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

Perry Nuclear Power Plant
Docket No. 50-440
Response to Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions"

Ladies and Gentlemen:

Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," dated September 30, 1996, requested licensees to evaluate their plant design and determine: (1) if containment air cooler cooling water systems are susceptible to either waterhammer or two-phase flow conditions during postulated accident conditions, and (2) if piping systems that penetrate the containment are susceptible to thermal expansion of fluid so that overpressurization of piping could occur.

In addition, within 120 days of the date of the GL, licensees were requested to submit a written summary report stating actions taken in response to the requested actions, conclusions 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, and corrective actions that were implemented or are planned to be implemented.

Attachment 1 to this letter is the written summary report. Attachment 2 identifies the regulatory commitments being made in this letter.

If you have questions or require additional information, please contact Mr. Henry L. Hegrat, Manager - Regulatory Affairs, at (216) 280-5606.

Very truly yours,

Lew W Myers

A072/,

KMN:sc

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III

I, Lew W. Myers, being duly sworn state that (1) I am Vice President, Nuclear of the Centerior Service Company, (2) I am duly authorized to execute and file this certification on behalf of The Cleveland Electric Illuminating Company and Toledo Edison Company, and as the duly authorized agent for Duquesne Light Company, Ohio Edison Company, and Pennsylvania Power Company, and (3) the statements set forth herein are true and correct to the best of my knowledge, information and belief.

Lew W Myers

Lew W. Myers

Sworn to and subscribed before me, the 28th day of January, 1997.

Jane E Mott

JANE E. MOTT
Notary Public, State of Ohio
My Commission Expires Feb. 20, 2000
(Recorded in Lake County)

CODED/8838/SC

Response to Generic Letter 96-06

BACKGROUND

Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," dated September 30, 1996, requires that within 30 days of the date of the GL, a written response be submitted to the Nuclear Regulatory Commission (NRC) 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.

Licensees were requested to evaluate their plant design and determine: (1) if containment air cooler cooling water systems are susceptible to either waterhammer or two-phase flow conditions during postulated accident conditions, and (2) if piping systems that penetrate the containment are susceptible to thermal expansion of fluid so that overpressurization of piping could occur.

In addition, within 120 days of the date of the GL, licensees were requested to submit a written summary report stating actions taken in response to the requested actions, conclusions 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, operability evaluations of affected systems and components, and corrective actions that were implemented or are planned to be implemented.

REQUESTED ACTION 1

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

Background

This issue was previously addressed at the Perry Nuclear Power Plant (PNPP) in response to NRC Information Notice (IN) 96-45, "Potential Common-Mode Post-Accident Failure of Containment Coolers," which was issued by the NRC on August 12, 1996. The result of the engineering evaluation for IN 96-45 concluded that the concerns raised in the IN were not applicable to PNPP.

Actions Taken at PNPP in Response to GL 96-06 Requested Action 1

Design Engineering at PNPP has initiated additional engineering evaluation of this issue as a result of GL 96-06. Concerns identified in GL 96-06 were reviewed for applicability to PNPP.

Conclusions

The containment cooling systems at PNPP relevant to the concerns identified in GL 96-06 are the Drywell Cooling System (M13) and the Containment Vessel Cooling System (M11). These systems are non-safety-related and are not assumed available for heat removal post accident. These systems are designed to maintain the containment and drywell atmospheres within their design requirements of 95°F and 145°F, respectively, during normal operation. These systems are not required to provide for heat removal during a design basis Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB). The potential exists for two-phase flow in the M13 system; however, since there is no reliance placed on heat removal of these systems, evaluation has concluded that the emergence of two-phase flow in the respective containment cooling systems is of no safety significance.

Discussion

Cooling water for the M11 air coolers in the air handling units is provided by the Containment Vessel Chilled Water System (P50) during normal operation. The P50 containment isolation valves are designed to automatically close in the event that a BOP LOCA signal (High Drywell Pressure or Reactor Pressure Vessel (RPV) Level 2) is received. The PNPP Plant Emergency Instructions (PEIs) direct the operation of containment coolers in the event that the containment average temperature cannot be maintained below 95°F, which assumes that the coolers are available. Therefore, at some point after an event which isolates the cooling water system from the containment and heats the containment atmosphere, it is possible that the containment coolers can be put into service by overriding isolation interlocks. The potential for the events in GL 96-06 are not applicable to this system because the containment design temperature of 185°F (the calculated post-accident containment temperature is <150°F per calculation NEDC-31940) will not support flashing of the P50 water in the containment cooler.

The M13 system is designed to maintain ambient air temperatures in the drywell within design temperature limits during normal operation, and also during the time the plant is shutdown and maintenance is occurring in the drywell. The cooling water for the M13 air handling units is provided by the Nuclear Closed Cooling (NCC) System (P43). The P43 system is isolated from the drywell when a LOCA signal (High Drywell Pressure and/or RPV Level 1) is received. In the event that the drywell temperature exceeds 145°F, PEIs direct the operator to initiate all available drywell cooling systems, which may include restoration of the P43 system post-LOCA. PEI-SPI 2.1 provides direction to restore NCC flow by overriding isolation interlocks, provided that there are no known leaks in the system. System Operating Instruction (SOI)-M13 contains a prerequisite that NCC flow be established prior to starting the drywell fans. Therefore, procedurally, the system does not force the drywell atmosphere across the cooling coils prior to NCC flow being established as described in GL 96-06.

Post-accident, the drywell coolers could be put into operation, assuming that the coolers are available. The PEI which permits the use of the drywell coolers were developed from

the Boiling Water Reactor Owners' Group (BWROG) Emergency Procedure Guidelines. The instructions are based on symptoms versus events and are predicated on the assumption that equipment may be used if available. The PEIs state that the operation of drywell coolers be used in the event that the drywell average temperature cannot be maintained below 145°F, which assumes that they are available. Therefore, at some point after an event which isolates the P43 system from the drywell and heats the drywell atmosphere, the drywell coolers can be put into service by overriding interlocks. In the unlikely event that a breach inside drywell occurs in the P43 system before, or as it is placed into service, the system will pump water into the drywell through the ruptured pipe. This will be detected in the control room by an NCC Surge Tank Level Low alarm, sensed by 1P43-N285 (ARI-H13-P970-1). Additionally, depending on the type of event, a leak in the P43 system inside drywell could also be detected by the drywell sump alarm sensed by 1E31-N093 (ARI-H13-P601-18), if not already alarming as part of the event. At this point, operator training requires the cause of the leakage to be investigated, ultimately leading to isolating loads, as required, which are thought to be the cause of the leakage, until the surge tank level can be maintained. In addition, the P43 system contains relief valves inside the drywell and in the containment to prevent piping damage due to isolation initiated by a LOCA signal. This minimizes the chance for damage to the system in the event that the system is started post-accident.

The concern identified in GL 96–06 dealt with the automatic sequencing of the cooler fans and the cooling water flow. With the fan coasting down prior to automatic re-establishment of the cooling flow, the stagnant water in the cooling coil was heated by the air flow and boiled, creating a steam void. The void collapsed when cooling water was re-established. This specific scenario is not safety significant at PNPP.

From a design basis standpoint, the P43 system is not required for safe shutdown during a LOCA or MSLB. The P43 system is isolated from the drywell and containment during a LOCA or MSLB. This isolates flow into and out of the drywell. In the event that the NCC pumps trip, which could be expected during a loss of power, the "A" and "B" NCC pumps will automatically restart if they had been running, but the isolation valves will not auto reposition open after they had automatically closed. The M13 system is classified as non-safety-related and is not required to shutdown the plant during a LOCA or for containment heat removal. As stated above, the P43 system contains relief valves inside the drywell and in the containment to prevent piping damage due to isolation initiated by a LOCA signal. Therefore, there are no safety concerns due to failure of the drywell coolers or other portions of the system.

At PNPP, the safety-related containment heat removal is accomplished by the Residual Heat Removal (E12) system in the Suppression Pool Cooling and/or Containment Spray mode(s) of operation. The heat exchangers for this system are located outside of the containment in the Auxiliary Building. The cooling flow for the heat exchanger is provided by the Emergency Service Water (ESW) System (P45). Suppression Pool water is circulated in the shell side of the heat exchanger and lake water is circulated through the tube side. The P45 system initiates with the Residual Heat Removal (RHR) system such that both systems begin to circulate water through the heat exchanger at about the same time. Two-phase flow in the heat exchanger is not a concern because

both flow paths operate simultaneously. Post LOCA, the RHR pump starts after a 5 second delay and the required RHR (Low Pressure Coolant Injection) flow to the reactor is at 27 seconds. In the event of a concurrent Loss-of-Offsite-Power (LOOP)/LOCA, the total system response time (i.e., from receipt of an accident signal to when the required flow is established) is required to be <37 seconds, including the time to start the emergency diesel generator (10 second maximum). It is expected that for most of this 37 seconds, the RHR pump will be in minimum flow recirculation which branches from the main line prior to the heat exchangers. The flow from the ESW pump is expected to start at 20 seconds (18.5 second time delay + 1.5 second for the ESW pump discharge valve to open 5%) from the receipt of the LOCA signal. In the event of a concurrent LOOP/LOCA, a 10 second maximum delay due to emergency diesel generator startup is added to the 20 second ESW pump start. Therefore, post-accident, the ESW flow is established when suppression pool water flows through the heat exchanger.

Discussion of NRC Information Notice 96-60

Subsequent to the issuance of GL 96-06, NRC IN 96-60, "Potential Common-Mode Post-Accident Failure of Residual Heat Removal Heat Exchangers," was issued to alert licensees to a potential common-mode post-accident failure of (Boiling Water Reactor) BWR RHR heat exchangers. The concern is that the heat input to the heat exchanger from the RHR side, with the lack of cooling flow, allowed the ESW in the heat exchanger tubes to boil, creating a void. Upon initiation of the cooling flow, approximately 10 minutes later, the steam voids collapse and have the potential to create a significant hydrodynamic load. This event is not applicable to PNPP because the ESW system starts at approximately the same time (within seconds) the RHR system initiates, such that the tube-side cooling water (ESW) is always running when the hot shell side water (RHR) is initiated. ESW flow is also routed to the Emergency Closed Cooling System (ECCS) heat exchangers and the Diesel Generator Jacket Water heat exchanger.

Operability Evaluation

The containment cooling systems at PNPP relevant to the concerns identified in GL 96-06 are the M11 and M13 systems. These systems are non-safety-related and are not assumed available for heat removal post accident. The M11 and M13 systems are designed to maintain the containment and drywell atmospheres within their design requirements of 95°F and 145°F, respectively, during normal operation. These systems are not required to provide for heat removal during a design basis LOCA or MSLB. In addition, there are no Technical Specification operability requirements for these systems. The potential for two-phase flow in the M13 system exists; however, since there is no reliance placed on heat removal by these systems, evaluation has concluded that the emergence of two-phase flow in the respective containment cooling systems is of no safety significance.

Corrective Actions

The containment cooling systems at PNPP are non-safety-related systems, and are not assumed available for heat removal post accident. Since engineering evaluation has concluded that the appearance of two-phase flow in these systems is not a safety concern, there is no need for corrective actions.

REQUESTED ACTION 2

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

Background

Operating Experience Report PS-5711, dated August 2, 1996, identified a containment vessel overpressure concern at Beaver Valley Power Station Unit 1. Independent Safety Engineering Group (ISEG) personnel subsequently generated Potential Issue Form (PIF) 96-2741 to investigate this issue for applicability to PNPP. PIF 96-2741 identified two similar penetrations (P301 and P311) for the Fuel Pool Cooling and Nuclear Closed Cooling systems, respectively, which were potentially applicable. Initial assessment of PIF 96-2741 identified 14 penetrations with potential overpressure concerns. The immediate review at that time concluded that these 14 penetrations have documented leakage rates and would not overpressurize. Furthermore, the initial PIF review concluded that even under a worst case scenario where the valves did not exhibit leakage and the given penetrations were to pressurize to the extent of failure, it is expected that the ruptures would be locally limited. A rupture of this nature would be expected to occur either locally inboard or locally outboard of the penetration, but not occurring simultaneously in both places. A degradation of this nature would not affect the ability of a containment penetration to function as a radiological barrier since one of the isolation valves (both must be 100 percent leak tight for the rupture to occur) would remain and perform its function of containment isolation.

Actions Taken at PNPP in Response to GL 96-06 Requested Action 2

As a result of GL 96-06, Design Engineering has performed a more thorough review of containment vessel penetrations. This review included isolated piping inside containment immediately upstream of penetration check valves to assure a relief path for the fluid is provided. Penetrations with air or gas as a medium were excluded because pressure increases in these penetrations are not as severe as in the case of a heated liquid. Additionally, penetrations with operating temperatures greater than the maximum post accident ambient temperatures were excluded. The results of the review for GL 96-06 has concluded that 11 penetrations are susceptible to the thermal expansion of fluid.

Conclusions

Preliminary calculations have been performed for the susceptible penetrations to establish the maximum permitted changes in temperatures. The specific operating and environmental conditions of one hour duration or more for each penetration were compared to the maximum temperature limits calculated with an ASME Appendix F stress limit. Based on this preliminary stress approach, three penetrations may be eliminated from the list of susceptible penetrations. The table at the end of this attachment provides a list of the 11 containment vessel penetrations that are susceptible to thermal expansion of fluid.

The effective ASME Code of Construction (Section III) for PNPP is the 1974 Edition through the Winter 1975 Addenda. Review of the ASME Code Subsection NC Subsubarticle NC-7110, "Scope," recognized that individual components which are isolable from normal system overpressure protection shall be reviewed to determine whether additional individual overpressure protection is necessary. Subparagraph NC-3621.2 requires the piping system to be designed to withstand the increased pressure due to fluid expansion. Other than this general guidance, there is no specific criteria defined for addressing isolated sections of piping which may be subject to environmental heating.

Operability Evaluation

This issue deals with the potential post LOCA thermal overpressurization of double isolated containment penetrations. When a section of water filled piping that is isolated by closed valves is heated, the water in the pipe and the pipe will expand. However, the expansion of the pipe is significantly less than that of water. The difference between the expansion of the water and the pipe will result in pressure build-up inside the pipe. The piping systems in question are not credited to mitigate design basis accidents, thus the operability concern is with respect to containment integrity. Engineering has reviewed the valve leakage test results for the susceptible valves and has concluded that leakage through the valve seat or bonnet should prevent piping failure due to overpressurization, and therefore containment integrity is maintained. Discussions with other utility representatives, and discussion with design engineering consulting representatives, indicate that a strain based analysis may demonstrate the piping will yield under the increased pressures without rupture. Also, a more detailed transient analysis which considers post accident ambient and piping temperatures with respect to time, as well as thermal expansion of the pipe, would yield more realistic heatup rates, and hence lower pipe stress. From an operability perspective, the risk due to failure resulting from environmental heating is judged to be minimal, since pipe expansion and leakage through the valve seat, packing, and potentially any gasket bolted joint where bolt relaxation may occur, would relieve pressure. A review of leak rate testing data for the subject penetrations supported that valve seat leakage exists. This leakage is expected to increase at elevated internal piping pressures, thereby further assisting in pressure relief. This led to the determination that no safety related system or containment integrity would be lost, and therefore operability is maintained.

Corrective Actions

The list of 11 susceptible penetrations is still under evaluation. As such, activities to determine the corrective actions for the affected penetrations are on-going. Long term solutions for the affected penetrations will be checked with generic industry activities and partnerships, such as the Nuclear Energy Institute (NEI), Electric Power Research Institute (EPRI), and the Boiling Water Reactor Owners' Group, in an effort to develop a long term solution. Consequently, the details of the final resolution to GL 96-06 will be provided by May 31, 1997.

As-Left LLRT Test Data

Description	Penetration Number	Valve Number	Medium	Valve Type	Test Date	As-Left LLRT Leakage Rate [SCCM] *
Condensate	P-111	1P11-F0080	water	butterfly	2/3/96	2
Condensate	P-111	1P11-F0090	water	butterfly	2/3/96	3.5
Fuel Pool Cooling	P-301	1G41-F0140	water	butterfly	1/29/96	[illegible]
Fuel Pool Cooling	P-301	1G41-F0145	water	butterfly	1/29/96	87.5
Demineralized Water	P-309	1P22-F0010	demin. water	gate	2/21/96	2
Demineralized Water	P-309	1P22-F0577	demin. water	gate	2/13/96	33.68
Nuclear Closed Cooling	P-311	1P43-F0140	water	butterfly	2/25/96	2
Nuclear Closed Cooling	P-311	1P43-F0215	water	butterfly	2/25/96	29.5
Chilled Water	P-405	1P50-F0140	chilled water	butterfly	3/14/96	4.4
Chilled Water	P-405	1P50-F0150	chilled water	butterfly	3/14/96	3.02
Fire Protection	P-406	1P54-F0726	water	gate	1/28/96	2.03
Fire Protection	P-406	1P54-F0727	water	gate	1/28/96	2
Post Accident Sampling	P-413	1P87-F0049	water	globe	2/7/96	2
Post Accident Sampling	P-413	1P87-F0052	water	globe	2/7/96	85.56
Post Accident Sampling	P-413	1P87-F0055	water	globe	2/7/96	2
Post Accident Sampling	P-413	1P87-F0046	water	globe	2/7/96	103.15
Equipment Drain	P-417	1G61-F0080	water	gate	3/7/96	2
Equipment Drain	P-417	1G61-F0075	water	gate	2/28/96	2
Floor Drains	P-418	1G61-F0165	water	gate	3/15/96	2.44
Floor Drains	P-418	1G61-F0170	water	gate	3/15/96	2
Backwash to RW	P-420	1G50-F0277	water	gate	2/25/96	2.82
Backwash to RW	P-420	1G50-F0272	water	gate	2/25/96	10.14
Reactor Water Cleanup	P-424	1G33-F0034	water	gate	3/14/96	2
Reactor Water Cleanup	P-424	1G33-F0028	water	gate	3/14/96	2

* NOTE: All testing performed at 7.8 - 8.58 psig with air.

The following table identifies those actions which are considered to be regulatory commitments. Any other actions discussed in this document represent intended or planned actions, are described for the NRC's information, and are not regulatory commitments. Please notify the Manager - Regulatory Affairs at the Perry Nuclear Power Plant of any questions regarding this document or any associated regulatory commitments.

Commitments
The list of 11 susceptible penetrations is still under evaluation. As such, activities to determine the corrective actions for the affected penetrations are on-going. Long term solutions for the affected penetrations will be checked with generic industry activities and partnerships, such as the Nuclear Energy Institute (NEI), Electric Power Research Institute (EPRI), and the Boiling Water Reactor Owners' Group, in an effort to develop a long term solution. Consequently, the details of the final resolution to GL 96-06 will be provided by May 31, 1997.