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ComEd

June 4, 1997

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: Commonwealth Edison Company (ComEd) Response to Nuclear Regulatory Commission (NRC) Generic Letter (GL) 96-06, "ASSURANCE OF EQUIPMENT OPERABILITY AND CONTAINMENT INTEGRITY DURING DESIGN-BASIS ACCIDENT CONDITIONS," dated, September 30, 1996.
LaSalle County Station, Units 1 and 2
Facility Operating License NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

References:

1. NRC Generic Letter 96-06, "ASSURANCE OF EQUIPMENT OPERABILITY AND CONTAINMENT INTEGRITY DURING DESIGN-BASIS ACCIDENT CONDITIONS," dated, September 18, 1996

2. John Hosmer Letter to U.S. NRC, Response to Generic Letter 96-06, "ASSURANCE OF EQUIPMENT OPERABILITY AND CONTAINMENT INTEGRITY DURING DESIGN-BASIS ACCIDENT CONDITIONS," dated, October 28, 1996

3. John Hosmer Letter to U.S. NRC, Response to Generic Letter 96-06, "ASSURANCE OF EQUIPMENT OPERABILITY AND CONTAINMENT INTEGRITY DURING DESIGN-BASIS ACCIDENT CONDITIONS," dated, January 28, 1997

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4. John Hosmer Letter to U.S. NRC, Response to Generic Letter 96-06, "ASSURANCE OF EQUIPMENT OPERABILITY AND CONTAINMENT INTEGRITY DURING DESIGN-BASIS ACCIDENT CONDITIONS," dated, May 2, 1997

In the Reference 1 Letter the NRC staff notified all addressees about safety-significant issues that could affect containment integrity and equipment operability during accident conditions. In NRC Generic Letter 96-06 the NRC staff required all addressees to submit within 30 days of the generic letter a written response indicating if the requested actions would be completed and submitted within the requested time period. Reference 2 transmitted ComEd's 30 day response to the Generic Letter.

Reference 3 transmitted ComEd's 120-day response summarizing our actions taken in response to the requested actions. In that letter ComEd committed to provide additional details on implementing hardware changes at each of our sites. In reference 4, LaSalle County Station restated that it would provide a detailed response by May 30, 1997, which will describe the analysis and any design changes required. On May 30, 1997, we requested from the NRR Project Manager, an extension to our response submittal date to June 6, 1997. This request was granted. These details for LaSalle County Station are provided in the attachment.

To the best of my knowledge and belief, the information contained herein is true and accurate.

If there are any questions or comments concerning this letter, please refer them to Perry Barnes, Regulatory Assurance Supervisor, at (815) 357-6761, extension 2383.

Respectfully,

W. T. Subalusky
Site Vice President
LaSalle County Station

Enclosure

cc: A. B. Beach, NRC Region III Administrator
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LaSalle County Station
NRC Dockets 50-373 and 50-374

Response to NRC Generic Letter 96-06
"Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions", dated September 30, 1996

Nuclear Regulatory Commission (NRC) Request:

"Within 120 days of the date of this generic letter, addressees are requested to submit a written summary report stating actions taken in response to the requested actions noted below, conclusions that were reached relative to susceptibility for waterhammer and two-phase flow in the containment air cooler cooling water system and overpressurization of piping that penetrates containment, the basis for continued operability of affected systems and components as applicable, and corrective actions that were implemented or are planned to be implemented. If systems were found to be susceptible to the conditions that are discussed in this generic letter, identify the systems affected and describe the specific circumstances involved."

NRC Requested Actions:

Addressees are required to determine:

(1) if cooling water systems serving the containment air coolers are susceptible to either waterhammer or two-phase flow conditions during either a loss-of-coolant accident (LOCA) or a main steamline break (MSLB); and

(2) if isolated water-filled piping systems that penetrate the containment are susceptible to thermally induced overpressurization that could lead to a breech of containment integrity via bypass leakage.

Response to Item 1:

The response to Item 1 was previously provided in ComEd's letter dated January 28, 1997 from John Hosmer to the U.S. NRC.

Response to Item 2:

The potential for thermally-induced overpressurization exists when liquid-filled piping inside containment is isolated during a LOCA/MSLB. A review of all piping that penetrates containment was performed. This review identified the following systems as having liquid filled piping that may be susceptible to this phenonom:

Low Pressure Core Spray
High Pressure Core Spray
Residual Heat Removal/Low Pressure Core Injection
Main Steam Drains
Standby Liquid Control
Feedwater
Residual Heat Removal/Shutdown Path
Reactor Recirculation Loop Sampling
Reactor Water Cleanup
Reactor Core Isolation Cooling
Chilled Water
Reactor Building Closed Cooling Water
Reactor Recirculation Flow Control Valve Hydraulic Piping
Residual Heat Removal Shutdown Cooling
Control Rod Drive

Currently both units at LaSalle County Station are shutdown. Unit 1 is in a forced outage and Unit 2 is in a refueling outage. Currently, there are no operability concerns due to thermally induced overpressurization, because primary containment integrity is not required under the current plants' modes. However, Operational Conditions 1, 2, or 3 of Technical Specification 3.6.3 will not be entered until the design changes described below are implemented. Both Units are currently shutdown and have been shutdown since the issuance of the generic letter.

A detailed screening was performed of the above systems to determine which systems have containment valve arrangements that may be susceptible to thermal overpressurization. Containment valve arrangements that utilize air operated, motor operated, or solenoid operated inboard and outboard isolation valves were determined to be potential candidates. Check valves were excluded, because this type of valve would open and release the pressure increase prior to an overpressurization condition. Containment valve configurations that utilized a manual valve on the inboard side of containment were excluded, because these valves are administratively controlled to be open during plant operation; therefore, the overpressurization condition would not occur.

Several penetrations were determined to be susceptible to the overpressurization conditions as discussed in the generic letter. The penetration descriptions are as follows and are applicable for Units 1 and 2:

Penetration Number	Description
M-25, M-26, M-27, M-28	Supply & Return Chilled Water piping for the Primary Containment Ventilation Heat Exchanger Coil
M-16, M-17	Supply & Return Reactor Building Closed Cooling Water Piping for the seals of the Reactor Recirculation Pump
M-7	Residual Heat Removal (RHR) Piping shutdown cooling mode RHR Pump Suction Piping from Reactor Recirculation
M-49, M-50	Reactor Recirculation Flow Control Valve Hydraulic Piping
M-30	Suction Piping to the Reactor water Cleanup (RT) Recirculation Pumps
M-36	Reactor Recirculation Sample Line Piping

These penetrations generally have a containment isolation valve on each side of the containment wall which is either open or closed during normal plant operation and automatically closes on a containment isolation signal. These penetrations could be potentially heated during a LOCA or a MSLB inside containment, either of which would provide the containment isolation signal.

It has been shown via generic analysis performed for ComEd Nuclear Plants that the piping stress design allowable will be exceeded and permanent strain may develop; however, ultimate strength for the materials involved would not be exceeded, if no changes to the piping systems were implemented. The piping would maintain its structural integrity. Pressure relief would occur through the weakest link, which was determined to be the valves in the system. Pressure relief would occur through the valve seat, packing, or body to flange gaskets. There are two valves used for primary containment. Either the inboard or outboard valve seat, packing, or body to flange gaskets will be the weak link and the other valve will maintain primary containment. Based on the above discussion the penetrations will continue to maintain pressure under accident conditions and will remain operable.

An extensive review of the isolated piping sections that affect the containment integrity was performed and corrective actions are being determined. LaSalle intends to install physical changes such as drilling holes in the valve disc, adding a relief valve, or a rupture type disc chamber configuration in selected applications that provide overpressure protection prior to unit startup.

Overview Methodology

A series of thermal transient analysis to determine the expansion rate of entrapped fluid was performed. The pipe segments to be analyzed penetrate the containment and are isolated during a Design Basis Accident. The postulated pipe breaks that were considered in the analysis include the recirculation line break, a main steam line break, and a small bore high energy line break. The bounding temperature profile occurs from a high energy line break in the reactor recirculation piping. However, after a period of time the small line break becomes the bounding temperature. Therefore the temperature profile for the reactor recirculation break with a current LOOP and the 0.1 ft^2 liquid line break were both analyzed to determine the worst case scenario. Once a bounding containment profile was determined, the remainder of the analysis was divided into three sections: (1) the calculation of the outside heat transfer coefficient using the Uchida approximation, (2) the calculation of the inside heat transfer coefficient by free convection, and (3) the calculation of the volumetric expansion of the water resulting from the corresponding heat transfer rates.

A summary of each affected penetration is provided below:

Penetration M-36; Reactor Recirculation Sample Line Piping
This penetration is bounded by the Recirculation System Sample Valves B33-F019 and B33-F020. Valve F019 is located inside containment and F020 is located outside containment. During normal plant operation, the isolation valves are open and the line has continuous flow. As a result the piping will be normally hot. These valves would be expected to close due to a Group 1 isolation signal during a LOCA or MSLB; however, thermal pressurization of the isolated volume would not be expected due to its higher initial temperature.

If the line is isolated for any reason prior to a LOCA or MSLB, the containment isolation valves and piping between them will cool down and the isolated volume may be susceptible to thermally induced pressurization.

Corrective Actions:

A design change will be implemented to provide a pressure relieving device for this volume prior to startup.

Penetration M-30; Suction Piping to the Reactor Water Cleanup (RT) Recirculation Pumps
The section of the RT piping that may be susceptible to thermally induced pressurization due to post-LOCA or MSLB heat-up is the containment penetration piping located between valves G33-F001 and G33-F004. Valve F001 is located inside containment and Valve F004 is located outside containment. During normal plant operation the isolation valves are open and the line has continuous flow. As a result the piping will be normally hot. If the isolation valves close due to a LOCA or MSLB, the temperature of the isolated line would not be expected to increase due to its higher initial temperature; and therefore, will not be subject to thermally induced pressurization.

If the line is isolated for any reason prior to the LOCA or MSLB, the containment isolation valves and associated piping will be cooler and the isolated volume may be susceptible to thermally induced pressurization depending upon the duration of isolation.

Corrective Actions:

A design change will be implemented to provide a pressure relieving device for this volume. Installation of this change will be completed prior to unit startup.

Penetrations M-16 and M-17; Supply & Return to the Reactor Building Closed Cooling Water Piping for the Seals of the Reactor Recirculation Pump

These penetrations have motor-operated gate valves inside and outside containment. The section of the RBCCW piping that may be susceptible to thermally induced pressurization due to post-LOCA or MSLB heat-up is the containment penetration piping located between supply valves WR029 and WR179 and return valves WR040 and WR180. Valves WR029 and WR040 are located outside containment and Valves WR179 and WR180 are located inside containment. The temperature of the supply and return lines are significantly lower than the maximum containment temperature during a DBA or MSLB. Therefore, a pressure relieving device is required for this volume.

Corrective Actions:

A design change will be implemented to provide a pressure relieving device for this volume. Installation of this change will be completed prior to unit startup.

Penetrations M-25, M-26, M-27, M-28; Supply & Return to the Chilled Water Piping for the Primary Containment Ventilation Heat Exchanger Coils

The primary containment penetrations associated with the chilled water pumps have motor operated gate valves on the outboard side of containment and motor operated butterfly valves on the inboard side. These sections of chilled water piping are susceptible to thermally induced pressurization. The specific piping susceptible due to post-LOCA or MSLB heat-up is the containment penetration piping located between supply valves VP063A and VP113A and VP063B and VP113B and return valves VP053A and VP114A and VP053B and VP114B. The temperature of the supply and return lines are significantly lower than the maximum containment temperature during a LOCA or MSLB. Therefore, a pressure relieving device is required for this volume.

Corrective Actions:

A design change will be implemented to provide a pressure relieving device for this volume. Installation of this change will be completed prior to unit startup.

Penetration M-7; Residual Heat Removal (RHR) Piping for Shutdown Cooling Mode RHR Pump Suction Piping from Reactor Recirculation

This penetration is bounded by the motor operated gate valves E12-F009 and F008. Valve E12-F009 is located inside containment and F008 is located outside containment. During normal plant operation, the isolation valves are closed and have no flow. As a result the piping will be normally cold. This section of piping is susceptible to thermally induced pressurization since it is significantly lower than the maximum containment temperature during a LOCA or MSLB. Therefore, a pressure relieving device is required for this volume.

Corrective Actions:

A design change will be implemented to provide a pressure relieving device for this volume. Installation of this change will be completed prior to unit startup.

Penetrations M-49, M-50; Reactor Recirculation Flow Control Valve B33-F060 Hydraulic Piping

These penetrations are bounded by solenoid operated valves that are open during normal plant operation. The affected piping contains Frequel EHC fluid. Frequel EHC is a synthetic phosphate ester fluid that looks and feels like a light mineral oil with good lubricating properties and excellent stability. This fluid is not to be confused with petroleum based hydraulic fluid. The fluid is fire resistant and can contain as much as 20% water during operation. Frequel consists of 15 - 25% triphenyl phosphate, 40 - 50% butylated triphenyl phosphate mixture, and 40 - 50 % trixylenyl phosphate. The possible presence of as much as 20% water adversely affects the volumetric expansion of this material. In determining the thermal relief rate for the isolated piping, the volumetric expansion of pure water was used because of maximum expansion.

The maximum operating temperature of the fluid per design in the affected piping is 150°F. This temperature is significantly lower than the maximum containment temperature a LOCA or MSLB. Therefore, a pressure relieving device is required for this volume.

Corrective Actions:

A design change is required to provide a pressure relieving device for this volume. However, it is not appropriate to relieve Frequel EHC inside the drywell. The appropriate fix for these penetrations has not been determined; but, will be implemented prior to unit startup.