Omaha Public Power District

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June 28, 1982
LIC-82-249

Dr. Stephen H. Hanauer, Director
Division of Safety Technology
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Reference: Docket No. 50-285

Dear Dr. Hanauer:

Pressurized Thermal Shock

As a result of the meeting between the Combustion Engineering Owners Group (CEOG) and the NRC staff on June 23, 1982, Omaha Public Power District (OPPD) would like to provide you with additional information regarding a number of features of the Fort Calhoun Station which would tend to limit or mitigate the effects of potential pressurized thermal shock (PTS) events. First, however, the District would like to take this opportunity to discuss some thoughts concerning the overall resolution of the PTS issue.

The District has reached the conclusion that no near-term concerns regarding PTS exist for the Fort Calhoun reactor vessel. Others evaluating PTS, including the Advisory Committee on Reactor Safeguards (ACRS), appear to have reached similar conclusions on pressurized water reactors (PWR's) in general. The fact that no near-term concerns exist, however, does not address the question of how the PTS issue should be resolved in the long term. We have a number of suggestions on a recommended approach for resolving the issue. We believe such an approach would also enable the NRC to develop a reasonable schedule under which such resolution could take place.

Although there is in excess of twenty (20) EFPY available on the Fort Calhoun reactor vessel before a PTS concern could exist, the District nevertheless agrees that it is in our best interest to resolve this issue in a timely manner. We therefore suggest that some type of screening criteria would be appropriate so that regulatory reviews of PTS could focus on those plants which might develop a potential PTS problem first. The NRC staff had proposed a specific value of RT_{NDT} as a PTS limit. While we would disagree with the use of any single RT_{NDT} value as a regulatory limit, using a RT_{NDT} value as a screening criteria to determine which reactor vessels would warrant additional consideration could aid in resolving the PTS issue.

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The value to be chosen, however, should not be developed in an arbitrary manner or imply that there is a near-term safety concern when a review of the work performed to date leads to the opposite conclusion. In our view, any properly developed screening criteria must not only identify individual vessels, but provide an indication of the number of years available for a plant to determine if a safety question did exist as well as what actions could be taken before they exceeded the screening value. It should also exclude those vessels which would not expect to experience a PTS problem.

The District also suggests that the criteria should reflect a realistic assessment of plant operating histories. Combustion Engineering (CE) has evaluated two (of the seven) significant overcooling events presented by the NRC at the June 9, 1982 industry meeting on PTS (i.e., Ginna and Rancho Seco). The evaluation of the Ginna event indicated a RT_{NDT} of greater than 360°F would have been needed before a condition in which $K_I = K_{IC}$ existed. The evaluation of Rancho Seco indicated a RT_{NDT} value of 315°F would have been needed. (This does not mean to imply crack initiation would have occurred in either case. In these two cases, $K_I = K_{IC}$ does not imply crack initiation, since a warm prestress condition existed in both events.) We believe that these values reflect industry experience more accurately than the initial RT_{NDT} value of 230°F proposed by the NRC and recommend that the NRC conduct similar evaluations in developing its own RT_{NDT} value. We also believe any screening criteria should reflect at least major design differences which affect the plant sensitivity to overcooling transients (e.g., once through as opposed to U-tube steam generators).

A related question to how one develops a RT_{NDT} screening value is how the current RT_{NDT} value for the vessels under consideration is developed. We believe that the information submitted to date, in CEN-189 and the June 23, 1982 NRC meeting with the CEOG, supports the use of a best estimate value for determining an initial value of RT_{NDT}.

The use of a best estimate initial RT_{NDT} value is also justified because of the conservative method used to predict the RT_{NDT} shift. We believe the best method available today for determining RT_{NDT} shift for high copper - high nickel vessels is Regulatory Guide 1.99. We also believe that the Regulatory Guide 1.99 curves represent an upper bound for RT_{NDT} shift predictions. Combining an upper two-sigma value of initial RT_{NDT} with an upper bound shift curve would result in a RT_{NDT} value which is unrealistically high.

The use of the best estimate value is further supported if, as anticipated, the NRC screening criteria considers only crack initiation or the present ability of linear elastic fracture mechanics (LEFM) to predict crack arrest since margin is available beyond LEFM methodology. It should be noted that prediction of crack initiation does not imply vessel failure. Additional margins available in the vessel will be quantified by presently planned EPRI programs.

As noted earlier, we would also like to discuss the actions that OPPD is taking to resolve the PTS issue for Fort Calhoun. While we have calculated that the Fort Calhoun vessel can withstand what we consider to be a bounding PTS transient at end of life fluence, we are nonetheless implementing a fuel management scheme to reduce the fluence to the vessel wall. Such a program would serve to increase the time available to evaluate any other programs that could minimize the impact of potential PTS events.

We are also undertaking measures to reduce the potential impact of overcooling transients today. A safety grade auxiliary feedwater activation system which controls the level (high and low) in the steam generators and isolates the affected steam generator on a steam line break was installed at Fort Calhoun in December 1981. We have also completed an extensive training program for our operations staff regarding PTS and have upgraded our emergency operating procedures. The effectiveness of this training program was reviewed by the NRC's contractor personnel from Battelle Pacific Northwest Laboratories (PNL) during the period of June 8-10, 1982. The PNL report dated June 16, 1982 documents the NRC contractor's findings. The effectiveness of this program is demonstrated by the absence of any recommendations for improvements to the Fort Calhoun PTS training program. OPPD is also participating in the CEOG efforts to perform an overall long-term upgrade of our emergency operating procedures.

OPPD is installing a Safety Parameter Display System (SPDS) at Fort Calhoun. This information system will assist the operator in controlling all off normal events, including potential PTS transients. In addition, we are planning an in-service inspection of the reactor vessel at our next refueling outage. This inspection will increase the confidence we have regarding the conservative assumptions that have been made in LEFM analysis regarding crack size, orientation, and location.

In the longer term, we would expect that ongoing EPRI programs will help quantify the margin available today and aid in resolving questions in areas of uncertainty such as mixing. Time should be allowed for the completion of such programs before additional plant modifications are required for plants approaching a specified RT_{NDT} value.

Based on all the information available to us today, OPPD believes that the actions we are taking at Fort Calhoun are sufficient to resolve the PTS issue for our plant. We also believe that the effects of these actions on Fort Calhoun should be considered by the NRC in any actions that they may recommend in the near future. If a screening criteria indicated further actions may be desired for Fort Calhoun or other plants, we believe that it would be inappropriate to make additional changes (beyond those currently planned) without a plant specific assessment to determine if a valid safety concern existed on those plants.

Since no near-term safety problem exists, each plant could confirm their acceptability for continued safe operation within some specified

period of time. That time period would be determined by the time available before a plant exceeded the screening criteria. One framework for such a plant specific evaluation, based on an assessment of crack initiation dependent on RT_{NDT}, was presented to the NRC by the CEOG on June 23, 1982. Such a framework would appear to be consistent with an implementation of an overall safety goal and would also have the advantage of recognizing plant specific differences including design modifications.

Sincerely,

W. C. Jones
Division Manager
Production Operations

cc: Mr. Robert A. Clark, Chief
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