



PECO NUCLEAR

A Unit of PECO Energy

Station Support Department

NRC GL 96-06

PECO Energy Company
965 Chesterbrook Boulevard
Wayne, PA 19087-5601

January 28, 1997

Docket Nos. 50-277
50-278
50-352
50-353

License Nos. DPR-44
DPR-56
NPF-39
NPF-85

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Peach Bottom Atomic Power Station, Units 2 and 3
Limerick Generating Station, Units 1 and 2
Response to NRC Generic Letter 96-06
"Assurance of Equipment Operability and
Containment Integrity During Design-Basis
Accident Conditions."

Gentlemen:

On September 30, 1996, the NRC issued Generic Letter (GL) 96-06 "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions." The NRC requested that all addressees submit, within 30 days of the date of the GL, a written response, and within 120 days a written summary report. PECO Energy satisfied the 30 day response by letter to the NRC dated October 29, 1996. Attached is the PECO Energy summary report stating actions taken in response to the GL requested actions, conclusions that were reached relative to susceptibility, the basis for continued operability, any corrective actions that have been taken or are planned to be implemented, and a description of the specific circumstances involving systems found to be susceptible relative to GL 96-06. This summary report satisfies PECO Energy Company's 120 day required response, stipulated in GL 96-06, for Peach Bottom Atomic Power Station, Units 2 and 3, and Limerick Generating Station, Units 1 and 2.

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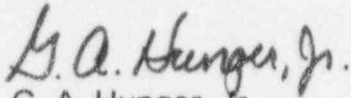
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Contained in Attachment 1 of this letter is PECO Energy's written summary report. The report is being submitted under affirmation, and the required affidavit is enclosed.

If you have any questions regarding this submittal, please contact us.

Very truly yours,



G. A. Hunger, Jr.
Director - Licensing

cc: H. J. Miller, Administrator, Region I, USNRC
W. L. Schmidt, USNRC Senior Resident Inspector, PBAPS
N. S. Perry, USNRC Senior Resident Inspector, LGS
R. R. Janati, Pa Bureau of Radiation Protection

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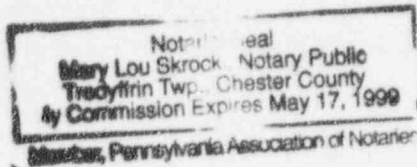
COUNTY OF CHESTER :

D. B. Fетters, being first duly sworn, deposes and says: that he is Vice President of PECO Energy Company, the Applicant herein; that he has read the enclosed response to NRC Generic Letter 96-06 "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," for Peach Bottom Atomic Power Station, Units 2 and 3, Facility Operating License Nos. DPR-44 and DPR-56, and Limerick Generating Station, Units 1 and Unit 2, Facility Operating License Nos. NPF-39 and NPF-85, and knows the contents thereof; and that the statements and matters set forth therein are true and correct to the best of his knowledge, information, and belief.



Subscribed and sworn to
before me this ^{28th} day
of January 1997.

Mary Lou Skrocki
Notary Public



ATTACHMENT 1

PEACH BOTTOM ATOMIC POWER STATION

UNITS 2 and 3

Docket Nos.

50-277

50-278

License Nos.

DPR-44

DPR-56

LIMERICK GENERATING STATION

UNITS 1 AND 2

Docket Nos.

50-352

50-353

License Nos.

NPF-39

NPF-85

Generic Letter 96-06

SUMMARY REPORT

"Assurance of Equipment Operability and Containment Integrity
During Design-Basis Accident Conditions"

Report - 10 pages

On September 30, 1996, the NRC issued Generic Letter (GL) 96-06 "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions." The following is PECO Energy Company's summary report stating actions taken in response to the GL requested actions, conclusions that were reached relative to susceptibility, the basis for continued operability, any corrective actions that have been taken or are planned to be implemented, and a description of the specific circumstances involving systems found to be susceptible relative to GL 96-06. This summary report satisfies PECO Energy Company's 120 day required response, stipulated in GL 96-06, for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, and Limerick Generating Station (LGS), Units 1 and 2.

Generic Letter 96-06 identified three issues of concern, restated here for convenience:

1. Cooling water systems serving the containment air coolers may be exposed to the hydrodynamic effects of waterhammer during either a loss-of-coolant accident (LOCA) or a main steam line break (MSLB). These cooling water systems were not designed to withstand the hydrodynamic effects of waterhammer and corrective actions may be needed to satisfy system design and operability requirements.
2. Cooling water systems serving the containment air coolers may experience two-phase flow conditions during postulated LOCA and MSLB scenarios. The heat removal assumptions for design-basis accident scenarios were based on single-phase flow conditions. Corrective actions may be needed to satisfy system design and operability requirements.
3. Thermally induced over-pressurization of isolated water-filled piping sections in containment could jeopardize the ability of accident-mitigating systems to perform their safety functions and could also lead to a breach of containment integrity via bypass leakage. Corrective actions may be needed to satisfy system operability requirements.

The following summary report was prepared following considerable effort put forth by PECO Energy Company in order to support the 120 day response period. As a result of this effort a total of two (2) LGS containment penetrations per unit and nine (9) PBAPS containment penetrations per unit were determined to be potentially susceptible to thermally induced pressurization; however, all penetrations are considered OPERABLE. Operability determinations were performed in accordance with the guidance provided in GL 91-18 "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability."

While Operable, the final resolution of GL 96-06, issue 3 is not yet complete. The proposed timelines of all final actions or solutions to eliminate any susceptible components or systems is commensurate with the safety significance of the issues and the operability assessment contained in the following report. Our intent is either to make appropriate physical changes that provide overpressurization protection or use more specific analytical

considerations that confirm that overpressurization is not a concern on a case-by-case basis. These final actions or solutions will be determined by May 31, 1997. A completion schedule will also be determined by May 31, 1997. All final actions or solutions will be complete prior to unit restart following the first refueling outage occurring after February 1998.

PECO Energy Company will continue to support the BWR Owners Group activities in an effort to develop appropriate solutions to the issues identified in GL 96-06.

SUMMARY REPORT

In response to the concerns identified in GL 96-06, PECO Energy established a team consisting of design engineers and plant operations engineers, to review system design and operation at LGS Units 1 & 2 and PBAPS Units 2 & 3, with the following results.

ISSUES 1 and 2 - Waterhammer and Two-Phase Flow in the Cooling Water Systems Serving the Containment Air Coolers

Both LGS and PBAPS are General Electric Company Boiling Water Reactor (BWR /4) plants. The containment air coolers at LGS and PBAPS do not provide a safety function for control of containment temperature following any design basis event.

LGS Issues 1 and 2

At LGS, normal cooling water to the containment air coolers is provided by the non-safety related Drywell Chilled Water System (DCWS). In the event of a design basis event which requires containment isolation, or in the event of a loss of power, the Primary Containment Isolation Valves (PCIVs) for the cooling water supply and return will automatically close. There is no design basis event which requires operators to re-establish drywell cooling using the DCWS.

Existing emergency procedures direct plant operators to establish drywell cooling using any means available if drywell temperature or pressure become a concern. This may include the re-establishment of DCWS flow to the containment air coolers. Review of equipment arrangement and maximum expected conditions following a large break LOCA or Main Steam Line Break (MSLB) LOCA in containment has concluded that waterhammer in the drywell cooling lines is not credible at LGS due to the elevated head tank in the DCWS, which provides adequate static pressure to the system to preclude steam generation at expected temperatures. For small break LOCAs, operators will re-establish DCWS using existing emergency procedures before containment temperatures reach a point where steam generation could occur.

The containment air coolers were not sized for thermal performance during the extreme conditions that might exist in the drywell following a design basis event. Since the performance of the containment air coolers is not safety related, a decrease in performance does not have an adverse effect on the plant's design.

basis.

Operability Determination Issue -1

The Limerick Generating Station Drywell Chilled Water System (DCWS) is a non-safety related system with the exception of the primary containment isolation valves and the DCWS Containment Penetrations. It is a closed loop system whose primary function is to provide a reliable source of chilled water to components and to enable the Drywell HVAC System to maintain ambient temperatures at acceptable levels during normal plant operation. The system is isolated automatically upon receipt of a Containment Isolation Signal. Potential waterhammer in the DCWS is not credible. The Containment Isolation portion of DCWS is able to perform its safety function which is to isolate upon receipt of a Containment Isolation Signal and will ensure that the chilled water piping inside the Drywell has no adverse effects on adjacent safety related equipment in the event of a Design Basis Accident (DBA). The DCWS and Containment Isolation System are considered OPERABLE with respect to Issue 1 of GL 96-06.

Operability Determination Issue - 2

The Limerick Generating Station Drywell Chilled Water System (DCWS) is a non-safety related system with the exception of the primary containment isolation valves and the DCWS Containment Penetrations. The system is isolated automatically upon receipt of a Containment Isolation Signal. The DCWS is not safety related and has no post-LOCA design cooling functions. The DCWS is considered OPERABLE with respect to Issue 2 of GL 96-06.

PBAPS Issues 1 and 2

At PBAPS, normal cooling water to the containment air coolers is provided by the non-safety related Drywell Chilled Water System (DCWS). The PCIVs for the DCWS do not receive any automatic isolation signal. However, in the event of a loss of off-site power (LOOP), a three-way valve will align the Reactor Building Closed Cooling Water (RBCCW) system to provide cooling to the containment air coolers. A head tank in the RBCCW system is elevated such that static pressure in the cooling water pipe servicing the containment air coolers prohibits steam generation at expected temperatures following a large break LOCA or MSLB LOCA. The RBCCW pumps are provided with AC power from one of the emergency AC buses, which are sequenced to connect to the onsite AC power (emergency diesel generators). This will re-establish flow to the containment air coolers before containment temperatures reach a point where steam generation would occur.

When containment pressure and the DCWS header pressure are equal, indicating a potential breach in the system piping, existing emergency procedures direct plant operators to close the DCWS PCIVs. Once isolated, operations will not re-open the PCIVs due to the suspected breach.

Existing emergency procedures direct plant operators to establish drywell cooling using any means available if drywell temperature becomes a concern (except for the suspected breach discussed above). This may include the re-establishment of DCWS or RBCCW flow to the containment air coolers. Review

of equipment arrangement and maximum expected conditions following a large break LOCA or MSLB LOCA in the containment has concluded that waterhammer in the drywell cooling lines is not credible at PBAPS due to the elevated head tank in the RBCCW system, or the pressure in the DCWS expansion tank, since each would provide adequate pressure to the unit coolers to preclude steam generation at the expected temperatures. For small break LOCAs, operators will re-establish drywell cooling using existing emergency procedures before containment temperatures reach a point where steam generation could occur.

In addition, the containment air coolers were not sized for thermal performance during the extreme conditions that might exist in the drywell following a design basis event. Since the performance of the containment air coolers is not safety related, a decrease in performance does not have an adverse effect on the plant's design basis.

Operability Determination Issue - 1

The Peach Bottom Atomic Power Station Drywell Chilled Water System (DCWS) is a non-safety related system with the exception of the primary containment isolation valves and the DCWS Containment Penetrations. It is a closed loop system whose primary function is to provide a reliable source of chilled water to components and to enable the Drywell HVAC System to maintain ambient temperatures at acceptable levels during normal plant operation. The system is isolated manually from the Control Room. During a loss of off-site power (LOOP), the DCWS does not operate; however, the Reactor Building Closed Cooling Water System (RBCCW) provides cooling water flow to components inside Primary Containment. Operator direction is provided in station procedures as to when and how to isolate DCWS to containment. Potential waterhammer in the DCWS is not credible. The Containment Isolation portion of DCWS is able to perform its safety function (which is to isolate) and will ensure that the chilled water piping inside the Drywell has no adverse effects on adjacent safety related equipment in the event of a Design Basis Accident (DBA). The DCWS and Containment Isolation System are considered OPERABLE with respect to Issue 1 of GL 96-06.

Operability Determination Issue - 2

The Peach Bottom Atomic Power Station Drywell Chilled Water System (DCWS) is a non-safety related system with the exception of the primary containment isolation valves and the DCWS Containment Penetrations. The system is isolated manually from the control room. The DCWS is not safety related and has no post-LOCA design cooling functions. The DCWS is considered OPERABLE with respect to Issue 2 of GL 96-06.

ISSUE 3 - Thermally Induced Pressurization of Isolated Water-Filled Pipe Penetrating Containment

A review of all primary containment penetrations was performed for each unit to identify penetrations that may be susceptible to the proposed overpressurization. Two concerns are addressed: potential impact on any active safety function and

potential impact on the passive safety function of containment integrity.

The screening criteria for component and system susceptibility was performed in two phases. The first phase involved a screening of all penetrations to eliminate from further review penetrations with no potential for such overpressurization. Penetrations which normally process gas, steam, or a mixture of steam and water were omitted from further review provided that there was no potential for lines to be liquid filled during startup, power operation, or shutdown. Penetrations which are not capable of being isolated were also eliminated from further review provided the line is not a closed loop within containment. Finally, penetrations which are provided with thermal relief were also eliminated from further review. The second phase of the review considered physical layout of the piping, design features of the isolated devices, and the operation of the system.

Results of these reviews are provided below.

LGS Issue -3

Review of all liquid water and gas filled piping which penetrates containment, for each unit, concluded that there is no isolated water or gas filled section of pipe inside containment which, upon either normal or design basis containment heat-up, might inhibit the performance of any design basis active safety function.

A review of all pipe penetrations concluded that only two (2) penetrations per unit of the more than 180 penetrations per unit are potentially susceptible to thermally induced pressurization. These penetrations are identified and discussed in the following paragraphs.

•Penetrations X-23 & X-24

These penetrations contain the Reactor Enclosure Cooling Water (RECW) supply and return to the Recirculation Pump motor oil and seal coolers. The two normally open motor-operated gate valves outside containment close on a containment isolation signal and fail as-is on loss of power. The cooling function of this line is not safety related. Only the containment penetration and isolation function are safety related to ensure containment integrity.

Operability Determination Issue -3

A review of all penetrations at Limerick Generating Station for both Units 1 & 2 was performed to identify penetrations that are potentially susceptible to thermal expansion induced overpressurization. The RECW lines that run through penetrations X-23 and X-24 normally carry cold water, are not insulated and automatically isolate on a containment isolation signal. Current Local Leak Rate Testing (LLRT) results for HV-013-106/206 and HV-013-107/207 show leakage providing reasonable assurance that overpressurization will not occur. The lines inside containment serving the Recirculation pump motor oil and seal coolers are long. Although there are many places inside containment where a minor release may occur to relieve overpressurization, the piping is expected to yield (relieving stress) before breaching. The pipe yielding would increase the

pipe volume, decreasing the fluid pressure to below the pipe rupture pressure. Should a breach occur, there is reasonable assurance that the breach will not occur inside and outside containment simultaneously. Additionally, if a breach were to occur, the most likely piping breach will occur inside containment. This was concluded since the pipe inside containment is non-safety related and the piping at the penetration and to the PCIV is safety-related, and the length of pipe outside containment is short relative to the inboard length. Welds at the penetration and to the PCIV are safety-related welds and their integrity is verified by radiograph. At Limerick Generating Station, thermally induced overpressurization of isolated water-filled piping sections in containment will not jeopardize the ability of accident-mitigating systems to perform their safety functions or contribute to a breach of containment integrity in the event of a Design Basis Accident (DBA). All plant penetrations including X-23 and X-24 are considered OPERABLE with respect to Issue 3 of GL 96-06.

An Engineering Change Request (ECR) has been initiated to consider a final solution to avoid any thermally induced overpressure condition in the RECW lines inside containment servicing the Recirculation Pump motor oil and seal coolers.

PBAPS Issue - 3

Review of all liquid water and gas filled piping which penetrates containment, for each unit, concluded that there is no isolated water or gas filled section of pipe inside containment which, upon either normal or design basis containment heat-up, might inhibit the performance of any design basis active safety function.

Review of all pipe penetrations at each unit has concluded that only nine (9) of the more than 160 penetrations per unit are potentially susceptible to thermally induced pressurization. Of the nine penetrations, six penetrations are potentially susceptible to overpressurization after operator remote-manual isolation. The station emergency procedures may direct isolation during a DBA. The six penetrations are: N23, N24 (RBCCW to Reactor Recirculation Pump motor oil and seal coolers ; N-53, N-54, N-55, N-56 (Drywell Chilled Water to the Drywell Unit Coolers). The remaining three penetrations that were determined potentially susceptible to overpressurization automatically isolate. The three penetrations are: N-12 (Normal RHR Shutdown Cooling Letdown); N-18, N-19 (Normal Drywell Sump Pump-out). All of these nine penetrations are identified and discussed in the following paragraphs.

•Penetrations N-23 & N-24

These penetrations contain the Reactor Building Closed Cooling Water (RBCCW) supply and return to the Recirculation Pump motor oil and seal coolers. The normally-open, motor-operated gate valve outside containment on each penetration does not automatically close on a containment isolation signal and fails as-is on loss of power. The cooling function of this line is not safety related. Only the containment penetration and isolation function are safety related to ensure containment integrity.

Although this line does not automatically isolate on containment isolation or loss of power, existing procedures will direct operators to isolate this line during an event if plant conditions warrant.

Operability Determination Issue - 3

Penetrations N-23 and N-24 are for the RBCCW to the Reactor Recirculation Pump motor oil and seal coolers. Overpressurization is only possible if an operator isolates the system by closing the isolation valves. Current LLRT results for MO-2(3)-35-2(3)373 and MO-2(3)-35-2(3)374 show leakage providing reasonable assurance that overpressurization will not occur. The lines inside containment serving the Recirculation Pump motor oil and seal coolers are long. Although there are many places inside containment where a minor release may occur to relieve overpressurization, the piping is expected to yield (relieving stress) before breaching. The pipe yielding would increase the pipe volume, decreasing the fluid pressure to below the pipe rupture pressure. Should a breach occur, there is reasonable assurance that the breach will not occur inside and outside containment simultaneously. Additionally, if a breach were to occur, the most likely piping breach will occur inside containment. This was concluded since the pipe inside containment is non-safety related and the piping at the penetration and to the PCIV is safety-related, and the length of pipe outside containment is short relative to the inboard length. Welds at the penetration and to the PCIV are safety-related welds and their integrity is verified by radiograph. Thermally induced overpressurization of this isolated water-filled piping section in containment will not jeopardize the ability of accident-mitigating systems to perform their safety functions or contribute to a breach of containment integrity in the event of a Design Basis Accident (DBA). Therefore, these penetrations can perform their containment isolation function and have been determined to be OPERABLE.

An Engineering Change Request (ECR) has been initiated to consider a final solution to avoid any thermally induced overpressure condition in the RBCCW lines inside containment servicing the Recirculation Pump motor oil and seal coolers.

•Penetrations N-53, N-54, N-55, & N-56

These penetrations contain the Drywell Chilled Water System (DCWS) supply and return to the Drywell Area Cooling Coils, Recirculation Pump Motor Air Coolers, and the Drywell Equipment Sump Cooler.

The normally-open, motor-operated gate valve outside containment on each penetration does not automatically close on a containment isolation signal and fails as-is on loss of power. The cooling function of these lines is not safety related. Only the containment penetration and isolation function are safety related to ensure containment integrity.

Although these lines do not automatically isolate on containment isolation or loss of power, existing procedures will direct operators to isolate these lines during an event if plant conditions warrant.

Operability Determination Issue -3

Penetrations N-53, N-54, N-55, and N-56 are for the Drywell Chilled Water System. These lines do not automatically isolate, but fail open to the RBCCW system on loss of power. Outboard isolation valves may be closed manually from the control room, following a design basis accident. Current LLRT results for MO-2(3)-44A-2(3)201A/B and for MO-2(3)-44A-2(3)200A/B show leakage providing reasonable assurance that overpressurization will not occur. The lines inside containment serving the drywell area cooling coils are long. Although there are many places inside containment where a minor release may occur to relieve overpressurization, the piping is expected to yield (relieving stress) before breaching. The pipe yielding would increase the pipe volume, decreasing the fluid pressure to below the pipe rupture pressure. Should a breach occur, there is reasonable assurance that the breach will not occur inside and outside containment simultaneously. Additionally, if a breach were to occur, the most likely piping breach will occur inside containment. This was concluded since the pipe inside containment is non-safety related and the piping at the penetration and to the PCIV is safety-related, and the length of pipe outside containment is short relative to the inboard length. Welds at the penetration and to the PCIV are safety-related welds and their integrity is verified by radiograph. Thermally induced overpressurization of this isolated water-filled piping section in containment will not jeopardize the ability of accident-mitigating systems to perform their safety functions or contribute to a breach of containment integrity in the event of a Design Basis Accident (DBA). Therefore, these penetrations can perform their containment isolation function and have been determined to be OPERABLE.

An Engineering Change Request (ECR) has been initiated to consider a final solution to avoid any thermally induced overpressure condition in the DCWS lines inside containment.

•Penetration N-12

This penetration contains the Shutdown Cooling Letdown line of the Residual Heat Removal system. The two normally closed motor-operated gate valves are arranged with one inside containment and the other outside containment. The only safety function of these valves and this penetration is containment isolation and integrity.

Operability Determination Issue -3

Penetration N-12 is the normal RHR Shutdown Cooling twenty-inch letdown line. Both PCIVs are normally closed by Operations during plant startup when water temperature is cool. Current LLRT results for MO-2(3)-10-17 and MO-2(3)-10-18 show leakage providing reasonable assurance that overpressurization will not occur. This twenty-inch pipe is also insulated, which would decrease pipe heat-up rates from outside sources. If overpressurization did occur, the piping is expected to yield (relieving stress) before breaching. The pipe yielding would increase the pipe volume, decreasing the fluid pressure to below the pipe rupture pressure. Should a breach occur, there is reasonable assurance that the breach will not occur inside and outside containment simultaneously. Additionally, if a breach were to occur, the most likely piping breach will occur

inside containment. This was concluded since the pipe inside containment is non-safety related and the piping at the penetration and to the PCIV is safety-related, and the length of pipe outside containment is short relative to the inboard length. Finally, the short run of free piping between the penetration and the outboard isolation valve is stiffened by the penetration anchor and valve body. Therefore, this penetration can perform its containment isolation function and has been determined to be OPERABLE.

An Engineering Change Request (ECR) has been initiated to consider a final solution to avoid any thermally induced overpressure condition in the Shutdown Cooling Letdown line inside containment.

•Penetrations N-18 & N-19

These penetrations contain the Drywell Floor and Equipment Drain pump discharge to the Floor and Equipment Drain Collector Tanks, respectively. The two, normally-open, air-operated globe valves on each penetration are located outside containment and close on a containment isolation signal and fail closed on loss of power. Only the containment penetration and isolation function are safety-related to ensure containment integrity.

Thermally induced pressurization for these penetrations may occur if water is trapped between the inboard pump discharge check valve and the outboard isolation valve. The discharge line is normally open and vented by means of a vacuum relief valve downstream of the outboard isolation valve. This allows the penetration piping to drain when the pumps are not operating. Therefore, thermally induced overpressurization of these penetrations can only occur if the pump is operating at the time the design basis accident occurs.

Operability Determination Issue -3

Penetrations N-18 and N-19 for the Floor Drain and Equipment Drain pump-out lines are considered identical for this review. The outboard globe valves, AO-2(3)-20-082/083 and AO-2(3)-20-094/095 are air open and spring close. Current LLRT results for the globe valves show leakage. The leak rate on the check valves is unknown. The leakage rates of the outboard globe valves provide reasonable assurance that overpressurization will not occur. If the leak rate on the outboard globe valves was insufficient, and both check valves did not leak, the system piping and valves are expected to yield relieving stress and preventing a breach. Should a breach occur, there is reasonable assurance that the breach will not occur inside and outside containment simultaneously. Additionally, if a breach were to occur, the most likely piping breach will occur inside containment. This was concluded since the pipe inside containment is non-safety related and the piping at the penetration and to the PCIV is safety-related, and the length of pipe outside containment is short relative to the inboard length. Therefore, these penetrations can perform their containment isolation function and have been determined to be OPERABLE.

An Engineering Change Request (ECR) has been initiated to consider a final term solution to avoid any thermally induced overpressure condition in the Drywell Floor and Equipment Drain pump discharge lines inside containment.

ADDITIONAL ASSESSMENT INFORMATION

A probabilistic evaluation of containment integrity during severe accidents was performed for LGS and PBAPS in response to GL 88-20 (Individual Plant Examination). The failure of containment isolation due to the possible dynamic effects from a large break LOCA on closed loop systems penetrating containment was considered, but determined to be probabilistically negligible compared to other containment failure or containment isolation failure causes. The results of this PSA analysis were considered when evaluating the timeliness of future actions or solutions not yet completed.