Westinghouse Electric Corporation | Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

US Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C., 20555

NS-NRC-90-3557
December 19, 1990

50-364

Attn: J.E. Richardson
Director, Division of Engineering and Systems Technology

Subject: Steam Generator Tube Deformation

Dear Mr. Richardson:

A meeting was held between members of your staff and Alabama Power Company on November 7, 1990, concerning the potential for steam generator tube collapse in the Farley 2 steam generators during a combined LOCA + SSE event. At that meeting, your staff requested a summary of all Westinghouse activity, completed to date, which address the generic implications of the tube collapse issue and a description of the Westinghouse action plan to address the potential for steam generator tube distortion and potential collapse in tubes experiencing cracking at the tube support plate elevations. Your staff also requested background information summarizing the circumstances which led to the identification of the potential issue of steam generator tube deformation. Consequently, in response to its request, this correspondence provides a) additional background information concerning the identification of this issue, b) a discussion of the impact on regulatory compliance (GDC 2, 10 CFR Part 50, Appendix K) of steam generator tube collapse, and, c) a summary of the Westinghouse plan to address the potential for tube collapse in tubes experiencing cracking at the tube support plate elevations. The Westinghouse plan involves a recommendation to utilities to implement an eddy current inspection near wedge locations at the tube support plate elevations to demonstrate the continued maintenance of steam generator tube integrity during subsequent plant operation.

Issue Background

The effect of tube deformation and flow area reduction in the steam generator was analyzed and evaluated for some plants by Westinghouse in the late 1970's and early 1980's. The combination of LOCA and SSE loads led to the following calculated phenomena:

1. LOCA and SSE loads cause the steam generator tube bundle to vibrate.

2. The tube support plates may be deformed as a result of lateral loads at the wedge supports at the periphery of the plate. The tube support plate deformation may cause tube deformation

280008

9012310089 901219
PDR ADOCK 05000364
P PDR

DF01
1/0
Add: J.E. Richardson 1
S. Hoffman PD2-1 1
Ltr

3. During a postulated large LOCA, the primary side depressurizes to containment pressure. Applying the resulting pressure differential to the deformed tubes causes some of these tubes to collapse, and reduces the effective flow area through the steam generator.

4. The reduced flow area increases the resistance to venting of steam generated in the core during the reflood phase of the LOCA, increasing the calculated peak cladding temperature (PCT).

The ability of the steam generator to continue to perform its safety function was established by evaluating the effect of the resulting flow area reduction on the LOCA PCT. The postulated break examined was the steam generator outlet break, because this break was judged to result in the greatest loads on the steam generator, and thus the greatest flow area reduction. It was concluded that the steam generator would continue to meet its safety function because the degree of flow area reduction was small, and the postulated break at the steam generator outlet resulted in a low PCT.

In April of 1990, in considering the effect of the combination of LOCA + SSE loadings on the steam generator component, it was determined that the potential for flow area reduction due to the contribution of SSE loadings should be included in other LOCA analyses. With SSE loadings, flow area reduction may occur in all steam generators (not just the faulted loop). Therefore, it was concluded that the effects of flow area reduction during the most limiting primary pipe break affecting LOCA PCT, i.e., the reactor vessel inlet break (cold leg break LOCA), had to be evaluated to confirm that 10CFR 50.46 limits continue to be met and that the affected steam generators will continue to perform their intended safety function.

Consequently, the action was taken to address the safety significance of steam generator tube collapse during a cold leg break LOCA. The effect of flow area reduction from combined LOCA and SSE loads was estimated. The magnitude of the flow area reduction was considered equivalent to an increased level of steam generator tube plugging. Typically, the area reduction was estimated to range from 0 to 7.5%, depending on the magnitude of the seismic loads. Since detailed non-linear seismic analyses are not available for Series 51 and earlier design steam generators, some area reductions had to be estimated based on available information. For most of these plants, a 5 percent flow area reduction was assumed to occur in each steam generator as a result of the SSE. For these evaluations, the contribution of loadings at the tube support plates from the LOCA cold leg break was assumed negligible, since the additional area reduction, if it occurred, would occur only in the broken loop steam generator. The resulting PCT penalty was estimated to be from 0 to 50^{0}F, depending on the evaluation model used. All plants have been assessed a PCT penalty and no plants were found to exceed the 2200^{0} F PCT limit of 10 CFR50.46 (b)(1). This information was communicated to all customers by November 7, 1990 in the form of a customer information letter.

Regulatory Assessment

Westinghouse recognizes that, for most plants, as required by GDC 2, "Design Basis for Protection against Natural Phenomena", that steam generators must be able to withstand the effects of combined LOCA + SSE loadings and continue to perform their intended safety function. It is judged that this requirement applies to undegraded as well as locally degraded steam generator tubes. Compliance with GDC 2 is addressed below for both conditions.

For tubes which have not experienced cracking at the tube support plate elevations, it is Westinghouse's engineering judgment that the calculation of steam generator tube deformation or collapse as a result of the combination of LOCA loads with SSE loads does not conflict with the requirements of GDC 2. During a large break LOCA, the intended safety functions of the steam generator tubes are to provide a flow path for the venting of steam generated in the core through the RCS pipe break and to provide a flow path such that the other plant systems can perform their intended safety functions in mitigating the LOCA event. Tube deformation has the same effect on the LOCA event as the plugging of steam generator tubes. The effect of tube deformation and/or collapse can be taken into account by assigning an appropriate PCT penalty, or accounting for the area reduction directly in the analysis. Evaluations completed to date show that tube deformation results in acceptable LOCA PCT. From a steam generator structural integrity perspective, Section III of the ASME Code recognizes that inelastic deformation can occur for faulted condition loadings. There are no requirements that equate steam generator tube deformation, per se, with loss of safety function. Cross-sectional bending stresses in the tubes at the tube support plate elevations are considered secondary stresses within the definitions of the ASME Code and need not be considered in establishing the limits for allowable steam generator tube wall degradation. Therefore, for undegraded tubes, for the expected degree of flow area reduction, and despite the calculation showing potential tube collapse for a limited number of tubes, the steam generators continue to perform their required safety functions after the combination of LOCA + SSE loads, meeting the requirements of GDC 2.

During the November 7, 1990 meeting with Alabama Power Company and your staff on this subject, a concern was raised that tubes with partial wall cracks at the tube support plate elevations could progress to through-wall cracks during tube deformation. This may result in the potential for significant secondary to primary inleakage during a LOCA event; it was noted that inleakage is not addressed in the existing ECCS analysis. Westinghouse did not consider the potential for secondary to primary inleakage during resolution of the steam generator tube collapse issue. This is a relatively new issue, not previously addressed, since cracking at the tube support plate elevations had been insignificant in the early 1980's when the tube collapse issue was evaluated in depth. There is ample data available which demonstrates that undegraded tubes maintain their integrity under collapse loads. There is also some data which shows that cracked tubes do not behave significantly differently from uncracked tubes when collapse loads are applied. However, cracked tube data is available only for round or slightly ovalized tubes.

Consequently, an action plan has been formulated to assure that cracked tubes at support plate locations which may be deformed under LOCA and/or SSE are removed from service. This action plan is described in more detail below.

It is important to recognize that the core melt frequency resulting from a combined LOCA + SSE event, subsequent tube collapse, and significant steam generator tube inleakage is very low, on the order of 10^{-8} /RY or less. This estimate takes into account such factors as the possibility of a seismically induced LOCA, the expected occurrence of cracking in a tube as a function of height in the steam generator tube bundle, the localized effect of the tube support plate deformation, and the possibility that a tube which is identified to deform during LOCA + SSE loadings would also contain a partial through-wall crack which would result in significant inleakage. To further reduce the likelihood that cracked tubes would be subjected to collapse loads, eddy current inspection requirements can be established as described below. The inspection plan would reduce the potential for the presence of cracking in the regions of the tube support plate elevations near wedges that are most susceptible to collapse which may then lead to penetration of the primary pressure boundary and significant inleakage during a LOCA + SSE event.

Westinghouse Action Plan

As noted above, detailed analyses which provide an estimate of the degree of flow area reduction due to both seismic and LOCA forces are not available for all steam generators. The information that does exist indicates that the flow area reduction may range from 0 to 7.5 percent, depending on the magnitude of the postulated forces, and accounting for uncertainties. It is difficult to estimate the flow area reduction for a particular steam generator design, based on the results of a different design, due to the differences in the design and materials used for the tube support plates.

While a specific flow area reduction has not been determined for some earlier design steam generators, the risk associated with flow area reduction and tube leakage from a combined seismic and LOCA event has been shown to be exceedingly low. Based on this low risk, it is considered adequate to assume, for those plants which do not have a detailed analysis, that 5 percent of the tubes are susceptible to deformation. The risk of inleakage can be reduced further by identifying eddy current inspections for the affected tubes. Westinghouse actions to develop these inspection plans, and to account for the effect of area reduction in the safety analysis, are described below.

1. Westinghouse will identify those tubes which require inspection for cracks. Five percent of the tubes will be identified, unless more detailed information exists. The schedule for the completion of this work is the end of June 1991.

2. A customer information letter will be issued conservatively recommending that:

   A. The eddy current inspection sample near wedges should be completed during each scheduled steam generator inspection for the tubes identified in item 1) above. Standard bobbin coil inspections of this wedge region sample will be recommended. This recommendation would apply to plants with identified tube cracking at the tube support plate elevations. This eddy current inspection sample would also minimize the potential for excessive primary to secondary leakage during a SSE event.

   B. Any tubes with crack indications found during the eddy current inspection of the wedge region sample should be removed from service.

   It is expected that customer information letters will be issued in the first and second quarters of 1991. Plants with scheduled spring outages will be provided with inspection plans as required.

3. The effect of potential steam generator area reduction on the cold leg break LOCA peak cladding temperature has been either analyzed or estimated for each Westinghouse plant. A value of 5 percent area reduction has been applied, unless a detailed non-linear analysis is available. The results of this evaluation have been communicated to all affected utilities.

Conclusions

Briefly summarizing, there are two issues associated with steam generator tube collapse during a combined LOCA + SSE event. For undegraded tubes at the tube support plate elevations, there is the potential for a reduction in flow area through the steam generator tube bundle due to tube distortion and/or collapse. For tubes with partial through-wall cracks, there is the additional potential that the cracks may propagate through-wall in a collapsed tube and result in significant secondary to primary inleakage. It has been shown that the risk associated with a combined seismic and LOCA event leading to significant inleakage is very low.

Westinghouse has concluded that the potential for flow area reduction does not prevent the steam generator from fulfilling its safety function, as long as tube integrity is maintained.

In order to further increase the likelihood that tubes affected by seismic and/or LOCA loads will maintain their integrity, Westinghouse will recommend that utilities with possible steam generator tube cracking at support plates implement an inspection plan to identify and plug these tubes.

Should you have any questions concerning this information, please do not hesitate to contact me at (412) 374-4868 or M. Y. Young at (412) 374-5081.

Sincerely,

W. J. Johnson, Manager
Westinghouse
Nuclear Safety Department