CP&L

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William R. Campbell
Vice President
Brunswick Nuclear Plant

JAN 28 1997

SERIAL: BSEP 97-0029

U. S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
GENERIC LETTER 96-06, "ASSURANCE OF EQUIPMENT OPERABILITY AND CONTAINMENT INTEGRITY DURING DESIGN-BASIS ACCIDENT CONDITIONS"

Gentlemen:

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 requests 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;

(2) If piping systems that penetrate the containment are susceptible to thermal expansion of fluid so that overpressurization of piping could occur.

For the 120-day response, licensees were 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 are found to be susceptible to the conditions that are discussed in this generic letter, licensees are requested to identify the systems affected and describe the specific circumstances involved.

Enclosure 1 provides the Brunswick Plant summary report of the 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 the corrective actions that are planned to be implemented. Enclosure 2 identifies the list of regulatory commitments contained in this response.

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Please refer any questions regarding this submittal to Mr. Mark A. Turkal at (910) 457-3066.

Sincerely,

William R. Campbell

William R. Campbell

WRM/wrm

Enclosures
1. 120-Day Response To Generic Letter 96-06
2. Regulatory Commitments

William R. Campbell, 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 officers, employees, and agents of Carolina Power & Light Company.

Gwendolyn R. McGlotte
Notary (Seal)

My commission expires: August 12, 2001

cc: U. S. Nuclear Regulatory Commission
ATTN.: Mr. Luis A. Reyes, Regional Administrator
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The Honorable R. Hunt
Chairman (Acting) - North Carolina Utilities Commission
P.O. Box 29510
Raleigh, NC 27626-0510

ENCLOSURE 1

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
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(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 licensee's postulated accident conditions, the NRC recommends that these items 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, licensees 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.

The generic letter requests two responses, the first within 30 days of the generic letter date and the second within 120 days of the generic letter date. In the 30-day response, addressees are requested to indicate: (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. The 30-day response was submitted on October 30, 1996.

In the 120-day response, licensees 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 are found to be susceptible to the conditions that are discussed in this generic letter, licensees are requested to identify the systems affected and describe the specific circumstances involved.

NRC REQUEST 1:

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

RESPONSE:

The Reactor Building Closed Cooling System (RBCCW) at Brunswick is not susceptible to either water hammer or two-phase flow conditions during postulated accident conditions.

System Description:

The closed loop RBCCW system provides cooling water to non-safety related loads in the primary containment including drywell cooling units, reactor recirculation pump coolers and motor coolers, drywell equipment drain tank heat exchanger, mechanical penetration cooling and normal sample system panel cooling.

The drywell cooling unit cooling coils have pneumatically-operated flow control valves that are interlocked with the control system for the drywell blowers and damper. Each cooling unit is provided with two cooling coils and two control valves at the coil inlet. Placing a particular cooling unit blower into service opens the associated pair of control valves and taking it out of service closes them. This precludes the possibility of inadvertently operating a unit without necessary cooling water supply. The control valves are arranged for fail-safe operation so that each individual valve will open on a loss of respective electric power or air supply.

The drywell equipment header can be isolated by the motor-operated return and supply valves, RCC-V28 and RCC-V52. These isolation valves, however, are normally open and do not receive automatic isolation signals. A check valve RCC-V53 is also installed on the supply header. The penetration cooling system isolates upon a loss of normal AC power.

The RBCCW circulation pumps will automatically trip on a loss of off-site power (LOOP) concurrent with a loss of coolant accident (LOCA). The pumps cannot be restarted until logic trips are reset. The Service Water supply valves to the RBCCW heat exchangers, SW-V103 and SW-V106, actuate to the closed position on a LOCA or LOOP signal providing automatic isolation of SW Flow to RBCCW Heat Exchangers.

Pressure indicators provide local indication of the suction and discharge pressures of each pump and indication of the discharge header pressure both locally and on the control board.

The RBCCW surge tank is located in the Reactor Building on the refueling floor elevation. The tank has a capacity of 600 gallons and has several functions including providing sufficient net positive suction head for the circulating pumps, volume change compensation for system heatup and cooldown and makeup water to the system.

The tank is equipped with a vent to atmosphere, a makeup water line from the Demineralized Water System, an overflow line and a drain line to the Radwaste System, and a level gauge sight glass. The tank also has a level switch that will annunciate high and low level alarms should an abnormal level exist.

Safety Function:

The RBCCW system is non safety related and is not required to be operable for accident mitigation. Emergency Operating Procedures, however, use RBCCW containment cooling capabilities to reduce containment temperature if available. As such the RBCCW system is desirable to have available post accident. The RBCCW system is also supplied with single containment isolation valves. The system is considered a closed loop inside of containment and as such the piping within the drywell is considered the first containment barrier.

In the event of a LOOP/LOCA, the RBCCW circulation pumps and drywell fan coolers will trip. Penetration cooling supply within the drywell will isolate; however, the remainder of the system remains open. Head pressure supplied by the RBCCW surge tank is greater than 55 psia at the RBCCW supply and return penetrations. (This is the highest elevation of RBCCW piping in the drywell excluding the penetration cooling header). Saturation temperature at 55 psia is approximately 287°F.

For a LOCA, after the initial blowdown, (t ≈ 50 seconds) the drywell temperature plateaus at approximately 270°F until long term containment spray / suppression pool cooling is established (t = 600 seconds). Drywell temperature is postulated to rapidly decrease after initiation of long term cooling.

Conclusion:

Based upon the design and operation of the RBCCW system at Brunswick, two phase flow or pipe voiding are not expected upon restart of the RBCCW pumps as head pressures provided by the RBCCW surge tank maintain the system fluid below the boiling point.

NRC REQUEST 2:

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

RESPONSE 2:

Brunswick Nuclear plant has performed a penetration review for primary containment to identify piping susceptible to overpressurization. The review identified three penetrations potentially vulnerable to a water solid volume within the drywell being subjected to an increase in pressure due to ambient heating in a post-LOCA environment. The three penetrations identified are: (1) Penetration X-12, RHR Shutdown Cooling Suction Line, (2) Penetration X-18, Drywell Floor Sump Pump Discharge Line, and (3) Penetration X-19, Drywell Equipment Sump Pump Discharge Line.

Penetration X-12, RHR Shutdown Cooling Suction Line

Description:

Normally closed Containment Isolation Valves (CIV's) 1(2)-E11-F008 and 1(2)-E11-F009 are 20 inch 600 pound Anchor Darling split wedge gate valves with pressure sealed bonnets. 1(2)-E11-F009 and approximately 30 feet of insulated piping are installed within the drywell. 1(2)-E11-F008 is located immediately outside of the drywell where the line is uninsulated. The piping is designed to a specification of 1,147 psig and 562°F. The valves were procured with a hydrostatic test conducted at 2,175 psig and seat leakage testing with an acceptance criteria of leakage not to exceed 40 cc/hour.

Operability Assessment:

Safety Function:

The RHR shutdown cooling suction line is used to remove decay heat from the reactor vessel during normal shutdowns. Long te   re cooling post accident uses the RHR pumps to circulate water from the suppression pool through the low pressure coolant injection lines to flood the core. Valves 1(2)-E11-F008 and 1(2)-E11-F009 perform a dual passive safety function in the normally closed position as Containment Isolation Valves (CIV's) and low pressure to high pressure system isolation valves (PIV's). For post-LOCA, the RHR shutdown cooling suction line serves a containment isolation function only .

Seat Leakage Testing:

Historical Local Leakage Rate Test (LLRT) results for the Shutdown Cooling Suction Line identify that leakage typically exists through either one or both of the CIV's for both units. The most recent Pressure Isolation Valve (PIV) testing using water at normal operating reactor pressures identified leakage for 1-E11-F008 at 10 ml/min and 2-E11-F008 at 650 ml/min.

Pressurization Characteristics:

A thermal/pressurization analysis was performed for the shutdown cooling suction line. The heat transfer rate due to piping insulation results in a gradual pressurization with a maximum pressure of 3,800 psi achieved approximately 168 hours after the accident, assuming zero out-leakage from the volume between the CIVs. This gradual pressurization would support that a minimal leakage rate from the piping system would mitigate the pressure increase. The analysis is based on a conservative assumption for the initial temperature profile of the trapped water volume. The initial temperature profile chosen assumes no thermal mixing for the dead legged line from the Reactor Recirculation system to the inboard CIV on the shutdown cooling suction line. In actuality, the initial temperature profile of the fluid at the 1(2)-E11-F009 valve is expected to be higher at the start of the event than assumed in the analysis, resulting in a lower maximum pressure.

An operability review was performed on this line using the conservative value of 3800 psi for the maximum pressure. The shutdown cooling suction line is seamless schedule 80 pipe, equivalent to either A-106 Grade B or A-333 Grade 6 material. Analysis showed that the piping meets short term structural integrity criteria. A hoop stress check was performed and the piping was deemed operable on the basis of calculation conservatisms for pressure determination.

Valve pressure retaining materials and dimensions were reviewed. Information Notice 96-08, "Thermally Induced Pressure Locking of a High Pressure Coolant Injection Gate Valve" identified failure of a smaller but similarly designed valve to the RHR shutdown cooling suction line CIVs with a pressure seal bonnet. The retaining ring for the pressure sealed bonnet was found bent. Although deformation of the retaining ring was observed and may be expected at valve internal threshold pressures of 3,000 to 7,000 psig, no leakage from the pressure seal bonnet was observed.

Operability Conclusion:

Pressurization of the line is not anticipated due to the leakage exhibited by the valves during both LLRT and PIV testing. However, if pressurization were to occur, the associated piping and valves are expected to withstand the postulated event with no safety impact on the plant.

Penetration X-18, Drywell Floor Sump Pump Discharge Line

Description:

Normally open Containment Isolation Valves 1(2)-G16-F003 and 1(2)-G16-F004 are 3-inch 150 pound Anchor Darling gate valves with bolted bonnets. Both of the CIV's are located immediately outside of primary containment. The floor drain sump has two sump pumps with 2-inch bolted bonnet 600 pound piston type discharge check valves (1(2)-G16-F001A & B) and normally open 600 pound manual discharge isolation gate valves (1(2)-G16-F002A & B) downstream of the check valves. The two discharge lines connect to a common 3-inch discharge header by a flanged connection. Approximately 36 feet of the uninsulated 3-inch common discharge header runs through containment to the penetration. The piping is designed to a specification of 200 psig and 390° F. Trapped water between the pump discharge check valves and the inboard containment isolation valve may subject the piping to pressurization due to ambient heating. The containment isolation valves were procured with a hydrostatic test conducted at 425 psig and seat leakage testing with an acceptance criteria of leakage not to exceed 6 cc/hour.

Penetration X-19 Drywell Equipment Sump Pump Discharge Line

Description:

For the attributes of this issue, Penetration X-18 and X-19 are equivalent with the exception that the common pump discharge line has an alternate path to recirculate back to the sump through a drain tank heat exchanger. This flow path is normally isolated by a 3-inch 150 pound motor operated wedged gate valve.

Penetrations X-18 & X-19, Drywell Floor/Equipment Sump Pump Discharge Lines

Operability Determination:

Safety Function:

The drywell floor and equipment drain sumps accumulate leakage during normal operations. This leakage is quantified and compared to the Technical Specification allowable leakage criterion. The normally open containment isolation valves receive a Group 2 isolation signal to close on Reactor Low Level 1 or High Drywell Pressure. Containment sumps are isolated and are not opened for any safety related function post LOCA. As such, the drywell floor and equipment drain sump lines have an active safety function for containment isolation only.

Local Leakage Rate Testing:

For the drywell floor and equipment drain penetrations, both CIV's on the penetration would need to leak in order to have a pressurization relief path. LLRT results indicated that two of the four sump drain line penetrations have some leakage identified past the CIV seats.

An outstanding corrective maintenance work order currently exists for seat leakage identified on the 1-G16-F015, Drywell Equipment Drain Tank Inlet Valve. This seat leakage would be expected to provide a pressure relief path back to the sump for penetration 1-X-19.

Although LLRTs are conducted with low pressure air and relief capabilities required for the components in question are at potentially high pressure water environments, the LLRT results do confirm that some leakage paths do exist in realistic applications. As stated in the penetration descriptions, seat leakage to some degree was acceptable for fabrication and testing of the new components.

The sump pump discharge check valves are not classified as CIV's and are not leak rate tested. System performance parameters do not rely on these valves being leak tight. Due to their function, these lines may be subjected to gritty particles being pumped out of the sumps. The possible presence of particulate in the fluid can accelerate seat wear or the probability of partial valve closure due to the particulates being trapped on the seating surfaces. Corrective maintenance history reviewed for these lines supports this discussion.

Pressurization Characteristics:

The analysis performed to determine pressurization of the subject lines showed different characteristics than for the RHR penetration. The floor and equipment drain penetrations, due to the length of uninsulated pipe in the drywell, are subjected to more rapid pressurization rates. Maximum pressure for these penetrations was analyzed to occur at approximately 8 minutes after the DBA LOCA. The volume of the piping system, however, is relatively small and as such minor leakage from the piping system would provide sufficient volumetric changes for pressure relief.

Self Relieving Capabilities:

In the event that valve seat leakage would not relieve pressures generated by post accident ambient heating, the floor and equipment drain penetration piping contains numerous bolted

bonnet valves and flanged connections that are characteristically self-relieving when subjected to elevated pressures.

This characteristic is supported by a technical paper written by Oak Ridge National Laboratories on ANSI B16.5 flanged joint connections. For example, a 3-inch 150 pound flange of similar configuration to that installed in the floor drain piping was pressure tested to the point of leakage with bolting preloads documented. The test found that with nominal bolting torque , the flange began to leak at internal pressures of 3,200 psi. The results of this test are conservative as no bending moments were applied on the flanged connection during the test. Carolina Power & Light Company has performed an analysis of piping representative of the penetration X-18 floor drain configuration at a pressure of 3,200 psi and determined that the stresses meet short-term structural integrity criteria.

The penetration X-19 equipment drain configuration uses 2-inch 150-pound flanges. This flange configuration was not reported as being tested in the Oak Ridge National Laboratories report; however, the 2-inch flange would also be expected to exhibit self-relieving characteristics.

A calculation has been performed by CP&L which compares body-to-bonnet bolt yield thresholds for the sump pump discharge check valves inside the drywell relative to the bolt yield thresholds for the containment isolation valves. Based on the calculation, CP&L has concluded that the sump pump discharge check valves will relieve at pressures significantly lower than those for the containment isolation valves.

Packing

Packing also is characteristically self-relieving when subjected to elevated pressures. In the event the inboard CIV packing was to become the pressure relief path, the valves are supplied with packing leak off lines that are routed to the reactor building equipment drain tank, located within secondary containment, where the leaking fluid would be sparged under water.

Operability Conclusion:

Pressurization of the subject lines is not anticipated as certain boundary valves for the trapped water are not held to the rigorous seat leakage standards like containment isolation valves. LLRT results for two of the penetrations also support the probability that trapped water will leak from the boundary in question.

In the unlikely event that pressurization should occur, the drywell floor and equipment drain piping has numerous bolted bonnet valves and flanged connections within the drywell that would exhibit self-relieving characteristics at elevated pressures prior to piping rupture. These leak paths would not inhibit the containment isolation safety function of the drain system. Although the containment isolation valves outside of containment have similar pressure relief characteristics, analysis shows that the most probable relief path will be within the drywell. In the event of a packing leak outside of containment, the containment isolation valves have packing leakoff lines installed that would contain the leakage to the reactor building equipment drain tank.

Conclusions:

Carolina Power & Light Company's detailed evaluations of the Brunswick Plant for the issues described in NRC Generic Letter 96-06 have not been completed. The Company has determined that the Reactor Building Closed Cooling Water System (RBCCW) is not susceptible to either water hammer or two-phase flow conditions during postulated accident conditions. The closed loop RBCCW system provides cooling water to non-safety related loads in the primary containment including drywell cooling units, reactor recirculation pump coolers and motor coolers, drywell equipment drain tank heat exchanger, mechanical penetration cooling and normal sample system panel cooling. The Company has also completed an extensive review of isolable piping sections that affect containment (drywell) integrity. While the piping penetrations potentially susceptible to overpressurization have been determined to be operable, long-term resolution has not yet been completed.

The Company intends to either install physical changes that provide overpressurization protection or use specific analytical considerations on a case-by-case basis that address and confirm that overpressurization is not a long-term concern. The Company will continue to support industry activities and partnerships, such as NEI, EPRI, and the BWR Owners' Group, in an effort to develop a long-term solution. By May 30, 1997, CP&L will identify specific corrective actions, including potential physical changes, to address final resolution of NRC Generic Letter 96-06 for the Brunswick Plant.

ENCLOSURE 2

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324
LICENSE NOS. DPR-71 AND DPR-62
GENERIC LETTER 96-06, "ASSURANCE OF EQUIPMENT
OPERABILITY AND CONTAINMENT INTEGRITY DURING
DESIGN-BASIS ACCIDENT CONDITIONS"

LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by Carolina Power & Light Company in this document. Any other actions discussed in the submittal represent intended or planned actions by Carolina Power & Light Company. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Manager-Regulatory Affairs at the Brunswick Nuclear Plant of any questions regarding this document or any associated regulatory commitments.

Commitment	Committed date or outage
1. Identify specific corrective actions, including potential physical changes, to address final resolution of NRC Generic Letter 96-06 for the Brunswick Plant.	5/30/97