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ACCESSION NBR:9702040117 DOC.DATE: 97/01/28 NOTARIZED: YES DOCKET #
FACIL:50-400 Shearon Harris Nuclear Power Plant, Unit 1, Carolina 05000400
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SUBJECT: Responds to GL 96-06, "Assurance of Equipment Operability & Containment Integrity During Design Basis Accident Conditions."

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CP&L

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William R. Robinson
Vice President
Harris Nuclear Plant

JAN 28 1997

SERIAL: HNP-97-011
10 CFR 50.54(f)

United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
NRC GENERIC LETTER 96-06, "ASSURANCE OF EQUIPMENT OPERABILITY AND CONTAINMENT INTEGRITY DURING DESIGN BASIS ACCIDENT CONDITIONS"
120 DAY RESPONSE

Dear Sir or Madam:

Carolina Power & Light Company (CP&L) hereby responds to NRC Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions," dated September 30, 1996 for the Harris Nuclear Plant (HNP).

Generic Letter 96-06 requested licensees to 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.

The generic letter requested a written response within 30 days indicating: (1) whether or not the requested actions would be completed, (2) whether or not the requested information would be submitted, and (3) whether or not the requested information would be submitted within the requested time period.

In a letter dated October 30, 1996, CP&L provided the required 30 day response (Serial: HNP-96-186) stating that a written summary report would be submitted describing the actions taken in response to the NRC request, the conclusions reached relative to susceptibility for waterhammer and two-phase flow in the containment air cooler system and overpressurization of piping which penetrates containment, the basis for continued operability of affected systems and components as applicable, and the corrective actions implemented or planned. CP&L committed to provide

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this information on or before January 28, 1997. This information is provided in the enclosure to this letter.

Please refer any questions regarding this submittal to Ms. D. B. Alexander at (919) 362-3190.

Sincerely,

W. R. Robinson

W. R. Robinson

KWS/kws

Enclosure

W. R. Robinson, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief; and the sources of his information are employees, contractors, and agents of Carolina Power & Light Company.

Darlene S. Yarborough
Notary (Seal)

My commission expires: 2-6-2000

DARLENE S. YARBOROUGH NOTARY *** PUBLIC HARNETT COUNTY, NC

c: Mr. J. B. Brady, NRC Sr. Resident Inspector
Mr. N. B. Le, NRC Project Manager
Mr. L. A. Reyes, NRC Regional Administrator

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File: H-X-0545

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
NRC GENERIC LETTER 96-06, "ASSURANCE OF EQUIPMENT OPERABILITY AND CONTAINMENT INTEGRITY DURING DESIGN BASIS ACCIDENT CONDITIONS"

On September 30, 1996, the U.S. Nuclear Regulatory Commission (NRC) issued Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions." The generic letter requested that licensees determine the following:

(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 to the individual licensee's postulated accident conditions, the NRC recommended review of the above items with respect to the scenarios referenced in the generic letter. If systems were found to be susceptible to the conditions discussed in this generic letter, licensees were expected to assess the operability of affected systems and take corrective action as appropriate in accordance with the requirements of 10 CFR Part 50 Appendix B and the plant Technical Specifications.

The generic letter required a written response within 30 days of the date of the generic letter, 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. CP&L provided the required 30 day response in a letter dated October 30, 1996 (Serial:HNP-96-186).

The generic letter also required a written summary report within 120 days of the date of the generic letter, describing the actions taken in response to the requested actions, the 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 as applicable, and the corrective actions implemented or planned. Licensees were requested to identify the systems susceptible to the overpressurization conditions and describe the specific circumstances involved.

The following information describes CP&Ls response to each requested action, including results, conclusions, and corrective actions.

Requested Action 1:

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

Response:

Summary

CP&L has evaluated the susceptibility of the Containment Fan Cooler Units (CFCUs) at HNP to waterhammer and two-phase flow during postulated post-accident conditions. A loss-of-coolant accident (LOCA) and a main steam line break (MSLB) inside containment were considered for this evaluation. A LOCA was determined to be most limiting with regard to the heat release rates to containment. The effects of a LOCA were compared to the effects of a LOCA occurring at the same time as a loss of offsite power (LOOP). It was determined that a LOOP event has the greatest effect on waterhammer susceptibility.

HNP experienced simulated LOOP events (without a LOCA) between 1986 and 1989 as part of the Station Electrical Blackout testing as well as the Emergency Diesel Generator Operability testing. During the performance of the Emergency Diesel Generator Operability test on the "A" train in 1989, a single support was damaged while other supports in the same proximity were displaced. Stress calculations performed on the piping and supports indicate that the piping can withstand future hydrodynamic effects of a waterhammer event caused by a LOOP. This evaluation has determined that the effects of a LOOP/LOCA event is bounded by the LOOP only event. Although the evaluation determines that void formation and subsequent waterhammer could occur, the system capability to perform the required safety functions will be maintained. Details of this evaluation are provided as follows:

System Descriptions and Safety Functions

The CFCUs are designed to remove heat from containment following a design basis accident, thereby reducing containment air temperature and pressure. The CFCUs receive cooling water to the coils via the Normal Service Water (NSW) or Emergency Service Water (ESW) systems. The CFCUs receive cooling water from the ESW system post-accident.

There are four CFCUs at HNP, with two fans in each unit (i.e., eight fans total). Four of the eight fans are required after a safety injection signal (one fan in each cooler running in low speed). Considering single failure criteria, one train of containment fan coolers (one fan running in each of two fan coolers) with one train of containment spray, can provide 100 percent of the heat removal capability required post-accident.

Power to the CFCUs and the ESW pumps and associated valves is provided from the safety buses.

Waterhammer Evaluation

Waterhammers Associated with LOOP/LOCA Events

CP&L has analyzed the sequence of postulated accident events (LOCA and MSLB) to determine if cooling water void formation and subsequent void collapse and water hammer could occur. The limiting accident for this evaluation was determined to be a LOCA (double ended guillotine break of the Reactor Coolant Pump suction line), based on the difference in heat transfer rates during the two accidents (LOCA vs. MSLB).

The most limiting system configuration is when the ESW discharge is aligned to the auxiliary reservoir. This alignment would result in the greatest initial column separation and therefore the greatest potential for waterhammer. The following analyses were performed with the system aligned in this configuration.

1. Coast down - Immediately following the LOOP/LOCA event, the ESW pumps and CFCU fans would begin to coast down. ESW pump flow coast down would occur and approach no flow conditions in 10 seconds following the LOOP. The rate at which the CFCU fans are coasting down is such that they are still turning when the ESW pumps get a restart signal from the sequencer.

2. Drain down - During ESW pump coast down, a void will form in the inlet and outlet piping to the CFCUs due to column separation. The water column on each side of the CFCU will decrease until a hydraulic equilibrium condition is reached. This solid state hydraulic balance is reached at about elevation 287 feet. The column will remain at this level unless it is affected by air leakage, gases coming out of solution, or repressurization due to steam formation in the fan cooler.

3. Repressurization - During a (LOOP/LOCA) event, during the fan coastdown and following the drain down period, any water remaining in the coolers will begin to boil. The steam generation begins to fill the void formed during the draindown. This boiling is initially delayed by the condensation of the initial volumes of steam as it enters the ESW discharge header. Repressurization will cause the water column to increase due to steam bubble growth. During the drain down and repressurization phase, water in the horizontal sections of piping would be partially drained and exposed to steam void formation. As steam enters the void space formed at the top of the pipe, the remaining water could cause a rapid condensation of the steam with subsequent trapping of steam bubbles. This rapid condensation could collapse the void, thereby inducing a waterhammer. Analysis indicates that the magnitude of the pressure pulses from these potential condensation induced waterhammers would be less than those from the column closure waterhammers.

4. Refill - At about 25 seconds into the postulated accident scenario, the ESW pumps have restarted and water is rapidly refilling the void created during the draindown/repressurization period. The refill process would cause two column closure waterhammers. The first would

occur in the 10 inch piping upstream of the CFCUs and the second would occur in the 10 inch piping downstream of the CFCUs. Analysis indicates that the magnitude of these pressure pulses is bounded by the magnitude of the column closure pressure pulses from a LOOP only event.

Effects of Non-Condensables

The presence of non-condensables has proven to be a significant mitigator to waterhammer events. Non-condensables have two principle effects: (1) they reduce the sonic velocity in water, and (2) in a bubble collapse or column rejoining waterhammer, the non-condensables act to cushion the pressure pulse. Because of non-condensables contained in HNP service water and in the steam expected to be generated, the pressure pulse resulting from LOOP/LOCA event related waterhammers would be significantly reduced compared to LOOP only events.

Waterhammers Associated with the LOOP Event Only

The waterhammers associated with a LOOP only event would be similar to the refill waterhammers during a LOOP/LOCA event. The coast down, drain down, and refill discussion from the LOOP/LOCA evaluation also applies to the LOOP. The repressurization would not occur and the column separation advance associated with the steam and bubble growth would not take place. The two column closure waterhammers associated with the pump restart would occur in the 10 inch pipe upstream and downstream of the coolers. Although the column separation would be slightly larger, the ratio of impact velocity to system velocity is the same for a LOOP with and without a LOCA because the piping lengths are so long in comparison to the void size. Therefore, pressure pulses for the LOOP/LOCA event are bounded by the pressure pulses for a LOOP only event, especially considering the effects of non-condensables associated with the LOOP/LOCA event.

Tests Resulting In Waterhammer

As discussed above, the waterhammer pressure pulses associated with a LOOP/LOCA are bounded by the waterhammer pressure pulses experienced from a LOOP only. A review of the service water system configuration reveals that the ESW discharge aligned to the auxiliary reservoir is the worst case scenario for column separation water hammer. This is due to the elevation difference between the auxiliary reservoir and the supply and return piping for the CFCUs on the operating deck of containment. There is evidence of several test events in this configuration. From October 1986 until December 1989, OST-1823 (1A-SA Emergency Diesel Generator Operability Test) and OST-1824 (1B-SB emergency Diesel Generator Operability Test) were performed. During this test, the 'A' and 'B' trains of service water to the CFCUs are aligned to the auxiliary reservoir. Power is removed from the service water pumps and valves, allowing the same draindown to occur as would happen with a LOOP. Power is then restored to the pumps and valves via the emergency sequencer. This results in a column closure waterhammer like that expected with a LOOP. During the waterhammer that occurred during a LOOP test in 1989, a single support was damaged, and three supports were shifted. The piping and supports were found to be acceptable for operability. The supports were repaired and the test system configuration was modified in 1989 to avoid such loading in the future. System

walkdowns performed subsequent to these tests indicate that there was no evidence of permanent deformation or piping damage. A stress analysis was performed where the impact of past loading on fatigue life has been determined to be insignificant and substantial margin remains for any postulated future events.

A service water system walkdown was performed in 1996, in accordance with NUREG/CR-5220, with the results being that piping, supports, and components did not experience permanent deformation during the performance of the above mentioned Operations Surveillance Tests.

Conclusion

CP&L has evaluated its service water system and containment fan cooling units for susceptibility to waterhammer in accordance with NRC Generic Letter 96-06 and determined that conditions do exist where voiding and subsequent waterhammer could occur.

Analyses of waterhammer events as they related to the service water piping supplying the containment fan cooler units were performed. The conclusions from this analysis are that waterhammers resulting from a LOOP coincident with a LOCA are no more severe than waterhammers resulting from a LOOP alone. A bounding event has occurred at HNP as a result of plant testing during the period from 1986 until 1989.

An event which occurred in 1989 damaged one support and displaced three other supports. The piping and supports were found to be acceptable for operability. The supports were repaired and the tests were modified in 1989 to avoid such loading in the future. System walkdowns performed on the system subsequent to these tests indicate that there is no evidence of permanent deformation or piping damage at this time. A stress analysis was performed where an impact of past loading on fatigue life has been determined to be insignificant and substantial margin remains for any postulated future events.

This analysis concludes that the piping design basis requirements have been satisfied by appropriate consideration of water hammer loading. Accordingly, CP&L ensures that the CFCUs and their associated service water system at HNP are capable of performing their required safety functions and that containment integrity will be maintained. Details of this determination are contained in plant ESR # 97-00008 Revision 0.

Two-Phase Flow - Operability Determination

Postulated accident events were evaluated for HNP to determine if two-phase flow in the CFCUs could occur. First, using the most limiting accident (LOOP/LOCA) and system alignment as a basis, this evaluation determined that two-phase flow could occur. The two-phase flow effects on the cooling capacity of the CFCUs were then evaluated. Appropriate containment analysis was performed to determine the effect on containment temperature and pressure (long and short term).

Preliminary results indicate that some two-phase flow could occur for as long as 110 seconds after the accident initiation. Preliminary containment analysis assumed that the CFCU's fans did not operate during the 110 second period and indicated that there are no significant effects on short or long term containment pressure or temperature response. Note that significant heat removal will, in fact, occur during the period beginning with CFCU operation until two-phase flow stops at 110 seconds.

Conclusion

CP&L has determined that two-phase flow in the service water flow through the CFCUs could exist, but for no longer than 110 seconds after the LOOP/LOCA accident is initiated. Using this as a design input for performing the appropriate containment analysis, preliminary reports indicate that assuming no CFCUs for that period of time has minimal effect on containment temperatures and pressures. Therefore, although engineering analysis is continuing, the Containment Cooling and Service Water Systems are determined to be able to perform their respective safety functions and containment integrity will be maintained. CP&L intends to complete the two-phase flow analysis and identify specific corrective actions to address final resolution of NRC Generic Letter 96-06 by the end of the next refueling outage at HNP (RFO7 for HNP is currently scheduled to begin April 1997). Details of this determination are contained in plant ESR # 97-00043 Revision 0.

<u>Requested Action 2</u>:

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

<u>Response</u>:

CP&L reviewed mechanical containment penetrations listed in the HNP FSAR Table 6.2.4-1, Containment Isolation System Data and design drawing 2165-G-064. Penetrations that are normally isolated or serve to isolate containment during shutdown or post-accident conditions were reviewed to determine their susceptibility to thermal expansion that could result in piping overpressurization. CP&L also reviewed the piping systems that penetrate containment to determine their susceptibility to thermal expansion of the piping fluid. The reviews considered the maximum heat input to the penetrations during a LOCA or MSLB event.

Evaluation of Penetrations

CP&L reviewed a total of 126 mechanical containment penetrations. The results of the review are summarized as follows:

- Thirty-two (32) penetrations do not automatically isolate on a containment isolation signal. These penetrations are not considered susceptible to thermal overpressurization.

- Fifty-three (53) pneumatic (air filled) and spare/capped penetrations are not considered to be susceptible to thermal expansion.

- One (1) penetration with a relief valve is not considered susceptible to thermal expansion.

- Twenty-three (23) penetrations with check valves that would relieve pressure from the penetration into the connecting piping within containment are not considered susceptible to thermal expansion.

- One (1) penetration at normal Reactor Coolant System operating temperature is not considered susceptible to further post-accident heat-up .

- Six (6) penetrations at normal Steam Generator operating temperature are not considered susceptible to further post-accident heat-up.

- Six (6) penetrations would not overpressurize based on evaluation of the penetration isolation valves. These valves rely on spring actuator forces to close. Valves of this type would relieve due to a pressure buildup in the penetrations as that pressure balances with the operator's spring force. This conclusion is based on engineering judgment. The containment isolation function of these penetrations is not degraded, because the leakage of the isolated fluid from the penetrations does not create a leak path from inside containment to the outside.

Final detailed analysis of the self-relieving capability of each valve will be completed by the end of the next refueling outage currently scheduled to begin in April, 1997.

- Four (4) penetrations were identified that isolate with no apparent means to relieve a potential pressure buildup.

The CP&L evaluation of the four (4) penetrations that could be susceptible to overpressurization is described below:

Penetrations Susceptible to Overpressurization

Four (4) penetrations, M-44, M-45, M-74, and M-91, were identified that have no apparent means of pressure relief to preclude their susceptibility to overpressurization. The piping/components could be subjected to a temperature increase, resulting in a pressure increase within the piping.

Two penetrations, M-44 and M-45, are used during refueling for fuel pool cooling and clean-up. CP&L will revise appropriate operating procedures to require that these penetrations are drained before they are closed after use for fuel pool cooling and clean-up. These two penetrations are considered operable until the procedure revision is implemented, based on the seat design of these diaphragm valves that would relieve as the pressure increases in the penetrations. Implementation of the revised procedures will be completed prior to exiting the next refueling outage currently scheduled to begin in April, 1997.

CP&L will also revise FSAR Table 6.2.4-1 to change the indication of fluid for penetrations M-44 and M-45 from W (water) to A (air) after implementation of the revised procedure.

Penetration M-74 connects to the non-safety containment sump pump discharge piping. Engineering analysis is being performed for penetration M-74. The preliminary analysis shows this penetration to be acceptable based on containment induced heat-up, thermally induced pressure build-up, and resultant acceptable pipe stresses. The penetration is determined to be operable for containment isolation based on the conclusion of this preliminary analysis.

Penetration M-91 connects to the non-nuclear safety normal service water discharge piping from the non-nuclear safety containment fan coolers. Engineering analysis is being performed for penetration M-91. This penetration is considered operable based on a history of acceptable leakage through the penetration's isolation valves as documented by LLRT test data. Therefore, in the event of thermal expansion of the piping fluid, the valve leakage would serve to relieve pressure from between the isolation valves (1SW-240 and 1SW-242).

CP&L will complete the final engineering analysis for penetrations M-74 and M-91 and identify any specific corrective actions to address final resolution of NRC Generic Letter 96-06 by the end of the next refueling outage at HNP. Details of this determination are contained in plant ESR # 96-00537 Revision 1.

Evaluation of Closed Loop Piping Systems Inside Containment

CP&L reviewed the piping systems that penetrate containment and form a closed loop within containment to determine their susceptibility to thermal expansion of the piping fluid. The results of the review are summarized below:

- Sixty-three (63) penetrations either do not continue into containment, do not form closed loops within containment, or are pneumatic (i.e., air-filled) and are not considered susceptible to thermal expansion.

- Thirty-seven (37) closed loops with relief valves are not considered susceptible to overpressurization.

- One (1) closed loop normally operates at temperatures above the postulated containment accident environment and is not considered susceptible to thermal expansion.

- Thirteen (13) closed loops are directly connected with the RCS such that pressure is controlled with the RCS are not considered susceptible to thermal expansion.

- One (1) closed loop is drained during normal operation and is not considered susceptible to thermal expansion.

- Six (6) closed loops that were environmentally qualified for post-accident containment are not considered susceptible to overpressurization.

- One (1) closed loop is isolated by a spring-to-close diaphragm valve that would leak due to a pressure buildup in the closed loop as that pressure balances with the operator's spring. The leak path is to the Pressurizer Relief Tank. Leakage below the seat of the diaphragm valve would occur such that overpressurization of the closed piping loop would not occur. This conclusion is based on engineering judgment. Final detailed analysis of this piping will be completed by the end of the next refueling outage.

- Four (4) closed loops of non-safety piping (Equipment Drain System, Safety Injection System test piping, Demineralized Water System, and Fire Protection Sprinkler System) could be susceptible to overpressurization, but would have no safety consequences should failure occur.

The CP&L evaluation of the four (4) closed loop piping systems susceptible to overpressurization is described below:

Closed Loop Piping Systems Susceptible to Overpressurization

The four non-safety piping sections identified, portions of the Equipment Drain System, Safety Injection System test piping, Demineralized Water System, and Fire Protection Sprinkler System, were evaluated for effects of their potential failure. No adverse consequences were identified.

Furthermore, these piping sections are not relied upon for any post accident actions, and containment integrity would not be challenged in the event of their failure. No corrective actions are determined necessary regarding the susceptibility of these non-safety piping sections to overpressurization. Details of this determination are contained in plant ESR # 96-00537 Revision 1.