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VICE PRESIDENT
NUCLEAR OPERATIONS

September 29, 1982

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Virgil C. Summer Nuclear Station
Docket No. 50/395
Operating License No. NPF-12
Thermal Sleeves

Dear Mr. Denton:

In response to the Virgil C. Summer Nuclear Station Operating License Condition 2.C.7, the following information is provided. Westinghouse performed a stress evaluation of the Virgil C. Summer Nuclear Station main reactor coolant loop piping with the thermal sleeves removed and the original welding surface ground flush for (a) the 3" normal and alternate charging from the chemical and volume control system, (b) the 6" safety injection system high and low head connections and (c) the 14" pressurizer surge line connection. This evaluation, which considered all design transients and mechanical loads specified in the piping design specification, demonstrates the structural integrity and the ASME Code compliance of the subject nozzles without thermal sleeves.

Due to the similarities in the geometry of all subject nozzles and the similarities in the thermal sleeve designs, the same analytical techniques were applied to all nozzles. The evaluation was separated into two basic regions on the nozzle: (1) the location of the nozzle to pipe field weld at the "safe-end" of the nozzle, and (2) the remaining body of the nozzle including the crotch region.

The stress analysis performed on the nozzles can be summarized as follows. The detailed geometry and material of the nozzle, without a thermal sleeve, was obtained from the appropriate specifications and drawings. Then a detailed two-dimensional finite element model was developed for the nozzle and appropriate representative portions of the large header pipe and attached branch pipe.

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Using piping design specifications containing operating transient descriptions developed on the basis of the systems, criteria, the temperature transients, fluid velocities, number of occurrences, etc., were summarized for all applicable transients, and appropriate loading conditions were developed for the heat transfer analysis using the finite element model. The analysis included a time-history thermal loading for a sufficient duration of time to insure that the maximum stress intensities were calculated for all locations.

Using the same finite element model, stress intensities were calculated from the pipe wall temperature distribution obtained from the heat transfer analysis for all critical locations. The actual fatigue evaluation of the component incorporates the methods and guidelines specified in the ASME Boiler and Pressure Vessel Codes, Section III.

In the analysis of the nozzle without thermal sleeves, two locations were found where maximum peak stress intensity and fatigue usage occurred: (1) the thick part of the nozzle near the crotch region, and (2) the nozzle to the branch pipe field weld. This second region was found to be critical after stress intensification factors were applied to the weld location, as specified in the ASME Code. Assuming the as-welded conditions, a stress concentration factor of 1.7 was applied on top of the calculated values. At the crotch region, a factor of only 1.1 was applied due to the ground flush condition at the weld location.

To complete the fatigue calculation, the external loadings on the nozzle, as calculated for the Virgil C. Summer Nuclear Station, were incorporated and a usage factor was calculated for each nozzle.

The cumulative usage factors calculated on the basis described above, and the external loadings based on the Virgil C. Summer Nuclear Station specific as-built information, indicates that all critical locations meet the ASME Code requirements. Therefore, it is concluded that the nozzles are qualified to withstand all applicable design transients and will maintain their structural integrity without thermal sleeves for the plant design life.

Documentation relating to the reactor coolant branch nozzle analysis has been updated consistent with the removal of the reactor coolant loop piping branch nozzle thermal sleeves.

If you have any questions, please let us know.

Very truly yours,

O. W. Dixon, Jr.

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