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April 15, 1997

U. S. Nuclear Regulatory Commission
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Washington, DC 20555

Subject: Oconee Nuclear Station
Docket Nos. 50-269, 270, 287
Supplemental Response to Generic Letter 96-06:
Assurance of Equipment Operability and Containment
Integrity During Design-Basis Conditions

Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Conditions," was issued on September 30, 1996. GL 96-06 requested licensees to determine if containment air cooler cooling water systems are susceptible to either waterhammer or two-phase flow conditions during postulated accident conditions and to determine if piping systems that penetrate containment are susceptible to thermal expansion of fluid that could lead to overpressurization of piping. Duke Power responded to GL 96-06 in submittals to the NRC dated October 29, 1996 and January 28, 1997.

In the January 28, 1997, submittal, Duke Power committed to provide additional information regarding the analysis of thermally induced overpressurization of closed sections of piping by April 15, 1997. Attached is a summary of the additional analyses that have been performed regarding thermally induced overpressurization of closed sections of piping.

The January 28, 1997, Duke Power submittal also stated that additional thermal-hydraulic analyses would be performed to assess the waterhammer issues described in GL 96-06. Duke Power stated that the results of these analyses, along with a schedule for completion of the piping structural analyses and any modifications that may be necessary, will be submitted to the staff by August 1, 1997.

Duke Power and the staff have had two conference calls to discuss the waterhammer analyses performed in response to GL

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96-06. During these conference calls, Duke Power stated that it has hired a consultant to further quantify the potential for condensation-induced water hammer in the Low Pressure Service Water (LPSW) System. In addition, Duke Power described the results of a LPSW System test during the recent Unit 3 refueling outage to quantify the pressure increase caused by column-rejoining waterhammers. The results of this testing will be considered in the final evaluation to assure that the predicted pressure increase from a column-rejoining waterhammer is conservative with respect to the most limiting conditions postulated during a design basis accident. A failure modes and effects analysis is being documented to confirm that the most limiting conditions have been assumed in the analyses.

The results of these additional waterhammer analyses for the LPSW System will be submitted to the staff by June 30, 1997. In addition, as stated in the January 28, 1997 submittal, a schedule for the piping structural analyses and any modifications to the LPSW System, if necessary, will be provided by August 1, 1997.

Please address any questions to J. E. Burchfield, Jr. at (864) 885-3292.

Very truly yours,

J W Hampton

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Evaluation of potential thermal overpressurization of Reactor Building penetrations

In our January 28, 1997 submittal, Duke Power concluded that overpressurization of fluid system penetrations at Oconee will not occur. In this submittal, all of the penetrations were reviewed to determine the basis that thermal overpressurization was not a concern. The penetrations were found to fit into one of the following categories:

A. Penetrations with relief devices installed between the isolation valves. Thermal overpressure protection is provided by the existing relief devices.

B. Penetrations where both containment isolation valves are located outside containment and temperature variations are minimal. Thermal overpressure protection is not required.

C. Penetrations that utilize an inboard check valve for containment isolation, through which the pressure increase could flow unimpeded to a relief path inside the reactor building. Thermal overpressure protection is not required.

D. Penetrations that are inservice and provide an accident mitigation function during a LOCA. Thermal overpressure protection is not required.

E. Penetrations which contain air, steam or other compressible gas mixtures that are not exposed to sufficient heat so as to experience overpressurization. Thermal overpressure protection is not required.

F. Penetrations that normally operate at fluid temperatures higher than peak containment accident temperature. Thermal overpressure protection is not required.

G. Penetrations that are flanged inside the reactor building, but open-ended outside the reactor building such as the fuel transfer tubes. Thermal overpressure protection is not required.

H. Penetrations which utilize at least one soft seated valve capable of slight leakage or displacement to accommodate thermal expansion following an accident. This feature is attributed to diaphragm, butterfly, and

soft seated valves. Thermal overpressure protection is not required.

I. Penetrations that rely on valve leakage and piping expansion for thermal overpressurization protection.

As stated in the January 28, 1997 submittal, valve leakage is a technically acceptable method of preventing thermal overpressure where small, but identifiable, leakage is detected by local valve leak rate testing. We will continue to verify thermal overpressure protection using this methodology until a better, long term alternative has been found. We are currently performing analyses to determine if an elastic-plastic analysis of the penetration piping will produce acceptable results. The status of this effort is described below.

Since the January submittal, five penetrations per unit with hard seated valves (a total of 15 penetrations) were selected to evaluate the effects of fluid thermal expansion on the piping pressure boundary. The new analysis assumes that these hard seated isolation valves do not leak.

These penetrations have been evaluated using elastic-plastic techniques. A simplified analysis of the piping systems was performed. In this analysis an ANSYS finite element model of the weaker sections of piping was performed with an imposed internal pressure to simulate the expansion of water as it is heated. This model was used to determine the increased piping volume as a function of pressure, up to the plastic limit of the piping.

Based on the total volume between the isolation valves, the increased volume of the water was calculated assuming the entire volume contained in the penetration was heated from an ambient temperature of 70°F to the maximum post-LOCA temperature of 290°F. This increased volume was then assumed to occur in the weakest section of piping between the isolation valves. If enough plastic expansion would occur in the existing length of weaker pipe to allow the displacement of the increased liquid volume, the pressure boundary of the piping remains intact.

Twelve of the fifteen penetrations have been evaluated to be acceptable based on this simplified analysis. The remaining three penetrations will require additional analysis to determine if the pressure boundary will remain intact assuming no valve leakage. Refinements will be made

to the simplified analysis. These refinements include taking credit for thermal expansion of the piping due to the increased ambient temperature, the compressibility of water, the increased volume of all of the piping (not just the weakest pipe) due to the pressure increase, and a more detailed evaluation of the temperature of the fluid in the piping (since not all of the piping is exposed to the post-accident Reactor Building environment).

The refined analyses for the three penetrations that did not meet the simplified analysis have not been completed. It is anticipated that the final results will show that the pressure boundary of all of the pipe will withstand the pressurization caused by the thermal expansion of the fluid.

We anticipate completing this analysis by June 15, 1997. Selection of a long-term resolution to thermal overpressurization of penetration piping, with a schedule for any necessary modifications, should be completed by June 30, 1997.

Evaluation of potential overpressurization between valves LP-1 and LP-2, and between LP-103 and LP-104

Possible overpressure of the piping caused by thermal expansion of the fluid trapped between closed valves LP-1 and LP-2 (in the secondary post-accident boron dilution flow path) and between LP-103 and LP-104 (in the primary post-accident boron dilution flow path) has been evaluated.

As stated in our January 27, 1997 submittal, periodic valve stroking procedures leave the volume between valves LP-103 and LP-104 drained. Since the line is not completely filled with water, no thermal overpressure protection is required.

As committed in the January submittal, a small volume of water is drained from the pipe between valves LP-1 and LP-2 prior to unit restart. Since this leaves the pipe with an adequate "air cushion" to allow expansion of the liquid without overpressurizing the pipe, no additional overpressure protection is required. Duke Power is considering other options, including modifications, to accommodate post-accident thermal expansion of water in this piping so that draining can be eliminated. No additional stress analyses of the piping between valves LP-1 and LP-2 to eliminate the need to partially drain this piping is currently planned.

Based on the information in the January 28, 1997 submittal and the information presented above, we conclude that thermally induced overpressurization of isolated piping systems as described in Generic Letter 96-06 is not an operability concern at Oconee Nuclear Station.