NUREG-0623

Generic Assessment of Delayed Reactor Coolant Pump Trip during Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors

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Generic Assessment of Delayed Reactor Coolant Pump Trip during Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors

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B. Sheron

Bulletins and Orders Task Force
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ABSTRACT

Recent analyses by the PWR vendors have indicated that excessive fuel cladding temperatures could result if the reactor coolant pumps remained running or their trip was delayed during a small break loss-of-coolant accident.

The review of these analyses and analysis models by the NRC staff have resulted in the conclusion that early, automatic pump trip during a small break LOCA is required in order to prevent excessive fuel cladding temperatures.

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1.0 Introduction

Immediately after the accident at Three Mile Island, operation of the reactor coolant pumps during a small break loss-of-coolant accident (LOCA) was considered not only acceptable but a preferred mode of operation. Bulletins 79-05A, 79-06A, and 79-06B were issued immediately after the TMI-2 accident which instructed the operators to keep the pumps running in the event of high pressure injection (HPI) actuation. More extensive evaluations and analyses performed by both the PWR vendor organizations (Babock & Wilcox, Combustion Engineering, and Westinghouse) and the staff subsequent to the TMI-2 accident have concluded that either delayed trip or continuous operation of the reactor coolant pumps during a small break LOCA can lead to predicted fuel cladding temperatures in excess of current licensing limits.

For small breaks between about 2 to 6 inches, operation of the pumps results in liquid discharge from the break beyond the time that the system would have drained down to allow steam discharge from the break had the pumps been immediately tripped. This, in turn, leads to periods of time in which the system liquid inventory is less than the inventory for the case when the pumps are tripped. If the pumps are now tripped during this period of lower system inventory, a longer duration and greater degree of core uncovery can occur.

In response to these findings, the NRC issued Office of Inspection Bulletins 79-05C (B&W Plants) and 79-06C (CE & W Plants) requiring an immediate manual pump trip following reactor trip and indication of HPI actuation. The bulletins also required that certain LOCA analyses be performed regarding delayed reactor coolant pump trip. The purpose of this report is to present the results of the staff review of the vendor analyses and the conclusions drawn by the staff.

2.0 Summary and Conclusions

The staff has reviewed the analysis models and analyses performed in support of predicting the behavior of PWRs during small break LOCAs in which the pumps remain running or the pump trip is delayed.

As a result of this review, the staff has concluded that sufficient uncertainty exists in the thermal-hydraulic phenomenological modeling such that the quantitative results of these small break analyses with the pumps running cannot be accepted at this time (i.e., the specific bounds of the critical break size/critical trip time map). However, the staff does believe that the predicted overall qualitative behavior, supplemented with a basic understanding of the phenomena in question, is sufficient to conclude that small break LOCAs with the pumps operational or with delayed trip can result in more severe consequences than when the pumps are tripped early in the accident.

Therefore, we have concluded that tripping of all of the reactor coolant pumps early in a small break accident is required at this time to preclude the occurrence of excessive fuel cladding temperatures.

While analyses of non-LOCA transients which exhibit behavior similiar to a small break LOCA indicate the results to be aggravated by pump trip, they have still been shown to be acceptable. However, the staff concludes that the probability of tripping the pumps during non-LOCA transients should be minimized to the extent practicable.

We have also concluded that the tripping of the pumps in accordance with the above two requirements (trip pumps for LOCAs, keep pumps running for non-LOCA transients) should be performed automatically and the equipment used to perform this function should be safety grade.

3.0 Background

The question of whether the reactor coolant pumps should remain running or be tripped at the onset of a small break loss-of-coolant accident (SBLOCA) evolved as a direct result of followup analyses for the Three Mile Island accident.

Although pump operation during the initial phases of the accident did not significantly affect the amount of liquid mass lost from the primary system, evaluation of plant data subsequent to the accident revealed two significant facts: (1) when the reactor coolant pumps were running, measured coolant temperatures indicated that the core was adequately cooled, and (2) that when the reactor coolant pumps were shut off, natural circulation did not occur and core damage resulted. The main text of this report discusses the ability to cool the reactor core during a small break LOCA with the pumps running. A discussion of the inability to establish natural circulation in the Three Mile Island plant subsequent to coolant pump trip is discussed in Appendix A to this report.

3.1 Bulletins 79-05A, 79-06A, and 79-06B

Based on the above two facts, the NRC issued Bulletin 79-05A on April 5, 1979, for operating plants designed by the Babcock & Wilcox Company, and Bulletins 79-06A and 79-06B on April 14, 1979, for operating plants designed by Westinghous and Combustion Engineering, respectively. The bulletins stated, in part, "that in the event of HPI initiation, with the reactor coolant pumps (RCPs) operating, at least one pump per coolant loop shall remain operating." These three bulletins are provided as Appendix B to this report.

Industry response to these bulletins was mixed. Owners of Babcock & Wilcox and Combustion Engineering plants followed the directive of the bulletins and both revised their emergency procedures and retrained their operators to assure the pumps would not be tripped during a SBLOCA. Westinghouse did not agree with the bulletins, however. They argued that because all transient and accident analyses previously performed for safety analysis reports assumed the reactor coolant pumps were tripped at the time of reactor coolant trip, and because these analyses showed acceptable core cooling, the pumps should be tripped early into the accident. Their objection to running the pumps was that this represented an accident scenerio not previously analyzed, and existing small break emergency procedures for Westinghouse plants required pump trip at 1250 psig.

As a result of the Westinghouse objection to the bulletin, approximately half of the owners of Westinghouse plants took exception to the bulletin while the other half followed the bulletin directive.

Those utilities which did not follow the bulletins retained their original procedure which required the reactor coolant pump to be tripped at a pressure of 1250 psig and upon confirming safety injection actuation. Prior to the issuance of the bulletins, neither Babcock & Wilcox nor Combustion Engineering plants had any emergency procedures on recommended or required reactor coolant pump operation for transients and accidents.

As previously stated, operation of the reactor coolant pumps during a SBLOCA is not analyzed as part of standard plant safety analyses. A more complete discussion on the basis for this is provided in Section 4.1. Because of this, as well as a lack of supporting analyses by those Westinghouse plant owners who took exception to this aspect of the bulletin, the NRC required the licensees to perform additional analyses on the effect of pump operation during a SBLOCA (Ref. 1).

3.2 Analysis Results

On July 18, 1979, Babcock & Wilcox presented to the NRC staff the preliminary results of their analyses of the effect of pump operation during a SBLOCA (Ref. 2). Their results showed that if the pumps remained running through the entire accident, the core remained acceptably cool. However, continuous operation of the coolant pumps resulted in the primary coolant system evolving to a high system void fraction* early in the accident and remaining relatively high until the system depressurized enough to actuate the low pressure injection system (LPIS) and recover the system liquid inventory. Their analyses also showed that water injected by the core flood tanks was of little benefit because of the slow depressurization and subsequent slow injection rate. This evolution of a high system void fraction early in the accident is shown in figure 3.2-1, and the corresponding depressurization rates are shown in figure 3.2-2. Both figures are from reference(5).

Because the system void fraction evolved to such a high value, B&W examined what would happen if the reactor coolant pumps were tripped at some time into the accident when the system void fraction was high. They arbitrarily assumed that the pumps were tripped when the system void fraction was 90 percent. At the time of pump trip, the liquid that was previously dispersed around the primary system through pumping action now collapsed down to the low points of the primary system, such as the bottom of the vessel and steam generators. This resulted in a significant uncovery of the reactor core, resulting in an insufficient amount of liquid being available to provide acceptable core cooling.

Due to design features as well as the temporal behavior of the system void fraction, B&W concluded that unacceptable consequences would result from delayed reactor coolant pump trip only for a range of small break sizes and a range of pump trip delay times after accident initiation.

Subsequent to receiving this information from Babcock & Wilcox, and in discussions with both Westinghouse and Combustion Engineering regarding similar calculations, it was concluded that a similar "window" of break sizes and delays in pump trip time existed for their plants that would

* System void fraction is the ratio of the volume of steam in the primary system to the volume of steam plus liquid in the primary system.

BREAK SPECTRUM-AVERAGE SYSTEM VOID FRACTION
WITH THE RC PUMPS OPERATIVE AND 2 HPI PUMPS

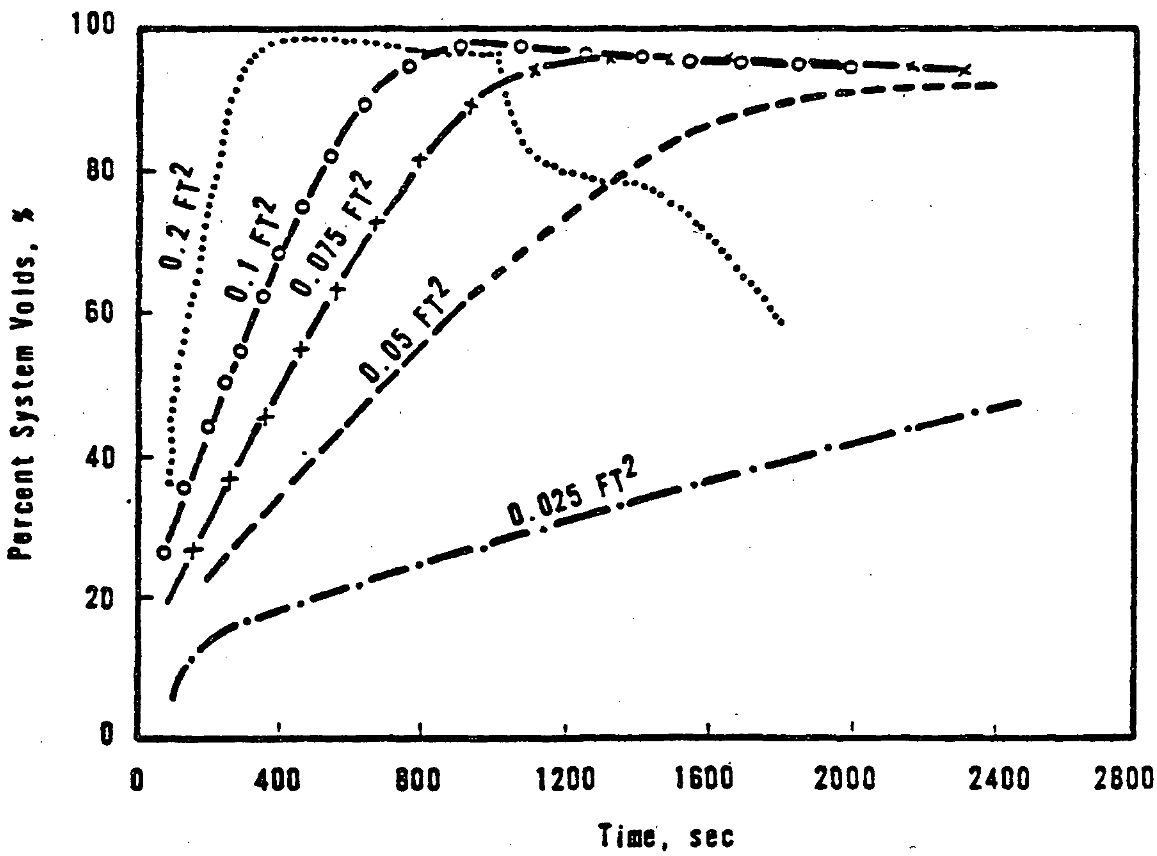


FIGURE 3.2-1
(FROM REFERENCE 5)

BREAK SPECTRUM-RC PRESSURE WITH
THE RC PUMPS OPERATIVE AND 2 HPI PUMPS

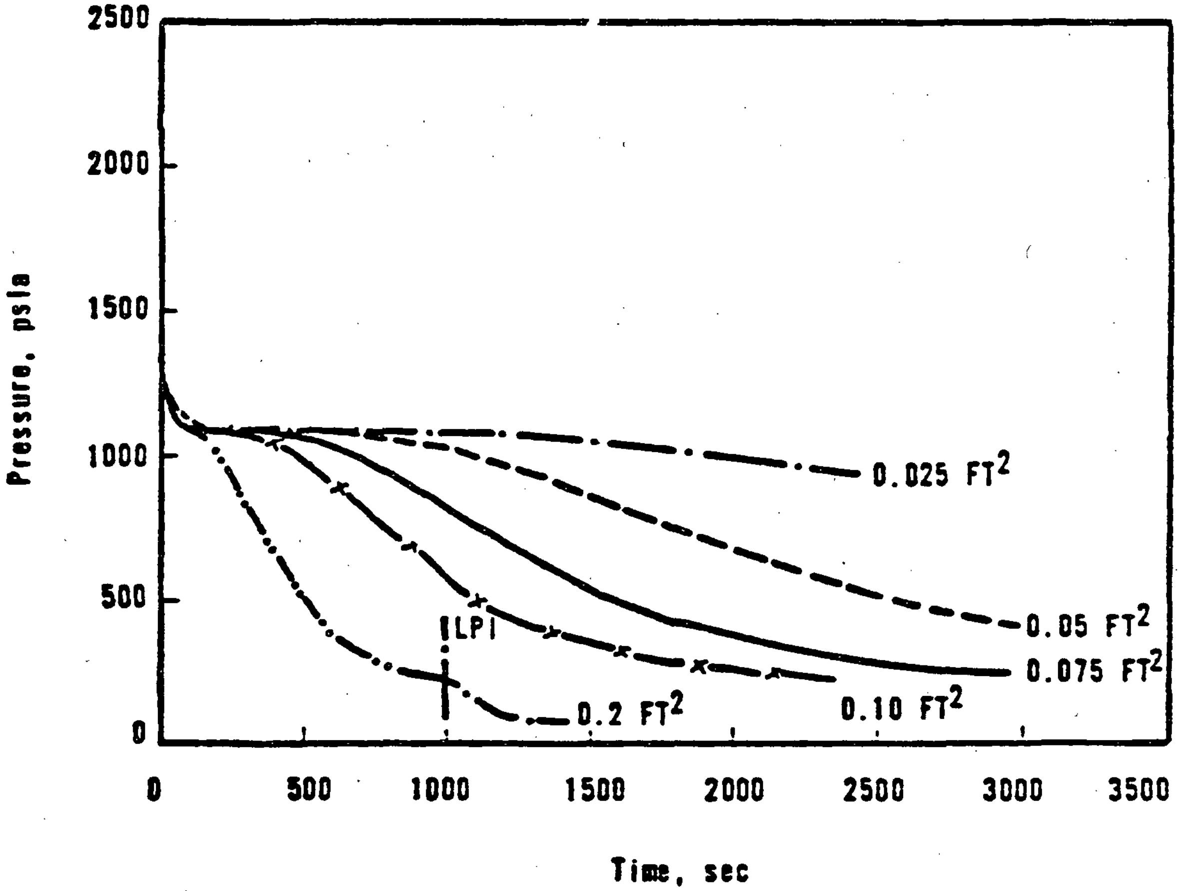


FIGURE 3.2-2
(FROM REFERENCE 5)

lead to calculated peak cladding temperatures which exceed the licensing limit of 2200°F.*

Based on the results of their evaluations, on July 20, 1979, Babcock & Wilcox issued a letter to all of its customers advising them to trip all of the reactor coolant pumps in the event of a reactor trip and safety injection signal on low reactor coolant system pressure. A copy of this letter is provided in Appendix C to this report.

Evaluation of all of the analyses available at the time, along with the B&W recommendation to immediately trip the reactor coolant pumps following HPI actuation to assure that peak cladding temperatures did not exceed licensing limits, resulted in the staff conclusion to issue Bulletins 79-05C and 79-06C on July 26, 1979.

3.3 Bulletins 79-05C and 79-06C

These bulletins, provided in Appendix D to this report, required the following short and long term actions to be taken by the nuclear plant operators (owners).

3.3.1 Short-Term Actions

(1) In the interim, until the design change required by the long-term action of this bulletin has been incorporated, institute the following actions at your facilities:

(a) Upon reactor trip and initiation of HPI caused by low reactor coolant system pressure, immediately trip all operating RCPs.

(b) Provide two licensed operators in the control room at all times during operation to accomplish this action and other immediate and followup actions required during such an occurrence. For facilities with dual control rooms, a total of three licensed operators in the dual control room at all times meets the requirements of this bulletin.

(2) Perform and submit a report of LOCA analyses for your plants for a range of small break sizes and a range of time lapses between reactor trip and pump trip. For each pair of values of the parameters, determine the peak cladding temperature (PCT) which results. The range of values for each parameter must be wide enough to assure that the maximum PCT or, if appropriate, the region containing PCTs greater than 2200°F is identified.

(3) Based on the analyses done under item 2, above, develop new guidelines for operator action for both LOCA and non-LOCA transients, that take into account the impact of RCP trip requirements. For Babcock & Wilcox designed reactors, such guidelines should include appropriate requirements to fill the steam generators to a higher level, following RCP trip, to promote natural circulation flow.

(4) Revise emergency procedures and train all licensed reactor operators and senior reactor operators based on the guidelines developed under item 3, above.

*This limit, and others, is specified in 10 CFR 50.46.

(5) Provide analyses and develop guidelines and procedures related to inadequate core cooling (as discussed in Section 2.1.9 of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations") and define the conditions under which a restart of the RCPs should be attempted.

3.3.2 Long-Term Actions

(1) Propose and submit a design change which will assure automatic tripping of the operating RCPs under all circumstances in which this action may be needed.

(2) The schedule for responding to the above items of the bulletins was also specified as follows.

3.3.3 Schedule

The schedule for the short-term actions of this bulletin is:

Item 1: Effective upon receipt of this bulletin;
Item 2: Within 30 days of receipt of this bulletin;
Item 3: Within 30 days of receipt of this bulletin;
Item 4: Within 45 days of receipt of this bulletin; and
Item 5: October 31, 1979 (as noted in Table B-2 of NUREG-0578 under item 3).

This report addresses the resolution of items 1 through 4 of the short-term actions and single long-term action.

3.4 Summary of Responses to Bulletins 79-05C and 79-06C Short-Term Items 1-4 and Long-Term Item

(1) Trip all RCPs on reactor trip and SI actuation. Have two operators in control room at all times to do this.
All operating plants complied immediately with this directive. This compliance was verified by inspectors from the NRC's regional offices of the Office of Inspection and Enforcement. The regional inspectors were briefed on the bulletin and its basis by representatives from NRC's Office of Nuclear Reactor Regulation.*

(2) Submit report on LOCA analyses for range of break sizes and time lapses between reactor trip and pump trip.
In response to this item, all three PWR vendors submitted reports documenting the information requested. These reports are listed as References 3, 4, and 5 of this report. The results of the analyses provided in these reports are discussed in Section 4.0.

(3) Develop operator action guidelines for pump trip for both LOCA and non-LOCA transients.
Proposed guidelines for CE plants were presented in Reference 2. For Westinghouse plants, the guidelines originally presented in the small break analysis report (Reference 6) were considered consistent with the pump trip guidance; therefore, additional guidelines were not proposed. B&W issued guidelines consistant with the bulletin directive to the B&W owners on August 30, 1979 (reference 7)

*While Bulletins 79-05C and 79-06C were issued by the Office of Inspection and Enforcement, the technical basis was developed by the Office of Nuclear Reactor Regulation.

(4) Revise emergency procedures and train operators based on item 3 guidelines.

As described for item 3, above, Westinghouse Plant owners have indicated that because they believe no revision to the emergency procedure guidelines is necessary, all retraining of operators for the procedures developed from the Reference 6 guidelines would be sufficient to meet this requirement of the bulletin.

Combustion Engineering plants owners have not yet responded to item 4 of the bulletin. They are awaiting staff approval of the item 3 proposed guidelines prior to the retraining of any operators. B&W owners have committed to retrain operators by 9/7/79.

Long-term item - propose system for automatically tripping pumps for SBLOCA.

Both Westinghouse and Combusion Engineering plant owners proposed that because of the calculated time available for pump trip, this action need not be automatic but could be performed manually by the operators. Some B&W plant owners, however, have proposed an automatic pump trip.

4.0 Analysis Methods and Results

4.1 Previous Analytical Studies

Since safety analyses were first performed, the large break usually has shown to be the most limiting of the loss-of-coolant accidents postulated for large, commercial reactors. In general, cladding temperatures were calculated to be higher for the large break than for the small break. Also, because of the fast transient nature of the large break LOCA (i.e., rapid depressurization, flow reversals, etc.), the analytical and empirical models used in the anlyses were subject to considerable technical review and evaluation as compared to small break LOCA models.

These major considerations pointed out above, as well as many of the minor areas which are not mentioned, led to a general concensus that large breaks were more limiting than small breaks.

The recent consideration that the delayed trip of the reactor coolant pumps during a LOCA could provide more limiting consequences than the immediate trip is not new. In fact, (II)(3) of Appendix K to 10 CFR 50 specifically states that the effect of pumps running versus pumps tripped should be evaluated as part of required modeling sensitivities. Sensitivity studies on the effects of pumps running versus pumps off have been performed by both the vendors and the staff for the large break LOCA. These studies concluded that pump trip with reactor trip produced more limiting results than the assumption of pumps running for most plants.* Moreover, the assumption of pump trip with reactor trip was consistent with the assumption of loss of offsite power and failure of an emergency diesel generator to start at the initiation of the accident, which in turn produces more limiting results than assuming offsite power is available.

*Large break LOCA analyses for Westinghouse two-loop plants have shown that continuous pump operation during the accident led to slightly higher predicted peak cladding temperatures.

Because the more limiting assumption of loss of offsite power in conjunction with the LOCA was consistent with the results that pump trip at reactor trip was more limiting for most large break LOCAs, it was generally assumed that this scenario would also produce the most limiting results for small break LOCAs without the benefit of supporting analyses.

4.2 General Description of Thermal-Hydraulic Phenomena With Delayed Pump Trip

4.2.1 Case 1: Immediate Pump Trip

For a small break LOCA in which the reactor coolant pumps are assumed to be tripped immediately, system recovery is based on the emergency core cooling system eventually providing more water to the primary system than is leaving the system out of the break. For the larger end of the small break spectrum, in which some core uncovery is predicted, the time at which core level recovery is started is generally keyed to the time the liquid level in the system is lowered enough so that steam rather than water is discharged through the break. Because the volumetric flow rate of steam out of the break is about five times greater than liquid flow at the representative pressures, transition of the break flow from liquid to steam results in a rapid system depressurization. This, in turn, produces greater HPI (high pressure injection) flow, and will also actuate the passive core flooding system when the pressure drops below the actuation pressure.

Although there are differences in plant design among the three PWR's, in general the worst small break location with the pumps tripped is calculated to occur on the bottom of a cold leg pipe for all plant types. The reason is that this location usually requires the greatest amount of liquid to be discharged from the system before break uncovery, and hence the greatest amount of core uncovery.

4.2.2 Case 2: No Pump Trip

In the event of a SBLOCA in which the reactor coolant pumps are not tripped, primary system thermal-hydraulic performance is not well known at this time. Information submitted by the three PWR vendors has indicated that predicted system performance may be highly dependent upon the thermal-hydraulic models used in the calculations, as well as physical differences among the three PWR designs. Of particular concern is whether the liquid and vapor phases of the coolant flow in the individual components (vessel, pipes, etc.) of the primary system are better represented as separated (heterogeneous) or mixed (homogeneous) phases. A more detailed discussion of modeling uncertainties is discussed in Section 4.3 of this report.

In the event that the pumps are able to circulate a two-phase mixture throughout the primary system for the entirety of the accident, then adequate core cooling of the fuel by either a two-phase mixture or high velocity steam is predicted. (It is noted that prior to turning the last pump off at Three Mile Island, measured coolant temperatures did not exceed saturation, indicating good cooling of the fuel by a two-phase mixture.) However, if the pumping action is not calculated to maintain a two-phase or high velocity steam flow through the core, then excessive cladding temperatures could result.

Regardless of which models better represent reality, a general conclusion can be drawn from the anlyses of all three PWR vendors. This is that when the pumps are running, a significantly higher amount of primary system liquid inventory is calculated to exit the break than the case when the pumps are immediately tripped. This is attributed to the role of primary coolant pump operation in redistributing the liquid mass in the system such that the break is fed with liquid for a longer period of time than would be calculated with the pumps immediately tripped. Hence, the system will evolve to a much higher system void fraction when the pumps remain running. In particular, these differences are expected to become significant when the pump running case continues to supply a two-phase mixture to the break after a pump trip case would predict break uncovery and steam discharge.

4.2.3 Case 3: Delayed Pump Trip

As discussed in Case 2, above, as well as in Section 3.2, if the system evolves to a high void fraction because of pump operation, and then the reactor coolant pumps are tripped for any reason, the void collapse could lead to substantial core uncovery and high cladding temperatures in excess of the 2200°F licensing limit. As a result of their analyses, all three PWR vendors have identified a range of break sizes and pump trip delay times in which the calculated peak cladding temperature exceeds the licensing limit. A more detailed discussion of these results is provided in Section 4.4.

4.3 Modeling Methods

This section discusses some of the key phenomena that are expected to exist during a small break LOCA, and to what degree present small break models consider these phenomena. A summary of how these phenomena are treated in each PWR vendor's small break analysis model that is used when the pumps are assumed to remain running is provided in Table 4.3-1. Also included in this table are the modeling methods employed by EG&G, Idaho in performing audit calculations for the NRC staff.

4.3.1 Two-Phase Equilibrium Modeling

All of the computer codes used by the PWR vendors to predict thermal-hydraulic behavior for small break LOCAs assume thermodynamic equilibrium within a given fluid control volume. This assumption requires that all steam and liquid calculated to exist within a control volume be at the same temperature and pressure. This requires the assumption of instantaneous mass and energy transfer to maintain equilibrium (for example, superheated steam and subcooled or saturated water would not be allowed to exist within the same control volume).

4.3.2 Homogeneous Modeling

Within a fluid control volume, the fluid within that volume can be represented homogeneously or as separated phases. By homogeneously, it is meant that the steam and water are completely mixed within the control volume, and that the mixture behaves as a single fluid with weighted-average properties (density, etc.) of the mixture. This type of modeling is usually used for regions of high velocity two-phase flow where the momentum forces acting on the fluid are high compared to gravity forces.

4.3.3 Heterogeneous (Phase Separation) Modeling

When the gravity forces acting on a fluid are significant compared with other forces, a separation of the liquid and vapor phases would be expected.

TABLE 4.3-1

MODEL DIFFERENCES DURING SBLOCA WITH PUMPS RUNNING

Item	MODEL W	CE	B&W	RELAP/MOD-7
Cold Leg Pump Discharge Pipe	Stratified flow	Homogeneous flow	Homogeneous flow	Heterogeneous flow
Downcomer	Heterogeneous model	Model switches from homogeneous to heterogeneous model when drift velocity criteria met	Homogeneous flow	Heterogeneous flow
Core	Heterogeneous flow	Heterogeneous flow	Homogeneous flow	Heterogeneous flow
Hot Leg Pipe	Homogeneous flow for CL breaks; heterogeneous control volumes for hot leg breaks; no counter-current flow allowed for either case	Heterogeneous flow with provisions for draining	Homogeneous flow	Heterogeneous flow. No counter-current flow allowed
Steam Generator Hot Side Tubes	Homogeneous flow	Drift flux model - allows liquid fallback to hot leg if possible	Homogeneous flow	Heterogeneous flow - no vertical slip or fluid runback to hot leg
Steam Generator Cold Side Tubes	Homogeneous flow	Homogeneous flow	Homogeneous flow	Heterogeneous flow - no vertical slip
Cold leg Loop Seal (suction pipe)	Homogeneous flow	Homogeneous flow	Homogeneous flow	Homogeneous flow

TABLE 4.3-1 (Continued)

Model/Method	W	CE	B&W	EG&G Idaho
ECC Injection	No injection assumed in broken loop for cold leg breaks	No spillage assumed for hot leg breaks - no injection assumed in broken loop for cold leg breaks	-30% spillage of ECC water injected in broken loop	Consistent with vendor assumptions
ECC Injection Location	Downcomer/ lower plenum node (cold leg by design)	Downcomer (cold leg by design)	Cold Leg (cold leg by design)	W - upper downcomer CE - cold leg B&W - cold leg by design)
Quench Behavior During Recovery	No carryover accounted for	No carryover accounted for	No carryover accounted for	No carryover accounted for
Steam Super-Heat Calculation	Superheating considered (description proprietary)	Axial coolant nodes in core; superheating of each node allowed	No superheat calculated due to single control volume model of core; All core heat added to liquid phase; Separate heatup model calculates superheat but uses CRAFT mixture level	3 axial coolant nodes in core; superheating of each node allowed
Core fluid quality	Thermodynamic equilibrium assumed - actual *quality not* calculated	Thermodynamic equilibrium assumed - actual quality *not calculated*	Thermodynamic equilibrium assumed - actual *quality not* calculated	Thermodynamic equilibrium assumed - actual quality *not calculated*

In this case, free vapor is calculated to collect at the top of a fluid control volume, while a two-phase mixture layer (i.e., steam bubbles dispersed in liquid) is calculated to exist in the lower part of the fluid control volume. Within the two-phase mixture layer, steam can be generated by heat addition or flashing, and it is calculated to separate from the two-phase mixture via a bubble rise or drift velocity model.

One of the advantages of a phase separation model is that exit flows from a control volume will take on the properties of of the phase which is predicted to exist at that flow path elevation.

A pictorial representation of both homogeneous and phase separation models is provided in Figure 4.3-1.

4.3.4 ECC Injection

There are a number of uncertainties introduced into small break analyses through uncertainties in ECC injection modeling. Because of the equilibrium assumptions made in the analysis codes, subcooled water injected into a steam-filled pipe would be calculated to instantaneously condense all of the steam it was capable of condensing. This rapidly lowers the calculated pressure at the injection location and can produce calculational instabilities in the computer code. To minimize the calculational instabilities, some vendors model the injection location either in the downcomer (CE) or in the lower plenum (W). This is done so that the water is injected into a region with liquid and therefore the rapid condensation of steam is eliminated. In reality, all three vendors have their high pressure injection in the cold leg pipe, and both CE and Westinghouse have accumulator injection in the cold leg. B&W plants have the accumulator injection in the upper downcomer annulus.

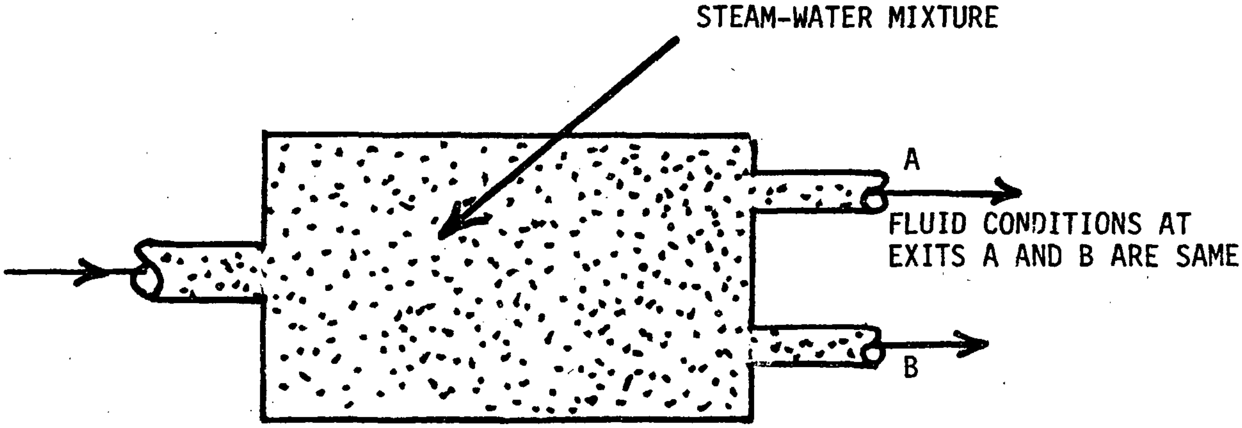
As part of the staff audit calculations (see Section 4.5), a sensitivity study (with pumps tripped) on safety injection location was performed. The results are shown in Figure 4.3-2 and illustrate the sensitivity of the pressure to injection location. Figure 4.3-3 shows the sensitivity of the amount of ECC water that is calculated to be injected as a result of the system pressure sensitivity to injection location.

Also, in conjunction with this is the uncertainty introduced by the size of the fluid control volume chosen. The amount of steam available for condensation within a control volume is dependent upon the size of the control volume.

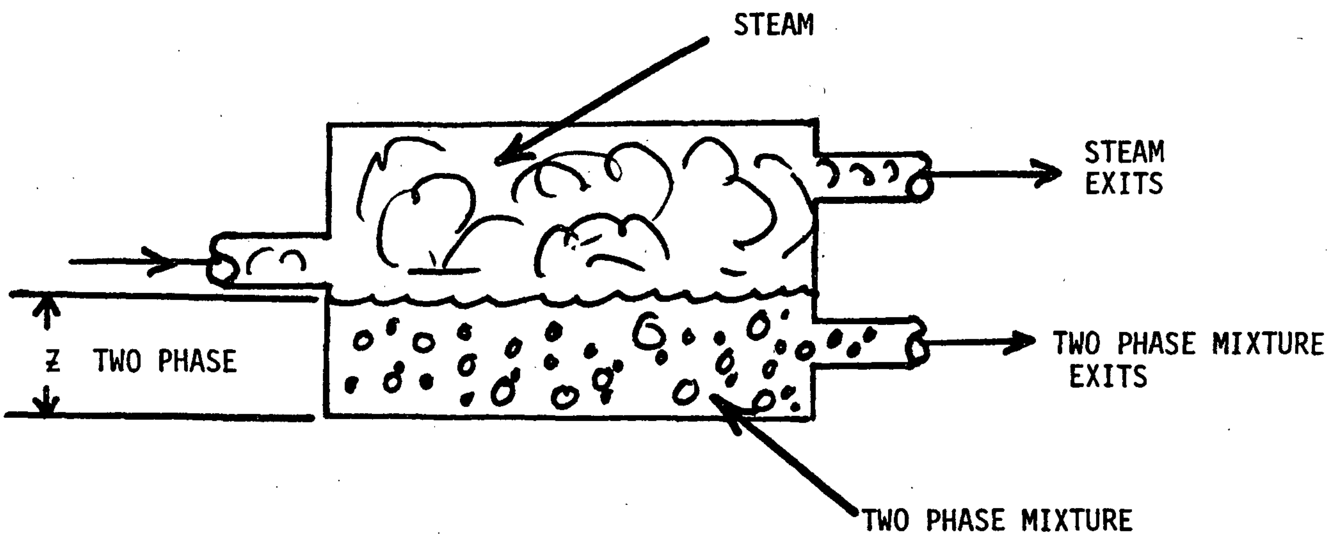
The above concerns result in an uncertainty in the local pressure at the injection location. Since the amount of injection flow is determined by the pressure difference between the accumulator tank (or safety injection discharge) and the local pressure in the pipe, uncertainties in local pressure will result in uncertainties in the amount of ECC water injected into the system. An example of system pressure decreases at the time of safety injection tank activation can be seen in Figure 4.3-4 as calculated by CE.

4.3.5 Pump Performance and Discharge

Pump performance during single phase operation is characterized by a set of homologous curves which relate pump speed, flow and developed head. Under conditions in which a two-phase mixture is being pumped, however,



HOMOGENEOUS CONTROL VOLUME



HETEROGENEOUS CONTROL VOLUME

FIGURE 4.3-1

HOMOGENEOUS AND HETEROGENEOUS (PHASE SEPARATION)
FLUID CONTROL VOLUME REPRESENTATIONS

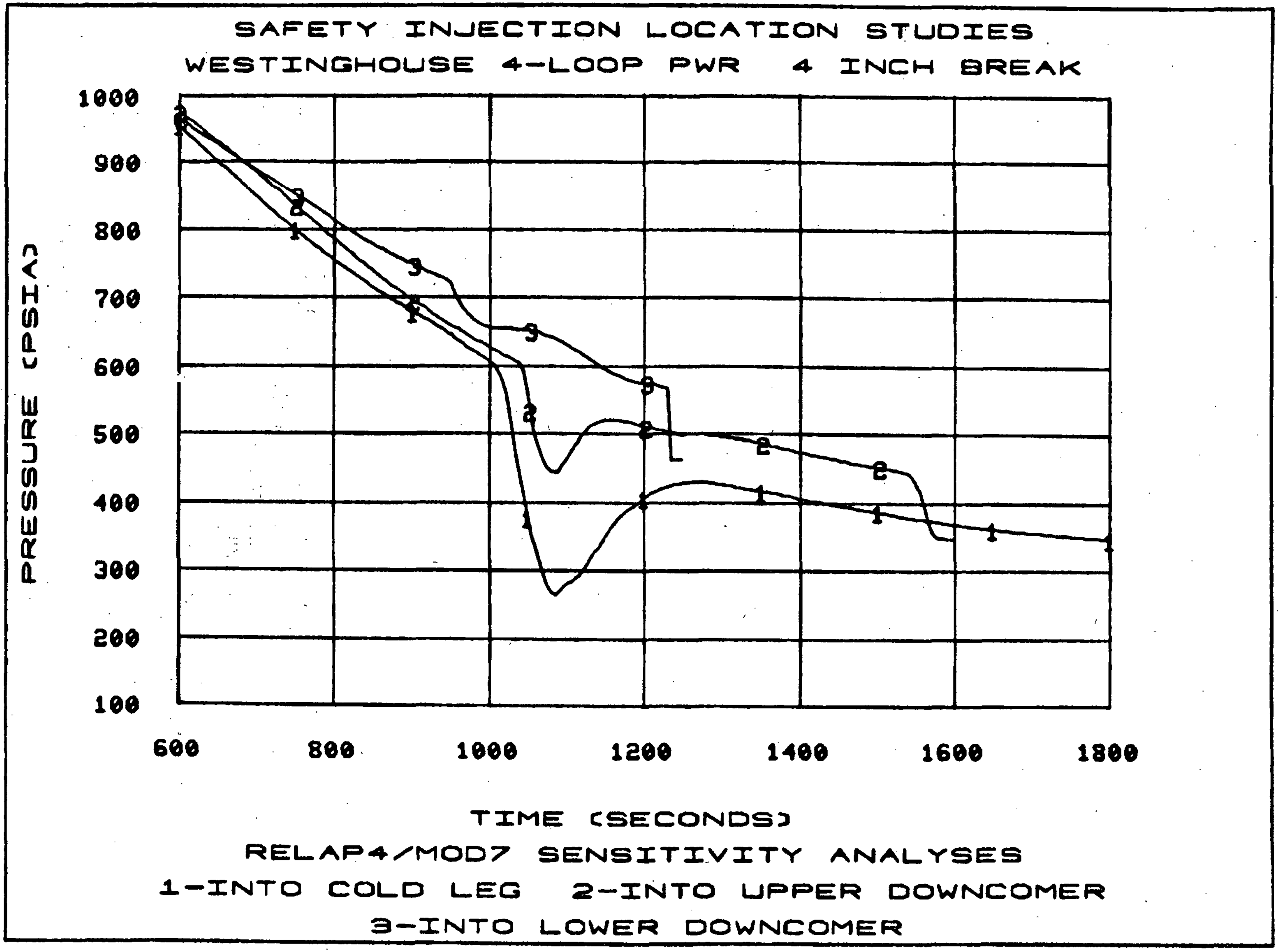


FIGURE 4.3-2

ECC INJECTION LOCATION SENSETIVITY

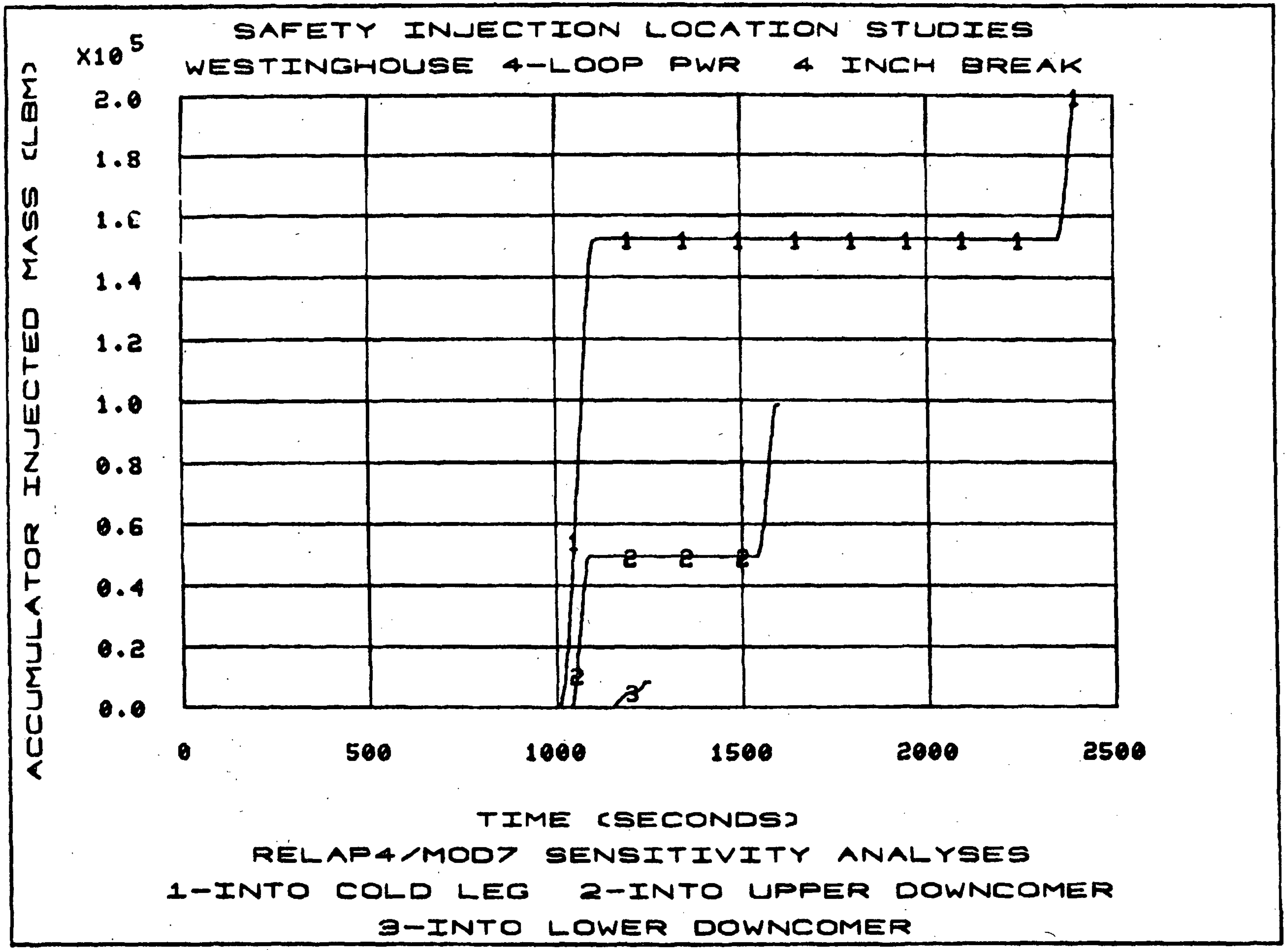


FIGURE 4.3-3

EFFECT OF ECCS MODEL INJECTION LOCATION ON AMOUNT OF ECC INJECTED

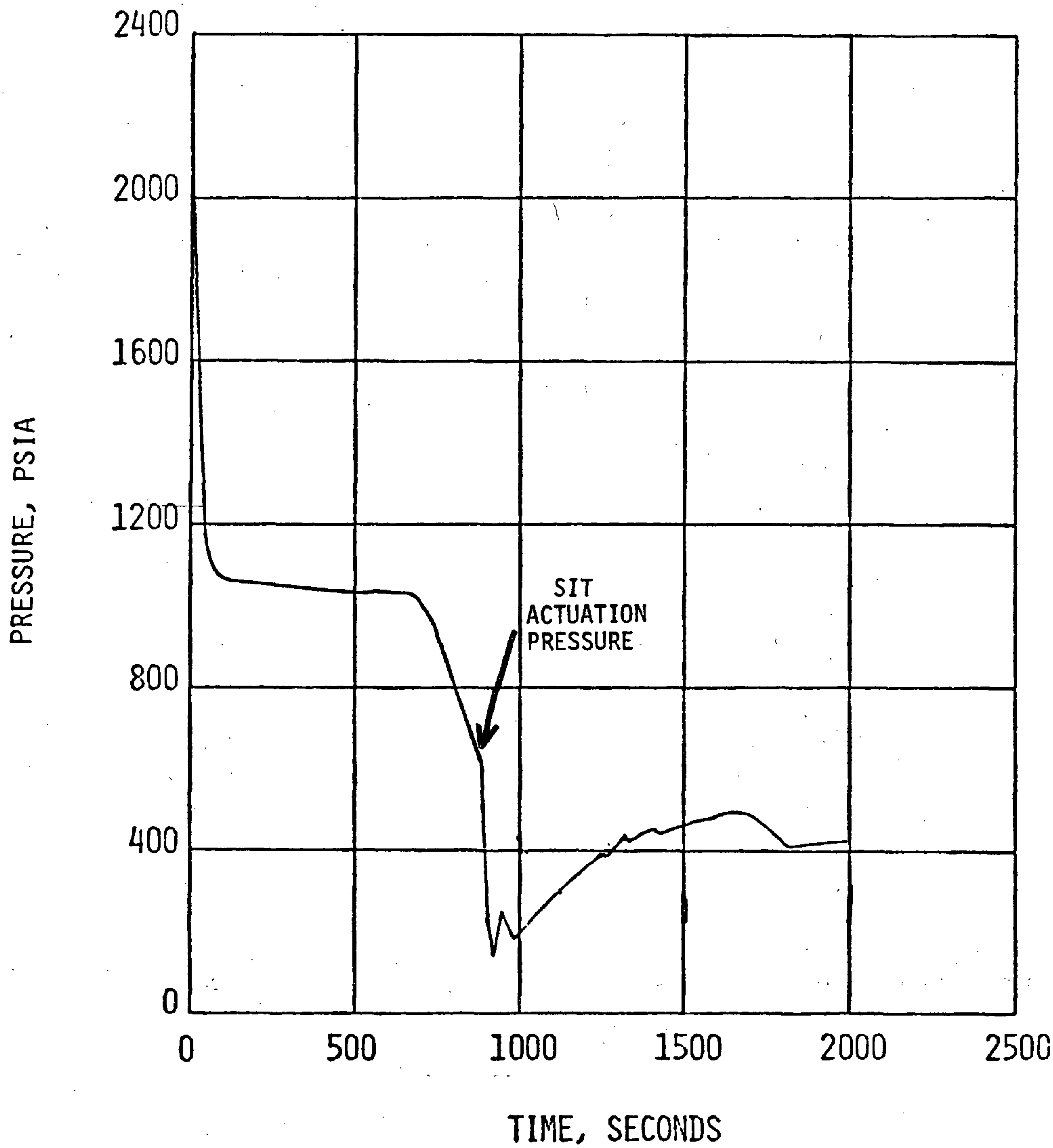


FIGURE 4.3- 4

EXAMPLE OF CALCULATED SYSTEM PRESSURE DECREASE AT TIME OF SAFETY INJECTION TANK ACTUATION (600 PSIG SIT'S) the homologous curves derived for single-phase conditions are no longer directly applicable, but must be modified to reflect the degradation in performance that occurs.

In general, the pump head and torque degrade very rapidly once the flow becomes two-phase. As the flow evolves into a pure steam flow, the head and torque recover and again the single-phase homologous curves are applicable for the pure vapor phase.

Pump head and torque degradation are typically represented by degradation multipliers which multiply the single-phase head and torque values. There is considerable uncertainty in these degradation multipliers. This is because of scatter in the available data and, because full-scale pumps are not used, the prototypicality of the pumps used to obtain the data is questionable.

Two concerns have been identified regarding pump performance effects in a small break LOCA. The first regards the discharge flow regime from the pumps. The pumps could be envisioned to behave as a centrifugal separator in which case the liquid flow at the discharge could be stratified. Another approach would be to envision the pump as a "mixer," in which case the pump discharge would appear as a homogeneous mixture. Thus, if a small break was postulated on the bottom or top of the cold leg pipe, whether the flow was separated or homogenized would have a significant effect on the mass and energy flow leaving through the break. The second concern regards the uncertainty in developed head in the high void fraction and steam flow regimes.

Both Westinghouse and Combustion Engineering assume phase separation in the vessel and downcomer when the pumps are running. This leads to a calculated depression of the downcomer level. Westinghouse has calculated that the pump will depress the downcomer mixture level enough to allow steam to flow down the downcomer and up through the core. This steam flow from the downcomer is relatively high and is calculated to produce good core cooling if the pumps remain in operation during the entire accident. Combustion Engineering, however, does not calculate that sufficient pump head can be delivered in some cases to depress the downcomer level low enough to pass steam into the core. As a consequence, Combustion Engineering calculates that for certain break sizes, the cladding temperatures will exceed licensing limits even if the pumps remain in operation throughout the entire small break LOCA.

There are two main factors which influence whether the pump can depress the level in the downcomer enough to pass steam into the core. These are the pump head and the elevation of the flow path from the downcomer to the core inlet. There is approximately a 1½-foot difference in elevation of the downcomer to core flow path with respect to the cold leg centerline between CE and Westinghouse plants. However, the uncertainty in pump head for high void and steam flow conditions is expected to overshadow this elevation difference. As such, the ability of Westinghouse plants to cool the core if the pumps run through the entire accident is as yet uncertain.

Because the B&W model represents all fluid control volumes as homogeneous, the detailed concerns regarding level depression are not appropriate, although overall modeling of the level depression phenomenon is. The large difference in phenomenological behavior between heterogeneous and homogeneous modeling is underscored in this area.

An additional concern has also been raised by C. Michelson of TVA, an ACRS consultant, regarding pump integrity during two-phase operation. This concern is that, during periods of steam flow, the delivery of a water slug into the pump could produce loadings which would damage or rupture the primary coolant boundary. This effect has not been addressed by the three PWR vendors.

4.3.6 Hot Leg and Steam Generator Hydraulic Behavior

When the reactor vessel is assumed to behave as a phase separation control volume, the steam flow entering the hot leg and steam generator will be due to core boil-off and any steam pumped into the core from the downcomer. Depending upon modeling assumptions, component elevations, etc., it is possible that the liquid in the hot leg side of the steam generator (for U-tube steam generators only) either carried in from the hot leg or produced from condensing steam will not have sufficient velocity to carry over the top of the U-tubes and fall into the pump suction piping in the cold leg. (For B&W plants with the "candy cane" hot leg, the inability of liquid to carry over the top of the "candy cane" would depend on the same principles as applied to a U-tube generator, except condensation effects would not be applicable.) Instead, this liquid would be expected to drain back out of the steam generator and back into the horizontal portion of the hot leg and/or the reactor vessel.

The uncertainty of whether liquid is carried over the U-tubes (or "candy cane" bend) or if it drains back is compounded by the competing effects on break flow, as described below:

(1) Hot Leg Breaks

For hot leg breaks, and in particular those on the bottom of the hot leg pipe, the inability to carry water over the top of the steam generator and the draining back of this water increases the liquid supply to hot leg breaks. However, the draining of liquid also reduces the liquid supplied to the cold leg and thus produces earlier pump head degradation. Earlier pump head degradation will reduce the elevation of the two-phase level in the inner vessel and possibly produce earlier hot leg uncovery, which would tend to switch the break flow from two-phase to steam earlier.

(2) Cold Leg Breaks

By allowing water to drain back into the hot leg, less water is available to feed a cold leg break. Thus, less inventory loss would be expected.

Westinghouse has submitted a more advanced analysis code, NOTRUMP (Ref. 8) for small break analysis. Preliminary calculations of small breaks in the cold leg indicate that when draining is allowed, it provides a liquid supply to the core, as well as producing a faster transitioning of the break from liquid to steam, and the consequences in terms of mass inventory lost and cladding temperatures are less severe than those calculated when no draining is allowed. No hot leg breaks were calculated with this code, and no experimental verification was presented. However, it does qualitatively show the potential importance of the draining behavior on small break LOCAs.

At present, both the Westinghouse and B&W models with pumps running do not allow for draining of the steam generators back into the hot leg and vessel. CE's model does allow for liquid draining in the steam generator if it is calculated to occur, and their hot leg model does have the provision for draining a liquid back to the vessel.

Because Combustion Engineering has found the hot leg small breaks with pumps running to be most limiting, while both Westinghouse and B&W have found the cold leg small breaks to be most limiting, the dependency of the conclusions on the drain assumption and drain model is considered significant.

4.3.7 Quench Behavior During Core Recovery

Presently, all of the vendor small break models do not permit liquid to exit the top of the core if the two-phase mixture level is calculated to be below the top of the core. Moreover, the models assume that cladding quench is instantaneous at all elevations in the core below the two-phase mixture level.

During the reflood portion of a large break LOCA, much experimental data (i.e., FLECHT) has been collected which shows that during the refilling of a hot, uncovered core, there are significant amounts of liquid carried out of core (carryover) due to entrainment from the quench process. Moreover, the elevation below which the cladding is quenched can be significantly below the elevation of the two-phase mixture level.

Some vendors have calculated core recovery rates in excess of 10 inches per second for certain break sizes. At present, there is considerable uncertainty as to whether the predicted recovery rates are realistic or whether they occur because of unrealistically high amounts of accumulator water calculated to be injected due to pressure uncertainties (see Section 4.3.4 discussion). In addition, it is questionable if the cladding would actually quench as fast as the two-phase mixture level is calculated to cover the core or whether there would be significant amounts of water carried out of the core due to entrainment from the quench process.

An illustration of this rapid quenching as calculated by Westinghouse is shown in Figure 4.3-5. As can be seen, rapid quenches are predicted to occur at approximately 700 and 1200 seconds. From the pressure history for this analysis shown in Figure 4.3-6, it can be seen that both quenches are the result of the accumulator injection.

4.3.8 Steam Superheat and Core Fluid Quality

Section 4.3.7 discussed the potential for liquid entrainment at the two-phase mixture interface. If liquid entrainment occurs, then less than the maximum amount of steam superheating will occur downstream of the mixture interface. Part of the heat transferred above the mixture level will be used to evaporate the entrained liquid, while the remainder will be used to superheat the steam. In addition, some desuperheating will take place due to heat transfer from the steam to the liquid.

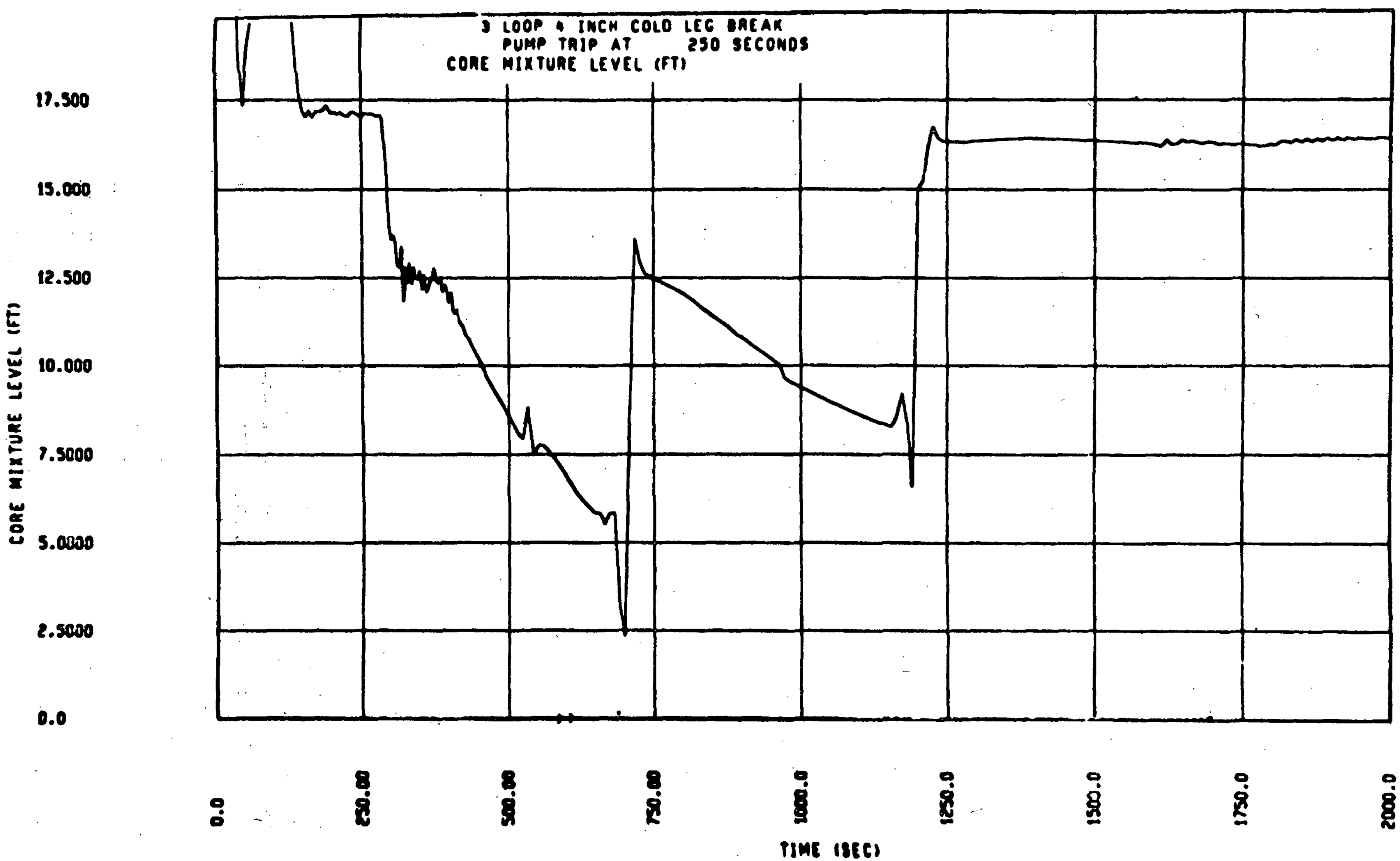


FIGURE 4.3-5

ILLUSTRATION OF CALCULATED LEVEL RECOVERY DURING SMALL BREAKS

(FROM WCAP-9584)

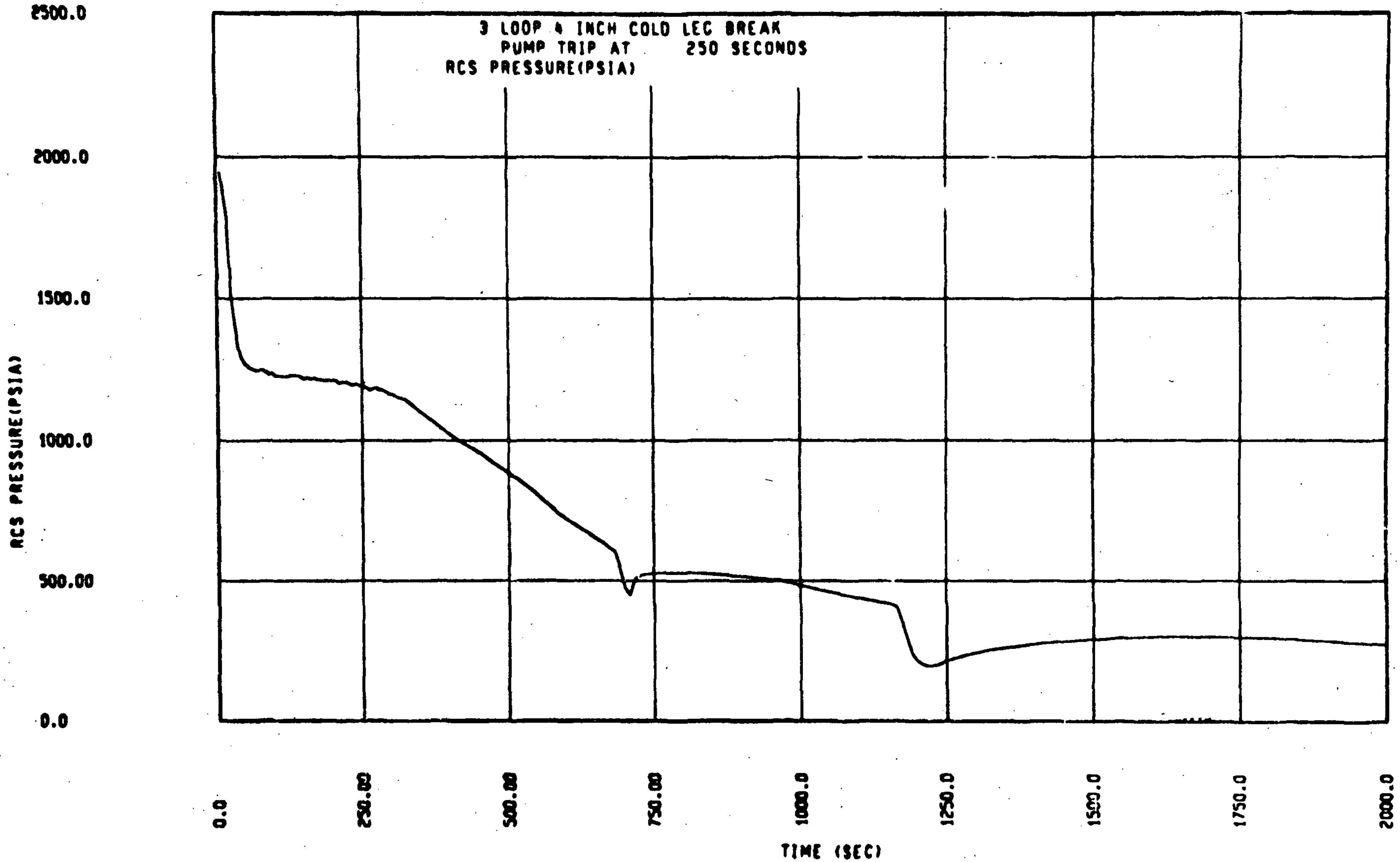


FIGURE 4.3-6
ILLUSTRATION OF PRESSURE HISTORY DURNING SMALL BREAKS (CALCULATED)
(FROM WCAP-9584)

These effects can be competing regarding the effect on fuel clading temperature. While entrainment produces greater core uncovery, it also reduces the steam superheat downstream of the two-phase mixture level, and thus the net effect on cladding temperatures could be positive or negative.

All three PWR vendors account for superheating of steam in their small break models as restricted by the equilibrium assumption. Thus, even if entrainment were significant, the present codes could not adequately account for it unless a more refined heat transfer model based on applicable data (such as the FLECHT correlation, which is based on low pressure reflood date) is developed.

4.3.9 Initial Condition Uncertainty

Small break models used by the vendors typically represent large fluid volumes, such as plenums, as one mixed volume at a uniform temperature. These large volumes, however, can have complex structure within them (such as the upper plenum and upper head), which may produce regions of stagnant fluid at higher temperatures than the mixed mean temperature due to hotter fluid exiting the hot regions of the core. If such regions did exist, they can potentially affect the hydraulic behavior by changing the predicted depressurization characteristics.

4.4 Analysis Results

Each vendor provided an anlysis of delayed reactor coolant pump trip using a model they considered met the requirements of Appendix K to 10 CFR 50. The purpose of the analysis was to identify that combination of break size-pump trip delay time pairs in which the peak cladding temperature exceeded 2200°F. The significant results of this study are discussed in the following sections, and Table 4.4-1 summarizes the analysis results.

4.4.1 Break Size-Pump Trip Delay Time

As a result of their analyses, each vendor concluded that a range of break size-pump trip delay time combinations existed in which the peak cladding temperature was calculated to exceed 2200°F. For B&W plants, the range of break sizes was approximately 0.025 ft^2 to 0.25 ft^2, and the range of unacceptable pump trip delay times ranged from 2 minutes to 2 hours. This result is shown graphically in Figure 4.4-1, which is reproduced from Reference 5. Because the size and location of a small break would be unknown to the operators, the most significant aspect of Figure 4.4-1 is the maximum time available for the operators to trip the coolant pumps. From this figure, it can be seen that the maximum calculated time available for pump trip to preclude excessive cladding temperatures for any size break is between 2 and 3 minutes. From this figure, it is also seen that for pump operation throughout the entire accident, the results are calculated to be acceptable for all break sizes.

Westinghouse calculations for a typical three-loop plant showed that the range of break sizes for which delayed pump trip could be unacceptable were between 2-inch and 4-inch diameter holes, or a break area range of 0.02 ft^2 to 0.087 ft^2. A graphical representation of these results is shown in Figure 4.4-2.

TABLE 4.4-1

VENDOR ANALYSES OF SMALL BREAK LOCAs WITH DELAYED REACTOR COOLANT PUMP TRIP - SUMMARY OF RESULTS

	Break Location	Break Size	Maximum Time Available For Pump Trip	Effect of Continuous Pump Operation	Effect of Tripping One Pump In Each Loop
B&W	RESULTS NOT SENSITIVE DUE TO HOMOGENEOUS MODELING ASSUMPTION	LIMITING BREAK SIZE ABOUT 0.02 - 0.2 FT^2	~ 3 MINUTES (BASED ON PRELIMINARY CALCULATIONS)	ACCEPTABLE CORE COOLING	NO EVALUATION
CE*	FOUND HOT LEG BREAKS LIMITING/ SOME COLD LEG BREAKS COULD EXCEED 2200°F	LIMITING BREAK SIZE ABOUT 0.02 - .1 FT^2	6 MINUTES AFTER TRIP + SIAS FOR EM ANALYSIS 10 MINUTES AFTER TRIP + SIAS FOR BE ANALYSIS	1.0 FT^2 BREAK IN HOT LEG LEADS TO PCTs > 2200°F	ACCEPTABLE CORE COOLING FOR BE ANALYSIS PROVIDED TWO PUMPS TRIPPED WITHIN 5 MINUTES AFTER BREAK
WESTINGHOUSE	COLD LEG BREAKS LIMITING, NO HOT LEG BREAKS ANALYZED RESULTED IN PCTs > 2200°F	LIMITING BREAK SIZE 0.02 - .05 FT^2	10 MINUTES FOR ALL PLANT TYPES (2, 3, 4 LOOPS)	ACCEPTABLE CORE COOLING	NO EVALUATION

*CE analyses performed for plants with 200 psi SITs, 1200 psi HPSI pumps; analyses considered conservative WRT plants with 600 psi SITs and/or higher head HPSI pumps.

CRITICAL REGION FOR RC PUMPS TRIP, BREAK SIZE VS TIME

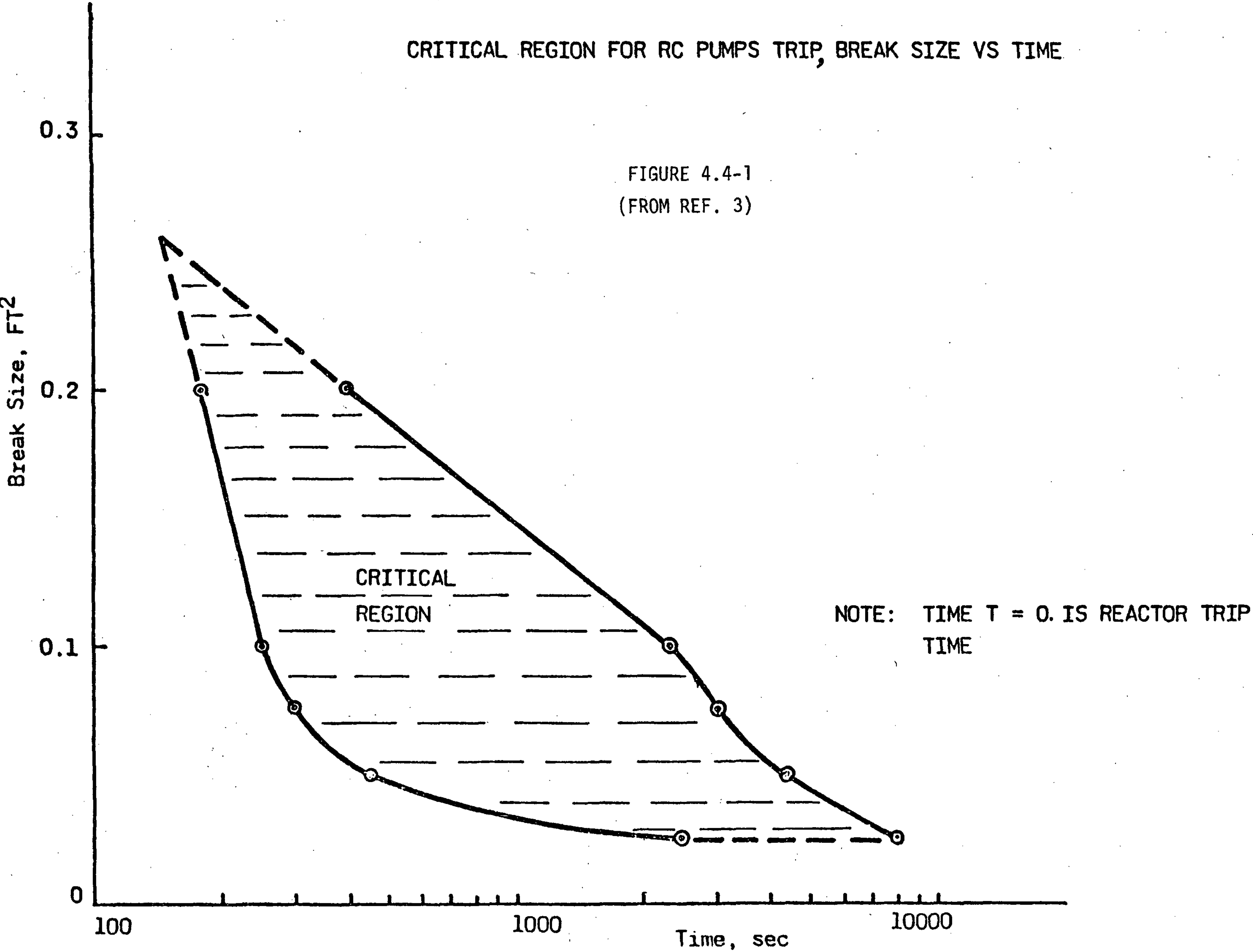


FIGURE 4.4-1
(FROM REF. 3)

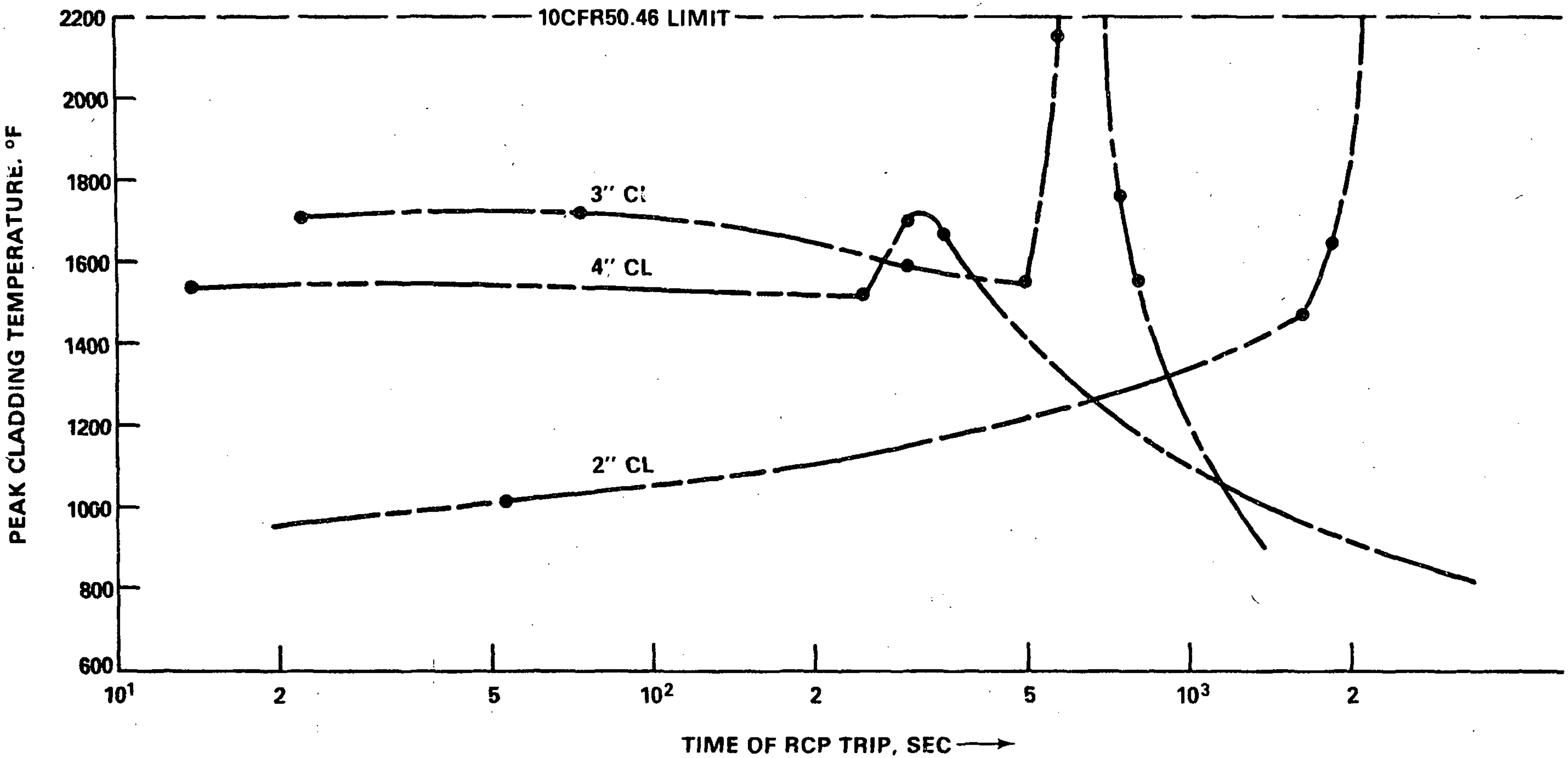


Figure 4.4-2. Effect of Pump Trip Time on Peak Cladding Temperature for Westinghouse 3-Loop Plant
(Prepared From Table 3.2-1 of WCAP-9584)

These analyses showed that the maximum delay time available for coolant pump trip was 600 seconds, or 10 minutes, for the 3-inch break. (This is seen in Figure 4.4-2 as that delay time when the peak cladding temperature exceeds 2200°F.) Westinghouse also calculated that if the pumps remained operational throughout the entire accident, no excessive cladding temperatures would be calculated. It can be inferred from Figure 4.4-2 that the maximum available time for the operator to trip the pump could be less than 10 minutes for breaks between 3 inches and 4 inches.

While Westinghouse has not performed calculations for break sizes between 3 and 4 inches to refine the maximum available trip delay time, staff bounding estimates indicate the maximum available time to be about 5 minutes. However, as will be discussed in other sections, the requirements and methods for automatic pump trip will assure that these maximum available trip times are never exceeded.

Combustion Engineering has also concluded that the range of break sizes for which tripping some or all of the coolant pumps is necessary is about 0.02 ft^2 to 0.1 ft^2. Their analyses indicate that using conservative, Appendix K assumptions, the maximum available pump trip delay time is 6 minutes, and 10 minutes if "best estimate" assumptions are used. Contrary to the conclusions of Westinghouse and Babcock & Wilcox, however, Combustion Engineering concluded that for a certain range of break sizes, continuous operation of four reactor coolant pumps throughout the accident could lead to excessive cladding temperatures. Therefore, for all small break LOCAs, tripping of all or some of the reactor coolant pumps was required.

4.4.2 Break Location

Both Westinghouse and Babcock & Wilcox concluded that a break in the cold leg discharge piping with the pumps running resulted in the limiting consequences. Westinghouse concluded that for all of the hot leg breaks analyzed, none was calculated to exceed the 2200°F licensing limit. For Babcock & Wilcox, the assumption of homogeneity throughout the entire system resulted in analyses which were, for the most part, insensitive to break location. The cold leg was found to be the most limiting because the ECC water injected into the broken loop cold leg was assumed to spill out the break. (No spillage is assumed for hot leg breaks.)

Combustion Engineering concluded that while there were some breaks in the cold leg piping which might lead to cladding temperatures in excess of 2200°F, the breaks postulated in the hot leg with delayed pump trip or pumps running were the most limiting with regard to peak cladding temperatures.

The reason why Combustion Engineering found hot leg breaks the most limiting while Westinghouse did not find them limiting is not entirely clear to the staff at this time. It is felt that the provision for steam generator and hot leg draining back to the vessel in the Combustion Engineering model promotes liquid inventory accumulation in the hot leg piping and therefore would provide liquid to a hot leg break longer than would be calculated by the Westinghouse model with no provision for draining. Likewise, the Westinghouse model is felt to promote more liquid accumulation in the cold legs than the Combustion Engineering model, and therefore cold leg breaks are expected to be more limiting in the Westinghouse analyses.

4.4.3 Continuous Pump Operation

4.4.3.1 All Pumps Operating

Both Westinghouse and Babcock and Wilcox concluded that if the reactor coolant pumps remain running through the entire accident for all break sizes, acceptable core cooling would result. Combustion Engineering, however, concluded that for a small range of break sizes (about 0.07 ft^2), continuous operation of all four pumps would lead to unacceptable peak cladding temperatures.

The reason for this is the pump is not calculated to depress the liquid in the downcomer low enough such that steam can pass from the downcomer to the core inlet for a certain range of break sizes . Because there is no steam supply from the lower plenum, the steam flow rate over the exposed fuel rods during the period of core uncovery is due to decay heat boiloff only. Decay heat steaming rates are low, and therefore both steam superheat and cladding temperatures are high. When steam flow from the downcomer to the core is calculated to occur, the core steaming rates over the exposed fuel rods are much higher and provide adequate cooling of the fuel rods. It is noted that Westinghouse calculates that the downcomer level is depressed low enough to allow steam flow to the core for all break sizes.

4.4.3.2 One Pump per Loop Operation for CE 2x4 Plants

Because pump operation provides forced circulation, it is preferable to run the reactor coolant pumps whenever possible. Combustion Engineering examined the possibility of running only one pump per loop during a small break LOCA. These analyses showed that if two pumps (one per loop) were tripped within 5 minutes of the break initiation, cladding temperatures would not exceed 1300°F. Moreover, if the remaining two pumps were tripped at any time after 5 minutes, the maximum cladding temperatures would also be acceptable. In the event that the two pumps were not tripped within 5 minutes of the event, then the operators could still trip all four of the pumps within 10 minutes and not have unacceptable consequences. However, CE did not provide an analysis of two-pump trip using Appendix K assumptions, and the acceptability of and time available for two pump trip using these assumptions is not known.

4.4.4 Plant Types

Each of the reactor vendors have more than one plant type. Westinghouse performed the majority of their analyses on the three-coolant loop type plants. However, analyses were also performed for both four-coolant and two-coolant loop plants. The conclusions drawn by Westinghouse based on these analyses were that the behavior of the two- and four-loop plants was quite similiar in phenomena to the three-loop plants, although times for key phenomena to occur were different. However, it was concluded that the maximum available time for reactor coolant pump trip was also approximately 10 minutes for the two- and four-loop plants.

Combustion Engineering examined the effects of both higher safety injection tank pressure (600 psi vs. 200 psi) and the higher shutoff head safety injection pumps found on some Combustion Engineering plants.

For the 600 psi safety injection tanks, they found the tanks discharged much sooner in the accident and resulted in decreased depth and duration of core uncovery as compared to the 200 psi tank case.

Analyses with the high shutoff head injection pumps showed they provided greater safety injection flow than the nominal shutoff head case. Since this will always increase the primary system inventory with respect to the nominal case, this also reduces the depth and duration of core uncovery as compared to the nominal pump head case. Therefore, the low head safety injection pump, 200 psi safety injection tank cases were considered bounding compared to the other plant types.

The Babcock & Wilcox analyses were performed for the 177 fuel assembly lowered-loop design plants. One plant, Davis-Besse Unit 1, is of the 177 fuel assembly, raised-loop design. For the raised-loop design, water that would collapse back into the steam generator and pump suction piping on lowered-loop designs would collapse into the core on raised-loop designs when the pumps were tripped. Thus, the core inventory would be greater after pump trip for the raised-loop design.

Babcock & Wilcox also showed that the total high pressure injection flow delivered for Davis-Besse Unit 1 during a small break LOCA is greater than for the lowered-loop plants, even though the Davis-Besse Unit 1 pumps do not have as high a shutoff head as lowered-loop plant pumps do. Because of the raised-loop design and the higher volume safety injection pumps, Babcock & Wilcox concluded that results for Davis-Besse Unit 1 would be less severe than for a lowered-loop plant. However, it was concluded that Davis-Besse Unit 1 could not be shown to comply with the 2200°F licensing limit for all cases of delayed reactor coolant pump trip.

4.4.5 High Pressure Injection Operational Status

The assumed number of high pressure injection pumps available during the accident may significantly affect the results. None of the vendors considered the case of no HPI pumps available. It can be generally concluded, however, that the results would be more severe than when one or two HPI pumps were available.

Westinghouse's analyses were all based on evaluation model assumptions, and no cases were analyzed by Westinghouse where both HPI pumps were assumed available. Although Westinghouse generally conluded that having both HPI pumps available would reduce inventory losses and lower the cladding temperatures, it was not determined to what extent their overall conclusions might change.

Babcock & Wilcox performed extensive analyses assuming that two HPI pumps were available. They concluded that, because the results showed than an inventory problem existed with two HPI pumps, the assumption of one HPI pump availability would only make the problem worse.

Combustion Engineering concluded that with two HPI pumps available, all four reactor coolant pumps could run throughout the accident and a delayed trip at any time would not lead to unacceptable results.

4.4.6 Best-Estimate Analysis

Both Combustion Engineering and Babcock & Wilcox provided selected best-estimate analyses of small break LOCAs with the reactor coolant pumps running. These analyses include the following differences from evaluation model analyses:

- 1.0 x ANS decay heat vs. 1.2 x ANS decay heat;
- One ECCS train available vs. two ECCS trains available; and
- Expected peaking factors and axial power shapes vs. worst case allowable.

Combustion Engineering reported that for the best-estimate case, with all four pumps operating throughout the accident and two HPI pumps available, no core uncovery was calculated and cladding temperatures remained at about coolant saturation temperatures. This same best-estimate analysis, however, with a slightly smaller break size (0.07 ft^2 vs. 0.1 ft^2) and with one rather than two HPI pumps available, produced core uncovery and predicted cladding temperatures were above 2200°F (this large difference is due to the ability to pass steam from the downcomer to the core; see Section 4.3.5 for a more complete discussion of this). The best-estimate analyses also showed that if all four coolant pumps were tripped 10 minutes after accident initiation, and only one HPI pump was available, cladding temperatures were predicted to remain below 800°F due to a shallow core uncovery.

Babcock & Wilcox concluded that if the three conservatisms identified previously were eliminated from the analyses, cladding temperatures would not be calculated to exceed 2200°F for any pump trip delay time during a small break LOCA.

4.5 Staff Audit Calculations

As part of the overall audit analyses on small break behavior being performed for the NRC staff by EG&G, Idaho, three cold leg small break calculations were performed which assumed a delayed pump trip. The RELAP4/MOD7 computer model was used by EG&G for these analyses.

The plant used for the audit calculations was a typical Westinghouse four-loop plant with 15x15 fuel. The calculations assumed a 4-inch (0.087 ft^2) break in the cold leg pump discharge pipe. One calculation assumed the pumps remained running throughout the entire accident. The second calculation assumed the pumps were tripped at 511 seconds after the break initiation (estimated to be the worst case), and the third calculation assumed the pumps tripped at 760 seconds after the break initiation (or 90 percent system void).

Figure 4.5-1 shows the temporal cladding behovior for the average fuel rod. Ongoing analyses for the hot fuel rod behavior were not completed for all cases as of this writing. However, for Case 4 shown in Figure 4.5-1, the hot rod peak cladding temperature was conservatively calculated to be 2095°F when the average rod peak cladding temperature is shown to be approximately 1325°F at approximately 950 seconds. The closest corresponding case analyzed by Westinghouse was a 4-inch cold leg break in a four-loop plant with a pump trip at 450 seconds after break initiation. Westinghouse calculated a peak cladding temperature of 1794°F for this case.

The staff analyses bear out the trends observed by all of the vendors that delayed reactor coolant pump trip during a SBLOCA can prolong the period in which liquid inventory is available to the break and can lead to more severe consequences than the immediate pump trip case. The limited, scoping, audit calculations performed show qualitative agreement with the results obtained by the vendors. However, because the staff model is

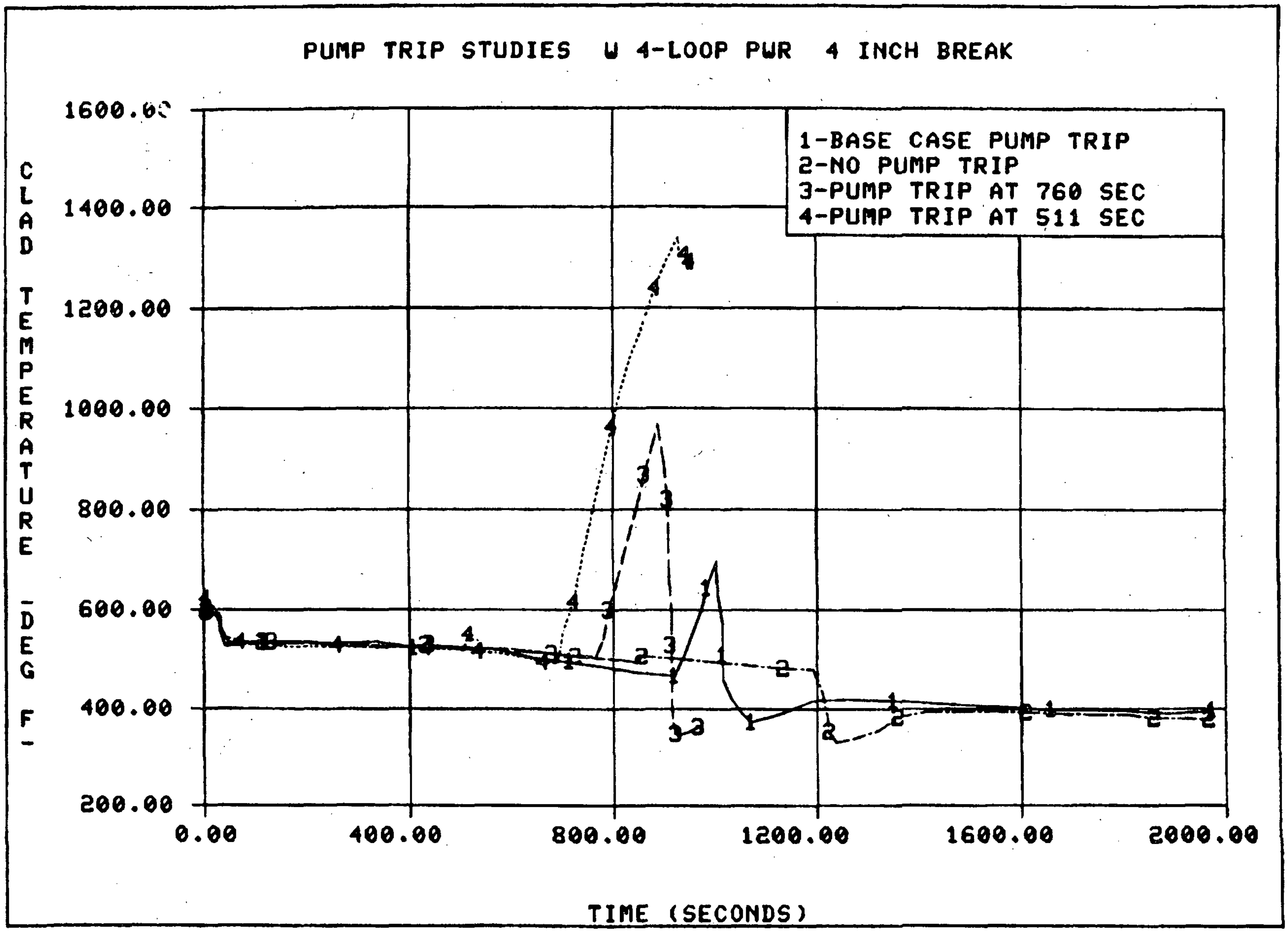


FIGURE 4.5-1

STAFF AUDIT CALCULATION RESULTS - EFFECT OF DELAYED PUMP TRIP ON SBLOCA subject to the same limitations and model uncertainties as the vendor models, the validity of the results is equally questionable.

4.6 Experimental Verification

4.6.1 Present Status

There is considerable experimental information available to support a number of the individual models used in the calculation of small break LOCAs with the pumps tripped. These include, for example, level swell information obtained from simulated core uncovery tests performed in 1973 by Westinghouse.

However, very little experimental information is available for integral systems verification of small break models. At present, only two small break tests (S-02-6 and S-07-10B) have been performed in the Semiscale facility and one isothermal small break test (L3-0) in the LOFT facility. In all these tests, the pumps were tripped at the start of the accident simulation. The relation of these tests to small break model verification can be found in Appendix VIII of NUREG-0611 (Ref. 9).

The PWR industry has not proposed any integral systems tests in any facilities in the short term to verify the overall small break calculational models with pumps running. Because of the staff position that, for small break LOCAs, reactor coolant pump trip should be performed automatically, present schedules (see Section 7.3) for design and installation of automatic pump trips may reduce or preclude the usefulness of any data obtained in the short term to substantiate allowable modes of pump operation during SBLOCAs.

4.6.2 Future Plans

At present, NRC plans to sponsor a series of scoping small break tests with the pumps running in the Semiscale MOD-3 facility. These tests will be run in order to gain a better understanding of mass distributions in the system as well as the appropriateness of certain aspects of small break LOCA models with the pumps running.

In spring of CY '80, tests will be run in the LOFT facility to gain a better understanding of small break behavior with the pumps running.

The staff will require that pretest predictions be performed by the PWR vendors and fuel suppliers for the first test in this series in which the pumps are operational. Pretest predictions are necessary because present requirements for pumps trip during a SBLOCA do not result in immediate pump trip. Given the expected system behavior, and present uncertainty margins, there is considerable probability that pump operation could continue for a significant period of time before trip setpoints were achieved during a SBLOCA.

5.0 Impact on Non-LOCA Transients

It has been shown in the preceding sections that in the event of a small break LOCA, certain combinations of break sizes and reactor coolant pump trip delay times can lead to excessive cladding temperatures.

One way to preclude the possibility of this occurrence would be to trip the reactor coolant pumps early in the accident before a significant system void fraction evolved.

This was the basis for the bulletin directive to trip all reactor coolant pumps upon reactor trip and actuation of the safety injection system on low pressure.

A concern with this approach (early pump trip), however, is that some non-LOCA transients exhibit system depressurization similiar to a small break LOCA during the initial phases of the transient, and compliance with the bulletin requirements could result in unnecessary pump trip for a non-LOCA transient, with the consequences becoming more severe than if the pumps remained running. Because of this concern, each PWR vendor evaluated the impact of an early reactor coolant pump trip on non-LOCA transients. The only transients which were evaluated were those which could cause a depressurization of the primary system to the safety injection actuation pressure setpoint.

5.1 Transients Which Produce Primary System Depressurization

A small break LOCA is characterized by a drop in primary system pressure due to a net mass loss from the system. There are three main classes of transients which can produce depressurization similiar to a small break LOCA. These are as follows:

- Increased secondary system heat removal;
- Reactor coolant system inventory decrease; and
- Reactor coolant system pressure control malfunction.

Both Combustion Engineering and Babcock & Wilcox identified the steam line break as that accident which would produce the greatest overcooling (increased secondary heat removal) of the primary system and thus the largest depressurization. The staff agrees with this conclusion.

5.2 Vendor Analyses

Combustion Engineering has concluded that by tripping the pumps for a steam line break accident, and assuming (1) a stuck control rod, (2) one high pressure injection pump failure, and (3) an EOC (end-of-cycle) core condition, the margin to fuel failure is decreased due to the flow reduction prior to the post-trip return to power, although the consequences were still found to be acceptable.

Combustion Engineering also pointed out that secondary steam releases due to isolation of a ruptured steam generator tube would be increased radiological releases due to potentially longer primary system cooldown periods. These results however, were also found to be acceptable.

A final conclusion reached by Combustion Engineering addressed premature tripping of the pumps. Specifically, they concluded that pump trip should be delayed at least 5 seconds after insertion of the control rods is verified in order to allow core heat fluxes to subside and avoid DNB (departure from nucleate boiling).

Westinghouse did not submit a formal assessment of the effect of reactor coolant pump trip on non-LOCA transients. Prior to issuance of Bulletin 79-06C, Westinghouse instructed its plant owners to trip the reactor coolant pumps at 1250 psi plus indicated pressure measurement inaccuracy. Moreover, Westinghouse has analyzed the steam line break accident assuming both offsite power available (pumps available) and offsite power lost (pumps tripped) in SAR analyses. These analyses conclude the acceptability of the accident with or without pump trip. Westinghouse did not address the consequences of premature pump trip as it affects the DNB margin. However, present pump trip criteria discussed in other sections of this report preclude premature tripping of the pumps.

Babcock & Wilcox examined the impact of pump trip on two steam line break accidents. These were a 0.6 ft^2 split, and a 12.2 ft^2 double-ended rupture. The 0.6 ft^2 split was chosen as it is the smallest break which results in a high flux reactor trip. Two analyses were performed, one in which two of the four pumps remained running (one per loop), and another in which all four pumps were tripped at HPI initiation.

For both cases, the system remained 30°F to 120°F subcooled, with no adverse affects due to the pumps being tripped.

The 12.2 ft^2 double-ended rupture was chosen to produce an excessive cooldown. In fact, this break size is beyond the current design bases of the plant since it requires the postulated double failure of both steam generator isolation valves.

As in the 0.6 ft^2 case, this event was analyzed twice, once with the pumps running and once with the pumps tripped.

The case with the pumps tripped produced the more severe consequences, in that a greater amount of steam was calculated to be produced in the hot leg containing the pressurizer. This steam was calculated to reduce the natural circulation flow in the affected loops to 5 to 10 percent of the unaffected loop natural circulation flow. The loop without the pressurizer was calculated to remain subcooled, although this subcooling margin was slight. The steam in the pressurizer loop was calculated to eventually condense and full flow natural circulation was established. While no calculations were presented on steam line breaks with expected failure (i.e., one isolation valve failure) the acceptable results shown by the 12.2 ft^2 rupture calculation indicate that analysis of breaks of smaller size will also be found acceptable, with less reduction of natural circulation in the pressurizer loop.

6.0 Staff Review and Conclusions

The staff has reviewed the analyses models provided by each of the three PWR vendors in support of evaluations of small break LOCAs with delayed reactor coolant pump trip. We have also reviewed the conclusions drawn from analyses performed with these models and the impact of reactor coolant pump trip on non-LOCA transients. Finally, we have reviewed the criteria and requirements for reactor coolant pump trip.

Based on these reviews, we have concluded the following:

(1) Based on the qualitative agreement among the PWR vendor analyses of the basic phenomena, as well as the inherent understanding of the physical processes involved, we concur in the conclusion reached by the PWR vendors that continued pump operation during a small break

LOCA can result in a greater mass inventory loss from the system than if the pumps were tripped.

(2) The ability to correctly represent the thermal-hydraulic behavior in key components within the primary system during a small break LOCA with the reactor coolant pumps running is questionable. Moreover, it is unclear at this time which models clearly result in conservative, bounding calculations. This is substantiated by the variety of different models used to represent the various primary system components in vendor analyses and the differences in the limiting small break analyses. It is our conclusion that this uncertainty in thermal-hydraulic modeling presently precludes the use of these models for quantitative determination of small break system behavior with the coolant pumps running. In particular, we cannot accept their use to substantiate allowable modes of pump operation during small break LOCAs.

(3) It is our conclusion that for the pumps running case, insufficient integral system experimental data presently exists to substantiate the quantiative results of the analysis codes. Moreover, we do not believe any proposed testing can be performed on a schedule compatibl with that necessary for short-term resolution, which includes the addition of equipment necessary to assure automatic tripping of the coolant pumps for small break LOCAs.

(4) From items (2) and (3), above, we find that tripping all of the reactor coolant pumps during small break LOCAs is required at this time, and that this pump trip should be automatically initiated from equipment that is safety-grade to the extent possible.

(5) The impact of an early pump trip on non-LOCA transients is not predicted to lead to unacceptable consequences. However, tripping the reactor coolant pumps for non-LOCA transients can aggravate the consequences of these transients and extend the time required to bring the plant into controlled shutdown condition. For B&W plants, tripping of the reactor coolant pumps during severe overcooling events increases the potential for interruption of natural circulation due to steam formation in the coolant loops.

Therefore, we conclude that the criteria and requirements for reactor coolant pump trip to be established from item (4), above, should minimize, to the extent practicable, the probability of initiating a reactor coolant pump trip for non-LOCA transients.

(6) The staff recognizes the potential desirability of running the reactor coolant pumps to provide forced circulation during small break LOCAs and we encourage the continued exploration by the industry of means by which this could be accomplished. For example, an increase in HPI capacity or two-pump operation as proposed by Combustion Engineering are a step in this direction.

(7) We will require verification of small break models with the pumps running against appropriate integral systems experimental tests. In particular, we will require that the PWR vendors and fuel suppliers

perform pretest predictions fo the LOFT SBLOCA test with pumps running scheduled to be performed in March of 1980.

7.0 Requirements for Pump Trip

7.1 Pump Trip Criteria

As previously stated, it is desirable to trip the reactor coolant pumps only for small break LOCAs, and to leave them running for non-LOCA transients. To accomplish this, a distinction of small break LOCA events based on the behavior of measured parameters is necessary. A number of parameters, used in coincident logic, have been suggested to make this distinction. These include:

- Low pressure;
- Low pump current;
- Safety injection actuation; and
- Time delay;

and are discussed briefly as follows.

7.1.1 Low Pressure

For non-LOCA transients, such as overcooling events, the initial pressure decrease is due to coolant contraction. This contraction is limited by the secondary system pressure, which in turn, determines the primary side temperature necessary to remove decay heat after a reactor trip. Pressure decreases due to moderate frequency overcooling events are therefore limited, and usually a pressure recovery follows the initial decrease because of either charging flow or safety injection flow. If the reactor coolant pumps were not tripped until the primary pressure dropped below some pressure which would not be expected to be reached during a majority of the non-LOCA transients, then the probability of pump trip for non-LOCA transients would be greatly reduced.

7.1.2 Low Pump Current

During the accident at Three Mile Island, the operators tripped one of the reactor coolant pumps because of low measured current to the pump motor. The pump motor current was low because of the steam voids being pumped. Thus, pump current would be an indicator of voids in the primary system. However, in order to use this signal for pump trip, the correlation between pump current and void fraction would have to be justified.

7.1.3 Safety Injection Actuation

Pump trip on safety injection actuation is required in Bulletins 79-05C and 79-06C. As a sole determinant of pump trip, or when used in coincidence with reactor trip, it could result in pump trip for a percentage of the non-LOCA depressurization transients.

7.1.4 Time Delay

This signal would work on the principle that the system pressure decrease for most non-LOCA transients is limited, and pressure recovery expected. Upon receipt of a safety injection actuation signal, a specified time would be allowed in which the pressure would have to recover to a certain value. If it did not recover within the allowable time, then the pumps would be tripped.

7.2 Short-Term Requirements

The following describe the short-term requirements for pump trip for each of the reactor vendors.

7.2.1 Control Room Operators

IE Bulletins 79-05C and 79-06C (item 1.B) require that two licensed operators be in the control room at all times (three for a dual control room) and that one of the two operators be designated to trip the reactor coolant pumps should the facility undergo a transient which results in a safety injection actuation signal due to low primary system pressure. The designated operator may perform any normal or routine control room duties at all other times. The licensee should confirm that an operator is designated to perform this action on each shift.

7.2.2 Westinghouse-Designed Plants

For the short-term, the staff has adopted the following position for manual pump trip requirements on Westinghouse-designed plants.

Staff Position on Pump Trip for Westinghouse Plants

We require that the reactor coolant pumps be tripped at a system pressure determined in the following manner:

(1) Secondary System Pressure - Based on the number and size of the secondary system safety valves, the secondary pressure will be established by determining the pressure setpoint for that valve in which the calculated steam relief is less than 60 percent of the valve's relief rating. If the calculated relief is greater than 60 percent of the rated capacity, then the next highest pressure setpoint should be used.

(2) Primary to Secondary Pressure Difference - To account for the pressure gradient needed for heat removal, pressure drop between the steam generator and safety valves, uncertainty of the safety valve setpoint, pressure drop from steam generator to measurement location, etc., the primary pressure for RCP trip should be the secondary pressure as established by (1), above, plus 100 psi if the calculated adjustments are 100 psi or less. If the adjustment are determined to be greater than 100 psi, the larger value should be used.

(3) Instrument inaccuracies appropriate to that time in the loss-of-coolant accident should be added to the primary pressure established in (2), above. The resulting pressure is the indicated pressure at which the operator should trip the RCPs.

7.2.3 Combustion Engineering-Designed Plants

Combustion Engineering has recommended that reactor coolant pump trip be manually initiated by the operator on receipt of reactor trip and safety injection actuation signals. Combustion Engineering is also evaluating the capability of their plants to accomodate a pump trip on reactor trip and a lower system pressure by a method similar to that established for Westinghouse as specified in Section 7.2.2, above.

The staff will accept the pump trip based on reactor trip and SI actuation for the short-term, since SI actuation pressure is approximately 1550 to

1600 psig for CE plants as compared to SI actuation pressures of about 1800 to 1900 psig for Westinghouse plants. It is expected that the pressure used for pump trip by Westinghouse will fall approximately in the range of the safety injection actuation pressure for both CE and B&W plants.

7.2.4 Babcock and Wilcox-Designed Plants

Babcock & Wilcox is also recommending that for the short term, pump trip be manually initiated on automatic actuation on low pressure of the safety injection system. In addition, Babcock & Wilcox and their plant owners are examining the possibility of a short-term manual trip requirement based on subcooling rather than automatic SI actuation on low pressure only. The staff agrees in principle with this approach, but final approval must wait until the details of such a method have been formally submitted and evaluated.

The staff finds the present short-term requirement for manual trip on automatic SI actuation on low pressure acceptable. B&W SI actuation setpoints are between 1500 psig and 1650 psig and are considered consistent with the setpoints at which the pumps would be tripped for both Westinghouse and Combustion Engineering plants.

7.2.5 Training Guidelines and Emergency Procedures

IE Bulletins 79-05C and 79-06C (items 3 and 4) requested the Westinghouse, Combustion Engineering, and Babcock & Wilcox plant licensees to:

(1) Develop new guidelines for LOCA and non-LOCA events based on LOCA analyses and RCP trip requirements, and

(2) Revise emergency procedures and train all licensed operators and senior reactor operators based on these new guidelines.

In general, the licensees have identified guidelines, procedures, and training for loss of coolant events in their responses to these items. This effort on LOCA events was already in progress at the time the bulletin was issued.

Because of the potential for initiating ECCS by other depressurization events such as overcooling because of a malfunction in the secondary system, the operator would have to trip the reactor coolant pumps before he could make a determination about what event is occurring. As a result, we require that the licensee have procedures and operator training to handle non-LOCA events which may also have ECCS actuation and reactor coolant pump trip.

The procedures for these non-LOCA events should include instructions on tripping the reactor coolant pumps, monitoring and initiating natural circulation, pressure control without the pressurizer spray, HPI termination criteria, and reactor coolant pump restart criteria. The licensees should confirm that these procedures for non-LOCA events are in place and the operators have been trained in their implementation.

7.3 Long-Term Requirements

7.3.1 Schedule

As stated in Section 6.0, it is required that tripping of the reactor coolant pumps for small break LOCAs be accomplished automatically. While the long-term requirement of Bulletins 79-05C and 79-06C required a schedule for the design of an automatic pump trip be submitted by the end of August 1979, both Combustion Engineering and Westinghouse owners concluded that automatic tripping was ~~not necessary and~~ therefore have submitted neither a schedule nor a design. Some Babcock & Wilcox owners have supplied preliminary designs for automatic pump trip based on low reactor coolant pump current.

The staff requires that a schedule for submittal of a design for automatic tripping of the reactor coolant pumps during a small break LOCA be provided by January 31, 1980. We also require that automatic pump trip systems be installed and operational by December 31, 1980. If operational considerations (e.g., refueling outage) support a later date, the staff will consider these on a case-by-case basis.

7.3.2 Automatic Reactor Coolant Pump Trip Requirements

The staff has reviewed the preliminary design descriptions of the automatic trip circuitry for the reactor coolant pump motors which have been provided by the B&W plant licensees. Westinghouse and Combustion Engineering plant licensees have not provided any design descriptions and have indicated their intention to retain the manual RCP trip rather than install an automatic trip. The staff requires that safety-grade RCP trip circuitry be installed at each operating facility to ensure that:

(1) A single failure in the added trip circuitry will not inadvertently trip the pumps.

(2) A single failure in the added trip circuitry will not preclude pump trip if needed.

The above requirements provide a more reliable pump trip than the existing reliance on manual action. The licensees are to submit proposed design modifications and schedules for implementing this requirement according to the schedules identified in Section 7.3.1.

The staff requires that sufficient information be provided by all licensees concerning RCP trip circuitry modifications to permit our assessment. This infomation shall include stating the degree of conformance with the applicable acceptance criteria listed in column 7.2 of Table 7-1 of the Standard Review Plan as well as providing justification for any nonconformance of the proposed design. The staff requires a discussion on the conformance of the design with the design requirements of Section 4 of IEEE Std 279-1971 and with Section 4 of IEEE Std 308-1974.

The licensees must provide detailed descriptions of any changes to and/or interfaces with the existing protection systems. Included should be diagrams (block, location, functional, and/or elementary wiring), as necessary, to clearly depict the changes and/or interfaces. In addition, an analysis which demonstrates that these changes and/or interfaces will not degrade the existing protection systems should be provided.

The staff requires that the licensees identify equipment which is identical to equipment utilized in existing safety-grade systems. Where the equipment is not identical, the differences should be briefly described. Descriptive information for the qualification requirements, test plan, test setup, test procedures, and acceptability goals and requirements should also be provided. Finally, a discussion of the criteria for the overall trip circuitry installation testing which will demonstrate that this circuitry has been installed properly should be presented.

8.0 References

1. Letter, R. W. Reid (NRC) to all operating [name of Vendor] plants, dated June 5, 1979.

2. "Summary of Meeting Held July 18, 1979, to Discuss Babcock and Wilcox Small Break Loss-of-Coolant Accident Analyses (Reactor Coolant Pumps Operating vs. Tripped)," dated August 15, 1979, issued by NRC/B&W Task Force.

3. "Analysis of Delayed Reactor Coolant Pumps Trip During Small Loss-of-Coolant Accident for Westinghouse Nuclear Steam Supply Systems," WCAP-9584, August 30, 1979.

4. "Response to NRC IE Bulletin 79-06C Items 2 and 3 for Combustion Engineering Nuclear Steam Supply Systems," CEN-115-P, August 1979.

5. "Analysis Summary In Support of An Early RC Pump Trip," and "Supplemental Small Break Analysis," submitted as Attachment B to B&W plant owners letters in response to Bulletin 79-05C.

6. "Report on Small Break Accidents for Westinghouse NSSS System," WCAP-9600, June 1979 (three volumes).

7. Letter, J. H. Taylor (B&W) to D. F. Ross (NRC) transmitting information sent to B&W plant owners in response to bulletin 79-05C, dated August 30, 1979.

8. "Study of Two-Phase Natural Circulation Following A Small LOCA Using the NOTRUMP Code," WCAP-9586, August 1979.

9. "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants," NUREG-0611 (to be published).

APPENDIX A

FAILURE OF NATURAL CIRCULATION DURING THE THREE MILE ISLAND ACCIDENT

A.1 Postulated Scenario

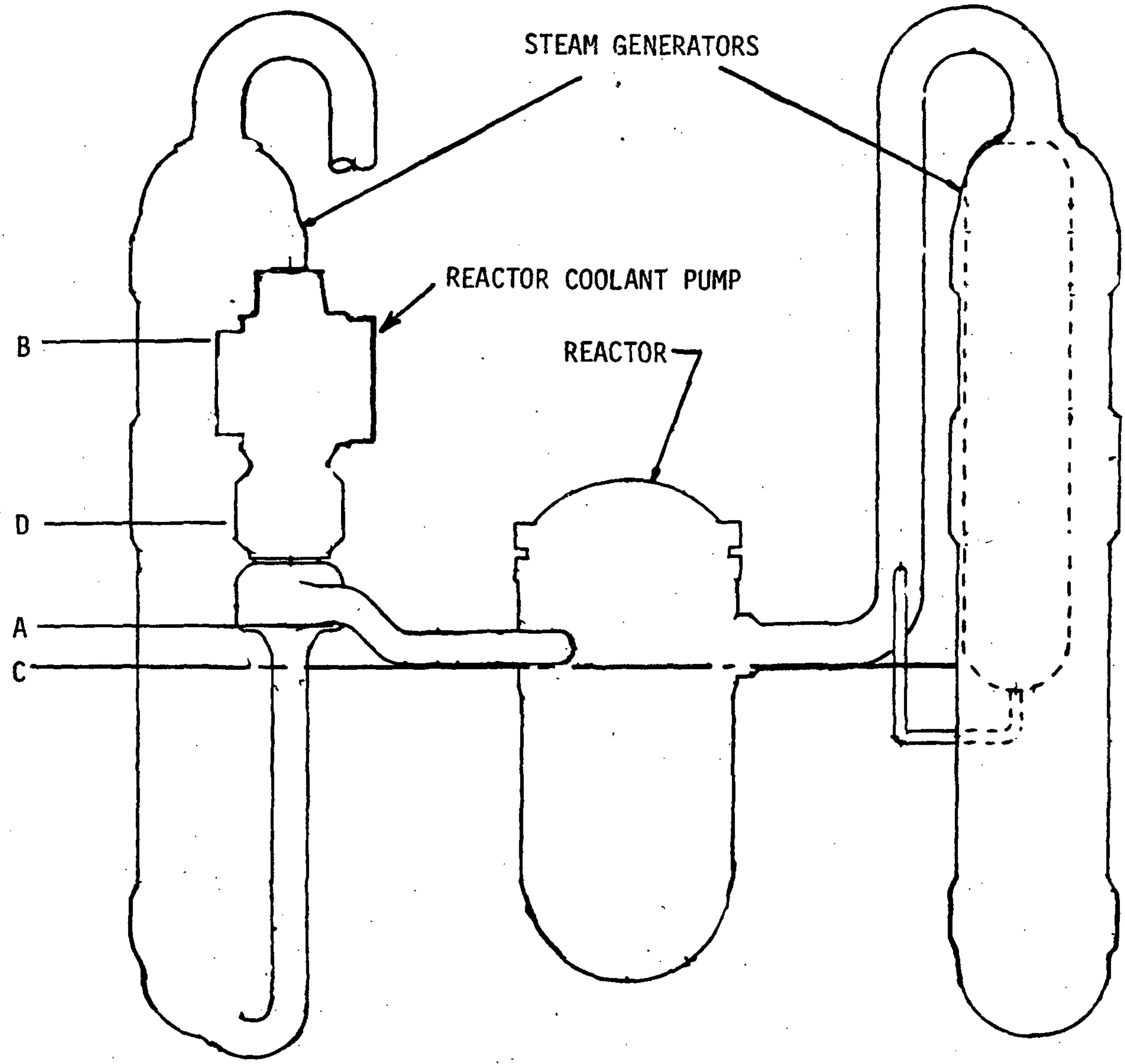
During the course of the accident at Three Mile Island, the operators stopped the reactor coolant pumps. Immediately after they were stopped, coolant temperatures in the hot leg piping were observed to rapidly increase, and it was during this period that a majority of the damage to the reactor core was postulated to occur. Because of this occurrence, concern was raised why natural circulation was not established after the pumps were tripped.

Based on examination of the component elevations in the plant, steam generator secondary level setpoints, and estimated primary system inventory, it is believed that the inability to achieve natural circulation flow can be qualitatively explained.

During the initial phase of the accident when the pumps were operating, the primary system evolved to a high system void fraction due to the continuous loss of inventory through the stuck open pilot-operated relief valve (PORV) without HPI makeup flow. Despite the high system void, operation of the pumps circulated the steam and water as a two-phase saturated mixture throughout the system and provided ample cooling of the fuel rods.

When the pumps were tripped, the steam and liquid phases separated, with the liquid falling to the lower elevations of the primary system. For the TMI-2 plant, this is the bottom of the steam generator, pump suction piping, and the bottom of the reactor vessel, as can be seen in Figure A-1. Also shown in Figure A-1 is the elevation (elevation C) of the automatic feedwater control level setpoint, which was set at 50 percent of the operating level when the pumps were tripped (since the TMI-2 accident, B&W has recommended that this level setpoint be increased to 95 percent of the operating range when the pumps are tripped).

After the pump trip, it is postulated that liquid existed in the bottom of the vessel and bottom of the steam generator and suction piping. Steam existed in the hot leg piping, upper portions of the core, vessel, steam generator, and also in the pump and cold leg piping. In order to initiate natural circulation, the liquid level in the suction pipe would have to increase such that it could flow through the pump and into the discharge pipe and vessel downcomer. To raise the liquid level in the suction pipe, the liquid level in the steam generator tubes must be raised to an elevation above that of the bottom of the pump discharge nozzle. This, in turn, can only be accomplished by establishing a condensing surface in the steam generator above this elevation (elevation A in Figure A-1). While the feedwater enters the steam generator through a sparger at elevation B (see Figure A-1), and would normally produce a condensing surface well above that needed to raise the water in the suction pipe into the pump, feedwater will only be supplied if it is replenishing liquid lost through boiling. Without the initial flow of liquid out of the steam generator to the



Elevation A - Bottom of pump discharge nozzle

Elevation B - Approximate elevation of feedwater sparger

Elevation C - 50% of operating level

Elevation D - 95% of operating level

FIGURE A-1

REACTOR COOLANT SYSTEM ARRANGEMENT FOR THREE MILE ISLAND, UNIT 2 - SELECTED ELEVATIONS
(from TMI-2 FSAR)

vessel, the stagnant primary liquid in the lower portion of the steam generator tubes eventually equilibrated with the secondary water which was held at the 50 percent level (elevation C in Figure A-1). When the heat transfer stopped, so did the boiling of the secondary water. This, in turn, stopped the feedwater demand, and the condensing surface due to sparger spray above the pump discharge nozzle was lost. The only condensing surface was now the secondary water level, and it was below the elevation necessary to allow water to flow through the pump to the vessel. Thus, liquid could not flow from the steam generator to the vessel, and steam produced in the core could not condense in the steam generator.

A.2 Adequacy of Corrective Action

Since the Three Mile Island accident, Babcock & Wilcox has included in their emergency guidelines the requirement for the operators to manually raise the steam generator secondary water level to 95 percent of the normal operating range in th event that the reactor coolant pumps are tripped during a small break loss-of-coolant accident. This is shown as elevation D in Figure A-1. As is seen, this level assures that a steam condensing surface will exist at elevations above the bottom of the pump discharge nozzle. Therefore, a sufficient static head of water will be available to establish natural circulation flow.

This action alone, however, would not have prevented the fuel damage from occurring at the Three Mile Island plant. Even though establishing two-phase natural circulation would have produced a heat removal path by steam flow in the core, this would have, in all likelihood, been insufficient to adequately cool the core, primarily because of an inadequate liquid inventory in the primary system.

Due to the uniqueness of the B&W lowered-loop design, the pre-TMI-2 inadequacies as well as the corrective action are only considered applicable for B&W lowered-loop plants.

APPENDIX B

OFFICE OF INSPECTION AND ENFORCEMENT BULLETINS

79-05A, 79-06A, AND 79-06B

WASHINGTON, DC 20555

APRIL 5, 1979

IE Bulletin 79-05A

NUCLEAR INCIDENT AT THREE MILE ISLAND - SUPPLEMENT

Description of Circumstances:

Preliminary information received by the NRC since issuance of IE Bulletin 79-05 on April 1, 1979 has identified six potential human, design and mechanical failures which resulted in the core damage and radiation releases at the Three Mile Island Unit 2 nuclear plant. The information and actions in this supplement clarify and extend the original Bulletin and transmit a preliminary chronology of the TMI accident through the first 16 hours (Enclosure 1).

1. At the time of the initiating event, loss of feedwater, both of the auxiliary feedwater trains were valved out of service.

2. The pressurizer electromatic relief valve, which opened during the initial pressure surge, failed to close when the pressure decreased below the actuation level.

3. Following rapid depressurization of the pressurizer, the pressurizer level indication may have lead to erroneous inferences of high level in the reactor coolant system. The pressurizer level indication apparently led the operators to prematurely terminate high pressure injection flow, even though substantial voids existed in the reactor coolant system.

4. Because the containment does not isolate on high pressure injection (HPI) initiation, the highly radioactive water from the relief valve discharge was pumped out of the containment by the automatic initiation of a transfer pump. This water entered the radioactive waste treatment system in the auxiliary building where some of it overflowed to the floor. Outgassing from this water and discharge through the auxiliary building ventilation system and filters was the principal source of the offsite release of radioactive noble gases.

5. Subsequently, the high pressure injection system was intermittently operated attempting to control primary coolant inventory losses through the electromatic relief valve, apparently based on pressurizer level indication. Due to the presence of steam and/or noncondensible voids elsewhere in the reactor coolant system, this led to a further reduction in primary coolant inventory.

Tripping of reactor coolant pumps during the course of the transient, to protect against pump damage due to pump vibration, led to fuel damage since voids in the reactor coolant system prevented natural circulation.

Actions To Be Taken by Licensees:

For all Babcock and Wilcox pressurized water reactor facilities with an operating license (the actions specified below replace those specified in IE Bulletin 79-05):

1. **(This item clarifies and expands upon item 1. of IE Bulletin 79-05.)**

   In addition to the review of circumstances described in Enclosure 1 of IE Bulletin 79-05, review the enclosed preliminary chronology of the TMI-2 3/28/79 accident. This review should be directed toward understanding the sequence of events to ensure against such an accident at your facility(ies).

2. (This item clarifies and expands upon item 2. of IE Bulletin 79-05.)

   Review any transients similar to the Davis Besse event (Enclosure 2 of IE Bulletin 79-05) and any others which contain similar elements from the enclosed chronology (Enclosure 1) which have occurred at your facility(ies). If any significant deviations from expected performance are identified in your review, provide details and an analysis of the safety significance together with a description of any corrective actions taken. Reference may be made to previous information provided to the NRC, if appropriate, in responding to this item.

3. (This item clarifies item 3. of IE Bulletin 79-05.)

   Review the actions required by your operating procedures for coping with transients and accidents, with particular attention to:

   a. Recognition of the possibility of forming voids in the primary coolant system large enough to compromise the core cooling capability, especially natural circulation capability.

   b. Operator action required to prevent the formation of such voids.

   c. Operator action required to enhance core cooling in the event such voids are formed.

(This item clarifies and expands upon item 4. of IE Bulletin 79-05.)

Review the actions directed by the operating procedures and training instructions to ensure that:

a. Operators do not override automatic actions of engineered safety features.

b. Operating procedures currently, or are revised to, specify that if the high pressure injection (HPI) system has been automatically actuated because of low pressure condition, it must remain in operation until either:

(1) Both low pressure injection (LPI) pumps are in operation and flowing at a rate in excess of 1000 gpm each and the situation has been stable for 20 minutes, or

(2) The HPI system has been in operation for 20 minutes, and all hot and cold leg temperatures are at least 50 degrees below the saturation temperature for the existing RCS pressure. If 50 degree subcooling cannot be maintained after HPI cutoff, the HPI shall be reactivated.

c. Operating procedures currently, or are revised to, specify that in the event of HPI initiation, with reactor coolant pumps (RCP) operating, at least one RCP per loop shall remain operating.

d. Operators are provided additional information and instructions to not rely upon pressurizer level indication alone, but to also examine pressurizer pressure and other plant parameter indications in evaluating plant conditions, e.g., water inventory in the reactor primary system.

5. (This item revises item 5. of IE Bulletin 79-05.)

Verify that emergency feedwater valves are in the open position in accordance with item 8 below. Also, review all safety-related valve positions and positioning requirements to assure that valves are positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance and testing, to ensure that such valves are returned to their correct positions following necessary manipulations.

6. Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to cause containment isolation of all lines whose isolation does not degrade core cooling capability upon automatic initiation of safety injection.

7. For manual valves or manually-operated motor-driven valves which could defeat or compromise the flow of auxiliary feedwater to the steam generators, prepare and implement procedures which:

   a. require that such valves be locked in their correct position; or

   b. require other similar positive position controls.

8. Prepare and implement immediately procedures which assure that two independent steam generator auxiliary feedwater flow paths, each with 100% flow capacity, are operable at any time when heat removal from the primary system is through the steam generators. When two independent 100% capacity flow paths are not available, the capacity shall be restored within 72 hours or the plant shall be placed in a cooling mode which does not rely on steam generators for cooling within the next 12 hours.

   When at least one 100% capacity flow path is not available, the reactor shall be made subcritical within one hour and the facility placed in a shutdown cooling mode which does not rely on steam generators for cooling within 12 hours or at the maximum safe shutdown rate.

9. (This item revises item 6 of IE Bulletin 79-05.)

   Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping of radioactive liquids and gases will not occur inadvertently.

   In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

   a. Whether interlocks exist to prevent transfer when high radiation indication exists, and

   b. Whether such systems are isolated by the containment isolation signal.

10. Review and modify as necessary your maintenance and test procedures to ensure that they require:

    a. Verification, by inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

    b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing.

    c. A means of notifying involved reactor operating personnel whenever a safety-related system is removed from and returned to service.

11. All operating and maintenance personnel should be made aware of the extreme seriousness and consequences of the simultaneous blocking of both auxiliary feedwater trains at the Three Mile Island Unit 2 plant and other actions taken during the early phases of the accident.

12. Review your prompt reporting procedures for NRC notification to assure very early notification of serious events.

For Babcock and Wilcox pressurized water reactor facilities with an operating license, respond to Items 1, 2, 3, 4.a and 5 by April 11, 1979. Since these items are substantially the same as those specified in IE Bulletin 79-05, the required date for response has not been changed. Respond to Items 4.b through 4.d, and 6 through 12 by April 16, 1979.

Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, DC 20555.

For all other reactors with an operating license or construction permit, this Bulletin is for information purposes and no written response is required.

Approved by GAO, B 180225 (R0072); clearance expires 7-31-80. Approval was given under a blanket clearance specifically for identified generic problems.

Enclosures:

1. Preliminary Chronology of TMI-2 3/38/79 Accident Until Core Cooling Restored.

2. List of IE Bulletins issued in last 12 months.

PRELIMINARY

CHRONOLOGY OF TMI-2 3/28/79 ACCIDENT UNTIL CORE COOLING RESTORED

TIME (Approximate)	EVENT
about 4 AM (t = 0)	Loss of Condensate Pump Loss of Feedwater Turbine Trip
t = 3-6 sec.	Electromatic relief valve opens (2255 psi) to relieve pressure in RCS
t = 9-12 sec.	Reactor trip on high RCS pressure (2355 psi)
t = 12-15 sec.	RCS pressure decays to 2205 psi (relief valve should have closed)
t = 15 sec.	RCS hot leg temperature peaks at 611 degrees F, 2147 psi (450 psi over saturation)
= 30 sec.	All three auxiliary feedwater pumps running at pressure (Pumps 2A and 2B started at turbine trip). No flow was injected since discharge valves were closed.
t = 1 min.	Pressurizer level indication begins to rise rapidly
t = 1 min.	Steam Generators A and B secondary level very low - drying out over next couple of minutes.
t = 2 min.	ECCS initiation (HPI) at 1600 psi
t = 4 - 11 min.	Pressurizer level off scale - high - one HPI pump manually tripped at about 4 min. 30 sec. Second pump tripped at about 10 min. 30 sec.
t = 6 min.	RCS flashes as pressure bottoms out at 1350 psig (Hot leg temperature of 584 degrees F)
= 7 min., 30 sec.	Reactor building sump pump came on.

TIME	EVENT
= 8 min.	Auxiliary feedwater flow is initiated by opening closed valves
t = 8 min. 18 sec.	Steam Generator B pressure reached minimum
t = 8 min. 21 sec.	Steam Generator A pressure starts to recover
t = 11 min.	Pressurizer level indication comes back on scale and decreases
t = 11-12 min.	Makeup Pump (ECCS HPI flow) restarted by operators
t = 15 min.	RC Drain/Quench Tank rupture disk blows at 190 psig (setpoint 200 psig) due to continued discharge of electromatic relief valve
t = 20 - 60 min.	System parameters stabilized in saturated condition at about 1015 psig and about 550 degrees F.
t = 1 hour, 15 min.	Operator trips RC pumps in Loop B
= 1 hour, 40 min.	Operator trips RC pumps in Loop A
t = 1-3/4 - 2 hours	CORE BEGINS HEAT UP TRANSIENT - Hot leg temperature begins to rise to 620 degrees F (off scale within 14 minutes) and cold leg temperature drops to 150 degrees F. (HPI water)
t = 2.3 hour	Electromatic relief valve isolated by operator after S.G.-B isolated to prevent leakage
t = 3 hours	RCS pressure increases to 2150 psi and electromatic relief valve opened
t = 3.25 hours	RC drain tank pressure spike of 5 psig
t = 3.8 hours	RC drain tank pressure spike of 11 psi - RCS pressure 1750; containment pressure increases from 1 to 3 psig
t = 5 hours	Peak containment pressure of 4.5 psig
t = 5 - 6 hours	RCS pressure increased from 1250 psi to to 2100 psi

TIME	EVENT
= 7.5 hours	Operator opens electromatic relief valve to depressurize RCS to attempt initiation of RHR at 400 psi
t = 8 - 9 hours	RCS pressure decreases to about 500 psi Core Flood Tanks partially discharge
t = 10 hour	28 psig containment pressure spike, containment sprays initiated and stopped after 500 gal. of NaOH injected (about 2 minutes of operation)
t = 13.5 hours	Electromatic relief valve closed to repressurize RCS, collapse voids, and start RC pump
t = 13.5 - 16 hours	RCS pressure increased from 650 psi to 2300 psi
t = 16 hours	RC pump in Loop A started, hot leg temperature decreases to 560 degrees F, and cold leg temperature increases to 400 degrees F. indicating flow through steam generator
Thereafter	S/G "A" steaming to condensor Condensor vacuum re-established RCS cooled to about 280 degrees F., 1000 psi
Now (4/4)	High radiation in containment All core thermocouples less than 460 degrees F. Using pressurizer vent valve with small makeup flow Slow cooldown RB pressure negative

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

April 14, 1979

IE Bulletin No. 79-06A

REVIEW OF OPERATIONAL ERRORS AND SYSTEM MISALIGNMENTS IDENTIFIED DURING THE THREE MILE ISLAND INCIDENT

Description of Circumstances:

IE Bulletin 79-06 identified actions to be taken by the licensees of all pressurized water power reactors (except Babcock & Wilcox reactors) as a result of the Three Mile Island Unit 2 incident. This Bulletin clarifies the actions of Bulletin 79-06 for reactors designed by Westinghouse, and the response to this bulletin will eliminate the need to respond to Bulletin 79-06.

Actions to be taken by Licensees:

For all Westinghouse pressurized water reactor facilities with an operating license (the actions specified below replace those identified in IE Bulletin 79-06 on an item by item basis):

1. Review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 3/28/79 accident included in Enclosure 1 to IE Bulletin 79-05A.

    a. This review should be directed toward understanding: (1) the extreme seriousness and consequences of the simultaneous blocking of both auxiliary feedwater trains at the Three Mile Island Unit 2 plant and other actions taken during the early phases of the accident; (2) the apparent operational errors which led to the eventual core damage; (3) that the potential exists, under certain accident or transient conditions, to have a water level in the pressurizer simultaneously with the reactor vessel not full of water; and (4) the necessity to systematically analyze plant conditions and parameters and take appropriate corrective action.

    b. Operational personnel should be instructed to: (1) not override automatic action of engineered safety features unless continued operation of engineered safety features will result in unsafe plant conditions (see Section 7a.); and (2) not make operational decisions based solely on a single plant parameter indication when one or more confirmatory indications are available.

c. All licensed operators and plant management and supervisors with operational responsibilities shall participate in this review and such participation shall be documented in plant records.

2. Review the actions required by your operating procedures for coping with transients and accidents, with particular attention to:

   a. Recognition of the possibility of forming voids in the primary coolant system large enough to compromise the core cooling capability, especially natural circulation capability.

   b. Operation action required to prevent the formation of such voids.

   c. Operator action required to enhance core cooling in the event such voids are formed. (e.g., remote venting)

3. For your facilities that use pressurizer water level coincident with pressurizer pressure for automatic initiation of safety injection into the reactor coolant system, trip the low pressurizer level setpoint bistables such that, when the pressurizer pressure reaches the low setpoint, safety injection would be initiated regardless of the pressurizer level. In addition, instruct operators to manually initiate safety injection when the pressurizer pressure indication reaches the actuation setpoint whether or not the level indication has dropped to the actuation setpoint.

4. Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to permit containment isolation whether manual or automatic, of all lines whose isolation does not degrade needed safety features or cooling capability, upon automatic initiation of safety injection.

5. For facilities for which the auxiliary feedwater system is not automatically initiated, prepare and implement immediately procedures which require the stationing of an individual (with no other assigned concurrent duties and in direct and continuous communication with the control room) to promptly initiate adequate auxiliary feedwater to the steam generator(s) for those transients or accidents the consequences of which can be limited by such action.

6. For your facilities, prepare and implement immediately procedures which:

   a. Identify those plant indications (such as valve discharge piping temperature, valve position indication, or valve discharge relief tank temperature or pressure indication) which plant operators may utilize to determine that pressurizer power operated relief valve(s) are open, and

   b. Direct the plant operators to manually close the power operated relief block valve(s) when reactor coolant system pressure is reduced to below the set point for normal automatic closure of the power operated relief valve(s) and the valve(s) remain stuck open.

7. Review the action directed by the operating procedures and training instructions to ensure that:

   a. Operators do not override automatic actions of engineered safety features, unless continued operation of engineered safety features will result in unsafe plant conditions. For example, if continued operation of engineered safety features would threaten reactor vessel integrity then the HPI should be secured (as noted in b(2) below).

   b. Operating procedures currently, or are revised to, specify that if the high pressure injection (HPI) system has been automatically actuated because of low pressure condition, it must remain in operation until either:

      (1) Both low pressure injection (LPI) pumps are in operation and flowing for 20 minutes or longer; at a rate which would assure stable plant behavior; or

      (2) The HPI system has been in operation for 20 minutes, and all hot and cold leg temperatures are at least 50 degrees below the saturation temperature for the existing RCS pressure. If 50 degress subcooling cannot be maintained after HPI cutoff, the HPI shall be reactivated. The degree of subcooling beyond 50 degrees F and the length of time HPI is in operation shall be limited by the pressure/temperature considerations for the vessel integrity.

c. Operating procedures currently, or are revised to, specify that in the event of HPI initiation with reactor coolant pumps (RCP) operating, at least one RCP shall remain operating for two loop plants and at least two RCPs shall remain operating for 3 or 4 loop plants as long as the pump(s) is providing forced flow.

d. Operators are provided additional information and instructions to not rely upon pressurizer level indication alone, but to also examine pressurizer pressure and other plant parameter indications in evaluating plant conditions, e.g., water, inventory in the reactor primary system.

8. Review all safety-related valve positions, positioning requirements and positive controls to assure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance, testing, plant and system startup, and supervisory periodic (e.g., daily/shift checks,) surveillance to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.

9. Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping, venting or other release of radioactive liquids and gases will not occur inadvertently.

In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

a. Whether interlocks exist to prevent transfer when high radiation indication exists, and

b. Whether such systems are isolated by the containment isolation signal.

c. The basis on which continued operability of the above features is assured.

10. Review and modify as necessary your maintenance and test procedures to ensure that they require:

a. Verification, by test or inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing.

c. Explicit notification of involved reactor operational personnel whenever a safety-related system is removed from and returned to service.

11. Review your prompt reporting procedures for NRC notification to assure that NRC is notified within one hour of the time the reactor is not in a controlled or expected condition of operation. Further, at that time an open continuous communication channel shall be established and maintained with NRC.

12. Review operating modes and procedures to deal with significant amounts of hydrogen gas that may be generated during a transient or other accident that would either remain inside the primary system or be released to the containment.

13. Propose changes, as required, to those technical specifications which must be modified as a result of your implementing the above items.

For all light water reactor facilities designed by Westinghouse with an operating license, respond to Items 1-12 within 10 days of the receipt of this Bulletin. Respond to item 13 (Technical Specification Change proposals) in 30 days.

Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, D.C. 20555.

For all other power reactors with an operating license or construction permit, this Bulletin is for information purposes and no written response is required.

Approved by GAO, B180225 (R0072); clearance expires 7/31/80. Approval was given under a blanket clearance specifically for identified generic problems.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

April 14, 1979

IE Bulletin No. 79-06B

REVIEW OF OPERATIONAL ERRORS AND SYSTEM MISALIGNMENTS IDENTIFIED DURING THE THREE MILE ISLAND INCIDENT

Description of Circumstances:

IE Bulletin 79-06 identified actions to be taken by the licensees of all pressurized water power reactors (except Babcock & Wilcox reactors) as a result of the Three Mile Island Unit 2 incident. This Bulletin clarifies the actions of Bulletin 79-06 for reactors designed by Combustion Engineering, and the response to this bulletin will eliminate the need to respond to Bulletin 79-06.

Actions to be taken by Licensees:

For all Combustion Engineering pressurized water reactor facilities with an operating license (the actions specified below replace those identified in IE Bulletin 79-06 on an item by item basis):

1. Review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 3/28/79 accident included in Enclosure 1 to IE Bulletin 79-05A.

    a. This review should be directed toward understanding: (1) the extreme seriousness and consequences of the simultaneous blocking of both auxiliary feedwater trains at the Three Mile Island Unit 2 plant and other actions taken during the early phases of the accident; (2) the apparent operational errors which led to the eventual core damage; (3) that the potential exists, under certain accident or transient conditions, to have a water level in the pressurizer simultaneously with the reactor vessel not full of water; and (4) the necessity to systematically analyze plant conditions and parameters and take appropriate corrective action.

    b. Operational personnel should be instructed to: (1) not override automatic action of engineered safety features unless continued operation of engineered safety features will result in unsafe plant conditions (see Section 6a.); and (2) not make operational decisions based solely on a single plant parameter indication when one or more confirmatory indications are available.

c. All licensed operators and plant management and supervisors with operational responsibilities shall participate in this review and such participation shall be documented in plant records.

2. Review the actions required by your operating procedures for coping with transients and accidents, with particular attention to:

a. Recognition of the possibility of forming voids in the primary coolant system large enough to compromise the core cooling capability, especially natural circulation capability.

b. Operation action required to prevent the formation of such voids.

c. Operator action required to enhance core cooling in the event such voids are formed. (e.g., remote venting)

3. Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to permit containment isolation whether manual or automatic, of all lines whose isolation does not degrade needed safety features or cooling capability, upon automatic initiation of safety injection.

4. For facilities for which the auxiliary feedwater system is not automatically initiated, prepare and implement immediately procedures which require the stationing of an individual (with no other assigned concurrent duties and in direct and continuous communication with the control room) to promptly initiate adequate auxiliary feedwater to the steam generator(s) for those transients or accidents the consequences of which can be limited by such action.

5. For your facilities, prepare and implement immediately procedures which:

   a. Identify those plant indications (such as valve discharge piping temperature, valve position indication, or valve discharge relief tank temperature or pressure indication) which plant operators may utilize to determine that pressurizer power operated relief valve(s) are open, and

   b. Direct the plant operators to manually close the power operated relief block valve(s) when reactor coolant system pressure is reduced to below the set point for normal automatic closure of the power operated relief valve(s) and the valve(s) remain stuck open.

6. Review the action directed by the operating procedures and training instructions to ensure that:

   a. Operators do not override automatic actions of engineered safety features, unless continued operation of engineered safety features will result in unsafe plant conditions. For example, if continued operation of engineered safety features would threaten reactor vessel integrity then the HPI should be secured (as noted in b(2) below).

   b. Operating procedures currently, or are revised to, specify that if the high pressure injection (HPI) system has been automatically actuated because of low pressure condition, it must remain in operation until either:

      (1) Both low pressure injection (LPI) pumps are in operation and flowing for 20 minutes or longer; at a rate which would assure stable plant behavior; or

      (2) The HPI system has been in operation for 20 minutes, and all hot and cold leg temperatures are at least 50 degrees below the saturation temperature for the existing RCS pressure. If 50 degress subcooling cannot be maintained after HPI cutoff, the HPI shall be reactivated. The degree of subcooling beyond 50 degrees F and the length of time HPI is in operation shall be limited by the pressure/temperature considerations for the vessel integrity.

c. Operating procedures currently, or are revised to, specify that in the event of HPI initiation with reactor coolant pumps (RCP) operating, at least one RCP ~~shall remain operating~~ in each loop as long as the pump(s) is providing forced flow.

d. Operators are provided additional information and instructions to not rely upon pressurizer level indication alone, but to also examine pressurizer pressure and other plant parameter indications in evaluating plant conditions, e.g., water, inventory in the reactor primary system.

7. Review all safety-related valve positions, positioning requirements and positive controls to assure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance, testing, plant and system startup, and supervisory periodic (e.g., daily/shift checks,) surveillance to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.

8. Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping, venting or other release of radioactive liquids and gases will not occur inadvertently.

   In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

   a. Whether interlocks exist to prevent transfer when high radiation indication exists, and

   b. Whether such systems are isolated by the containment isolation signal.

   c. The basis on which continued operability of the above features is assured.

9. Review and modify as necessary your maintenance and test procedures to ensure that they require:

   a. Verification, by test or inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing.

c. Explicit notification of involved reactor operational personnel whenever a safety-related system is removed from and returned to service.

10. Review your prompt reporting procedures for NRC notification to assure that NRC is notified within one hour of the time the reactor is not in a controlled or expected condition of operation. Further, at that time an open continuous communication channel shall be established and maintained with NRC.

11. Review operating modes and procedures to deal with significant amounts of hydrogen gas that may be generated during a transient or other accident that would either remain inside the primary system or be released to the containment.

12. Propose changes, as required, to those technical specifications which must be modified as a result of your implementing the above items.

For all light water reactor facilities designed by Combustion with an operating license, respond to Items 1-11 within 10 days of the receipt of this Bulletin. Respond to item 12 (Technical Specification Change proposals) in 30 days.

Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, D.C. 20555.

For all other power reactors with an operating license or construction permit, this Bulletin is for information purposes and no written response is required.

Approved by GAO, B180225 (R0072); clearance expires 7/31/80. Approval was given under a blanket clearance specifically for identified generic problems.

APPENDIX C

LETTER FROM BABCOCK AND WILCOX TO ALL B&W OPERATING PLANTS ON REVISED OPERATING GUIDELINES FOR RC PUMPS UPON RECEIPT OF ESFAS ACTUATION

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505
Telephone: (804) 384-5111

July 20, 1979

All B&W Operating Plants

Subject: Revised Operating Guidelines for RC Pumps
Upon Receipt of ESFAS Actuation

Gentlemen:

The Babcock & Wilcox Company has recently completed additional small break LOCA analyses which have identified a potentially unsafe condition. The details of the analyses were discussed with the owners' representatives on July 18, 1979. In summary, Appendix K criteria may be exceeded for a low probability event involving a certain range of small breaks (approximately 0.025 ft^2 to 0.2 ft^2) and the subsequent loss of RC pumps after the system void fraction has achieved a high value. Analyses have been performed which confirm a safe situation if the RC pumps are tripped promptly in the transient prior to establishing a high void fraction.

Based upon the above information, you are requested to promptly initiate the following B&W recommendation:

Upon receipt of an ESFAS actuation caused by low RC pressure, all operating RC pumps must be tripped immediately.

This revised criteria will of course alter the "immediate action" portion of the Small Break Operator Guidelines previously issued.

If there are further questions, please advise.

Very truly yours,

Service Manager

APPENDIX D

OFFICE OF INSPECTION AND ENFORCEMENT BULLETINS

79-05C and 79-06C

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

July 26, 1979

IE Bulletin Nos. 79-05C & 79-06C

NUCLEAR INCIDENT AT THREE MILE ISLAND - SUPPLEMENT

Description of Circumstances:

Information has become available to the NRC, subsequent to the issuance of IE Bulletins 79-05, 79-05A, 79-05B, 79-06, 79-06A, 79-06A (Revison 1) and 79-06B, which requires modification to the "Action To Be Taken By Licensees" portion of IE Bulletins 79-05A, 79-06A and 79-06B, for all pressurized water reactors (PWRs).

Item 4.c of Bulletin 79-05A required all holders of operating licenses for Babcock & Wilcox designed PWRs to revise their operating procedures to specify that, in the event of high pressure injection (HPI) initiation with reactor coolant pumps (RCPs) operating, at least one RCP per loop would remain operating. Similar requirements, applicable to reactors designed by other PWR vendors, were contained in Item 7.c of Bulletin 79-06A (for Westinghouse designed plants) and in Item 6.c of Bulletin 79-06B (for Combustion Engineering designed plants).

Prior to the incident at Three Mile Island Unit 2 (TMI 2), Westinghouse and its licensees generally adopted the position that the operator should promptly trip all operating RCPs in the loss of coolant accident (LOCA) situation. This Westinghouse position, has led to a series of meetings between the NRC staff and Westinghouse, as well as with other PWR vendors, to discuss this issue. In addition, more detailed analyses concerning this matter were requested by the NRC. Recent preliminary calculations performed by Babcock & Wilcox, Westinghouse and Combustion Engineering indicate that, for a certain spectrum of small breaks in the reactor coolant system, continued operation of the RCPs can increase the mass lost through the break and prolong or aggravate the uncovering of the reactor core.

The damage to the reactor core at TMI 2 followed tripping of the last operating RCP, when two phase fluid was being pumped through the reactor coolant system. It is our current understanding that all three of the nuclear steam system suppliers for PWRs now agree that an acceptable action under LOCA symptoms is to trip all operating RCPs immediately, before significant voiding in the reactor coolant system occurs.

Action To Be Taken By Licensees:

In order to alleviate the concern over delayed tripping of the RCPs after a LOCA, all holders of operating licenses for PWR facilities shall take the following actions:

Short-Term Actions

1. In the interim, until the design change required by the long-term action of this Bulletin has been incorporated, institute the following actions at your facilities:

   A. Upon reactor trip and initiation of HPI caused by low reactor coolant system pressure, immediately trip all operating RCPs.

   B. Provide two licensed operators in the control room at all times during operation to accomplish this action and other immediate and followup actions required during such an occurence. For facilities with dual control rooms, a total of three licensed operators in the dual control room at all times meets the requirements of this Bulletin.

2. Perform and submit a report of LOCA analyses for your plants for a range of small break sizes and a range of time lapses between reactor trip and pump trip. For each pair of values of the parameters, determine the peak cladding temperature (PCT) which results. The range of values for each parameter must be wide enough to assure that the maximum PCT or, if appropriate, the region containing PCTs greater than 2200 degrees F is identified.

3. Based on the analyses done under Item 2 above, develop new guidelines for operator action, for both LOCA and non-LOCA transients, that take into account the impact of RCP trip requirements. For Babcock & Wilcox designed reactors, such guidelines should include appropriate requirements to fill the steam generators to a higher level, following RCP trip, to promote natural circulation flow.

4. Revise emergency procedures and train all licensed reactor operators and senior reactor operators based on the guidlines developed under Item 3 above.

5. Provide analyses and develop guidelines and procedures related to inadequate core cooling (as discussed in Section 2.1.9 of NUREG-0578, "TMI 2 Lessons Learned Task Force Status Report and Short-Term Recommendations") and define the conditions under which a restart of the RCPs should be attempted.

Long-Term Action

1. Propose and submit a design which will assure automatic tripping of the operating RCPs under all circumstances in which this action may be needed.

Schedule

The schedule for the short-term actions of this Bulletin is:

Item 1: Effective upon receipt of this Bulletin,

Item 2: Within 30 days of receipt of this Bulletin,

Item 3: Within 30 days of receipt of this Bulletin,

Item 4: Within 45 days of receipt of this Bulletin,

Item 5: October 31, 1979 (as noted in Table B-2 of NUREG-0578, under Item 3).

A schedule for the long-term action required by this Bulletin should be developed and submitted within 30 days of receipt of this Bulletin.

Reports should be submitted to the Director of the appropriate NRC Regional Office with copies forwarded to the Director, Office of Inspection and Enforcement and the Director, Office of Nuclear Reactor Regulation, Washington, D. C. 20555.

Approved by GAO (R0072): clearance expires 7/31/80. Approval was given under a blanket clearance specifically for generic problems.

NRC FORM 335 (7-77) U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET**	1. REPORT NUMBER (Assigned by DDC) NUREG-0623
4. TITLE AND SUBTITLE (Add Volume No., if appropriate) Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors	2. (Leave blank)
	3. RECIPIENT'S ACCESSION NO.
7. AUTHOR(S) Brian W. Sheron	5. DATE REPORT COMPLETED MONTH November YEAR 1979
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555	DATE REPORT ISSUED MONTH November YEAR 1979
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	11. CONTRACT NO.
13. TYPE OF REPORT Technical Report	PERIOD COVERED (Inclusive dates)
15. SUPPLEMENTARY NOTES	14. (Leave blank)

16. ABSTRACT (200 words or less)

Recent analyses by the PWR vendors have indicated that excessive fuel cladding temperatures could result if the reactor coolant pumps remained running or their trip was delayed during a small break Loss-of-Coolant Accident.

The review of these analyses and analysis models by the NRC staff have resulted in the conclusion that early, automatic pump trip during a small break LOCA is required in order to prevent excessive fuel cladding temperatures.

17. KEY WORDS AND DOCUMENT ANALYSIS	17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

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