Iodine Instrumentation (NUREG 0578, Section 2.1.8C)

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

RESPONSE

Spectral measurements are currently performed on in-plant iodine cartridges. An evaluation will be performed to determine if this method can be utilized during accident conditions as part of our review for section 2.1.8a. Modifications or alternate provisions will be implemented as required.
Completion: Design review: January 1, 1980 Modifications: January 1, 1981 unless precluded by equipment or material unavailability

15. NRC Requirement - Transient and Accident Analysis (NUREG 0578, Section 2.1.9)

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a. Provide the analysis, mergency procedures, and training to substantj ily improve operator performance during a small break .oss-oi-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.

RESPONSE

General Electric Company, under the direction of the BWR Owners Group, is presently engaged in performing transient and accident analyses in response to NUREG 0578, section 2.1.3B (Inadequate Core Cooling) section 2.1.9 (Transients and Accidents), and the small break loss-of-coolant accident requirements specified by the Bulletins and Orders Task Force as requested in a letter from D. f. Ross to T. D. Keenan, dated October 1, 1979. The scope of these activities is as follows:

a. Analyses and guidelines for emergency procedures to substantially improve operator performance during a small break loss-of-coolant accident, were developed in response to questions of the Bulletins and Orders Task Force (B& OTF) letter dated July 13, 1979. Development of plant emergency procedures and operator training will take place after NRC acceptance of the General Electric analysis.

b. Analyses and guidelines for emergency procedures to assure that the operator can recognize and respond to conditions of inadequate core cooling are being developed in response to NUREG 0578, section 2.1.3B; and questions 13 and 14 of the B & O Task Force letter dated July 13, 1979. The schedule for this work is being coordinated with the B & O Task Force. Development of plant emergency procedures and operator training will take place after NRC acceptance of the General Electric response.

c. Analyses and guidelines for emergency procedures to substantially improve operator performance during transients and accidents, including events that are caused or worsened by inappropriate operator actions, are being developed in response to question 15 of the B & O Task Force letter dated July 13, 1979. Detailed guidelines are being developed for a complete loss of feedwater and for a stuck open relief valve. Guidelines for bringing the reactor from a post accident or post transient condition to cold shutdown are being developed. All other events analyzed in the Safety Analysis Report (SAR) which cause a reactor system transient are being examined to determine whether the SAR predictions

and operator actions would be significantly different from the real world due to the conservative assumptions employed in the SAR predictions. The schedule for this work is being coordinated with the B & O Task Force. Development of plant emergency procedures and operator training will take place after NRC acceptance of the response.

General Electric expects to complete the above studies by late November, 1979. The information derived from the preceding analyses will be included in the plant emergency procedures and operator training programs within three months following NRC acceptance of these studies.

16. NRC Requirement - Shift Supervision Responsibilities (NUREG 0578, Section 2.2.1A)

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).

RESPONSE

The Nuclear Regulatory Commission's position will be implemented as stated by January 1, 1980.

17. NRC Requirement - Shift Technical Advisor (NUREG 0578, Section 2.2.1B)

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.

RESPONSE

It is our considered opinion that the long term solution to improving the accident assessment capabilities of shift personnel is the upgrading of senior licensed operator qualifications and

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training. While this represents our long term objective, an interim method of providing the functions of the technical advisor is outlined in NUREG 0578, Section 2.2.1.b and discussed in Enclosure 2 "Alternative to Shift Technical Advisor" to your letter of September 13, 1979, will be necessary over the foreseeable future.

By January 1, 1980, an individual shall be designated as the Shift Technical Advisor on each shift. This individual shall fulfill the accident assessment function during a plant transient. The individual assigned to this position will have a bachelor's degree or equivalent in an engineering or scientific field. He will report as soon as he is alerted.

During a transient, the shift technical advisor will observe control room instrumentation and ECCS operation and availability to determine that the transient is proceeding as predicted. He will advise Shift Supervision of significant adverse conditions. After a stable condition has been achieved, he will aid shift personnel in analysis of the transient to determine cause and Technical Specification implications. At the request of Shift Supervision, the Shift Technical Advisor may also aid in reporting plant conditions to Plant Management and serve as liason with technical support personnel. The Shift Technical Advisor will have no duties or responsibilities for manipulation of controls or command of operations during the transient.

The individuals filling this position will receive special training in nuclear physics, chemistry, reactor thermodynamics, fluid mechanics, heat transfer, electrical circuitry and reactor control theory. Action has been initiated to accomplish some of this training prior to January 1, 1980. Completion of this training for all individuals assigned to this position will be completed prior to January 1, 1981.

The operating experience assessment function described in Enclosure 2 to your letter of September 13, 1979, will be provided by the Shift technical Advisors and supplemented by individuals possessing bachelor's degrees or equivalent in engineering or scientific fields from the Engineering and Research Department and Electric Production Department. These individuals would typically represent Mechanical Engineering, Electrical Engineering, Quality Assurance, and Licensing fields. Reports on these assessments will be provided to the Superintendent - Generation Division/Nuclear and to the Station Superintendent. This review process will be initiated January 1, 1980.

The above method represents our interim method of meeting the requirements of NUREG 0578, Section 2.2.1.b. Following implementation, other methods may be identified which enhance the program or are equally effective. Significant changes to the program described above will be discussed with the resident inspector prior to implementation. Any changes made will meet the criteria outlined at the NRC - Industry Topical Meeting in Bethesda on October 12, 1979.

18. NRC Requirement - Shift Turnover Procedures (NUREG 0578, Section 2.2.1C)

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.

RESPONSE

The plant procedures will be reviewed and revised to assure the following.

a. Shift turnover checklists, logs, status boards or comparable mechanisms will be developed for use by the control room operator and shift supervision to ensure the transfer of vital information.

b. Maintenance and testing of equipment vital to safe operation of the plant is performed with the knowledge and approval of the appropriate licensed control room operator. The checklists, status boards, or logs will be utilized to identify any equipment under maintenance or test that by themselves could degrade a system critical to the prevention and mitigation of operational transients and accidents, or initiate an operational transients. Some of this information will be supplied to the control room operators and supervisors, as appropriate, by the auxiliary operators and technicians for entry into the checklists and logs. Shift personnel meetings under the direction of shift supervision are normally held during each shift. The auxiliary operator's participation in these meetings includes review of the checklists and logs. These methods are considered more effective in the transfer of vital information during shift turnover than the use of separate logs by the auxiliary operators and technicians.

c. A senior member of the engineering staff holding an NRC issued operating license (for example: Engineer - Operations) will evaluate the effectiveness of ths shift and relief turnover procedure.

Completion: January 1, 1980

19. NRC Requirement - Control Room Access Control (NUREG 0578, Section 2.2.2A)

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.

RESPONSE

Access to the control room is limited to personnel whose presence in this area is required for the performance of plant operating and maintenance activities. The availability of a Technical Support Center and an Operational Support Center will further reduce the number of personnel present in the control room during emergency situations. The plant administrative procedures, as specified on page A-56 of NUREG 0578, will be reviewed and revised to incorporate the provisions of Section 2.2.2.A. Completion: January 1, 1980.

20. NRC Requirement - Onsite Technical Support Center (NUREG 0578, Section 2.2.2 B)

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 st tus and potential offsite impact in support of the control room command and control function. The center sho ld also be used in conjunction with implementation of onsite and offsite emergency plans, including communications with an offsite emergency response center. Provide at the onsite 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.

RESPONSE

A separate technical support center that has the capability to transmit plant status information to technical support personnel will be established. The location, description, and function of the center will be presented to the NRC emergency planning review team during a meeting wit the Philadelphia Electric Company, currently scheduled for lat October, 1979. The former Peach Bottom Unit 1 (HTGR) contr l room may be adapted for use as an interim technical support center by January 1, 19 0. Direct communications with the control room, and appropriate plant drawings will be provided at the interim center by this date. Additionally, we plan to provide, as soon as practicable, continuous direct and airborne radiation monitoring equipment for the interim technical support center. A feasibility study is in progress addressing the advisability of providing a closed circuit television link from the Unit 2 -3 control room, and output from the plant process computer terminal at the support center.

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The conceptual plans and implementation schedule for a permanent technical support center will be established by January 1, 1980. The Peach Bottom Emergency Plan will be revised by mid 1980, as requested, to incorporate the role and location of the technical support center.

21. NRC Requirement - Onsite Operational Support Center (NUREG 0578, Section 2.2.2 C)

    Each operating nuclear power plant should establish and maintain 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.

RESPONSE

An area(s) to be designated as the onsite operational support center will be established by January 1, 1980. The purpose of the center will be to provide a place for operating support personnel to report during an emergency situation in the event shift supervision deems it necessary to avoid unnecessary personnel congestion in the control room. The proposed location and description of the center will be discussed later this month at the meeting with the NRC emergency planning review team. The Peach Bottom Emergency Plan will be revised by mid 1980, as requested, to reflect the existence and role of the onsite operational support center.

22. NRC Requirement - Revise LCOs for Safety System Availability (NUREG 0578, Section 2.2.3)

    As indicated in the NRC letter of September 13, 1979 no action is required for the Peach Bottom Station at this time for this requirement.

23. NRC Requirement - Improve Containment Pressure, Water Level, and H2 Concentration Instrumentation - Enclosure 3 (September 13, 1979 letter)

    a. 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.

RESPONSE

A continuous indication of containment pressure will be provided in the control room. The indication will have a range from minus five psig to three times the design pressure of the containment and will meet the provisions of Regulatory Guide 1.97, including qualification, redundancy, and testability.
Completion: January 1, 1981, unless precluded by material or equipment unavailability.

b. 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.

RESPONSE

A continuous indication of containment water level will be provided in the control room. The indication will have a range from one foot above the bottom of the suppression pool to five feet above the normal water level of the suppression pool and will meet the provisions of Regulatory Guide 1.97, including qualification, redundancy, and testability.
Completion: January 1, 1981, unless precluded by material or equipment unavailability

c. 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.

RESPONSE:

The existing containment combustible gas concentration monitoring equipment will be re-evaluated. The existing hydrogen monitoring instrumentation covers the range specified by the NRC. Modifications and/or qualification programs will be undertaken as necessary to meet the design and qualification provisions of Regulatory Guide 1.97.
Completion: January 1, 1981, unless precluded by equipment or material unavailability.

24. NRC requirements - Remotely Operated High Point vents in the Primary Coolant System - Enclosure 4 (September 13, 1979)

Install reactor coolant system and reactor vessel head high point vents remotely operated from the control room.

RESPONSE

The issue of high point venting is generic to all BWR facilities. The General Electric BWR Owners Group has developed a position on this subject. We concur with the conclusions of the group that adequate reactor coolant system venting capability is provided by the existing plant design. Peach Bottom is provided with 5 power operated safey grade relief valves (ADS valves) which can be used to vent the reactor pressure vessel. The point of connection of the vent lines to the vessel is such that accumulation of gases above this elevation in the vessel will not inhibit natural circulation cooling of the reactor core. In addition, a RPV head vent, which is operable from the control room, could be used as a backup to the ADS valve venting capability.

a. We concur with the BWR Owners Group's conclusion that the single failure criteria for prevention of inadvertent actuation of these valves is not applicable to BWRs. Inadvertent opening of the safety relief valves is a design basis event and has been demonstrated to be a controllable transient.

b. Information regarding the design, qualification, power source, etc. of the safety/relief valves is presented in the Peach Bottom Final Safety Analysis Report. Sections 4.4, 6.4, 6.5, and 7.4.

c. An analysis demonstrating that the direct venting of non condensable gases will not result in violation of combustible gas concentration limits in containment is presented in FSAR supplement 1, response to question 14.7. The gas generation rates assumed in this analysis are in accordance with Safety Guide 7.

d. The review and revision of procedures in response to IE Bulletin 79-08, and discussed in our answer to section 2.1.3b, included plant procedures governing the operators use of the safety relief valves for venting the reactor pressure vessel.

25. NRC Requirements - Emergency Preparedness - Enclosure 7 (September 13, 1979 letter)

a. 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.

RESPONSE

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Revisions to the Peach Bottom Emergency Plan, considering the requirements of Regulatory Guide 1.101 revised draft of 10 CFR 50, Appendix E, and a document titled "Emergency Planning Acceptance Criteria for Licensensed Nuclear Power Plants: presented by the NRC at the August 20, 1979 Regional Meeting, are in preparation and will be presented at a meeting with the NRC review team scheduled for late October, 1979. The revised plan will be implemented, as requested, by mid 1980.

b. 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.

RESPONSE

The results of our engineering studies related to the NUREG requirements on instrumentation for post-accident sampling, high range radioactivity monitors, improved in plant radioiodine instrumentation and instrumentation for detection of inadequate core cooling will be factored into the emergency plan action level criteria. The revisions will be incorporated into the plan within three months following implementation of the modifications required by NUREG 0578, Section 2.1.8.

c. Determine that an emergency operations center for Federal, State and Local personnel has been established with suitable communcations 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.

RESPONSE

The proposed location and function of an offsite emergency operations center, as well as our proposal regarding the onsite technical support center, will be discussed at the meeting with the NRC emergency planning review team, scheduled for late October, 1979. The role and function of the center will be incorporated into the Emergency Plan by mid 1980, as requested.

d. Assure that improved licensee offsite monitoring capabilities (including additional thermoluminescent dosimeters or the equivalent) have been provided for all sites.

RESPONSE

The PBAPS offsite environmental radiological monitoring program will be reviewed to assure that the program is capable of assessing the offsite effects of an emergency situation. This review and necessary changes to the program will be accomplished by mid 1980, unless precluded by unavailability of equipment or material. The objective of the review will be to insure adequate coverage, both by media and sector.

e. Assess the relationship of State/Local plans to the licensees 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.

RESPONSE

The state and local (county) plans are being revised and draft copies have been sent to the NRC for review. Dialogue has been maintained between the licensee and the state regarding composition of emergency plans. The licensee plan will be compatible with the state and local plans, and NRC requirements by mid 1980.

f. 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

The emergency plans will be revised to require periodic joint exercises. An integrated government/licensee emergency exercise will be performed during 1980 in accordance with, and following approval of the revised emergency plans.