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March 18, 1983

Mr. R. C. DeYoung, Director
Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Unanalyzed Reactor Vessel Thermal Stress During Cooldown

Dear Mr. DeYoung:

The purpose of this letter is to describe an issue which has been processed as a preliminary safety concern at B&W. This concern has evolved out of further consideration of the 6/11/80, St. Lucie upper head voiding event and has the potential to be generic to all Pressurized Water Reactors.

The concern that thermal stresses, beyond those considered in the original design, may develop in the RV flanges and studs due to large axial temperature gradients across the reactor vessel flanges. These gradients could develop as a result of non-uniform cooling of the reactor coolant within the vessel. These non-uniform effects in the reactor coolant may occur once the reactor coolant pumps are secured and the decay heat removal system has been actuated in the normal cooldown mode or during a natural circulation cooldown. During either mode, a relatively stagnant area exists in the upper head region. Because of this stagnant flow region, there is poor thermal mixing between the fluid in the head and the fluid in the plenum and nozzle regions of the reactor vessel. Possible axial gradients between 150° and 200°F could produce thermal stresses in the vessel flange area or in the studs during either of these two cooling modes that might exceed code allowables when added to stresses already considered.

B&W's initial evaluation of this concern included calculating the temperature profiles of the reactor vessel head and flange and the shell flange during natural circulation cooldown. The average estimated cooldown rate of the reactor vessel head for B&W 177 FA plants after the reactor coolant pumps were secured was about 2°F/hr and decreased to less than 1°F after approximately 28 hours. The calculations assumed a large stagnant volume of

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fluid in the region from the plenum cover to the top of the reactor vessel head. Thus the thermal calculations took no credit for additional cooling that might result from convection in the fluid in the upper head region since no data was available as a basis for establishing this cooling. Only a very small amount of heat transfer was calculated to occur between the stagnant fluid and the lower temperature coolant flowing below. The balance of the heat loss was by conduction through the insulated head to the containment. Temperature differences of up to 200°F developed between the vessel shell at the nozzle belt region and the shell flange and upper head region. Only a small temperature difference was calculated to exist in the metal from the top of dome to the bottom of the head flange. As a result of these temperature gradients, high thermal stresses were postulated in the radius at the outer transition between the vessel flange forging and the nozzle belt forging and in the closure studs.

Although fluid temperature differences of greater than 150°F have been experienced in several plants during cooldown, the extent to which these fluid differences cause metal temperature differences specifically in the flange area is not known. Furthermore, the cooldown rate of the stagnant area as calculated is believed to be conservative. This implies that if the gradients are unacceptable they may be reduced simply by slowing down the cooldown rate of the reactor coolant during the latter portion of the cooldown operation after the reactor coolant pumps are secured.

As a part of the B&W investigation of this matter, we contacted EPRI/NSAC to determine whether they had investigated actual gradient effects in connection with the St. Lucie event. They had done some work using the MARC code which is a general purpose finite element code. This unpublished EPRI investigation used cooldown rates of the coolant in the vessel head region in the neighborhood of 20°F/hr. These significantly higher rates produced improved temperature gradients and the resultant vessel stresses were not excessive. Stresses in the reactor vessel studs were not investigated.

This phenomenon is not likely to be a serious near term safety concern but it does represent an unanalyzed situation with a potential for margin reduction over plant life.

The NRC Staff may have access to data from the St. Lucie or similar events that would enable a better assessment of actual cooldown rates of the fluid in the upper head region or temperature profiles of the metal and studs in RV head closure and nozzle belt regions. In lieu of such information it would seem that the most direct approach to further this investigation is to obtain actual reactor vessel and head flange metal temperatures during a natural circulation cooldown.

B&W has apprised the B&W operating plant owners of this matter and we are initiating discussions with them about how to proceed toward resolution. Any input the NRC can provide to help in technical evaluation of this issue would be appreciated.

If you have any questions on this matter please call me on (804-385-2817) or T. L. Baldwin (804-385-3142) of my staff.

Very truly yours,

J. H. Taylor
Manager, Licensing

JHT/fw
cc: R. B. Borsum - B&W Bethesda Office
W. Layman - EPRI/NSAC
R. H. Vollmer - NRR
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