UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 6, 1992

TO: ALL HOLDERS OF OPERATING LICENSES OR CONSTRUCTION PERMITS FOR PRESSURIZED WATER REACTORS (PWRs)

SUBJECT: RESOLUTION OF GENERIC ISSUE 79, "UNANALYZED REACTOR VESSEL (PWR) THERMAL STRESS DURING NATURAL CONVECTION COOLDOWN" (GENERIC LETTER 92-02)

The U.S. Nuclear Regulatory Commission (NRC) is providing this letter to inform addressees of (1) the NRC's resolution of Generic Issue 79, "Unanalyzed Reactor Vessel (PWR) Thermal Stress During Natural Convection Cooldown" and (2) the conclusions reached by the staff as the result of the evaluations performed to resolve this generic issue. No new requirements are being established and no specific action or written response is required.

Background

On May 5, 1981, the NRC issued Generic Letter (GL) 81-21, "Natural Circulation Cooldown," in response to a natural circulation cooldown (NCC) event that occurred at the St. Lucie Plant, Unit 1, on June 11, 1980. That event caused a void (steam bubble) to form in the reactor vessel head. In GL 81-21, addressed to all operating PWR power reactor licensees and applicants for operating licenses (except for St. Lucie, Unit 1), the NRC requested that addressees determine whether operator training and plant procedures were adequate to effect a controlled NCC from operating conditions to cold shutdown. The NRC requested addressees to demonstrate their capability by test or analysis or both in accordance with Section 50.54(f) of Title 10 of the Code of Federal Regulations (10 CFR 50.54(f)).

By letter of March 18, 1983, the Babcock & Wilcox Company (B&W) notified the NRC that large axial temperature gradients across the RV closure region may cause thermal stresses, beyond those considered in the original design of PWR vessels, to develop in the reactor vessel (RV) flanges and studs. This condition could be outside the design basis of the PWR RVs. During an NCC event, the upper head of a PWR vessel is likely to remain at a higher temperature than the cylindrical portion of the vessel because there is little or no mixing of the fluid in this region with the remainder of the fluid in the reactor vessel. Further, a steam bubble may develop in the top of the vessel as the reactor coolant system is depressurized. The NRC determined that this concern could be a generic safety issue and designated it as Generic Issue 79 (GI-79).

Discussion

B&W performed a detailed analysis of the B&W 177 Fuel Assembly Reactor Vessel (B&W 177) and submitted it to the NRC by letter of October 15, 1984. The NRC used an independent confirmatory analysis performed by the Brookhaven National Laboratory (BNL) in May 1989, to evaluate the B&W submittal regarding the

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stresses in the reactor vessel and the reactor vessel closure studs. The NRC staff also performed a detailed fracture mechanics evaluation of the nozzle shell course and the reactor vessel closure studs. The staff discussed these analyses in NUREG-1374, "An Evaluation of PWR Reactor Vessel Thermal Stress During Natural Convection Cooldown," May 1991, which is enclosed. The NRC concluded that the B&W 177 meets the currently applicable regulatory design stress and fracture prevention criteria for NCC transient conditions up to and including those used by the NRC and its contractor in these analyses, as shown in Figure 3 of NUREG-1374.

In 10 CFR 50.73(a)(2)ii(A) and (B), the NRC requires the licensee to submit a licensee event report for any event that resulted in the nuclear power plant being in an unanalyzed condition that significantly compromised plant safety or in a condition that was outside the design basis of the plant. The analysis noted above considers a B&W 177 to be in an analyzed condition and within its design basis for NCC events that are bounded by the NCC transient profile shown in Figure 3 of NUREG-1374.

The detailed analyses by B&W, NRC, and BNL indicated clearly the extremely complex nature of this type of analysis. This analysis included numerous thermal-hydraulic and mechanical modeling assumptions which, although considered to be conservative, were not confirmed by specifically measured data. Calculated stress results for the B&W 177 were as high as 98-percent of allowable values in the RV studs specified in the American Society of Mechanical Engineers (ASME) Code. While the Code allowable value includes margins, differences between the stresses calculated by B&W and those calculated by BNL, indicated that an RV could be in an unanalyzed condition for certain NCC events, particularly for events complicated by other factors such as an atmospheric dump valve that is stuck open.

The limitations of the analysis, as stated above, prevented the staff from making a definitive conclusion regarding compliance with the applicable regulatory criteria of B&W 177s that might experience an NCC that is outside the bounds of the analysis assumptions, or for B&W non-177s and other PWR vessels that may experience a significant NCC event in the future. However, the staff reviewed the results of the analyses and the qualitative extrapolation of those results and concluded the following:

1. The B&W 177 is considered analyzed for NCC events that are bounded by the NCC transient profile shown in Figure 3 of NUREG-1374.

2. It is extremely unlikely that a single NCC event will cause the failure of any U.S. PWR RV, even if a cooldown rate of 100 °F per hour is exceeded.

3. An NCC event that does not exceed a total cooldown of 100 °F, independent of rate, would not be expected to compromise the safety of any U.S. PWR RV. However, it may result in the RV being outside its documented design basis.

4. Exposure of U.S. PWR RVs to certain NCC transients, particularly transients complicated by other factors such as a stuck-open atmospheric dump valve, may result in a condition that is outside the documented design basis of the RV.

The NRC staff has further concluded that (1) NCC events of the type analyzed, which result in the plant being brought to a cold shutdown condition occur infrequently and (2) the actual severity of a specific NCC event will determine the need for (if any) and the extent of actions that may be required of any licensee following certain NCC events that may place a reactor vessel in an unanalyzed condition or outside its documented design basis. Therefore, no requirement for generic or plant-specific actions was deemed necessary for safety reasons.

Backfit Discussion

The NRC is establishing no new requirements in this generic letter and is requiring no specific action. Existing regulations address any calculations that may be required to be performed after an NCC event. Therefore, the NRC is not imposing a backfit.

This generic letter contains no requirements for collecting information and therefore is not subject to the requirements of the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.).

Although no response to this letter is required, if you have any questions regarding this matter, please contact the technical contact listed below.

Sincerely,

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosure:
NUREG-1374

Technical Contact:
J. D. Page, RES
(301) 492-3941

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