PDR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

CHAIRMAN

November 7, 1989

Ms. Barbara Ahern Risacher
Councilwoman, District A
County Council of Harford County
20 West Courtland Street
Bel Air, Maryland 21014

Dear Ms. Risacher:

I am responding to your letter of August 28, 1989, concerning the Nuclear Regulatory Commission's (NRC's) disposition of the recommendations in SECY-89-017, "Mark I Containment Performance Improvement Program," and the implications of our decision on containment improvements planned for the Peach Bottom Station, Units 2 and 3. The Commission, after carefully considering SECY-89-017, decided on a different course of action to ensure that the recommended improvements will be implemented, where justified, after due consideration has been given to plant-specific design differences.

In its January 23, 1989 paper, the staff recommended five improvements for Mark I containment plants: (1) improved hardened vent capability, (2) improved reactor pressure vessel depressurization system reliability, (3) an alternate water supply to the reactor vessel and drywell sprays, (4) extended emergency procedures and training, and (5) accelerated staff actions to implement the Station Blackout Rule. After considering these recommendations, the Commission directed the staff for the first item to follow an approach of approving installation of a hardened vent for licensees who, on their own initiative, elect to incorporate this plant improvement. The staff also was directed to initiate plant-specific backfit analyses for the remaining plants to evaluate the efficacy of requiring the installation of such vents. This directive has been communicated to nuclear power plant licensees by Generic Letter 89-16, "Installation of a Hardened Wetwell Vent," dated September 1, 1989 (Enclosure 1). As stated in detail in the Generic Letter, the staff believes that the available information provides strong incentive for installation of a hardened vent since it would contribute to improved accident management strategies and to a reduced likelihood of core melt. Licensees, including the Philadelphia Electric Company, have been requested to respond within 45 days of receipt of the letter with a description of their plans for addressing the resolution of this issue. The staff is currently reviewing Philadelphia Electric Company's response dated October 30, 1989 (Enclosure 3). The Commission has indicated its intent that the entire process, including implementation, be completed within three years.

With regard to public participation, the staff plans, in a related effort, to prepare a generic environmental assessment of containment venting using the improved hardware and procedures. However, with the environmental assessment incomplete, decisions concerning the appropriate staff actions and degree of public participation in this process have not yet been made.

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Regarding issues (2) through (4), the Commission believed that with the additional design-specific insights that will be gained from the Individual Plant Examination (IPE) program, licensees and the NRC staff will be in a better position to assess for each plant the risk significance of these issues relative to other identified plant vulnerabilities. Accordingly, the Commission concluded that issues (2) through (4) should be evaluated as part of the IPE program. Generic Letter 88-20, "Individual Plant Examination For Severe Accident Vulnerabilities," dated November 23, 1988, has been issued to request licensees to perform a systematic evaluation, defined as individual plant examinations, to identify and report any plant-specific vulnerabilities to severe accidents. Generic Letter 88-20 was augmented by a supplement dated August 29, 1989, to provide, among other things, a summary of the staff's conclusions and recommendations on issues (2) through (4) for consideration in each Mark I licensee's IPE (Enclosure 3).

With respect to the fifth issue, since the Station Blackout Rule was issued in the Commission's regulations (10 CFR 50.63) on July 21, 1988, all nuclear power plant licensees have submitted information regarding their implementation of the rule. The staff has developed a station blackout review process for all plants that reflects our objective of reducing the overall risk of station blackout expeditiously. Plants with Mark I containments, including the Peach Bottom plants, are included in the higher priority group of plants to be reviewed. The staff expects to complete the station blackout review for the Peach Bottom plant in the spring of 1990.

By letter dated September 25, 1989, the Philadelphia Electric Company informed you of its plans to respond to NRC's initiatives for Mark I containment improvements. The NRC will evaluate the utility's response to these initiatives and develop findings on the various aspects of the Peach Bottom containment performance.

I hope that this information will clarify the Commission's position on the Mark I Containment Performance Improvement program. If you have any further questions, please contact me or Mr. William T. Russell, Regional Administrator, NRC Region I Office, 475 Allendale Road, King of Prussia, Pennsylvania 19406.

Sincerely,

Kenneth M. Carr

Kenneth M. Carr

Enclosures:
1. Generic Letter 89-16, Installation of a Hardened Wetwell Vent
2. Philadelphia Electric Company's response to Generic Letter 89-16
3. Generic Letter 88-20, Supplement 1, Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR § 50.54(f)

ENCLOSURE 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 1, 1989

TO: ALL HOLDERS OF OPERATING LICENSES FOR NUCLEAR POWER REACTORS WITH MARK I CONTAINMENTS

SUBJECT: INSTALLATION OF A HARDENED WETWELL VENT (GENERIC LETTER 89-16)

As a part of a comprehensive plan for closing severe accident issues, the staff undertook a program to determine if any actions should be taken, on a generic basis, to reduce the vulnerability of BWR Mark I containments to severe accident challenges. At the conclusion of the Mark I Containment Performance Improvement Program, the staff identified a number of plant modifications that substantially enhance the plants' capability to both prevent and mitigate the consequences of severe accidents. The improvements that were recommended include (1) improved hardened wetwell vent capability, (2) improved reactor pressure vessel depressurization system reliability, (3) an alternative water supply to the reactor vessel and drywell sprays, and (4) updated emergency procedures and training. The staff as part of that effort also evaluated various mechanisms for implementing of these plant improvements so that the licensee and the staff efforts would result in a coordinated coherent approach to resolution of severe accident issues in accordance with the Commission's severe accident policy.

After considering the proposed Mark I Containment Performance Program (described in SECY 89-017, January 1989), the Commission directed the staff to pursue Mark I enhancements on a plant-specific basis in order to account for possible unique design differences that may bear on the necessity and nature of specific safety improvements. Accordingly, the Commission concluded that the recommended safety improvements, with one exception, that is, hardened wetwell vent capability, should be evaluated by licensees as part of the Individual Plant Examination (IPE) Program. With regard to the recommended plant improvement dealing with hardened vent capability, the Commission, in recognition of the circumstances and benefits associated with this modification, has directed a different approach. Specifically, the Commission has directed the staff to approve installation of a hardened vent under the provisions of 10 CFR 50.59 for licensees, who on their own initiative, elect to incorporate this plant improvement. The staff previously inspected the design of such a system that was installed by Boston Edison Company at the Pilgrim Nuclear Power Station. The staff found the installed system and the associated Boston Edison Company's analysis acceptable.

A copy of Boston Edison Company's description of the vent modification is enclosed for your information. For the remaining plants, the staff has been directed to initiate plant-specific backfit analyses for each of the Mark I plants to evaluate the efficacy of requiring the installation of hardened wetwell vents. Where the backfit analysis supports imposition of that requirement, the staff is directed to issue orders for modifications to install a reliable hardened vent.

The staff believes that the available information provides strong incentive for installation of a hardened vent. First, it is recognized that all affected plants have in place emergency procedures directing the operator to vent under certain circumstances (primarily to avoid exceeding the primary containment pressure limit) from the wetwell airspace. Thus, incorporation of a designated capability consistent with the objectives of the emergency procedure guidelines is seen as a logical and prudent plant improvement. Continued reliance on pre-existing capability (non-pressure-bearing vent path) which may jeopardize access to vital plant areas or other equipment is an unnecessary complication that threatens accident management strategies. Second, implementation of reliable venting capability and procedures can reduce the likelihood of core melt from accident sequences involving loss of long-term decay heat removal by about a factor of 10. Reliable venting capability is also beneficial, depending on plant design and capabilities, in reducing the likelihood of core melt from other accident initiators, for example, station blackout and anticipated transients without scram. As a mitigation measure, a reliable wetwell vent provides assurance of pressure relief through a path with significant scrubbing of fission products and can result in lower releases even for containment failure modes not associated with pressurization (i.e., liner meltthrough). Finally, a reliable hardened wetwell vent allows for consideration of coordinated accident management strategies by providing design capability consistent with safety objectives. For the aforementioned reasons, the staff concludes that a plant modification is highly desirable and a prudent engineering solution of issues surrounding complex and uncertain phenomena. Therefore, the staff strongly encourages licensees to implement requisite design changes, utilizing portions of existing systems to the greatest extent practical, under the provisions of 10 CFR 50.59.

As noted previously, for facilities not electing to voluntarily incorporate design changes, the Commission has directed the staff to perform plant-specific backfit analyses. In an effort to most accurately reflect plant specificity, the staff herein requests that each licensee provide cost estimates for implementation of a hardened vent by pipe replacement, as described in SECY 89-017. In addition, licensees are requested to indicate the incremental cost of installing an ac independent design in comparison to a design relying on availability of ac power. In the absence of such information, the staff will use an estimate of $750,000. This estimate is based on modification of prevalent existing designs to bypass the standby gas treatment system ducting and includes piping, electrical design changes, and modifications to procedures and training.

The NRC staff requests that each licensee with a Mark I plant provide notification of its plans for addressing resolution of this issue. If the licensee elects to voluntarily proceed with plant modifications, it should be so noted, along with an estimated schedule, and no further information is necessary. Otherwise, the NRC staff requests that the above cost information be provided. In either event, it requests that each licensee respond within 45 days of receipt of this letter.

This request is covered by Office of Management and Budget Clearance Number 3150-0011, which expires December 31, 1989. The estimated average burden hours are 100 person hours per licensee response, including searching data sources, gathering and analyzing the data, and preparing the required letters. These estimated average burden hours pertain only to the identified response-related matters and do not include the time for actual implementation of the requested actions. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for reducing this burden, to the Record and Reports Management Branch, Division of Information Support Services, Office of Information Resources Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555; and to the Paperwork Reduction Project (3150-0011), Office of Management and Budget, Washington, D.C. 20503.

If you have any questions regarding this matter, please contact the NRC Lead Project Manager, Mohan Thadani, at (301) 492-1427.

Sincerely,

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Description of Vent Modification at the Pilgrim Nuclear Power Station
2. List of Most Recently Issued Generic Letters

Enclosure 1

BOSTON EDISON

Pilgrim Nuclear Power Station
Rocky Hill Road
Plymouth, Massachusetts 02360

Ralph G. Bird
Senior Vice President — Nuclear

BECo 88-126
August 18, 1988

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

License DPR-35
Docket 50-293

REVISED INFORMATION REGARDING PILGRIM STATION
SAFETY ENHANCEMENT PROGRAM

Dear Sir:

Enclosed is a description of a revised design for the Direct Torus Vent System (DTVS) that was described in the "Report on Pilgrim Station Safety Enhancements" dated July 1, 1987 and transmitted to the NRC with Mr. Bird's letter (BECo 87-111) to Mr. Varga dated July 8, 1987. This revision supersedes in its entirety the Section 3.2 included in the July 1, 1987 report.

On March 7, 1988 Boston Edison Company (BECo) personnel met with Dr. Murley, Mr. Russell, and Dr. Thadani and provided a tour of SEP modifications and an informal presentation of the quantification of competing risks associated with venting the containment and conclusions drawn from these results. This presentation provided BECo the opportunity to respond to questions posed under Item 1 Section 3.2 - "Installation of A Direct Torus Vent System (DTVS)" in Mr. Varga's letter to Mr. Bird of August 21, 1987 "Initial Assessment of Pilgrim Safety Enhancement Program". The material presented was made available to the resident inspector and was included as Attachment II in NRC Inspection Report #88-12, dated May 31, 1988.

As you are aware from plant inspections we have installed the DTVS piping and portions of related control wiring. Currently, the DTVS is isolated from the Standby Gas Treatment System (SBGTS) by blind flanges installed in place of Valve AO-5025 and the DTVS rupture disk. This configuration was inspected by NRR in the performance of a technical review which focused on System, Mechanical Design and Structural Design issues. The review took place on March 2-3, 1988 as documented in NRC Inspection Report #88-07, dated May 6, 1988 and determined the installation configuration to be acceptable. We now plan to remove these blind flanges and proceed with installation of Valve AO-5025 and the DTVS rupture disk. We conclude the valve and rupture disk provide equivalent physical isolation of the DTVS piping from the SBGTS and appropriately ensure the operational integrity of the SBGTS under design basis accident conditions. Following completion of this work, we will perform a local leak rate test to verify that Valve AO-5025 is acceptably leak tight using the same method previously utilized in testing the blind flange. We also plan to complete all remaining electrical work on the DTVS in accordance with the revised design.

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On the basis of the revised Section 3.2, we conclude that the DTVS design as described in the enclosure does not require any change to the Technical Specifications and that we can proceed with installation without prior NRC approval.

Please feel free to contact me or Mr. J. E. Howard, of my staff at (617) 849-8900 if you have any questions pertaining to the design details of the DTVS.

R. G. Bird

Attachment: Section 3.2 Revision 1 "Installation Of A Direct Torus Vent System (DTVS)"

JEH/amm/2282

cc: Mr. D. McDonald, Project Manager
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, D.C. 20555

U. S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Senior NRC Resident Inspector
Pilgrim Nuclear Power Station

Attachment to BECo Letter 88-126

Section 3.2 Revision 1 "Installation Of A Direct Torus Vent System (DTVS)"

pages 14, 15, 16, 17, 18, 19, 19A, 19B

3.2 INSTALLATION OF A DIRECT TORUS VENT SYSTEM (DTVS)

3.2.1 Objective of Design Change

This design change provides the ability for direct venting of the torus to the main stack. Containment venting is one core damage prevention strategy utilized in the BWR Owners Group Emergency Procedure Guidelines (EPGs) as previously approved by the NRC and is required in plant-specific Emergency Operating Procedures (EOPs). The torus vent line connecting the torus to the main stack will provide an alternate vent path for implementing EOP requirements and represents a significant improvement relative to existing plant vent capability. For 56 psi saturated steam conditions in the torus, apporoximately 1% decay heat can be vented.

3.2.2 Design Change Description

This design change (Figure 3.2-1) provides a direct vent path from the torus to the main stack bypassing the Standby Gas Treatment System (SBGTS). The bypass is an 8" line whose upstream end is connected to the pipe between primary containment isolation valves AO-5042 A & B. The downstream end of the bypass is connected to the 20" main stack line downstream of SBGTS valves AON-108 and AON-112. An 8" butterfly valve (AO-5025), which can be remotely operated from the main control room, is added downstream of 8" valve AO-5042B. This valve acts as the primary containment outboard isolation valve for the direct torus vent line and will conform to NRC requirements for sealed closed isolation valves as defined in NUREG 0800 SRP 6.2.4. The new pipe is ASME III Class 2 up to and inclusive of valve AO-5025. Test connections are provided upstream and downstream of AO-5025.

The design change replaces the existing AC solenoid valve for AO-5042B with a DC solenoid valve (powered from essential 125 volt DC) to ensure operability without dependence on AC power. The new isolation valve, AO-5025, is also provided with a DC solenoid powered from the redundant 125 volt DC source. Both of these valves are normally closed and fail closed on loss of electrical and pneumatic power. One inch nitrogen lines are added to provide nitrogen to valves AO-5042B and AO-5025. New valve AO-5025 will be controlled by a remote manual key-locked control switch. During normal operation, power to the AO-5025 DC solenoid will also be disabled by removal of fuses in the wiring to the solenoid valve. This satisfies NUREG 0800 SRP 6.2.4, Containment Isolation System acceptance criteria for a sealed closed barrier. An additional fuse will be installed and remain in place to power valve status indication for AO-5025 in the main control room.

NUREG 0800, SRP 6.2.4, Item II.6.F allows the use of sealed closed barriers in place of automatic isolation valves. Sealed closed barriers include blind flanges and sealed closed isolation valves which may be closed remote-manual valves. SRP 6.2.4 calls for administrative control to assure that sealed closed isolation valves cannot be inadvertently opened. This includes mechanical devices to seal or lock the valve closed, or to prevent power from being supplied to the valve operator.

Consistent with SRP 6.2.4, valve AO-5025 will be a sealed closed remote manual valve under administrative control to assure that it cannot be inadvertently opened. Administrative control will be maintained by a key-locked remote manual control switch and a fuse removed to prevent power from being supplied to the valve operator. In accordance with NUREG 0737, Item II.E.4.2.7 Position 6, AO-5025 will be sealed closed and verified as such at least every 31 days.

A 20" pipe will replace the existing 20" diameter duct between SBGTS valves AON-108, AON-112 and the existing 20" pipe to the main stack. The existing 20" diameter duct downstream of AO-5042A is shortened to allow fitup of the new vent line branch connection.

A rupture disk will be included in the 8" piping downstream of valve AO-5025. The rupture disk will provide a second leakage barrier. The rupture disk is designed to open below containment design pressure, but will be intact up to pressures equal to or greater than those which cause an automatic containment isolation during any accident conditions.

The two Primary Containment Isolation Valves (PCIVs) AO-5042B and AO-5025 are placed in series with the rupture disk. No single operator error in valve operation can activate the DTVS. The rupture disk has a rupture pressure above the automatic containment high pressure trip point. Thus, the inboard PCIV (AO-5042B) will receive an automatic isolation prior to disk rupture. The inboard PCIV (AO-5042B) requires physical electrical jumper installation to open at primary containment pressure above the automatic high pressure trip point.

Valve AO-5025 will be closed whenever primary containment integrity is required and DC power to its solenoid control valve will be disconnected. Indication of valve position will be provided in the main control room even with the valve power removed. Use of the direct torus vent will be in accordance with approved EPG requirements and controlled by EOPs in the same manner as other existing containment vent paths. Prior to opening the vent valves the SBGT system will be shutdown and valves AON-108 and AON-112 (the outlet of SBGT) placed in a closed position.

New 8" vent pipe (8"-HBB-44), including valve AO-5025 is safety related. Vent piping downstream of AO-5025, including SBGTS discharge piping to main stack, is also safety related. All safety related piping will be supported as Class I. Nitrogen piping is non-safety related and will be supported as Class II/I.

The interpretation of the Class II/I designation through this report is given below:

All Class II items which have the potential to degrade the integrity of a Class I item are analyzed. Such Class II items do not require dependable mechanical or electrical functionality during SSE, only that all of the following conditions prevail:

1. The Class II items create no missiles which impact unprotected Class I items safety functions.

2. The Class II item does not deform in a way which would degrade a Class I item.

3. If the Class II item fails, then the Class I item is protected against the full impact of all missiles generated by the assumed failure of Class II items.

All electrical portions of this design are safety related except for the indicating lights on the MIMIC panel C904, the tie-ins to the annunciator, and interface with the plant computer.

3.2.3 Design Change Evaluation

3.2.3.1 Systems/Components Affected

Containment Atmospheric Control System (CACS)

The torus purge exhaust line inboard isolation valve AO-5042B and the associated 8" pipe are the components of the CACS affected by the design modification. With incorporation of the subject modification, the CACS will depend on both essential AC (for valve AO-5042A) and essential DC (for AO-5042B) to perform its purging function.

The new 8" torus vent line will be connected to existing 8" CACS piping between valves AO-5042B and AO-5042A.

Standby Gas Treatment System (SBGTS)

The SBGTS fan outlet valves (AON-108 and AON-112), ductwork from these valves to the 20" line leading to the main stack, and the 20" line leading to the main stack are the components of this system affected by the proposed change.

Valve AON-108 is normally closed, fail-open. Valve AON-112 is normally closed, fail-closed, and these valves are provided with essential DC power and local safety related air supplies.

Primary Containment Isolation System (PCIS)

Valve AO-5042B is affected by the change from AC to DC power for the solenoid and by replacement of the existing air supply with nitrogen. The addition of containment outboard isolation valve (AO-5025) will not affect the PCIS.

Primary Containment System (PCS)

Valve AO-5025 acts as the primary containment outboard isolation valve for the direct torus vent line and will conform to NRC requirements for sealed closed isolation valves as defined in NUREG 0800 SRP 6.2.4.

3.2.3.2 Safety Functions of Affected Systems/Components

Containment Atmospheric Control System

This system has the safety function of reducing the possibility of an energy release within the primary containment from a Hydrogen-Oxygen reaction following a postulated LOCA combined with degraded Core Standby Cooling System.

Standby Gas Treatment System

This system filters exhaust air from the reactor building and discharges the processed air to the main stack. The system filters particulates and iodines from the exhaust stream in order to reduce the level of airborne contamination released to the environs via the main stack. The SBGTS can also filter exhaust air from the drywell and the suppression pool.

Primary Containment Isolation System

This system provides timely protection against the onset and consequences of design basis accidents involving the gross release of radioactive materials from the primary containment by initiating automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceed pre-selected operational limits.

Primary Containment System

The primary containment system, in conjunction with other safeguard features, limits the release of fission products in the event of a postulated design basis accident so that offsite doses do not exceed the guideline values of 10 CFR 100.

3.2.3.3 Potential Effects on Safety Functions

Containment Atmospheric Control System, Standby Gas Treatment System, Primary Containment Isolation System and Primary Containment System

The improvements change the AO-5042B solenoid control from AC to DC enabling it to open (from its normally closed position) with no dependence on AC power availability. The existing air supply to AO-5042B is being replaced by nitrogen.

Ductwork at the outlet of the SBGTS is replaced with pipe and the new vent line is connected to the 20" line at the outlet of the SBGTS.

Addition of a new 8" vent line with containment isolation valve AO-5025 off the existing torus vent line could introduce a flow path under design basis conditions that could vent the containment directly to the stack bypassing the SBGTS.

3.2.3.4 Analysis of Effects on Safety Functions

An analysis of the effects on the safety functions of CACS, SBGTS, PCIS and PCS for the installation of the direct torus vent is described as follows:

The change from AC to DC control and the replacements of air with nitrogen on AO-5042B does not adversely affect the ability to open AO-5042B when the containment is being purged, or to isolate under accident conditions.

The modifications to the ductwork and 20" line leading to the main stack do not affect the design basis safety function of any of the safety related systems.

During normal plant operations, the CACS and the SBGTS do not use the torus 20" purge and vent line to perform their safety functions. The containment isolation valves are in their normally closed position, thus maintaining primary containment boundary integrity.

There are no adverse affects on the primary containment system by the addition of the DTVS. Valve AO-5025 will conform to NRC criteria for sealed closed isolation valves as defined in NUREG 0800 SRP 6.2.4 and will not affect design basis accidents. Use of the DTVS will be in accordance with the containment venting provisions of EPGs as approved by the NRC and controlled by EOPs in the same manner as other existing containment vent paths. The effects on the torus of the new 8" piping and AO-5025 have been evaluated for Mark I program loadings, using ASME BPVC Section III criteria. The remaining piping including the rupture disk was evaluated using ANSI B31.1 requirements.

During plant startup and shutdown (non-emergency condition) when the purge and vent line is in use, valve AO-5025 remains closed. In addition, the rupture disk downstream of valve AO-5025 will provide a second positive means of preventing leakage and prevent direct release up to the stack during containment purge and vent at plant startup or shutdown.

During containment high pressure conditions, the torus main exhaust line is automatically isolated by the PCIS. There is no change to the existing primary containment isolation system function for AO-5042A or AO-5042B. The sealed closed position of valve AO-5025 and the additional assurance added by the rupture disk downstream will prevent any inadvertent discharge up the stack for all design basis accident conditions.

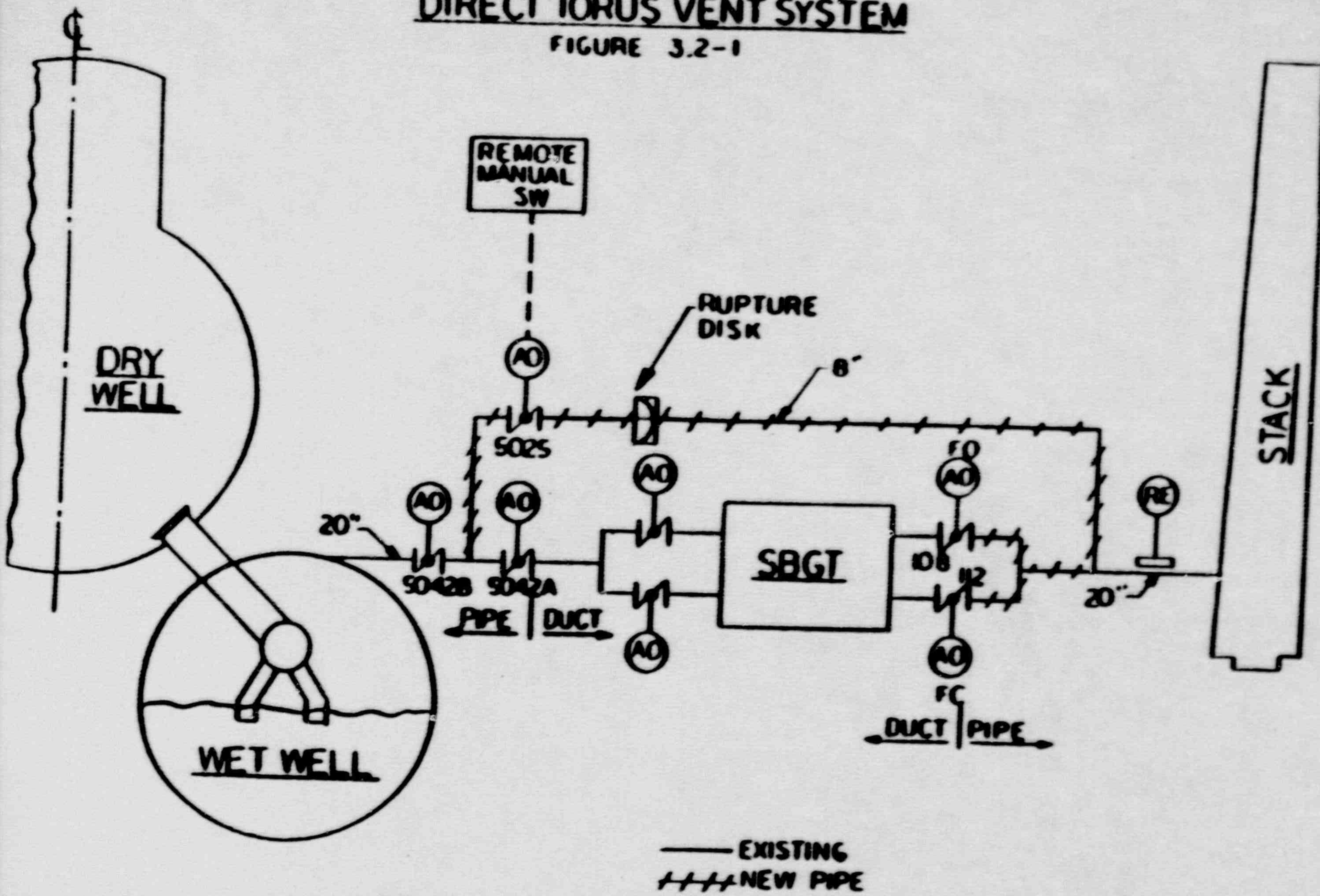
3.2.3.5 Design Change Evaluation Summary Conclusions

Installation of the DTVS does not adversely affect the safety functions of the CACS, SBGTS, PCIS or the integrity of primary containment or any other safety related systems.

Use of the DTVS will be in accordance with the containment venting provisions of EPGs as approved by the NRC and controlled by EOPs in the same manner as other existing containment vent paths. The DTVS provides an improved containment venting capability for decay heat removal which reduces potential onsite and offsite impacts relative to the existing containment venting capability.

DIRECT TORUS VENT SYSTEM

FIGURE 3.2-1



Enclosure 2

LIST OF RECENTLY ISSUED GENERIC LETTERS

Subject	Date of Issuance	Issued To
INSTALLATION OF A HARDENED WETWELL VENT (GENERIC LETTER 89-16)	09/01/89	ALL GE PLANTS
GENERIC LETTER 88-20 SUPPLEMENT NO. 1 (INITIATION OF THE INDIVIDUAL PLANT EXAMINATION FOR SEVERE VULNERABILITIES 10 CFR 50.54(f))	08/29/89	ALL LICENSEES HOLDING OPERATING LICENSES AND CONSTRUCTION PERMITS FOR NUCLEAR POWER REACTOR FACILITIES
EMERGENCY RESPONSE DATA SYSTEM GENERIC LETTER NO. 89-15 CORRECT ACCESSION NUMBER IS 8908220423	08/21/89	ALL HOLDERS OF OPERATING LICENSES OR CONSTRUCTION PERMITS FOR NUCLEAR POWER PLANTS
SUPPLEMENT 1 TO GENERIC LETTER 89-07, "POWER REACTOR SAFEGUARDS CONTINGENCY PLANNING FOR SURFACE VEHICLE BOMBS"	08/21/89	ALL LICENSEES OF OPERATING PLANTS, APPLICANTS FOR OPERATING LICENSES, AND HOLDERS OF CONSTRUCTION PERMITS
LINE-ITEMS TECHNICAL SPECIFICATION IMPROVEMENT - REMOVAL OF 3.25 LIMIT ON EXTENDING SURVEILLANCE INTERVALS (GENERIC LETTER 89-14)	08/21/89	ALL LICENSEES OF OPERATING PLANTS, APPLICANTS FOR OPERATING LICENSES, AND HOLDERS OF CONSTRUCTION PERMITS
GENERIC LETTER 89-13 SERVICE WATER SYSTEMS PROBLEMS AFFECTING SAFETY-RELATED EQUIPMENT	7/18/89	LICENSEES TO ALL POWER REACTORS BWRS, PWRS, AND VENDORS IN ADDITION TO GENERAL CODES APPLICABLE TO GENERIC LETTERS
GENERIC LETTER 89-12: OPERATOR LICENSING EXAMINATIONS	7/6/89	LICENSEES TO ALL POWER REACTORS BWRS, PWRS, AND VENDORS IN ADDITION TO GENERAL CODES APPLICABLE TO GENERIC LETTERS

ENCLOSURE 2

PHILADELPHIA ELECTRIC COMPANY

NUCLEAR GROUP HEADQUARTERS
955-65 CHESTERBROOK BLVD.
WAYNE, PA 19087-5691
(215) 640-6000

C. A. McNEILL, Jr.
EXECUTIVE VICE PRESIDENT - NUCLEAR

October 30, 1989

Docket Nos. 50-277
50-278

License Nos. DPR-44
DPR-56

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Peach Bottom Atomic Power Station, Units 2 and 3
Generic Letter 89-16, "Installation of a Hardened Wetwell Vent"

Gentlemen:

NRC Generic Letter (GL) 89-16, "Installation of a Hardened Wetwell Vent," dated September 1, 1989, required Philadelphia Electric Company (PECo) to submit a response which provides notification of our plans for addressing the hardened wetwell vent issue. GL 89-16 directed licensees with Mark I plants to voluntarily proceed with plant modifications and provide an estimated schedule in the response; otherwise, provide cost estimates for implementation of a hardened vent by pipe replacement for NRC staff use in performing plant-specific backfit analyses.

Our response, provided in the attachment, provides notification that we will proceed with plant modifications to improve the current venting capabilities at Peach Bottom Atomic Power Station. The establishment of criteria and schedule for assessing and implementing potential modifications is described in the response.

If you have any questions, or require additional information, please contact us.

Very truly yours,

Attachment

cc: W. T. Russell, Administrator, Region I, USNRC
T. P. Johnson, USNRC Senior Resident Inspector

8911140369 (3pp)

ATTACHMENT

PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3
RESPONSE TO GENERIC LETTER 89-16
"INSTALLATION OF A HARDENED WETWELL VENT"

Peach Bottom was chosen as a reference plant for the Reactor Safety Study (WASH-1400) and the recently published NUREG-1150, entitled Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants. NUREG-1150 provides a state-of-the-art understanding of severe accident risk and also provides an update of the risk from WASH-1400. Both studies concluded that the risk of a severe accident at Peach Bottom is extremely low. Changes in plant configuration and procedures, the evolution of Probabilistic Risk Assessment (PRA) methodology, and an increased understanding of severe accident phenomena have all contributed to a factor of 30 decrease in total core damage frequency (CDF) from that in WASH-1400 to that in NUREG-1150. In fact, the most dominant scenario from WASH-1400, the loss of decay heat removal (TW), has decreased three orders of magnitude due to a more realistic assessment of containment venting using existing equipment and successful injection following venting.

Peach Bottom has implemented Revision 3 of the BWROG Emergency Operating Procedures (EOP) Guidelines which incorporate the use of existing hardware to vent the primary containment. Detailed emergency procedures exist for each of the nine identified vent paths and are prioritized in order to minimize the impact of containment venting on the environment, personnel, and equipment. The hard-piped vent paths from the torus are prioritized first since they provide the best choice for satisfying the criteria of a scrubbed release with little impact on personnel and equipment. The hard-piped 6 inch Integrated Leak Rate Test (ILRT) flow path is the principal vent path capable of handling depressurization flow rates associated with decay heat. This particular flow path originates from the wetwell airspace and discharges outside the reactor building.

The emergency procedures regarding venting were used as the basis in NUREG-1150 in determining the probability of failing to successfully implement the requirements for venting. Given the time, procedures, and hardware available, a failure to vent rate of 1 in 100 (.01) was used to represent operator failure. In addition, an extensive NRC review of the venting capacity at Peach Bottom was conducted during the Emergency Operating Procedure Inspection (50-277(8)/88-200) in August, 1988. The inspection team concluded that PECo was capable of carrying out the provisions of the EOPs concerning primary containment venting using existing equipment except under the special conditions associated with station blackout.

To further improve the current venting capabilities at Peach Bottom, Philadelphia Electric Company (PECo) will proceed with plant modifications. The analysis in NUREG-CR-5225 Addendum 1 and SECY 89-017 indicate the greatest risk reduction potential from installation of a hardened vent is achieved in reducing even further the probability of the postulated loss of decay heat removal scenario (TW) assuming little credit for existing venting capability. PECo will use TW criteria (i.e., clean steam vent) as the assessment basis when determining the risk reduction potential of modifications at Peach Bottom. PECo will be working with the BWR Owner's Group to develop generic design criteria for the hardened vent. It is anticipated that the design criteria will be available for NRC review by April 30, 1990. Specific design details will be developed as PECo completes the appropriate portions of the Individual Plant Examination (IPE) for Peach Bottom and studies the possibility of systems interaction effects between the vent and existing plant design. Evaluation of containment venting impact on scenarios other than TW will be conducted during the Peach Bottom IPE process.

Using the risk reduction potential as a measure assures Philadelphia Electric will address and reduce the appropriate severe accident risk contributors while providing a plant-specific basis of assessing the most effective modifications. This maintains a continued PECo position of providing and enhancing the protection of the public health and safety.

The modifications will be implemented prior to restart following the second refueling outage (Reload 9) at each unit. These outages are currently projected to occur in the fall of 1992 for Unit 2 and fall of 1993 for Unit 3.

ENCLOSURE 3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

August 29, 1989

TO: ALL LICENSEES HOLDING OPERATING LICENSES AND CONSTRUCTION PERMITS FOR NUCLEAR POWER REACTOR FACILITIES

SUBJECT: INITIATION OF THE INDIVIDUAL PLANT EXAMINATION FOR SEVERE ACCIDENT VULNERABILITIES-10 CFR §50.54(f) - GENERIC LETTER NO. 88-20, SUPPLEMENT NO. 1

This letter announces the availability of NUREG-1335, "Individual Plant Examination: Submittal Guidance," (enclosed) and initiation of the Individual Plant Examination (IPE) process. In accordance with Generic Letter No. 88-20, licensees are requested to submit within 60 days from the date of the Federal Register notice announcing the availability of the enclosed guidance document, their proposed programs for completing their IPEs. The proposed programs should be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, and should:

1. Identify the method and approach selected for performing the IPE,
2. Describe the method to be used, if it has not been previously submitted for staff review (the description may be referenced), and
3. Identify the milestones and schedules for performing the IPE and submitting the results to the NRC.

NUREG-1335 was published in draft form in January 1989 and issued for public comment. All comments received, including those made during the IPE Workshop on February 28 through March 2, 1989, and staff responses to them, may be found in Appendix C of NUREG-1335. Licensees may find it useful in preparing their initial responses to review two options discussed on the matters of internal flooding and submittal format in Appendix C, in response to comments 5.1 and 11.3 respectively.

In accordance with a recent Commission decision on staff recommendations for enhancements to BWR Mark I plants, the staff plans to communicate directly with each licensee who possesses a Mark I plant on the matter of a hardened vent path. A summary of the staff's conclusions and recommendations for other potential Mark I enhancements is given in the enclosure hereto, for consideration in each Mark I licensee's IPE. Additional information is contained in SECY 89-017, "Mark I Containment Performance Improvement Program," dated January 23, 1989. The staff expects to issue conclusions and recommendations for all other plants and containment types in about 6 months for similar consideration in IPEs.

Regulatory Basis

Generic Letter 88-20 was issued pursuant to 10 CFR §50.54(f). A copy of the 10 CFR 50.54(f) evaluation which justified issuance of Generic Letter 88-20

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is in the Public Document Room. This supplement does not change the scope of Generic Letter 88-20. Therefore, there is no additional burden associated with this letter, and an OMB clearance number is not required.

Sincerely,

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. NUREG-1335, "Individual Plant Examination: Submittal Guidance," August 1989
2. Mark I Containment Performance Improvements
3. List of Most Recently Issued Generic Letters

NATIONAL SCIENCE FOUNDATION

Meeting

Name: Committee on Equal Opportunities in Science and Engineering.

Place: National Science Foundation, 1800 G Street, NW., Washington, DC 20550.

Dates: October [illegible], 19, 20, 1989.

Times/Rooms: October 18: Subcommittee on Persons with Disabilities 9:00 a.m.—12:00 p.m., Room 540.

October 18: Subcommittee on Minorities 1:30 p.m.—4:30 p.m., Room 540.

October 19: Full Committee Meeting 9:00 a.m.—5:00 p.m., Room 540.

October 20: Subcommittee on Women 9:00 a.m.—12:00 p.m.

Type of Meeting: Open.

Contact: Mary [illegible]. Kohlerman, Executive Secretary of the CEOSE, National Science Foundation, Room 635. Telephone Number: 202-357-7066.

Purpose of Meeting: To provide advice to the Foundation on policies and activities to encourage full participation of groups currently underrepresented in scientific, engineering, professional and technical fields.

Minutes: May be obtained from the Executive Secretary at the above address.

Agenda: To review progress by the subcommittees, become familiar with successful intervention programs, and to meet with the Director and other NSF staff.

M. Rebecca Winkler,

Committee Management Officer.

8-29-89

[FR Doc. 89-20684 Filed 8-31-89; 8:45 am]

BILLING CODE 7555-01-M

Instructional Materials Development Panel Meeting

The National Science Foundation announces the following meeting:

Name: Instructional Materials Development Panel Meeting.

Date and Time: September 22, 1989, from 8:30 a.m. to [illegible]:00 p.m.

Place: National Science Foundation, 1800 G. St. NW., Washington, DC 20550. Room #1242.

Type of Meeting: Closed Meeting.

Contact Person: Alice J. Moses, National Science Foundation, 1800 G. St. NW., Washington, DC 20550, Instructional Materials Development, Room 635-A Phone (202) 357-7066.

Minutes: May be obtained from the Contract person at the above address.

Purpose of Meeting: To attend Instructional Materials Development Panel and provide advice and recommendations concerning K-12 Math. Science and Technology education.

Agenda: To review and evaluate Instructional Materials Development proposals as part of the selection process for awards.

Reason for Closing: The proposals being reviewed include information of a propriety confidential including nature, including technical information; financial data such as salaries and personal information concerning individuals associated with the proposals. These matters are within exemptions ([illegible]) and (6) of 5 U.S.C. 552b(c), Government in the Sunshine Act.

Dated: August 29, 1989.

M. Rebecca Winkler,

Committee Management Office.

[FR Doc. 89-2[illegible]25 Filed 8-31-89; 8:45 am]

BILLING CODE [illegible]-01-M

NUCLEAR REGULATORY COMMISSION

Individual Plant Examination

AGENCY: Nuclear Regulatory Commission.

ACTION: Initation of the Individual Plant Examination for Severe Accident Vulnerabilities.

SUMMARY: This notice announces the availability of NUREG-1335, "Individual Plant Examination: Submittal Guidance," and initiation of the Individual Plant Examination (IPE) process. In accordance with Generic Letter No. 88-20, licensees are requested to submit within 60 days of this notice, their proposed programs for completing their IPEs. The proposed programs should be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555 and should:

1. Identify the method and approach selected for performing the IPE.

2. Describe the method to be used, if it has not been previously submitted for staff review (the description may be referenced), and

3. Identify the milestones and schedules for performing the IPE and submitting the results to the NRC.

A copy of the IPE submittal guidance (NUREG-1335) is available for inspection and/or copying in the NRC Public Document Room, 2120 L Street NW., Lower Level of the Gelman Building, Washington, DC.

FOR FURTHER INFORMATION CONTACT: John H. Flack, Office of Nuclear Regularory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Telephone (301) 492-3979.

Dated in Rockville, Maryland this 28th day of August, 1989.

For the Nuclear Regulatory Commission.

R. Wayne Houston,

Director, Division of Safety Issue Resolution, Office of Nuclear Regulatory Research.

[FR Doc. 89-20646 Filed 8-31-89; 8:45 am]

BILLING CODE 7590-01-M

[Docket No. 03[illegible]07099]

The Applied Radiant Energy Corp.; Issuance of Director's Decision Under 10 CFR Section 2.206

[License No. 45-[illegible]90-01]

Notice is hereby given that the Director, Office of Nuclear Material Safety and Safeguards, has taken action with regard to a petition for action under 10 CFR 2.206 received from Ms. Kristen Albrecht, Research Coordinator, National Coalition to Stop Food Irradiation, dated March 23, 1989, with respect to The Applied Radiant Energy Corporation (ARECO). The Petitioner requested that a proceeding be instituted to suspend the use of cesium-137 sealed sources at the ARECO facility.

The Director of the Office of Nuclear Material Safety and Safeguards has determined to deny the Petition. The reasons for this denial are explained in the "Director's Decision under 10 CFR 2.206," (DD-89-[illegible]) which is available for public inspection in the Commission's Public Document Room, 2120 L Street (Lower Level), NW., Washington, DC 20555. A copy of this decision will be filed with the Secretary for the Commission's review in accordance with 10 CFR Section 2.206(c) of the Commission's regulations. As provided by this regulation, the Decision will constitute the final action of the Commission twenty-five (25) days after the date of issuance of the decision unless the Commission on its own motion institutes a review of the decision within that time.

Dated at Rockville, Maryland this 24th day of August, 1989.

For the Nuclear Regulatory Commission.

Guy A. Arlotto,

Deputy Director, Office of Nuclear Material Safety and Safeguards.

[FR Doc. 89-2064[illegible] Filed 8-31-89; 8:45 am]

BILLING CODE 7590-01-M

Enclosure 2

Mark I Containment Performance Improvements

The NRC staff has identified certain containment performance improvements that would likely reduce the vulnerability of the Mark I containment to severe accident challenges (Ref. 1 and 2). The Commission expects that licensees of Mark I plants will seriously consider these improvements during their Individual Plant Examinations. It should be noted that these improvements should be considered in addition to improvements that stem from the evaluation and implementation of the hardened vent.

(a) Alternate Water Supply for Drywell Spray/Vessel Injection:

An important improvement would be to employ a backup or alternate supply of water and a pumping capability that is independent of normal and emergency AC power. By connecting this source to the low pressure residual heat removal system (RHR) system as well as to the existing drywell sprays, water could be delivered either into the reactor vessel or to the drywell, by use of an appropriate valving arrangement.

An alternate source of water injection into the reactor vessel would greatly reduce the likelihood of core melt due to station blackout or loss of long-term decay heat removal, as well as provide significant accident management capability.

Water for the drywell sprays would also provide significant mitigative capability to cool core debris, to cool the containment steel shell to delay or prevent its failure, and scrub airborne particulate fission products from the atmosphere.

A review of some BWR Mark I facilities indicates that most plants have one or more diesel driven pumps which could be used to provide an alternate water supply. The flow rate using this backup water system may be significantly less than the design flow rate for drywell sprays. The potential benefits of modifying the spray headers to assure a spray were compared to having water run out of the spray nozzles. Fission product removal in the small crowded volume in which the sprays would be effective was judged to be small compared with the benefit of having a water pool on top of the core debris.

(b) Enhanced Reactor Pressure Vessel (RPV) Depressurization System Reliability:

The Automatic Depressurization System (ADS) consists of relief valves which can be manually operated to depressurize the reactor coolant system. Actuation of the ADS valves requires DC power and pneumatic supply. In an extended station blackout after station batteries have been depleted, the ADS would not be available and the reactor would be re-pressurized. With enhanced RPV depressurization system reliability, depressurization of the reactor coolant system would have a greater degree of assurance. Together with a low pressure alternate source of water injection into the reactor vessel, the major benefit of enhanced RPV depressurization reliability would be to provide an additional source of core cooling which could significantly reduce the likelihood of high pressure severe accidents, such as from the short-term station blackout.

Another important benefit is in the area of accident mitigation. Reduced reactor pressure would greatly reduce the possibility of core debris being expelled under high pressure, given a core melt and failure of the reactor pressure vessel. Enhanced RPV depressurization system reliability would also delay containment failure and reduce the quantity and type of fission products ultimately released to the environment. In order to increase reliability of the RPV depressurization system, assurance of electrical power beyond the requirements of existing regulations may be necessary. Performance of the cables needs to be reviewed for temperature capability during severe accidents as well as the capacity of the pneumatic supply.

(c) Emergency Procedures and Training:

NRC has recently reviewed and approved Revision 4 of the BWR Owners Group EPGs (General Electric Topical Report NEDO-31331, BWR Owner's Group "Emergency Procedure Guidelines, Revision 4," March 1987).

Revision 4 to the BWR Owners Group EPG is a significant improvement over earlier versions in that they continue to be based on symptoms, they have been simplified, and all open items from previous versions have been resolved. The BWR EPGs extend well beyond the design bases and include many actions appropriate for severe accident management.

The improvement to EPGs is only as good as the plant-specific EOP implementation and the training that operators receive on use of the improved procedures. The NRC staff encourages licensees to implement Revision 4 of the EPGs and recognize the need for proper implementation and training of operators.

1. E. Claiborne et al., "Cost Analysis for Potential BWR Mark I Containment Improvements," Science and Engineering Associates Inc., NUREG/CR-5278, SEA 87-253-07-A:1, January 1989.

2. Wagner, K. C. et al., "An Overview of BWR Mark I Containment Venting Implications, Addendum 1: An Evaluation of Potential Mark I Containment Improvements, NUREG/CR-5225 Addendum 1, July 1989.

Enclosure 3

LIST OF RECENTLY ISSUED GENERIC LETTERS

Generic Letter No.	Subject	Date of Issuance	Issued To
88-20 SUPPLEMENT 1	GENERIC LETTER 88-20 SUPPLEMENT NO. 1 (INITIATION OF THE INDIVIDUAL PLANT EXAMINATION FOR SEVERE VULNERABILITIES 10 CFR 50.54(f))	08/29/89	ALL LICENSEES HOLDING OPERATING LICENSES AND CONSTRUCTION PERMITS FOR NUCLEAR POWER REACTOR FACILITIES
89-15	EMERGENCY RESPONSE DATA SYSTEM GENERIC LETTER NO. 89-15	08/21/89	ALL HOLDERS OF OPERATING LICENSES OR CONSTRUCTION PERMITS FOR NUCLEAR POWER PLANTS
89-07	SUPPLEMENT 1 TO GENERIC LETTER 89-07, "POWER REACTOR SAFEGUARDS CONTINGENCY PLANNING FOR SURFACE VEHICLE BOMBS"	08/21/89	ALL LICENSEES OF OPERATING PLANTS, APPLICANTS FOR OPERATING LICENSES, AND HOLDERS OF CONSTRUCTION PERMITS
89-14	LINE-ITEMS TECHNICAL SPECIFICATION IMPROVEMENT - REMOVAL OF 3.25 LIMIT ON EXTENDING SURVEILLANCE INTERVALS (GENERIC LETTER 89-14)	08/21/89	ALL LICENSEES OF OPERATING PLANTS, APPLICANTS FOR OPERATING LICENSES, AND HOLDERS OF CONSTRUCTION PERMITS
89-13	GENERIC LETTER 89-13 SERVICE WATER SYSTEMS PROBLEMS AFFECTING SAFETY-RELATED EQUIPMENT	7/18/89	LICENSEES TO ALL POWER REACTORS BWRS, PWRS, AND VENDORS IN ADDITION TO GENERAL CODES APPLICABLE TO GENERIC LETTERS
89-12	GENERIC LETTER 89-12: OPERATOR LICENSING EXAMINATIONS	7/6/89	LICENSEES TO ALL POWER REACTORS BWRS, PWRS, AND VENDORS IN ADDITION TO GENERAL CODES APPLICABLE TO GENERIC LETTERS
89-11	GENERIC LETTER 89-11: RESOLUTION OF GENERIC ISSUE 101 "BOILING WATER REACTOR WATER LEVEL REDUNDANCY"	6/30/89	ALL BWR PLANTS & ALL LISTINGS APPLICABLE TO GENERIC LETTERS & VENDORS, ETC.