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TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401
400 Chestnut Street Tower II

October 16, 1980

Mr. James P. O'Reilly, Director
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Region II - Suite 3100
101 Marietta Street
Atlanta, Georgia 30303

Dear Mr. O'Reilly:

OFFICE OF INSPECTION AND ENFORCEMENT BULLETIN 80-18 - NRC-OIE REGION II LETTER RII:JPO 50-327, SEQUOYAH NUCLEAR PLANT UNIT 1 - RESPONSE TO BULLETIN

Enclosed is our complete response to your letter dated July 24, 1980, which transmitted IE Bulletin 80-18 on Adequate Minimum Flow Through Centrifigal Charging Pumps. A partial response to the bulletin was submitted on September 22, 1980. The enclosed response incorporates all of the information transmitted by our September 22, 1980, letter.

TVA employees have expended approximately 55 manhours conducting the review and preparing the reports required by this bulletin. An additional 15 manhours are expected to be expended to complete the required modifications.

If you have any questions, please get in touch with D. L. Lambert at FTS 857-2581.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L M Mills

L. M. Mills, Manager
Nuclear Regulation and Safety

Enclosure
cc: Mr. Victor Stello, Director (Enclosure)
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Washington, DC 20555

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ENCLOSURE

SEQUOYAH NUCLEAR PLANT UNIT 1
RESPONSE TO IE BULLETIN 80-18
ADEQUATE MINIMUM FLOW TO CENTRIFUGAL CHARGING PUMPS

Response to Item 1 of the Bulletin

TVA has completed calculations to determine if the Sequoyah Nuclear Plant unit 1 charging system would maintain adequate pump flow during parallel safety injection operation and determined that adequate flow would not be maintained. The detailed calculations outlined by the Westinghouse Electric Corporation letter (NS-TMA-2245) are included as Attachment 1.

Response to Item 2 of the Bulletin

a. Modifications are planned for Sequoyah unit 1 as described under Interim Modification I of the Westinghouse letter attached to the bulletin. These modifications include:

   (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.

   These modifications are expected to be complete by November 15, 1980. In view of the startup test schedule, TVA does not believe this schedule for modifications has any significant safety implications.

b. As indicated in the Westinghouse Electric Corporation safety evaluation (Attachment 2), if manual operator action is taken to close the CCP miniflow valves when the RCS pressure drops to the calculated pressure for manual reactor coolant pump trip (1,500 psig), no significant change in peak clad temperature (PCT) would be observed. Since tripping of the reactor coolant pumps is itself a manual operator action, it is our opinion that the additional requirement of closing the CCP miniflow valves (two handswitches) will not burden the operator and can be accomplished in the time necessary.

c. The CCP miniflow valves are supplied with shutdown power via the diesel generators. The same post-accident monitoring instrumentation (powered by batteries and/or diesel generators) used to determine the reactor coolant pump trip pressure will be utilized to determine the need for opening or closing the CCP miniflow valves.

d. As indicated in the Westinghouse safety evaluation, the flow available from the CCP's with the modification in place, along with the operator action indicated in item 2.b above, will have a negligible effect on the safety-related analysis (note Attachment 3 for UHI plants).

e. Since the results of the safety-related analyses evaluated in item 2.d indicate the insignificant effects of the interim modification and procedure change, all technical specifications based on these remain valid.

ATTACHMENT 1

SEQUOYAH NUCLEAR PLANT UNIT 1
MINIMUM CENTRIFUGAL CHARGING PUMP FLOW DURING TWO PUMP PARALLEL SAFETY INJECTION CALCULATION FOR NRC IE BULLETIN NO. 80-18

Purpose

Check capability to provide minimum pump flow during parallel safety injection with two centrifugal charging pumps (CCP's).

References

1. NRC IE Bulletin No. 80-18.

2. Letter from T. M. Anderson, Westinghouse Water Reactor Division, to V. Stello, NRC, dated May 8, 1980, No. NS-TMA-2245.

3. Sequoyah Nuclear Plant Unit 1 Preoperational Test WG.1C data.

Calculations

Following the format suggested in Reference 2, using data from Reference 3.

Step 1: Maximum developed head pump flow = 2,600 psid = 6,006 ft. @ 73.1 gpm (pump 1A-1A)

Minimum developed head pump flow = 2,470 psid = 5,705.7 ft. @ 72.3 gpm (pump 1B-1B)

Step 2: Correction for testing error.

Test gauge accuracy = .25% x 3,000 psig = 7.5 psi (17.25 ft.) + 10 psi (23 ft.) reading accuracy = 40.25 ft.

Maximum pump = 6,046.25 ft. @ 73.1 gpm

Minimum pump = 5,665.45 ft. @ 72.3 gpm

Step 3: From construction of pump flow curves, attached, minimum pump = 5,670 ft. @ 60 gpm

Projection of weak pump head point on strong pump operating curve shows flow of 224 gpm.

Total flow from both CCP's guaranteeing 60 gpm to tie weak pump is 224 gpm + 60 gpm = 284 gpm

Step 4: Determination of injection piping head loss.

From Reference 3, runout head of pump 1A-1A = 480 psi
runout flow of pump 1A-1A = 490 gpm

$$K = \frac{\text{Developed Head}}{(\text{Runout Flow Rate})^2} = \frac{\Delta h}{Q^2} = \frac{1104 \text{ ft.}}{(490 \text{ gpm})^2} = 4.6 \times 10^{-3} \text{ ft/gpm}$$

The resistance of the injection piping (Δhf) at the total CCP flow required to maintain 60 gpm through the weak pump is:

$$\Delta hf = KQ^2 = (4.6 \times 10^{-3} \text{ ft/gpm})\ (284 \text{ gpm})^2 = 370.86 \text{ ft.}$$

Step 5: RCS head loss for 4-loop plant - 50 psid (116 ft.)

Step 6: Determining elevational head loss

RWST elevation	739' - 5 3/4"
CCP suction elevation	672' - 11"
RCS cold leg injection nozzle elevation	697' - 1 13/16"
Pressurizer safety valve elevation	757' - 2 3/16"
RWST to CCP suction	66.56'
Minus CCP suction to RCS	-24.23'
Minus RCS to pressurizer S.V. (60.03 ft. assuming a full pressurizer) (Corrected for density difference)	-43.30'
	- 0.97'

Step 7: Calculation of pressurizer safety valve pressure

Note: 1% setting tolerance

Relief pressurizer = 2,485 psig + 25 psig = 2,510 psig (5,798 ft.)

Step 8: Determination of maximum RCS pressurizer pressure at which 60 gpm minimum flow is maintained to weak CCP.

Maximum RCS pressurizer = (CCP developed head @ total CCP flow) - (injection piping head loss) - (Head loss through RCS) - (elevation head loss)

Maximum RCS pressurizer = 5,665.45 - 370.86 - 116 - .97 = 5,177 ft. = 2,241.5 psig

Conclusions

Comparing the maximum RCS pressurizer = 2,241.5 psig with the safety valve relief pressurizer = 2,510 psig, it is evident that the 60 gpm flow required for the weak CCP will not be maintained.

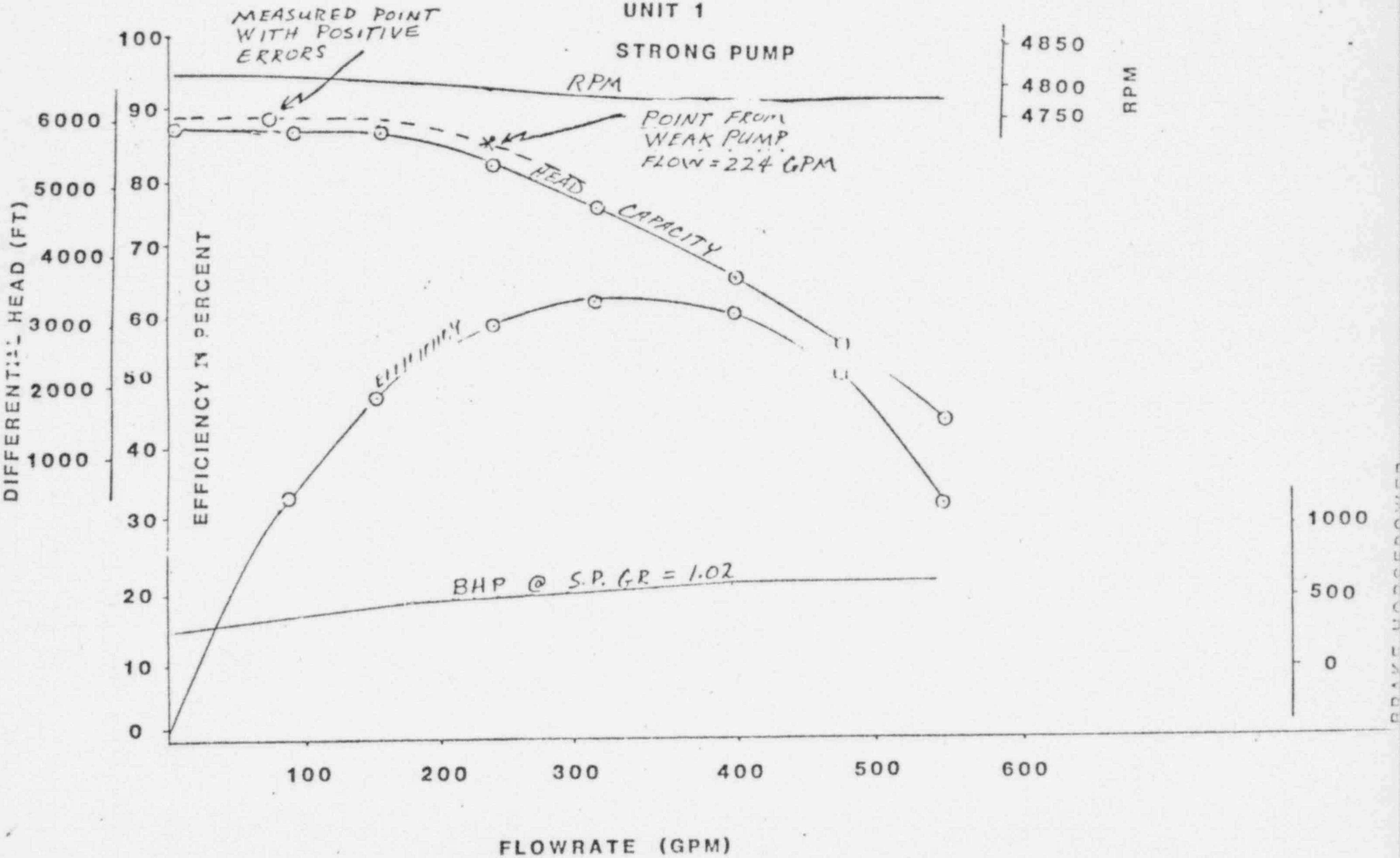
SRM:CLT
9/8/80

PUMP 1A-1A

SEQUOYAH NUCLEAR PLANT

UNIT 1

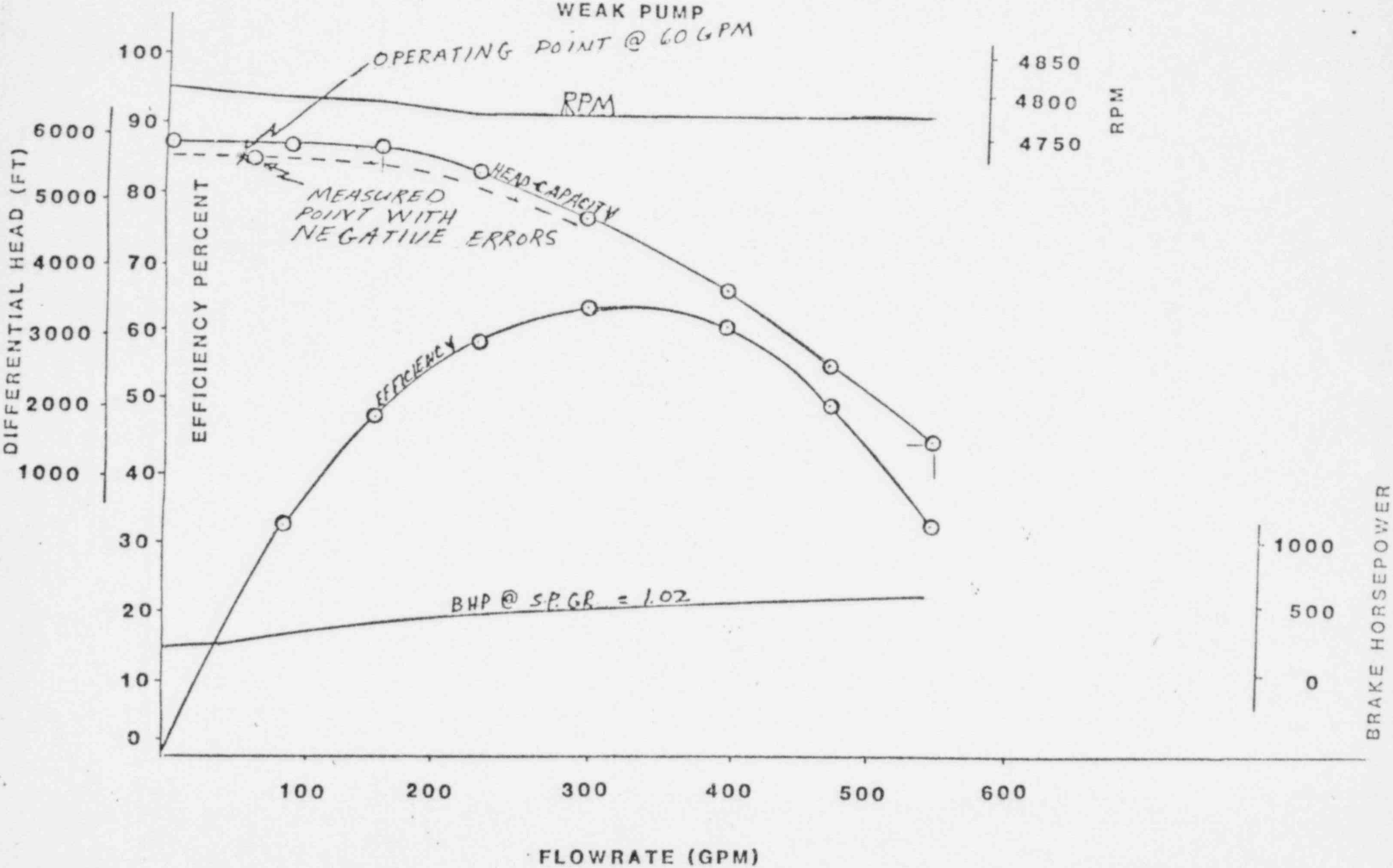
STRONG PUMP



PUMP 1B-1

SEQUOYAH NUCLEAR PLANT

WEAK PUMP



WESTINGHOUSE ELECTRIC CORPORATION SAFETY EVALUATION
CENTRIFUGAL CHARGING PUMP OPERATION
FOLLOWING SECONDARY SIDE HIGH ENERGY LINE RUPTURE

Reference 1: Westinghouse Letter NS-TMA-2245, 5/8/80.

Reference 1 notified the NRC of a concern for consequential damage of one or more centrifugal charging pumps (CCP) following a secondary system high energy line rupture. Reference 1 included a calculational method and sample calculation to permit evaluation of this concern on a plant specific basis. Should a plant specific problem be identified, Westinghouse provided several recommendations for the interim until necessary design modifications can be implemented to resolve the problem. These recommendations included two proposed interim modifications which included:

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

2. Modify 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 2000 psig.

Prior to making this recommendation, Westinghouse evaluated the impact of the recommended operating procedure modifications on the results of the various accidents which initiate safety injection and are sensitive to CCP flow delivery. The accidents evaluated in detail include secondary system ruptures and the spectrum of small loss of coolant accidents. The analytical results for steam generator tube rupture and large loss of coolant accident are not sensitive to a reduction in CCP flow of the magnitude that results from the recommended modifications. This letter functions to supplement Reference 1 and identify the sensitivity of the accident analyses to the recommended modifications. This evaluation is generic in nature.

Secondary System Rupture

Sensitivity analyses have been performed 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. Feedline Rupture

Following a feedline 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 inoperable. With minimum safeguards, the reactor coolant pressure will not reach the pressurizer safety valve setpoint until approximately 300 seconds. The time that the reactor coolant system pressure remains at the pressurizer safety valve setpoint is a function of the auxiliary feedwater flow injected into the non-faulted steam generators and the time at which the operator is assumed to take action. With the miniflow isolation valves open, the peak reactor coolant system pressure and the water discharged via the pressurizer safety valves are insignificantly changed from the FSAR results.

B. Steamline Rupture

The effects of maintaining the miniflow isolation valves in a normally open position was also investigated following a main steamline rupture. For the condition II "credible" steamline 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 condition III and IV main steamline 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.

Small Loss of Coolant Accidents

Sensitivity analyses have been performed to evaluate the impact of reduced safety injection flow on small break loss of coolant accidents (LOCAs). These analyses indicated that miniflow isolation can be delayed, but it must occur at some time into the small break LOCA transient in order to limit the peak clad temperature (PCT) penalty.

The proposed modification delays miniflow isolation and reduces SI flow delivered by approximately 45 gpm at 1250 psia during the delay time period. The impact of this modification was evaluated based on two isolation times: 1) The time equivalent to the RCP trip time, and 2) approximately 10 minutes in the transient, or just prior to system drain to the break for the worst small break sizes. The second time was evaluated to determine the impact if the operator does not isolate miniflow within the proposed prescribed time. The spectrum of small break sizes are considered to encompass all possible small break scenarios. Only cold leg break locations are considered since they will continue to be limiting in terms of PCT.

A. Very small breaks that do not drain the RCS or uncover the core, and maintain RCS pressure above secondary pressure (< ~2" diameter).

   For these break sizes, it is quite possible that the operator may never isolate the miniflow line, since the pressure setpoint will not be reached, and continued pumped SI degradation will persist. However, this will have no adverse consequences in terms of core uncovery and PCT. No core uncovery will be expected for the degraded SI case, similarly to the base comparison case with full SI. The only effect would be a slightly lower equilibration pressure for a given break size.

B. Small breaks that drain the RCS and result in the maximum cladding temperatures (2" < diameter < 6").

   This range of break sizes represents the worst small break size for most plants as determined utilizing the currently approved October 1975 Evaluation Model version, as shown in WCAP-8970-P-A. If miniflow is isolated at the RCP trip setpoint rather than the "S" signal, a reduction in safety injection flow of less than 45 gpm results, averaged for the approximately 50 second period of time separating the two events. This reduction in RCS liquid inventory results in core uncovery less than one second earlier, and has a negligible impact on PCT. If miniflow is isolated at the time of core uncovery, or approximately 10 minutes for break sizes in this range, a greater reduction in RCS liquid inventory results in a core uncovery 10 seconds earlier in the transients resulting in less than a 10°F PCT penalty for the worst size small break. This would not result in any present FSAR small break analysis becoming more limiting than the corresponding large break LOCA FSAR analysis.

If miniflow isolation does not occur at any time into the transient for this category of small LOCA, a PCT penalty of 200°F or more could occur.

C. Small break sizes larger than the worst break through the intermediate break sizes (≥ 6" diameter).

Break sizes in this range have been determined to be non-limiting for small break utilizing the currently approved October 1975 Evaluation Model, WCAP-8970-P-A. If miniflow isolation occurs at the RCP trip time for these break sizes, the negligible effect on PCT presented above also applies. Similarly, if isolation occurs prior to core uncovery, the small (< 10°F) PCT penalty will result as well. However, for these larger break sizes, the time of first core uncovery occurs prior to 10 minutes. If miniflow isolation is not performed until 10 minutes, reduced SI will be delivered during the core uncovery time, which can have a greater impact on PCT. Studies indicate a potential PCT penalty of 40°F resulting for these non-limiting break sizes if miniflow is not isolated until 10 minutes. This is not expected to shift the worst break size to larger breaks, since these breaks are typically hundreds of degrees less than smaller limiting small breaks analyzed with the currently approved Evaluation Model.

For all FSAR small LOCA analyses, one complete train failure is assumed. It is clear that two charging pumps without miniflow isolation provides more flow than one pump with miniflow isolation. The impact presented in this evaluation maintains the one train failure and assumes no miniflow isolation for the remaining pump. If both pumps were operating, the PCT results would be much lower than present FSAR calculations even if miniflow isolation is not assumed to occur for the two pump case. In this situation, the plant FSAR small break calculations remain conservative.

These sensitivity studies form the basis for the recommended interim modifications to the emergency operating procedures. The accidents evaluated are relatively insensitive to the recommended modifications. Further, the accidents evaluated will give results that satisfy acceptance criteria as long as the CCP miniflow is isolated within 10 minutes of event initiation. However, small LOCA sensitivity studies with one SI train operating confirm that small LOCA analyses require miniflow isolation within 10 minutes.

To comply with the recommended modifications, the operator can isolate miniflow at any point in the depressurization transient prior to RCS pressure reaching the RCP trip setpoint. Should a repressurization transient occur, the operator can open CCP miniflow at any point between the RCP trip setpoint and 2000 psig. Such operator actions will ensure that plant accidents satisfy acceptance criteria and protect the CCPs from consequential damage during the repressurization transient that accompanies a secondary system high energy line rupture at high initial power levels.

WESTINGHOUSE ELECTRIC CORPORATION SAFETY EVALUATION

CENTRIFUGAL CHARGING PUMP OPERATION

FOLLOWING SECONDARY SIDE HIGH ENERGY LINE BREAK (UHI PLANT SUPPLEMENT)

The small loss of coolant accident (LOCA) section of the main report was generated primarily for plant applications which do not include upper head injection (UHI) as part of the ECCS design. This supplement provides additional small LOCA information for UHI plants and, together with the main report, assesses the impact of delayed miniflow isolation for small LOCAs for UHI plants.

The model utilized to determine the SI sensitivities and to identify the worst small break size discussed in the main report was the October 1975 Model (WCAP-8970-P-A) version of the Evaluation Model. This model is not yet approved for UHI plant analyses. UHI small break analyses are performed with the December 1974 small break version. However, sensitivity studies performed to determine the effect of pumped SI on small break LOCA PCTs utilizing the December model yielded nearly identical results as presented in the main report. This is expected since the model changes included in the October model do not affect the basi vessel inventory and core boiloff relationships that determine the impact of changes in pumped safety injection on PCT.

An important difference in UHI plant small break analysis results as compared to similar non-UHI plant analysis results is the small break size resulting in the highest PCT. This break size is generally greater for UHI plants than for similar non-UHI plants because of the additional safety injection flow provided by the UHI accumulator at relatively high RCS pressures. The worst small break size for UHI plants may be a six inch diameter break or larger. The main report identified breaks of this size and larger as non-limiting small break sizes. While this is true for non-UHI plants, it is not accurate for typical UHI plant small break analyses. Therefore, the stated 40°F potential penalty for six inch breaks applies to the worst break for UHI plants for the case where miniflow isolation is delayed until 10 minutes. It is Westinghouse's opinion, however, that the stated penalty of 40°F is conservatively high and bounding for UHI plants, for the following reasons: a) The 40°F penalty was based on sensitivity studies performed assuming an approximate 20% reduction in total HPI flow. However, the anticipated 20% reduction actually applies only to the charging pumps. Intermediate head SI pumps are not affected. Therefore, total HPI for plants with intermediate head SI pumps, which includes all UHI plants, will result in less total degradation, and thus a smaller PCT penalty. The high pressure accumulator on UHI plants has a similar effect of reducing the total HPI degradation due to the delay in miniflow isolation. b) The UHI accumulator is a significant source of liquid mass inventory for breaks greater than or equal to six inches in diameter. This additional mass delays the core uncovery time as compared to the same size break occuring on a similar non-UHI plant, since more liquid mass must exit from the break prior to core uncovery. The delay in core uncovery results in clad heatup at a lower power level caused by the decay in residual core heat. Therefore, clad heatup rates are slower which also tends to reduce the sensitivity to changes in HPI delivery rate.

In conclusion, the sensitivity provided for six inch diameter and larger break sizes in the main report represents the worst break size range for UHI plants. The stated 40°F PCT penalty for breaks of this size resultant from a 10 minute delay in miniflow isolation is a conservatively high and bounding value for UHI plants, for the reasons stated above. If miniflow is isolated at the time of RCP trip, the negligible impact on PCT discussed in the main report applies for UHI plants as well. The <10°F penalty resultant if miniflow isolation occurs prior to core uncovery also applies to UHI plants, with the added benefit that this event occurs later in a UHI plant transient than for a non-UHI plant transient of the same break size, allowing more time for the operator to act.