Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515-5701

January 28, 1997

ComEd

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

Subject: Braidwood Station Units 1 and 2
Byron Station Units 1 and 2
Dresden Station Units 2, and 3
LaSalle County Station Units 1 and 2
Quad Cities Station Units 1 and 2
Zion Station Units 1 and 2

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.

NRC Dockets 50-454 and 50-455
NRC Dockets 50-456 and 50-457
NRC Dockets 50-237 and 50-249
NRC Dockets 50-373 and 50-374
NRC Dockets 50-254 and 50-265
NRC Dockets 50-295 and 50-304

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 USNRC, Response to Generic Letter 96-06, "ASSURANCE OF EQUIPMENT OPERABILITY AND CONTAINMENT INTERGRITY DURING DESIGN-BASIS ACCIDENT CONDITIONS," dated October 28, 1996

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 addressess 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.

A072

In GL 96-06 the NRC staff requested that all licensees implement the following actions:

Addressees are requested to determine:

(1) if containment air cooler cooling water systems are susceptible to either waterhammer or two-phase flow conditions during postulated accident conditions;

(2) if piping systems that penetrate the containment are susceptible to thermal expansion of fluid so that overpressurization of piping could occur.In addition to the individual addressee's
postulated accident conditions, these items should be reviewed with respect to the scenarios referenced in the generic letter.

If systems are found to be susceptible to the conditions discussed in this generic letter, addressees are expected to assess the operability of affected systems and take corrective action as appropriate in accordance with the requirements stated in 10 CFR Part 50 Appendix B and as required by the plant Technical Specifications.

In GL 96-06, the NRC staff requested that all licensees provide a written response describing the following information:

1. Within 30 days from the date of this generic letter, addressees are required to submit a written response indicating: (1) whether or not the requested actions will be completed, (2) whether or not the requested information will be submitted and (3) whether or not the requested information will be submitted within the requested time period. Addressees who choose not to complete the requested actions, or choose not to submit the requested information, or are unable to satisfy the requested completion date, must describe in their response any alternative course of action that is proposed to be taken, including the basis for establishing the acceptability of the proposed alternative course of action and the basis for continued operability of affected systems and components as applicable.

2. 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 above, conclusions that were reached relative to susceptibility for waterhammer and two-phase flow in the containment air cooler cooling water system and overpressurizatrion 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.

ComEd has reviewed the requested information provided within Generic Letter 96-06. ComEd's 120-day response (Item 2 above) summarizing our actions taken in response to the requested actions as described above for each of our six stations is provided in Attachments 1 through 6 to this letter.

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

Please direct any questions concerning this response to this office.

Sincerely,

John B. Hosmer

John B. Hosmer
Vice President

Signed before me on this 28 day,
of January, 1996.

Sherry L. Butterfield
Notary Public

OFFICIAL SEAL
SHERRY L BUTTERFIELD
NOTARY PUBLIC STATE OF ILLINOIS
MY COMMISSION EXPIRES:05/23/00

cc: A. Beach, Regional Administrator-RIII
R. Capra, Director of Directorate III-2, NRR
G. Dick, Byron Project Manager, NRR
R. Assa, Braidwood Project Manager, NRR
J. Stang, Dresden Project Manager, NRR
D. Skay, LaSalle Project Manager, NRR
R. Pulsifer, Quad Cities Project Manager, NRR
C. Shiraki, Zion Project Manager, NRR
C. Phillips, Senior Resident Inspector (Braidwood)
S. Burgess, Senior Resident Inspector (Byron)
C. Vanderniet, Senior Resident Inspector (Dresden)
M. Huber, Senior Resident Inspector (LaSalle)
C. Miller, Senior Resident Inspector (Quad Cities)
R. Westberg, Acting Senior Resident Inspector (Zion)
Office of Nuclear Facility Safety - IDNS

Attachment 1
Braidwood Station
NRC Docket 50-456 and 50-457
Response to NRC Generic Letter 96-06,
"Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," dated September 30, 1996

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 above, 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 penetrated 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 requested to determine:

if containment air cooler cooling water systems are susceptible to either waterhammer or two-phase flow conditions during postulated accident conditions;

if piping systems that penetrate the containment are susceptible to thermal expansion of fluid so that overpressurization of piping could occur."

RESPONSE TO ITEM 1:

Containment air cooler cooling water systems were reviewed to determine if they were susceptible to either waterhammer or two-phase flow during postulated accident conditions. Analytical models have been developed using RELAP5 to evaluate the effects of post accident temperatures on the Reactor Containment Fan Coolers (RCFCs). Preliminary calculations have been performed using Loss of Coolant Accident (LOCA) containment temperature profiles and work is in progress to finalize the pressure loads calculation. The final calculations will include cases covering a Main Steam Line Break (MSLB) as well as a LOCA. The purpose of this calculation is to quantify the thermal hydraulic behavior of the service water side of the RCFCs during the initial phases of the design basis LOCA with a Loss of Offsite Power (LOOP). In this situation, the service water flow will decay rapidly upon loss of offsite power and allow boiling to occur in the RCFC tubes, prior to restart of the service water flow during the diesel generator loading sequence. This calculation is expected to demonstrate that, for a representative piping configuration, some voiding will occur, particularly in the upper coils, but that conditions leading to severe waterhammer are not anticipated.

The reason that low dynamic loads are expected for Braidwood is based on the physical characteristics of Braidwood's systems, with open loop cooling and water sources that have elevations higher than the top of the RCFCs. This geometry leads to a situation in the postulated scenario in which some voiding occurs in the fan cooler coils, but with heat transport to the vertical runs of piping leading to the entrance and exit of the coils. The pump restart then causes a collapse of the voids with water that is already at elevated temperatures, yielding relatively mild pressure changes. The preliminary results show that the two phase effects during pump start yield approximately 20 psi pressure loads, in addition to the normal pump start load. The RCFC cooling coils are designed for 200 psig, while the maximum pressure associated with the transient is 50 psig. This pressure is well within the design limits of the RCFC units. Final calculations will be completed by May 31, 1997.

In summary, Braidwood RCFCs are susceptible to the postulated conditions discussed in this Generic Letter (GL). An operability evaluation has been completed based on the preliminary calculations discussed above. Final calculations will be completed by May 31, 1997. No physical changes are expected at this time.

RESPONSE TO ITEM 2:

Piping systems that penetrate containment were evaluated to determine if they were susceptible to thermal expansion of fluid so that overpressurization of piping could occur. A screening evaluation was performed to identify containment piping penetrations potentially susceptible to thermal overpressurization as discussed in the Generic Letter. This evaluation resulted in the issuance of four preliminary calculations. The first calculation (Calculation 1) determined the maximum temperature change in water trapped between two closed valves during accident conditions for a variety of containment penetration configurations. The second calculation (Calculation 2) developed a methodology to calculate an incremental pressure increase per °F as a function of pipe outside diameter to wall thickness ratio and final water temperature. Pressure changes in the isolated pipe sections could then be predicted using the pipe material and size, the pressure increase per °F (from Calculation 2), and the calculated temperature rise (from Calculation 1). The third calculation (Calculation 3) determined the maximum permissible piping pressures for design, upset, emergency, and faulted conditions to determine the pressure retaining capability of isolated pipe sections under thermal pressurization conditions. The final calculation reviewed containment penetrations and compared the predicted pressure increase with the maximum permissible piping pressures (from Calculation 3).

Several penetrations were determined to be susceptible to the overpressurization conditions discussed in the Generic Letter. The penetration and system descriptions are as follows:

- P-5, P-6, P-8, and P-10 - Containment Chilled Water System
- P-11 - Reactor Coolant Drain Tank Pump Discharge
- P-30 - Containment Demineralized Water Supply
- P-32 - Fuel Pool Cooling Return to Refueling Cavity
- P-34 - Containment Fire Protection Supply
- P-37 - Reactor Coolant System Loop Fill Header
- P-44 - Primary Water Supply to Reactor Coolant Pump Seal #3 and Pressurizer Relief Tank
- P-47 - Containment Floor Drain Sump Pump Discharge
- P-55 - Safety Injection Accumulator Fill Line
- P-57 - Fuel Pool Cooling Suction from Refueling Cavity

These penetrations generally have a containment isolation valve on each side of the containment wall which is closed during normal plant operation (P-30, P-32, P-37, P-55, and P-57) or automatically closes on a containment isolation signal (P-5, P-6, P-8, P-10, P-11, P-34, P-44, and P-47). These penetrations could be potentially heated during a LOCA or a MSLB inside containment, either of which would provide the containment isolation signal.

The basis for operability has been determined for these affected penetrations which includes consideration of one or more of the following: expansion of the trapped fluid in voided areas of the isolated piping section, leakage from one of the containment isolation valves (seat, packing, or body to bonnet flange), consideration of existing insulation of piping inside containment or heat dissipation from piping outside containment to prevent the temperature induced pressure increase, lifting of air operated valves due to the pressure increase and plastic straining of the affected pipe to accommodate the pressure increase. Based on these considerations, maximum permissible pressures, as allowed by Generic Letter 91-18 for operability determination, are not expected to be exceeded.

An extensive review of isolated piping sections that affect containment integrity has been completed. While the susceptible penetrations have been determined to be operable, final corrective actions have not been determined. Braidwood intends to either install physical changes that provide overpressure protection or use penetration specific analytical evaluations that clearly address and confirm overpressurization is not a concern. Alternative solutions are being evaluated and additional time is necessary to assure the correct resolution for each isolated penetration is identified. Braidwood will provide a response by May 31, 1997 which will include details on what changes, if any, are being made to each affected penetration. At that time, the answers to questions on material availability, design package development and completion schedules, and schedule windows for implementation of the subject changes will be known. This will enable Braidwood to provide committed schedule dates for implementing the subject changes. Braidwood Station will continue to support industry activities and partnerships, such as NEI, EPRI, and BWROG, in an effort to develop a long term solution to this issue.

Attachment 2

Byron Station
NRC Docket 50-454 and 50-455
Response to NRC Generic Letter 96-06,
"Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions", dated September 30, 1996

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 above, conclusions that were reached relative to susceptibility for waterhammer and two-phase flow in the containment air cooler cooling water system and overpressurizatrion 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 requested to determine:

if containment air cooler cooling water systems are susceptible to either wate hammer or two-phase flow conditions during postulated accident conditions:

if piping systems that penetrate the containment are susceptible to thermal expansion of fluid so that overpressurization of piping could occur."

RESPONSE TO ITEM 1:

Containment air cooler cooling water systems were reviewed to determine if they were susceptible to either waterhammer or two-phase flow during postulated accident conditions. Analytical models have been developed using RELAP5 to evaluate the effects of post accident temperatures on the Reactor Containment Fan Coolers (RCFCs). Preliminary calculations have been performed using LOCA containment temperature profiles and work is in progress to finalize the pressure loads calculation. The final calculations will include cases covering a MSLB as well as a LOCA. The purpose of this calculation is to quantify the thermal hydraulic behavior of the service water side of the RCFCs during the initial phases of the design basis LOCA with a LOOP.

In this situation, the service water flow will decay rapidly upon loss of offsite power and allow boiling to occur in the RCFC tubes, prior to restart of the service water flow during the diesel generator loading sequence. This calculation is expected to demonstrate that for a representative piping configuration, that some voiding will occur, particularly in the upper coils, but that conditions leading to severe water hammer are not anticipated.

The reason that low dynamic loads are expected for Byron is based on the physical characteristics of Byron's systems, with open loop cooling and water sources that have elevations higher than the top of the RCFCs. This geometry leads to a situation in the postulated scenario in which some voiding occurs in the fan cooler coils, but with heat transport to the vertical runs of piping leading to the entrance and exit of the coils. The pump restart then causes a collapse of the voids with water that is already at elevated temperatures, yielding relatively mild pressure changes. The preliminary results show that the two phase effects during pump start yield approximately 20 psi pressure loads, in addition to the normal pump start load. The RCFC cooling coils are designed for 200 psig, while the maximum pressure associated with the transient is 50 psig, which is well within the design limits of the RCFC units.

In summary, Byron RCFCs are susceptible to the postulated conditions discussed in this Generic Letter (GL). An operability evaluation has been completed based on the preliminary calculations discussed above. Final calculations will be completed by May 31, 1997. No physical changes are expected at this time.

RESPONSE TO ITEM 2:

Piping systems that penetrate containment were evaluated to determine if they were susceptible to thermal expansion of fluid so that overpressurization of piping could occur. An evaluation was performed by ComEd and Bechtel to identify containment piping penetrations potentially susceptible to thermal overpressurization as discussed in the Generic Letter. This evaluation resulted in the issuance of four preliminary calculations. The first calculation (Calculation 1) determined the maximum temperature change in water trapped between two closed valves during accident conditions for a variety of containment penetration configurations. The second calculation (Calculation 2) developed a methodology to calculate an incremental pressure increase per °F as a function of pipe outside diameter to wall thickness ratio and final water temperature. Pressure changes in the isolated pipe sections could then be predicted using the pipe material and size, the pressure increase per °F (from Calculation 2), and the calculated temperature rise (from Calculation 1). The third calculation (Calculation 3) determined the maximum permissible piping pressures for design, upset, emergency, and faulted conditions to determine the pressure retaining capability of isolated pipe sections under thermal pressurization conditions. The final calculation reviewed containment penetrations and compared the predicted pressure increase with the maximum permissible piping pressures (from Calculation 3) to determine if maximum permissible pressures were exceeded.

Several penetrations were determined to be susceptible to the overpressurization conditions discussed in the Generic Letter. The penetration and system descriptions are as follows:

- P-5, P-6, P-8, and P-10 - Containment Chilled Water System
- P-11 - Reactor Coolant Drain Pump Discharge
- P-30 - Containment Demineralized Water Supply
- P-32 - Fuel Pool Cooling Return to Refueling Cavity
- P-34 - Containment Fire Protection Supply
- P-37 - Reactor Coolant System Loop Fill Header
- P-44 - Primary Water Supply to Reactor Coolant Pump Seal #3 and Pressurizer Relief Tank
- P-47 - Containment Floor Drain Sump Pump Discharge
- P-55 - Safety Injection Accumulator Fill Line
- P-57 - Fuel Pool Cooling Suction from Refueling Cavity

These penetrations generally have a containment isolation valve on each side of the containment wall which is closed during normal plant operation (P-30, P-32, P-37, P-55, and P-57) or automatically closes on a containment isolation signal (P-5, P-6, P-8, P-10, P-11, P-34, P-44, and P-47). These penetrations could be potentially heated during a loss of coolant accident or a main steamline break inside containment, either of which would provide the containment isolation signal.

Byron Operability Assessment 97-006 was completed January 23, 1997 to document the basis for continued operability of these affected penetrations and systems/components. The basis for operability of these affected penetrations includes consideration of one or more of the following: expansion of the trapped fluid in voided areas of the isolated piping section, leakage from one of the containment isolation valves (seat, packing, or body to bonnet flange), consideration of existing insulation of piping inside containment or heat dissipation from piping outside containment to prevent the temperature induced pressure increase, lifting of air operated valves due to the pressure increase, and plastic straining of the affected pipe to accommodate the pressure increase.

An extensive review of isolated piping sections that affect containment integrity has been completed. While these penetrations have been determined to be operable, final corrective actions have not been determined. Byron intends to either install physical changes that provide overpressure protection or use penetration specific analytical evaluations that clearly address and confirm overpressurization is not a concern. Alternative solutions are being evaluated and additional time is necessary to assure the correct resolution for each isolated penetration is identified. Byron will provide a response by May 31, 1997 which will include details on what changes are being made to each affected penetration.

Attachment 3

Dresden Station
NRC Docket 50-237 and 50-279
Response to NRC Generic Letter 96-06,
"Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions", dated September 30, 1996

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 above, conclusions that were reached relative to susceptibility for waterhammer and two-phase flow in the containment air cooler cooling water system and overpressurizatrion 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 requested to determine:

if containment air cooler cooling water systems are susceptible to either water hammer or two-phase flow conditions during postulated accident conditions:

if piping systems that penetrate the containment are susceptible to thermal expansion of fluid so that overpressurization of piping could occur."

RESPONSE TO ITEM 1:

The Drywell Cooling System at Dresden Station is non-safety related. It is not assumed to operate under postulated accident conditions. Therefore, the potential for water hammer or reduced Drywell cooling efficiency due to two-phase cooling water flow is not a Design Basis concern at Dresden.

Cooling water to the Drywell Coolers is supplied by the Reactor Building Closed Cooling System (RBCCW). The RBCCW system inside the primary containment also serves the non-safety related cooling needs of the Reactor Recirculating Pumps and the Drywell Equipment Sump Heat Exchanger. The portions of the RBCCW system at the containment penetrations between the inboard and outboard isolation valves are safety related. The rest of the RBCCW system inside the primary containment is a non-safety related closed loop. Although the RBCCW inside drywell does not serve any safety related cooling needs inside the drywell, it is not automatically isolated during a LOCA/MSLB. The cooling water continues to flow through the drywell until it is manually isolated from the control room due to RBCCW pump trip or the expansion tank (located outside drywell) low water level. This leads to 3 possible scenarios discussed below.

1. If the RBCCW system inside the drywell continues to operate following the accident, the heat transfer into the pipe will be removed by the water flow and voids will not build-up.
2. If the drywell isolation valves are closed, the heat transfer into the pipe will cause the water to expand and result in thermally induced pressurization. Thermally induced pressurization will also prevent voiding in the pipe.
3. If the RBCCW pump trips during the initial phase of the accident and the isolation valves are not manually closed, the system may be susceptible to voiding. However, the Dresden RBCCW system is a closed loop system with an expansion tank that is open to the atmosphere at the highest elevation of the system. The elevation difference between the water level in the expansion tank and RBCCW piping in the drywell is at least 50 ft and therefore the pressure inside the piping will always be above 36 psia. There is a potential for a small amount of voiding to occur in the cooler coils during this period. It is expected that any voids that form would be subsequently collapsed by the surrounding cool water at the inlet and outlet of the coolers and no significant water hammer or 2 phase flow would occur.

Therefore, the Dresden RBCCW piping inside drywell will not develop any significant voids due to the post accident drywell environment and, therefore, the water hammer and two phase flow issues identified in NRC GL 96-06 are not significant for Dresden Units 2 and 3.

RESPONSE TO ITEM 2:

Piping systems that penetrate containment were evaluated to determine if they were susceptible to thermal expansion of fluid so that overpressurization of piping could occur. An evaluation was performed to identify containment piping penetrations potentially susceptible to thermal overpressurization as discussed in the Generic Letter.

Several penetrations were determined to be susceptible to the overpressurization conditions discussed in the Generic Letter. The penetration and system descriptions are as follows:

X-139B and X-144 (Unit 2), and X-139C (Unit 3)	- Control Rod Drive
X-119 (Unit 2 and 3)	- Demin Water Piping
X-109B (Unit 2) and X-109A (Unit 3)	- Isolation Condenser Return
X-117 (Unit 2 and 3)	- Drywell Floor Drain
X-118 (Unit 2 and 3)	- Drywell Equipment Drain
X-122 (Unit 2 and 3)	- Reactor Recirculation System Sample Line
X-113 (Unit 2 and 3)	- Reactor Water Cleanup
X-111A and X-111B (Unit 2 and 3)	- Shutdown Cooling
X-123 and X-124 (Unit 2 and 3)	- Reactor Building Closed Cooling

These penetrations generally have a containment isolation valve on each side of the containment wall which is or may be closed during normal plant operation (X-139B, X-144, X-139C, X-119, X-109B, X-109A, X-117, X-118, X-122, X-113, X-111A, X-111B, X-123, and X-124). These penetrations could be potentially heated during a loss of coolant accident or a main steamline break inside containment, either of which would provide the containment isolation signal.

A Dresden Operability Assessment was completed to document the basis for continued operability of these affected penetrations and systems/components. The basis for operability of these affected penetrations includes consideration of one or more of the following: expansion of the trapped fluid in voided areas of the isolated piping section, leakage from one of the containment isolation valves (seat, packing, or body to bonnet flange), insulation of piping to delay the temperature increase, lifting of air operated valves due to the pressure increase, partial draining of the isolated portion, and plastic straining of the affected pipe to accommodate the pressure increase.

An extensive review of isolated piping sections that affect containment integrity has been completed. While these penetrations have been determined to be operable, final corrective actions have not been determined. Dresden intends to either install physical changes that provide overpressure protection or use penetration specific analytical considerations that clearly address and confirm overpressurization is not a concern. Alternative solutions are being evaluated and additional time is necessary to assure the correct resolution for each isolated penetration is identified. Dresden will provide a response by May 31, 1997 which will include details on what changes are being made to each affected penetration. It is noted that the next refueling outage for Dresden Unit 3 (D3R14) will start at the end of March 1997 and Dresden Unit 3 will not be able to implement the required changes during D3R14.

Attachment 4

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

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:

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

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 purpose of the Primary Containment Cooling System (VP) is to maintain a suitable environment inside the drywell for equipment operation and longevity. The VP System consists of two subsystems, Primary Containment Ventilation and Chilled Water Systems. The VP System is a non-safety-related system, has no safety design basis, and is designed to operate during normal plant operating conditions. The system is not required to operate under post accident conditions.

The only portion of the chilled water system that is safety-related is located between the primary containment isolation valves located in the supply and return piping to the heat exchanger coils. The balance of the chilled water system located inside primary containment is a non-safety-related closed loop. The valves are automatically isolated via a Group II isolation (i.e., high drywell pressure or low reactor level). Consequently, there are no post accident concerns with respect to Item 1 and the item does not apply to LaSalle County Station.

However, as stated in response to License Event Report (LER) 96-020-00 LaSalle County Station has committed to review systems important to safety or utilized within the actions of the emergency operating procedures to determine if the systems are susceptible to waterhammer. The chilled water system is included as part of our commitment in the LER and additionally will be evaluated for two-phase flow. The evaluation will be completed prior to startup of Units 1 and 2. This issue is tracked via the LaSalle County Station Nuclear Tracking System (NTS). The NTS Numbers are 373-180-96-020.05LER for Unit 1 and 373-180-96-020.06LER for Unit 2.

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 the following liquid-filled piping that penetrates containment was performed:

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
Chilled Water
Reactor Building Closed Cooling Water
Reactor Recirculation Flow Control Valve Hydraulic Piping
Residual Heat Removal Shutdown Cooling
Control Rod Drive

It was determined that there are several piping sections that potentially could experience thermally-induced overpressurization. These locations are listed below:

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

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 appropriate operability evaluations are performed.

Our approach will be consistent with the guidance in Generic Letter 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on resolution of Degraded and Nonconforming Conditions and on Operability." This item is tracked under NTS Number 373-104-96-00602.

We will provide a response by May 31, 1997 which describes results of our analysis, any design changes required, and the schedule for implementation of these design changes.

Attachment 5

Quad Cites Station
NRC Docket 50-254 and 50-265
Response to NRC Generic Letter 96-06,
"Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," dated September 30, 1996

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 above, conclusions that were reached relative to susceptibility for waterhammer and two-phase flow in the containment air cooler cooling water system and overpressurizatrion 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 requested to determine:

if containment air cooler cooling water systems are susceptible to either water hammer or two-phase flow conditions during postulated accident conditions:

if piping systems that penetrate the containment are susceptible to thermal expansion of fluid so that overpressurization of piping could occur."

RESPONSE TO ITEM 1:

The Drywell Cooling System at Quad Cites Station is non-safety related. It is not assumed to operate under postulated accident conditions. Therefore, the potential for water hammer or reduced Drywell cooling efficiency due to two-phase cooling water flow is not a Design Basis concern at Quad Cities.

The Quad Cities Emergency Operating Procedures recognize the benefits of Drywell Cooling operation and direct the system's use under specific post accident conditions. The potential for water hammer and two phase flow in the Drywell Cooling system piping was evaluated. This evaluation is summarized below:

Cooling water to the Drywell Coolers is supplied by the Reactor Building Closed Cooling System (RBCCW). The RBCCW system inside the primary containment also serves the non-safety related cooling needs of Reactor Recirculating Pumps and the Drywell Equipment Sump Heat Exchanger. The RBCCW pumps trip automatically under LOCA conditions. The Containment Isolation Valves associated with the RBCCW supply and return lines to the Drywell do not have an automatic isolation function and would remain open . If conditions indicative of a RBCCW line break are observed, procedures direct the operator to close the RBCCW isolation valves. Re-establishment of RBCCW flow to the Drywell would not be expected if the system was isolated due to a line break.

The Quad Cities RBCCW system is a closed loop system with an expansion tank at the highest elevation of the system that is open to the atmosphere. This tank is located more than 50 feet above the Drywell Coolers and would provide a static pressure head of approximately 36 psia to the RBCCW piping in the Drywell. There is a potential for a small amount of voiding to occur in the cooler coils during this period. It is expected that any voids that form would be subsequently collapsed by the surrounding cool water at the inlet and outlet of the coolers and no significant water hammer or 2 phase flow would occur.

RESPONSE TO ITEM 2:

Piping systems that penetrate containment were evaluated to determine if they were susceptible to thermal expansion of fluid so that overpressurization of piping could occur. An evaluation was performed to identify containment piping penetrations potentially susceptible to thermal overpressurization as discussed in the Generic Letter. This evaluation resulted in the issuance of four calculations. The first calculation determined the maximum temperature change in water trapped between two closed valves during accident conditions for a variety of containment penetration configurations. The second calculation developed a methodology to calculate an incremental pressure increase per °F as a function of pipe outside diameter to wall thickness ratio and final water temperature. Pressure changes in the isolated pipe sections could then be predicted using the pipe material and size, the pressure increase per °F and the calculated temperature rise. The third calculation determined the maximum permissible piping pressures for design, upset, emergency, and faulted conditions to determine the pressure retaining capability of isolated pipe sections under thermal pressurization conditions. The final calculation reviewed containment penetrations and compared the predicted pressure increase with the maximum permissible piping pressures to determine if maximum permissible pressures were exceeded.

Several penetrations were determined to be susceptible to the overpressurization conditions discussed in the Generic Letter. The penetration and system descriptions are as follows:

Penetration X-24 - Reactor Building Closed Cooling Water
*Penetration X-41 - Reactor Recirculation System Sample Valves
*Penetration X-12 - Reactor Shutdown Cooling Water Sunction Piping
*Penetration X-14 - Reactor Water Cleanup Piping
Penetration X-20 - Clean Demin Piping

These penetrations have containment isolation valves. These penetrations could be potentially heated during a loss of coolant accident or a main steamline break inside containment, either of which would provide a containment isolation signal. Penetrations designated by an asterik close automatically on an isolation signal.

An operability assessment was completed January 27, 1997, to document the basis for continued operability of these affected penetrations and systems/components. The basis for operability of these affected penetrations includes consideration of one or more of the following: expansion of the trapped fluid in voided areas of the isolated piping section, leakage from one of the containment isolation valves (seat, packing, or body to bonnet flange), insulation of piping to delay the temperature increase, lifting of air operated valves due to the pressure increase, and plastic straining of the affected pipe to accommodate the pressure increase.

A review of piping sections that affect containment integrity which can be isolated has been completed. While these penetrations have been determined to be operable, final corrective actions have not been determined. Alternative solutions are being evaluated and additional time is necessary to assure the correct resolution for each isolated penetration. Quad Cities will provide a response by May 31, 1997, which will include details on what changes are being made to each affected penetration.

Attachment 6

Zion Station
NRC Dockets 50-295 and 50-304
Response to NRC Generic Letter 96-06,
"Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," dated September 30, 1996

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 above, 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 requested to determine:

if containment air cooler cooling water systems are susceptible to either water hammer or two-phase flow conditions during postulated accident conditions:

if piping systems that penetrate the containment are susceptible to thermal expansion of fluid so that overpressurization of piping could occur."

RESPONSE TO ITEM 1:

Potential Water Hammer Transient

This issue has been previously addressed at Zion. The results of the evaluation are included in the AEA O'Donnell report entitled "Evaluation of the Reactor Containment Fan Cooler and Associated Service Water Piping and Components for the Loss of Offsite Power Event", issued in January, 1992. The results concluded that the loads associated with a LOCA/LOOP event were within design limits. The NRC had a number of questions prior to IN 96-45 and subsequently GL 96-06. As a result of NRC discussions, analyses were started in mid 1996 to further evaluate the potential for water hammer in the Zion Station service water (SW) system supply and return piping for the RCFCs.

To address the above concerns, a RELAP model was developed to model heat transfer to the RCFC's during the LOOP/LOCA in order to predict the size and location of voids prior to pump restart. For the hydraulic transient effects, HYTRAN, a Sargent and Lundy computer program was utilized.

While still in progress, the preliminary results indicate that operability can be demonstrated (pressure retaining boundary, structural stability and required flow area) based on the fact that the calculated pipe stresses are within operability limits.

Potential for Two-Phase Flow

Once the service water piping refills upon pump restart and the RCFC fans restart, the RCFC cooling water exit temperature will be maintained at an elevated level due to the high condensation heat transfer rates. Consequently, a review is being performed to determine if two-phase flow, and/or flashing occurs at high-resistive components (orifices and throttle valves downstream of the RCFCs) due to hydraulic friction losses combined with the increased vapor pressure of the cooling water.

A calculation is in progress using the RELAP model developed to analyze the initial part of the LOOP/LOCA transient and will be used to determine the steady-state cooling water flow rates following pump restart. Steady-state heat transfer to the RCFC coils and containment piping will be modeled.

The results of the evaluation in progress are expected to demonstrate that the required flow rate per RCFC unit is achieved and two-phase flow will not occur. Based on the RCFC outlet pressures from the various pump scenarios contained in the flow model , the minimum pressure downstream of the throttle valves and orifice is approximately 4 psig where the RCFC discharge water mixes with other streams. The saturation temperature of water at this pressure is approximately 224 degrees F. The maximum temperature of water leaving an RCFC is expected to be less than this value based on the heat exchanger design calculations and containment heat removal requirements.

Zion Operability Assessments ER9604802 and ER9606789 document the basis for continued operability of this service water piping in accordance with GL 91-18. Zion will provide the details of the final resolution to Generic Letter 96-06 by May 31, 1997. After the detailed analysis is completed and the final loads are assessed, the extent of piping modifications necessary, if any, will be determined. Other system changes (e.g. vacuum breakers) and changes to operating configurations will also be considered. Additionally, when the detailed changes have been determined, we will evaluate the schedule for implementation based on material availability, design package development and issuance, and schedule windows needed for implementation.

RESPONSE TO ITEM 2:

Potential for Thermally-Induced Overpressurization of Isolated Piping

Thermally-induced overpressurization is applicable to liquid-filled piping system which penetrate containment. Documentation such as Piping and Instrumentation Diagrams, Piping Specifications, and Insulation Specifications were reviewed to identify piping sections that may be isolated during operation and subject to external heating during a LOCA/MSLB. Calculations were prepared to determine the temperature and pressure increases, determine and evaluate the affected systems and to determine operability.

As a result of the above review, some affected piping sections were found to be susceptible to overpressurization. They are as follows:

Penetration No.	System
P-1	Fire Protection
P-34	Demineralized Water
P-43	Waste Drain
P-88/89	Heating System
P-102	Primary Water
P-76	Accumulator Test Line
P-76	Reactor Coolant Loop Fill
P-76/86	Sample System

For these piping sections, operability of the piping was demonstrated by showing that the permanent strain developed would be on the order of 6%, which is well below the ultimate strain for the material. Additionally, the basis for operability include considerations of leakage through one of the isolation valves (seat, packing, or body to bonnet flange), consideration of insulation of piping inside containment or of heat dissipation for piping outside containment to prevent temperature enduced pressure increase, and lifting of air operated valves due to pressure increases.

Zion Operability Assessment ER9606789 documents the basis for continued operability of these affected penetrations and systems/components in accordance with GL 91-18. While these penetrations have been determined to be operable, final corrective actions have not been determined. Alternative solutions are being evaluated and additional time is necessary to assure the correct resolution for each isolated penetration is identified. Zion intends to either install physical changes that provide overpressure protection or use location specific analytical considerations that clearly address and confirm overpressurization is not a concern. Consequently, Zion will provide the details of the final resolution to Generic Letter 96-06 by May 31, 1997. Additionally, when the detailed changes have been determined, we will evaluate the schedule for implementation based on material availability, design package development and issuance, and schedule windows needed for implementation.