TENNESSEE VALLEY AUTHORITY
CHATTANOOGA, TENNESSEE 37401
400 Chestnut Street Tower II

February 14, 1983

U.S. Nuclear Regulatory Commission
Region II
Attn: Mr. James P. O'Reilly, Regional Administrator
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

Dear Mr. O'Reilly:

OFFICE OF INSPECTION AND ENFORCEMENT BULLETIN 80-20 - RII:JPO 50-327 AND 50-328 - SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2 - REVISED FINAL REPORT

On October 16, 1980, TVA responded to your letter dated July 24, 1980 which transmitted IE Bulletin 80-18 on Adequate Minimum Flow Through Centrifugal Charging Pumps. Supplemental responses were submitted on November 17, 1980 and July 2, 1981.

Enclosed is a revised final response to the subject bulletin to ensure that the centrifugal charging pumps minimum flow requirements are maintained at all times.

If you have any questions, please get in touch with R. H. Shell at FTS 858-2688.

To the best of my knowledge, I declare the statements contained herein are complete and true.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills

L. M. Mills, Manager
Nuclear Licensing

Enclosure
cc: Mr. Richard C. DeYoung, Director (Enclosure)
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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ENCLOSURE
REVISED FINAL RESPONSE TO OIE BULLETIN 80-18
SEQUOYAH NUCLEAR PLANT

Background

Westinghouse Electric Corporation notified NRC and all Westinghouse utilities pursuant to title 10 CFR 21 in a May 8, 1980 letter of a certain condition where the centrifugal charging pumps (CCPs) could be damaged. NRC consequently notified all utilities of the safety concern in IE bulletin 80-18. This concern exists for plants which utilize the CCPs as emergency core cooling system (ECCS) pumps where the CCPs are automatically started and where the CCP miniflow isolation valves are automatically isolated upon safety injection initiation.

Following a secondary side high energy line rupture and associated reactor trip, reactor coolant system (RCS) pressure and temperature initially decrease. Safety injection is actuated, and the CCPs start to increase RCS inventory. RCS pressure and temperature subsequently increase due to the loss of secondary inventory, steam line and feed line isolation, RCS inventory addition, and reactor core decay heat generation. The accident scenario may vary with rupture size and specific plant design, but it will develop into an RCS heatup transient with accompanying increase in RCS pressure. As RCS pressure increases, the pressurizer power-operated relief valves (PORVs) are designed to limit RCS pressure to 2350 psia. Although these valves are normally available, they are not designed as safety-related equipment. It can be postulated that, due to either loss of offsite power, adverse environment inside containment, the pressurizer PORV in manual mode, or the PORV block valve in a closed position due to PORV leakage, the pressurizer PORVs may not be available. As a result of the RCS heatup and inventory increase, the RCS pressure could rise to the pressurizer safety valve setpoint of 2500 psia within approximately 200 seconds and remain at that pressure until transient "turnaround." Transient "turnaround" can occur between 1800 and 4200 seconds depending on operator action and available equipment. During the initial portion of this transient, the SI termination criteria may not be satisfied. Consequently, the RCS pressure can reach the pressurizer safety valve relief pressure before CCP operation is terminated. During this period, the minimum flow required for CCP operation must be satisfied by flow to the RCS since the CCP miniflow isolation valves are automatically closed on safety injection initiation. This requires that the CCPs be able to deliver their minimum required flow to the RCS at the safety setpoint pressure.

TVA had evaluated this concern and consequently informed NRC of our interim modification in a letter from L. M. Mills to J. P. O'Reilly dated October 16, 1980. The modifications performed for Sequoyah units 1 and 2 as described under Interim Modification I of the Westinghouse letter attached to the bulletin included the following.

1. Verifying that the CCP miniflow return is aligned directly to the CCP suction during normal operation with the alternate return path to the volume control tank isolated (locked closed),

2. Removing the safety injection initiation automatic closure signal from the CCP miniflow isolation valves,

3. Modifying plant emergency operating procedures to instruct the operator to

   A. Close the CCP miniflow isolation valves when the actual RCS pressure drops to the calculated pressure for manual reactor coolant pump trip,

   B. Reopen the CCP miniflow isolation valves should the wide-range RCS pressure subsequently rise to greater than 2,000 psig.

Final Solution

We propose the following solutions to ensure the CCP minimum flow requirements are met at all times.

1. Verify that the CCP miniflow return is aligned directly to CCP suction during normal operation with the alternate return path to the volume control tank isolated (locked closed).

2. Remove the safety injection initiation automatic closure signal from the CCP miniflow isolation valves.

3. Modify plant emergency operating procedures to leave the CCP miniflow isolation valves open at all times.

The reduced charging flow due to the CCP miniflow isolation valves being left open will impact the large break LOCA, small break LOCA, feed line rupture, and steam line rupture analyses.

I. Secondary System Rupture

Sensitivity analyses have been performed by Westinghouse on a generic basis for secondary high energy line ruptures to evaluate the impact of reduced safety injection flow due to normally open miniflow isolation valves. These analyses indicate an insignificant effect on the plant transient response.

A. Feed Line Rupture

Following a feed line rupture, the reactor coolant pressure will reach the pressurizer safety valve setpoint within approximately 100 seconds assuming maximum safeguards with the power-operated relief valves unavailable. With minimum safeguards, the reactor coolant pressure will not reach the pressurizer safety valve setpoint until approximately 300 seconds. The time that the RCS pressure remains at the pressurizer safety valve setpoint is a function of the auxiliary feedwater flow injected into the nonfaulted steam generators and the time at which the operator is assumed to take action. With the

miniflow isolation valves open, the peak RCS pressure and the water discharged by way of the pressurizer safety valves are insignificantly changed from the FSAR results.

B. Steam Line Rupture

The effects of maintaining the miniflow isolation valves in a normally open position were also investigated following a main steam line rupture. For the condition II "credible" steam line rupture, the results of the transient with the miniflow valves open showed that the licensing criterion (no return to criticality after reactor trip) continues to be met. The conditions III and IV main steam line ruptures were also reanalyzed assuming the miniflow valves were open. The results of the analysis showed that, even with reduced safety injection flow into the core, no DNB occurred for any rupture.

II. Large Break Loss of Coolant Accident (LOCA)

The large break ECCS performance analysis was redone for Sequoyah unit 1 cycle 2. Safety injection flows are modeled in both the UHISATAN and UHIWREFLOOD computer codes for the large break LOCA analysis. The CCP flow being evaluated (refer to figure 1) is based on the following.

1. Actual CCP test data degraded by 5 percent,

2. CCP miniflow line open during the injection phase.

The limiting case identified in the cycle 2 large break analysis for Sequoyah unit 1 is the C_D = 0.8 DECLG break with imperfect mixing assumed in the reactor vessel upper head. For this case, the end-of-bypass (EOB) time is 47.893 seconds. The reduction in SI flow will therefore affect the limiting case break ECCS performance from 47.893 seconds through the time at which the calculated peak clad temperature (PCT) turns around. In the UHISATAN calculation for Sequoyah unit 1 cycle 2, pumped safety injection enters a fluid volume which represents the cold leg accumulator discharge line. From 47.893 seconds forward, the UHISATAN-assumed pumped injection flow overstates the amount of water delivered to the RCS by approximately 70 pounds total. The combined cold leg accumulator/pumped SI flow delivery rate is such that EOB time would be extended by only 0.03 seconds in order to achieve filling of the lower plenum. Based on the clad heatup rate in effect at end-of-SATAN, the PCT increase associated with an additional 0.03 seconds of blowdown is less than $1^{O}F$.

The effect of the reduced charging pump flow in refill/reflood is more than adequately accommodated within the current LOCA analysis. Overly conservative input which introduced excessive flow resistance in the RCPs was corrected in a new UHIWREFLOOD run together with the charging pump flow values. The new run produced a reflood transient with a greater flooding rate and a quicker filling of the downcomer. A 0.01-second delay in refill was seen when the reduced flows were used, but the clad heatup rate at that point in time was less than 20 degrees per second making the impact less

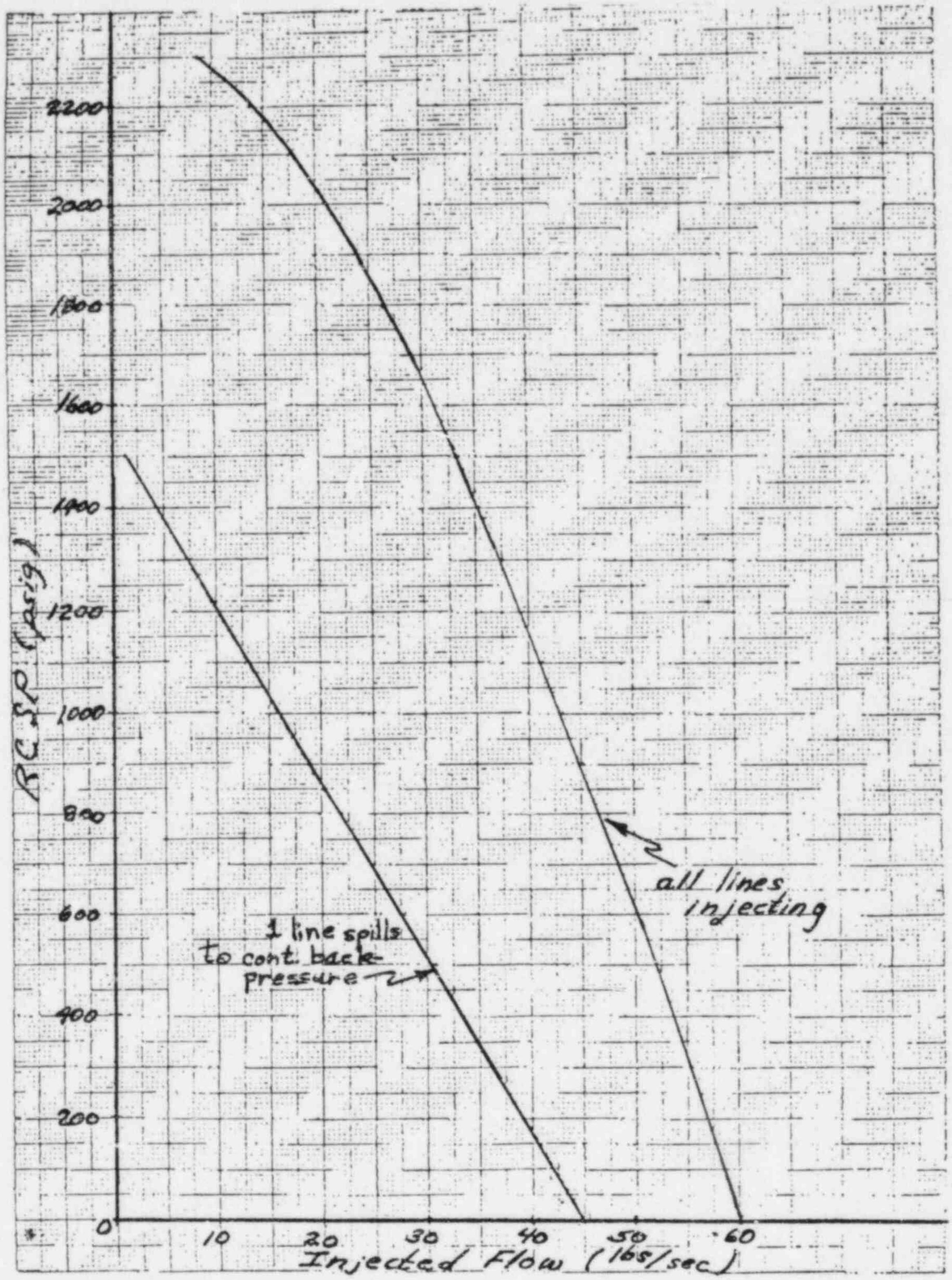
than a 1°F penalty in calculated PCT. The improvement in reflood rate continues throughout the new run producing a more rapid core recovery. The enhanced recovery transient will improve calculated PCT by more than the 2°F in penalties identified, so the analysis done for cycle 2 is bounding and remains applicable.

The present Sequoyah unit 1 cycle 2 PCT is 2137°F as performed using the approved UHI appendix K models reported in WCAP-8479, revision 2 which were updated to include NUREG-0630 fuel rod models as specified in the NRC staff SER on 1981 version of the Westinghouse ECCS evaluation model.

III. Small Break Loss of Coolant Accident (LOCA)

In the Sequoyah small break LOCA FSAR analysis, the 6-inch and 8-inch break sizes gave calculated PCT values only 11°F apart; both values were more than 400°F higher in calculated PCT than the 4-inch break. Since the core is uncovered for a far longer period of time in the 6-inch break case than in the 8-inch case, the impact of lower pumped safety injection should be greater for the 6-inch break. Core uncovery first occurs at 253.2 seconds in the FSAR 6-inch break case. At this time, the RCS pressure is approximately 800 psi, and the UHI accumulator has delivered basically all the water to the reactor vessel. Before this time, UHI water has enabled the core to remain covered; therefore, the impact of reduced SI flow to the cold legs will be small. Once the cold leg accumulator activates at approximately 400 psi, the pumped SI flow will be a small contributor to the total water flow being delivered to the RCS. Clearly, the period of interest in assessing the effect reduced SI flow has on the Sequoyah 6-inch break ECCS performance calculation is the period when system pressure is falling from 800 psi to 400 psi. At 600 psig, the short fall in CCP flow is such that the total pumped SI flow rate is reduced by slightly less than 6 percent. Since the core uncovery period in the 6-inch break case is similar to a non-UHI plant uncovery transient (i.e., UHI effects are minimal), the non-UHI plant small break sensitivity to changes in pumped injection flow may be applied. This sensitivity has been established as a 20°F increase in calculated PCT for each percent reduction in pumped SI flow rate. The impact of the Sequoyah short fall is therefore estimated as (20)6 or a 120°F increase in PCT. The present PCT as reported in the FSAR for a 6-inch break is 1475°F; therefore, a PCT margin of greater than 600°F still remains.

FIGURE 1



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