March 29, 2002

Mr. Paul D. Hinnenkamp
Vice President - Operations
Entergy Operations, Inc.
River Bend Station
P. O. Box 220
St. Francisville, LA 70775

SUBJECT: RIVER BEND STATION, UNIT 1 - COMPLETION OF LICENSING ACTIVITY FOR GENERIC LETTER (GL) 96-06, "ASSURANCE OF EQUIPMENT OPERABILITY AND CONTAINMENT INTEGRITY DURING DESIGN-BASIS ACCIDENT CONDITIONS" (TAC NO. M96858)

Dear Mr. Hinnenkamp:

The U.S. Nuclear Regulatory Commission (NRC) staff issued GL 96-06 on September 30, 1996, to all holders of operating licenses for nuclear power reactors, except for those licenses that have been amended to possession-only status. GL 96-06 requested information from licensees related to two concerns: (1) water hammer and two-phase flow in the cooling water systems that serve the containment air coolers, and (2) thermally induced overpressurization of isolated water-filled piping sections in containment. On November 13, 1997, the staff issued Supplement 1 to GL 96-06, informing licensees about ongoing efforts and new developments associated with GL 96-06 and providing additional guidance for completing corrective actions. You responded in letters dated October 30, 1996, January 28, 1997, June 17, 1997, July 21, 1998, November 12, 1998, May 1, 2001, and February 19, 2002. The results of the NRC's review of your responses to GL 96-06 follow.

<u>Water Hammer and Two-Phase Flow</u>

The containment structure at River Bend Station, Unit 1 (River Bend) is the General Electric Mark III design. The design includes a drywell containing the reactor vessel and recirculation loops, a suppression pool, and containment building which surrounds the drywell and suppression pool. The containment provides a secondary barrier to release of radioactive material that might leak from the drywell or suppression pool following an accident.

Cooling of the containment building is accomplished by the containment unit coolers (two safety-related and one non safety-related). During normal operation, cooling water to the containment unit coolers is supplied by the Ventilation Chilled Water System (VCWS). The VCWS is isolated from the containment unit coolers on a loss-of-coolant accident (LOCA) signal. The station operating procedure for the system provides guidance for filling and venting the VCWS prior to placing it in service from a previously drained condition, including a specific waterhammer precaution and venting instruction for the containment unit coolers before they would be returned to service.

During accidents, the safety-related containment unit coolers are provided with cooling water via the normal or standby service water systems. You evaluated temperatures within the containment cooler units for post LOCA conditions with and without off site power available. The containment was calculated to reach a maximum temperature of 125 degrees Farenheit (°F). No steam formation was calculated to occur in the containment cooler units.

Drywell cooling is accomplished by six non safety-related coolers that are located within the drywell. Cooling water for the drywell coolers is provided by service water systems. You calculated that, following a LOCA, the temperature within the drywell could rise to approximately 330 °F. The emergency operating procedures (EOPs) direct that the drywell unit coolers be manually placed in service when the drywell temperature exceeds 145 °F. The EOPs further direct that the drywell coolers not be placed in service should the drywell atmospheric temperature exceed 200 °F. This is because calculations by the licensee have shown that steam voids might form within the cooler units while they were isolated and result in water hammer, should service water flow be reestablished. By manually establishing service water flow to the drywell coolers only when the drywell temperature is no more than 200 °F, conditions leading to waterhammer will be avoided.

Although waterhammer will be prevented in the drywell coolers administratively following a LOCA, the licensee postulated a scenario by which the failure of the Division I diesel generator could cause the drywell cooler inlet isolation valves to fail in the open position. Under these conditions load sequencing to the standby service water pumps would allow a period of 40 seconds during which there would be no flow to the drywell cooler units. During this time, you calculated that steam could form within the cooler unit coils. Heat transfer from convection, radiation and condensation were included. Steam would form only in the drywell unit cooler tubes because of the short time before service water would be restored. The steam could then be collapsed by the incoming service water, sequenced on after 40 seconds, causing waterhammer. Using the methodology of NUREG/CR-5220, you calculated a peak waterhammer pressure of 1,262 pounds per square inch (psi). GL 96-06 referenced the use of NUREG/CR-5220 methodology for evaluating water hammer conditions. The NUREG/CR-5220 methodology is based on the Joukowski equation, which maximizes calculated pressures by giving no credit for the effect of air in reducing the magnitude of the water hammer. Air would be released during the boiling process and is expected to reduce significantly the actual water hammer pressure compared to that calculated by the Joukowski equation.

You analyzed the effect of a 1,262 psi pressure surge on the service water piping inside the drywell using the methodology of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code). The calculated stresses did not exceed the allowable faulted stress condition. Using the existing seismic stress report methodology for the relief valves in the system, the calculated stresses were found to be substantially smaller than the allowable stresses. Drywell cooler major components were analyzed and it was found that the stress did not exceed the allowable faulted stress condition for any of these components. Drywell service water isolation valves were analyzed for 1,500 psi and the resulting stresses were found to be acceptable.

Since voiding was only calculated for the drywell cooler coils themselves, condensation-induced water hammer that might occur during draining of long horizontal piping sections is not a concern.

In addition to water hammer, GL 96-06 is concerned with the occurrence of two-phase flow conditions within containment air coolers that might affect the assumptions used for heat removal during design basis accidents. Two containment air coolers at River Bend have a safety-related function. The maximum containment temperature will be limited to 125 °F so that two-phase flow conditions will not occur.

Based on the forgoing considerations the NRC staff concludes that the occurrence of a water hammer event such as will affect plant safety as postulated in GL 96-06 is highly unlikely at River Bend. Furthermore the NRC staff concludes that you have provided the required evaluations and have adequately addressed the issues raised in GL 96-06 regarding the potential for water hammer and two-phase flow.

Thermally Induced Overpressurization

In your letter dated January 28, 1997, you identified nine penetrations (one in the containment and eight in the drywell) potentially vulnerable to a water solid volume that may be subjected to an increase in pressure due to heating of the trapped fluid. You determined that the containment penetration is operable on the basis that system pressure would be relieved via two bourdon-tube pressure gages. When the pressure gages are overpressurized, the bourdon-tube within the gauge case will rupture and the system pressure will be relieved through a relief port on the instrument. You determined that the relief of the system pressure will occur below the pressure at which pipe stresses would exceed the design-basis ASME Code-allowable stresses. For your long term corrective action, in your letter dated June 17, 1997, you committed to install an ASME Code-qualified pressure relief device during the seventh refueling outage, which was scheduled for September 12, 1997 (this action has been completed).

Your letter dated January 28, 1997, stated that the eight drywell penetrations contain piping that range from 0.5 to 1.0 inch in diameter. Each penetration is equipped with a single outboard motor operated drywell isolation valve. You postulated that all the eight drywell isolation valves could fail in such a way as to allow leakage to pass from the drywell to the containment. You determined that the resulting drywell bypass leakage, in addition to the current measured drywell bypass leakage during the most recent surveillance test, was well within the limits of technical specification surveillance acceptance criteria and design-basis requirements. Furthermore, the subject piping is supported and restrained such that the postulated failure would not compromise any safety-related piping or equipment in the drywell. Your letter dated June 17, 1997, stated that based on your further review, the current plant configuration meets the drywell design and safety function requirements, and that River Bend design and testing documents would be updated to include postulated drywell bypass leakage due to a failure of all the eight penetrations. The NRC staff finds your evaluation reasonable and acceptable.

The NRC staff concludes that your corrective action and evaluation provide an acceptable resolution for the issue of thermally-induced pressurization of piping runs penetrating the containment.

Summary

The NRC staff has reviewed your responses to GL 96-06 and finds that all of the requested information has been provided, and that the responses are an acceptable resolution for the issues of water hammer and two-phase flow, and thermally induced overpressurization of piping runs penetrating the containment. Therefore, the NRC considers GL 96-06 to be closed for River Bend.

Sincerely,

/RA/

David J. Wrona, Project Manager, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-458

cc: See next page

Summary

The NRC staff has reviewed your responses to GL 96-06 and finds that all of the requested information has been provided, and that the responses are an acceptable resolution for the issues of water hammer and two-phase flow, and thermally induced overpressurization of piping runs penetrating the containment. Therefore, the NRC considers GL 96-06 to be closed for River Bend.

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