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MEMORANDUM FOR: Atomic Safety and Licensing Appeal Board for Rancho Seco

FROM: Gus C. Lainas, Assistant Director for Operating Reactors, Division of Licensing

SUBJECT: BOARD NOTIFICATION (BN-82-71) - RANCHO SECO

The Staff has prepared the enclosed report regarding "feed and bleed" cooling of the reactor core upon loss of feedwater to steam generators. The report is in response to a request for information from H. R. Denton dated April 29, 1982.

Feed and bleed cooling is recognized in the report to be an additional defense-in-depth method for providing core cooling in the unlikely event that emergency feedwater to the steam generators is lost. However, it is important to note that the feed and bleed method is not a preferred method of decay heat removal, but is a possible emergency alternative for primary system heat removal for events beyond the plant design basis.

Babcock and Wilcox originally performed the feed and bleed analysis so that inadequate core cooling procedures could be developed. These procedures instruct the plant operator to establish and maintain feed and bleed cooling following a complete loss of heat sink, until feedwater can be restored.

This information is being supplied to the ASLAB because this material is relevant to natural circulation, void formation and small-break LOCA, which were issues on the Rancho Seco hearing record. No new safety issues have been raised by this information, and it does not affect current staff positions.

Original signed by

Gus C. Lainas, Assistant Director
for Operating Reactors
Division of Licensing

Enclosure:
Report on NRC Staff Position
on Feed & Bleed Cooling

cc w/enclosure: See next page

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REPORT ON NRC STAFF POSITION ON

FEED AND BLEED COOLING

Item 1 A description of the staff position at the TMI-1 restart hearing on the role of "feed and bleed" during a SBLOCA

RESPONSE

The staff's position at the hearing was tnat feed and bleed cooling is not relied on for heat removal. This position was made clear to the ASLB in the TMI-1 restart hearing in (1) written testmony by NRC staff witness J. Wermiel and (2) oral testimony of W. Jensen as follows.

(1) Written Testimony of J. Wermiel in Response to Board Question 6: Question 6i. Will the reliability of the emergency feedwater system be greatly improved upon conversion to safety-grade, and is it the licensee's and staff's position that the improvement is enough such that the feed-and-bleed backup is not required.

(Witness Wermiel)

Response: Based on knowledge of the improvement in reliability gained by eliminating first order failure sources, it is the staff's judgment that the reliability of the emergency feedwater system will be improved once the fully safety-grade system is installed. The single failure problem associated with integrated control system/non-nuclear instrumentation described in the response to 6a and b above will be eliminated. In addition, various other hardware, procedural and administrative improvements as identified in the TMI-1 Restart SER, NUREG-0680 under Order Item 1a should enhance emergency feedwater system reliability. However, a quantitative reassessment of the reliability of the fully safety-grade EFW system has not been performed. The feed-and-bleed back-up is not required by the staff and, therefore, need not meet all requirements of a safety system. However, it is recognized as additional defense in depth for providing core cooling in the very unlikely event that emergency feedwater is lost, and the HPI pumps and primary safety valves which comprise the feed and bleed mode are required to be available by Technical Specifications.

(2) Oral Testminony of W. Jensen Regarding UCS Contentions 1 and 2

(Dr. Jordan) I would address the question then directly to Mr. Jensen. Did I misstate what you said? Do you believe that the high pressure injection system is important in that it not only supplies emergency cooling inventory but it also removes heat in the feed and bleed mode? That that is an important safety feature?

(The Witness) The high pressure injection system is an important safety feature for making up the coolant lost from a small break LOCA. The NRC does not rely on this system for heat removal in the feed and bleed mode by which core decay heat would be forced through the safety valve or the PORV. Instead, we rely on the heat removal from the emergency feedwater system.

(Dr. Jordan) Okay. That's fine.

(Ms. Weiss) If I can refer, Dr. Jordan, I think the exact question you are asking is answered on page 9 of the staff testimony in response to Board question number 6. I was going to read the sentence to you. (Wermiel testimony above)

The feed and bleed back up is not required by the staff and therefore need not meet all the requirements of the safety system. It's just simply a direct quote.

(Dr. Jordan) Yes. I remember that and thank you for pointing that out. I think that clears up the matter."

Item 2 An interpretation of the TMI-1 Licensing Board decision regarding the need for reliable and effective "feed and bleed" during SBLOCA

RESPONSE

There is an interest in whether the ASLB accepted the staff position on the reliance to be placed on feed and bleed cooling. We believe that the ASLB did not accept our position, regarding emergency feedwater reliability, as shown in the following excerpts from its decision. We believe however that the board did not err in declining to find that additional modifications to the emergency feedwater system are necessary at TMI-1 prior to restart.*

Page 224 of the TMI-1 Licensing Board decision acknowledges the NRC Staff position (see Item 1 above) by noting that:

"The Staff's position is that the loss of emergency feedwater following a main feedwater transient is not an accident which must be protected against with safety-grade equipment."

To us, this observation by the ASLB says that our position in Item 1 above was understood by the Board. At Page 242 of the decision the Board goes on to point to a precedent ruling made by the St. Lucie-2 Appeal Board for requesting additional reliability numbers from the staff. The TMI-1 Board noted that:

*NRC response to UCS' exceptions to the PID, filed with the Appeal Board in the TMI-1 Restart proceeding May 20, 1982.

"The (St. Lucie) Appeals Board decided that measures were required to mitigate such an event** should it occur. We believe that similar measures are necessary at TMI-1; that the reliability of the EFW system has not been demonstrated to be adequate by itself. However, the EFW system is backed up by the high pressure injection system, so that in the event of failure of the EFW system the core can be cooled by feed and bleed while repairs are being made to the EFW system."

We conclude from this statement that the TMI-1 Board has relied upon the availability of feed and bleed in reaching its finding that the TMI-1 design is acceptable. The question then is how the Board reached this conclusion in light of the Staff position (Item 1 above). The answer is summarized on page 250 of the TMI-1 Board decision where the Board states:

"We have relied on the staff figures on reliability of the EFW system <u>and our own estimates</u> (emphasis added) of the adequacy of the feed-and-bleed backup to arrive at our conclusion that the core is adequately protected from a loss of main feedwater transient, the dominant challenge to the EFW system."

**Complete loss of all AC power including both diesel generators.

We conclude that the Licensing Board reached the same conclusion as the staff (the TMI-1 design satisfies the Commission's regulations), although the board's basis for the conclusion is different. The basis for the staff position is summarized in Question 3 below. We have studied the Licensing Board decision to understand the basis for its conclusion. At paragraph 1056 we find the following:

"Since the EFW System is backed up by a safety-grade HPI, designed to protect the core in the event of a small break LOCA, we believe we can conservatively assume an additional safety factor of 100, or an overall probability of failure to protect the core of about 10^{-6}/yr. Lacking any demonstration that the above failure probabilities are grossly in error, we conclude that the EFW system, as modified, will, with the HPI backup, adequately protect the health and safety of the public."

During the TMI-1 hearing, the NRC Staff did not provide any detailed discussion, for or against, the above Licensing Board assessment. We do not have sufficient information regarding the uncertainties associated with of feed and bleed cooling to credit it with a 100 fold reduction in the probability of core melt.

Item 3a <u>A detailed explanation of the staff's technical basis for its position on "feed and bleed" at TMI-1.</u>

RESPONSE

It was the Staff's position during the TMI-1 hearing that the emergency feedwater (EFW) system is required to be available for decay heat removal in feedwater transients and certain small break loss-of-coolant accidents without feedwater. We also noted that should EFW be initially unavailable, there is at least 20 minutes time available to take action to establish EFW flow prior to uncovering of the core following a loss of main feedwater or certain small break loss of coolant accidents. The TMI-1 EFW system will, at the time of restart, meet the Commission's requirements for safety related equipment, in the event of small break LOCA and/or loss of main feedwater if credit for operator action is given (to initiate the system) within 20 minutes. The TMI-1 EFW system will be fully automatic for these events by the first refueling outage after restart. The staff recognizes that a feed and bleed capability exists at TMI-1 to provide additional defense in depth for decay heat removal should EFW fail. The inadequate core cooling procedures at TMI incorporate the feed and bleed process. Operators are trained in the use of these procedures at TMI-1 and feed and bleed is covered in the scope of OLB examinations of the TMI operators. It is usually covered in the simulator portion of the examination. Safety grade equipment to accomplish feed and bleed backup to EFW in the event of a complete loss of all feedwater is not required to be included within the design basis since the EFW system at the time of restart is sufficiently reliable to make a postulated loss of EFW system acceptably low.

Item 3b <u>Clarify the difference between the "feed and bleed" mode of cooling and the "boiler/condenser" mode of cooling</u>

RESPONSE

For small breaks below a certain size, the break area is not large enough to relieve all the energy generated by decay heat. For this condition, heat transfer through the steam generator is the preferred method of providing additional required energy removal capability. To accomplish this, emergency or auxiliary feedwater systems must be operating. Since the reactor coolant pumps are tripped for most small breaks, coolant flow through the core is by natural circulation. Feed & Bleed is a method by which decay heat is removed from the primary system if no feedwater were available so that natural circulation did not occur. The "boiler/condenser" mode of cooling is one of three modes of natural circulation cooling discussed below. Each mode represents a progressively degraded condition of the primary system in terms of system inventory. Thus it is possible for some small break scenarios to experience all three modes of natural circulation heat removal. In small break LOCA calculations by B&W* temporary interruption of all modes of natural circulation was predicted however, inventory loss in these three modes is not sufficient to cause extended core uncovery and fuel damage. It is not necessary that the primary system be refilled folllowing a LOCA in order to adequately cool the core. Analyses by B&W indicate that adequate decay heat can be removed under any of the following three natural circulation modes.

1. Single phase - In this mode the entire primary system remains in a subcooled liquid state. Core flow is maintained solely by density differences between hot and cold liquid.

2. Two phase continuous - This mode is similar to mode 1 except that the hot side is at saturation and at low steam quality. Bubbles are formed in the upper portion of the core and are swept, as part of a continuous two phase mixture, into the steam generator and condensed. During this time, some of the steam generated in the core will rise into the upper head and accumulate there as a single large bubble. For B&W plants this heat removal mode will persist until the liquid level drops below the hot leg U-bend.

*B&W report "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177-FA Plants" May 7, 1979.

3. Boiler/Condenser - When the hot leg U-bend is voided, liquid will not be carried into the steam generator. However, when sufficient steam has accumulated from boiling in the core such that a condensing surface is exposed within the steam generator tubes, heat will be removed by steam condensation on the tube walls. This method of heat removal is referred to as boiler/condenser. Thus a period will exist between formation of a bubble in the hot leg U-bends when mode 2 natural circulation is lost, and the uncovering of the steam generator condensing surface, during which no natural circulation would exist in B&W plants. The condensing surface is at a higher elevation than the core so that boiler/condenser natural circulation will be established in the event of a small break LOCA before the core could be uncovered. Boiler condenser natural circulation was demonstrated to be effective in LOFT* and Semiscale** experiments for U-tube steam generators.

*NUREG CR-1570 "Experimental Data Report for LOFT Nuclear Small Break Experiment L-3-7", August 1980.

**EGG-SEMI-5507 "Quick Look Report for Semiscale Mod-2A Test S-NC-2," July 1981.

If heat removal through the steam generator cannot be achieved due to loss of all feedwater (an event not required to be considered as a part of the design basis), "feed and bleed" can be used as an alternate heat removal method. The procedure involves energy removal by venting hot water and/or steam through the primary system PORVs and/or safety valves (bleeding), and replacing the vented coolant with cold HPI water (feeding).

Item 3c <u>Assessment of Current Status and Existing Information on "Feed and Bleed"</u>

RESPONSE

As you recall, in a recent communication to Dr. Henry Myers we noted that for a small break LOCA which is subsequently isolated, a phenomenon similar* to "feed and bleed" might ultimately occur as the means of decay heat removal if steam bubbles were trapped at the top of hot legs and did not rapidly condense even if emergency feedwater were available. This method of heat removal from the primary system might occur if the core were sufficiently cooled so that decay heat no longer boiled the incoming HPI water but forced it through

*The term "similar" is used, since in this case feedwater to the secondary side of the steam generator is assumed available, and no operator actions are assumed to initiate decay heat removal via the safety valves.

the safety valves as liquid. If boiling occurred in the core, the steam production would act to increase the bubble size in the hot-leg U-bends. If the hot leg bubble size increased sufficiently, a condensing surface on the steam generator tubes would be exposed. This would establish natural circulation in the boiler/condenser mode.

The bubbles could not expand sufficiently to uncover the core or to exhaust steam out of the pressurizer since the secondary system water level in the steam generators would be above the core and the pressurizer surge line entry elevation. Although our study of this scenario is recent and was not discussed during the TMI-1 hearing, no additional staff reliance on feed and bleed should be implied since if the feed and bleed process discussed above were insufficient to remove decay heat, sufficient coolant loss through the safety and relief valves would eventually reestablish natural circulation in the boiler/condenser mode. The letter to Dr. Myers is attached for further information on these recent developments.

All three PWR suppliers are developing emergency procedure guidance to licensees on how to use equipment to perform "feed and bleed" operations as a backup method of heat removal if all measures for feeding steam generators are lost. It is important to stress that at this time "feed and bleed" is not a preferred method of decay heat removal. The equipment used for feed and bleed operation was not designed for that purpose. Feed and bleed is only one possible emergency alternative for primary system heat removal for events beyond the design basis. All PWRs have in their proposed emergency guidelines, methods for use of decay heat removal schemes other than the design basis equipment. In particular, guidance is given to provide alternate sources of secondary cooling if main and auxiliary feedwater are unavailable (e.g., by depressurizing the secondary system and activating the condensate pumps). Operators would resort to feed and bleed only if no source of water is available to feed the steam generators. The NRC has no design requirements for these other alternate schemes, just as we have none for the "feed and bleed" capability. What is required for the design basis is a reliable auxiliary feedwater system to remove decay heat until the RHR system can be activated to ultimately achieve cold shutdown. However, to provide defense in depth, feed and bleed procedural instructions should be available to operators because the capability to feed and bleed exists.

As to the technical performance of "feed and bleed," we know it depends on the HPI pump performance characteristics, the PORV relieving capacity, and the plant power to volume ratio. Analyses have been conducted by all three PWR suppliers to examine "feed and bleed" capability for their designs. Also, NRC contractors at LANL and INEL have analyzed "feed and bleed" with the computer codes TRAC and RELAP. As noted previously, a B&W calculation for a TMI class plant showed that "feed and bleed" was an effective heat removal method even if no credit is taken for PORV actuation. This is because most B&W plants have HPI pumps with a very high shutoff head, and enough energy can be relieved at high pressure through the safety valves. It is important to note that the assessment of "feed and bleed" rests almost exclusively on analysis.

Analytical uncertainties related to such phenomena as non-equilibrium thermodynamics, bubble formation and repressurization caution against taking too much credit for analytical predictions of system behavior.

One LOFT experiment (L9-1/L3-3) explored "feed and bleed" in a limited way. After a simulated loss of feedwater, the PORV was latched open to allow depressurization. The results showed that depressurization to the HPI actuation point did indeed occur. However, HPI actuation was purposely not allowed to occur so that other accident mitigation schemes could be explored.

Item 4 Recommendations for Future Action

It is desirable to improve the experimental basis for understanding system behavior during "feed and bleed." This should improve the guidance in emergency procedures and training that is being developed under Task I.C.1 of NUREG-0737. To accomplish this, we are exploring ways to expand the current Semiscale test series to include "feed and bleed" experimental data. We expect shortly to issue a request to RES which will include these proposals.

The current Semiscale configuration cannot simulate the unique features of the B&W NSSS. You know from previous discussions that we have been trying to resolve the problem of uncertainties for the B&W analytical methods in predicting long term LOCA recovery under Task II.K.3.30 NUREG-0737. We are investigating the unique features of the B&W design and the lack of integral systems data (see attached letter to B&W owners). We will shortly transmit to all B&W owners our conclusion that such data are required. The basis for this conclusion is the need for additional verification of some aspects of the thermal-hydraulic behavior during natural circulation cooling of the B&W design with feedwater available during small break LOCAs, as well as uncertainty in the feed and bleed process. You will also recall that the ACRS letter of June, 1982 highlighted this problem for resolution prior to its concurrence on full power operation of Midland, a B&W reactor.

MAY 7 1982

Dr. Henry Myers
Subcommittee on Energy and the Environment
Committee on Interior and Insular Affairs
United States House of Representatives
Washington, D.C. 20515

Dear Dr. Myers:

In early April you requested the NRC staff to comment on a statement in paragraph 619 of the TMI Restart Partial Initial Decision. The statement read as follows:

"If, however, the voids are steam, as would be expected in a small-break LOCA, the bubble in the hot leg should be compressed and condensed as the primary system pressure is increased by operation of the HPI system."

The context of this statement in paragraph 619 is a discussion of the recovery from a small-break LOCA when ECCS injection flow exceeds break flow so that the reactor system is filling. The discussion addressed a Union of Concerned Scientists (UCS) concern that for this condition a steam bubble in the top of the hot leg U-bends might prevent the reestablishment of single-phase natural circulation.

Mr. Kammerer of NRC replied to your request in a letter of April 8, 1982. The response included the following comments:

"If a steam bubble exists and primary system pressure is raised, the bubble will be compressed and there will be condensation. The condensation occurs because as you raise system pressure system temperature drops below saturation. Condensation must occur to reach saturation conditions again. The bubble will not necessarily be completely condensed. This depends on bubble size and pressure change." (Sentence was underlined in your April 19, 1982 letter.)

In a letter of April 19, 1982, you asked if the quote underlined above from our April 8, 1982, response was correct. Specifically, you questioned "... whether, in the case of a small break LOCA, the heat transfer would be sufficient to keep the steam space from expanding into the core." The underlined quote is not correct because there must be heat transfer to a heat sink to absorb the heat of vaporization associated with the condensation of steam, and the heat transfer rate is important. Also, the system temperature does not actually drop; rather, the saturation temperature increases with increasing pressure. However, while the condensation rate of the hot leg bubble will affect the overall recovery behavior, it does not, by itself, govern the overall adequacy of decay heat removal during a small break LOCA in a plant designed by B&W. To be certain that there is no similar misunderstanding regarding this matter, we are providing a copy of this letter to the Licensing Board and will inform the Appeal Board of this issue during the upcoming appellate process.

PDR
~~8205210004~~

The response of a B&W designed reactor to steam accumulation at the top of the hot leg U-bends under off-normal conditions is complex. The detailed behavior of the reactor under these conditions is still being actively studied by B&W, the B&W plant owners, and the staff. Although uncertainties remain, including heat transfer rates for steam condensation, we have reached some general conclusions about the effect of these uncertainties and the potential for producing unacceptable core heatup. These conclusions are based on calculations by B&W and/or by a staff contractor of a number of postulated small break LOCAs. We have concluded that the uncertainties can be physically bounded and that these bounding assumptions do not produce unacceptable core heatup.

In the enclosure, the staff addresses your questions regarding the impact of steam condensation rates and supporting analyses and describes how the use of bounding assumptions would not result in unacceptable core heatup. Calculations related to these conclusions are identified.

There are still uncertainties in the thermal-hydraulic response of the B&W design under some of these conditions. We are pursuing with B&W and B&W plant owners the details of the thermal and hydraulic phenomena involved, including the possible need for additional sensitivity analyses and a small scale, integral systems test. Our objective is to confirm the analytical results described in the enclosure and to aid in our further analysis of more complex, multiple failure events being studied in the context of the new symptom-oriented, emergency procedure guidelines.

Sincerely,
Original Signed by
H. R. Denton
Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
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Enclosure

DYNAMIC RESPONSE OF B&W REACTORS TO SMALL BREAK LOCAS

Detailed analyses have been performed either by B&W or by staff contractors for the two classes of small break LOCAs: (1) those that can be subsequently isolated (e.g., letdown lines, PORVs), and (2) those that are not isolatable. They are discussed in that order, below.

Small Breaks Which Are Subsequently Isolated

We have had our contractor, the Los Alamos National Laboratory (LANL) perform an analysis of a small break in the cold leg of the B&W reactor coolant system that is subsequently isolated. The calculations were performed with the advanced TRAC computer code, and we have some preliminary results. We are not aware of any calculations that have been performed by B&W in which a small break LOCA was subsequently isolated.

In our analysis, the system was assumed to lose primary coolant from the break until the upper vessel head region, pressurizer, and hot leg U-bends were filled with steam. At that time, the break was assumed to be isolated by the operator. Approximately 1,000 seconds later, it was assumed the operator began a controlled secondary system depressurization. The analyses showed that the flow of cold water from the two high pressure injection pumps, coupled with controlled secondary system depressurization, would condense the hot leg bubbles and restore natural circulation. We are still evaluating the ability of the TRAC code to model the heat conduction in the liquid near the steam-liquid interface which is required to accurately calculate the primary system refilling process. However, steam condensation rates are not expected to influence the overall conclusion that no unacceptable core heatup would occur, as explained in the following paragraphs.

If the steam was condensed at 100% efficiency by the cold HPI water, the top of the hot leg U-bends would refill with liquid and single phase natural circulation would be restored. If, however, the steam condensation rate is very low, and, in fact, in the limit the steam is assumed not to condense at all, two possibilities would result.

The first possibility is that the HPI pumps would repressurize the system sufficiently to compress the steam to a small enough volume to allow liquid on the upstream side of the hot leg U-bend to spill over into the downstream side and resume natural circulation. If this did not occur, the HPI would continue to inject ECC water and repressurize the reactor coolant system until the pressure reaches either the PORV or the safety valve set pressure. We would expect that core cooling would be maintained by a "feed and bleed" process until the steam bubble at the top of the U-bend eventually condensed by heat transfer to the pipes and across the liquid vapor interface. Once the bubble condenses, single phase natural circulation throughout the steam generators would be resumed.

The other possible behavior if the steam condensation rate is low is associated with a design difference in B&W reactors. One plant designed by B&W, the Davis-Besse plant, does not have a "high head" HPI pump (one that can pump water into the primary system at or above the safety valve set point). The shutoff head of this pump is about 1700 psi. In the event the cold water from the HPI pumps does not condense the steam in the hot leg U-bend, the system may repressurize to above the shutoff head of the HPI pump, and eventually reach the PORV or the safety valve setpoint. The system will begin to lose primary coolant through the PORV or the safety valves and drain down. Once the primary coolant level on the downstream side of the hot leg U-bend extends into the steam generator tube region below the condensing surface of the secondary coolant, steam in the primary system will begin to condense, lowering the primary system pressure and closing the PORV or the safety valve.

As the system pressure decreases below 1700 psi, the HPI will actuate and begin to fill the primary system. This would result in covering the condensing surface in the steam generator, and producing another repressurization, which, in turn, could stop HPI flow and cause the PORV or the safety valve to open and release enough coolant to reestablish a condensing surface. A number of these cycles may occur before the charging system completely refills the system or before the steam bubble is condensed by heat transfer to pipes and across the liquid vapor interface. Since the condensing surface in the steam generators is above the elevation of the top of the core, natural circulation should be established before the hot leg steam bubble extends into the core.

Small Breaks Which Are Not Isolated

Small break LOCAs which are not subsequently isolated have been calculated by both B&W and by the staff's contractor, LANL. The analysis performed by B&W was for a 0.01 ft^2 cold leg break. They used a computer code that conforms to the requirements of Appendix K to 10 CFR 50. With the exception of the assumed number of HPI pumps available, the modeling assumptions required by Appendix K do not affect the thermal-hydraulic models of interest for the small break LOCA. Thus, the results of the hydraulic analysis should be realistic. The 0.01 square foot break was selected since this size is insufficient to remove decay heat via break flow (thus requiring the steam generators for decay heat removal). It was also predicted to result in the repressurization phenomenon for the reactor coolant system.

From these analyses, B&W concluded that a range of small break sizes could be postulated in which steam generated in the core would accumulate at the top of the hot leg U-bends and cause an interruption of natural circulation flow. The interruption of natural circulation flow would isolate the steam being produced in the reactor core from the steam generator heat sink. The net steam accumulation in the system was calculated to cause the primary system to repressurize. This repressurization was calculated to continue until the primary system coolant loss through the break was sufficient to uncover a steam condensing surface in the steam generators. It is expected that some steam generated in the core would flow into the upper elevations of the downcomer annulus via the vent valves and condense in the colder water in that region. However, cold water from one HPI pump was not calculated to be sufficient to condense all of the steam generated in the core.

Similar to the isolated break case previously discussed, the repressurization of the reactor coolant system caused by interruption of natural circulation would lead to a boiler-condenser mode of two-phase natural circulation, and subsequently reduce system pressure. Once the HPI flow is calculated to exceed the break flow, the system coolant inventory will stop decreasing and begin to increase. In figures 1, 2, and 3, the temporal behavior of system pressure, liquid level in the hot leg piping, and liquid level in the reactor vessel are shown for this case as calculated by B&W. The B&W analyses submitted to the staff terminate at about the time system inventory begins to increase. However, the continued recovery of the event is considered relevant to your concern, and is described further below.

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As the system refills, the steam condensing surface in the steam generator will again be recovered by liquid, and a steam bubble will be trapped at the top of the hot leg U-bend. The scenario is now expected to proceed similar to the isolated break case previously described. That is, if the steam is rapidly condensed during the refilling process, single phase natural circulation will be reestablished and primary system pressure will remain low with no significant repressurization. If the steam trapped at the top of the hot leg U-bend is not rapidly condensed, the system would repressurize until (1) the break flow exceeded the HPI flow and a condensing surface was reestablished, (2) the system repressurized and compressed the steam to a sufficiently small enough volume so that water upstream of the hot leg U-bend could spill over into the downstream side of the hot leg U-bend and reestablish natural circulation, or (3) the PORV/safety valve setpoint was reached. For the Davis-Besse plant, we believe only option 1 would occur since the HPI pumps are not sufficient to pump water into the system at or above the safety valve setpoint.

The staff has also been calculating and analyzing the response of B&W-designed reactors to small break LOCAs in which natural circulation is predicted to be interrupted by steam accumulation at the top of the hot leg U-bends. Our contractor at LANL has recently completed a few small break analyses and has looked at four recovery enhancement actions presently either proposed by B&W or being considered by the staff. These four options are: (1) high point vent operation, (2) momentary pump restart, (3) secondary side depressurization, and (4) ECC spray at the top of the hot leg U-bends. All of these options are being investigated to determine their ability to enhance the reestablishment of single phase natural circulation during the recovery portion of the accident. Initial results of our contractor's calculations show that for a realistically calculated small break (i.e., nominal decay heat, two HPI pumps available, etc.) in a B&W plant, with a break size in the range of that for which B&W predicts repressurization would occur, the system did not repressurize once the hot leg U-bends filled with steam. Although our evaluation is not yet complete, we believe that the reason the LANL calculation did not show a repressurization is because steam generated in the core vented to the upper reaches of the vessel annulus via the vent valves, and the flow from two HPI pumps was sufficient to condense all of the steam produced in the core.

The results of the TRAC analyses are shown in figures 4 through 7. In figures 4 and 5, the B&W results are overlayed to show the differences. We believe they can be attributed to two versus one HPI pump being available. The results of the analyses to investigate natural circulation recovery enhancement methods were only recently presented to the staff at a meeting with LANL. These analyses have not been documented by LANL in a formal report, and we have not reviewed them in any detail. However, based on information received at the meeting with the contractor, the results show that the hot leg U-bend was not refilled following recovery from the small break LOCA (1.75 inch diameter), and that none of the natural circulation recovery enhancement methods previously listed were effective. However, LANL reported that the core remained covered and decay heat was continuously removed from the core. They attributed the heat removal to internal recirculation (steam exiting the core is vented to the vessel upper annulus and condensed by the cold HPI water entering the downcomer). This situation physically could only persist until the decay heat was eventually removed entirely by the break flow or the system eventually was refilled and natural circulation reestablished.

Before widely disseminating the results of the LANL calculations, we have asked our Office of Regulatory Research to carefully document and evaluate them and assist us in confirming their validity. We believe this careful approach is justified because the analyses showed that core decay heat was continuously removed and that no core uncovery or heatup was predicted.

In summary, although we are continuing our evaluation of the rate of hot leg steam bubble condensation in the recovery from both isolated and unisolated reactor coolant system small-break LOCAs in reactors like TMI-1, we do not believe that steam bubbles present in the reactor coolant system resulting from small break LOCAs (either isolated or unisolated) in either the cold or hot legs of the primary system will result in unacceptable heatup of the core.

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Figure 1

CORE PRESSURE VS. TIME -
0.01 FT^2 BREAK AT PUMP
DISCHARGE-SYMMETRIC AFW

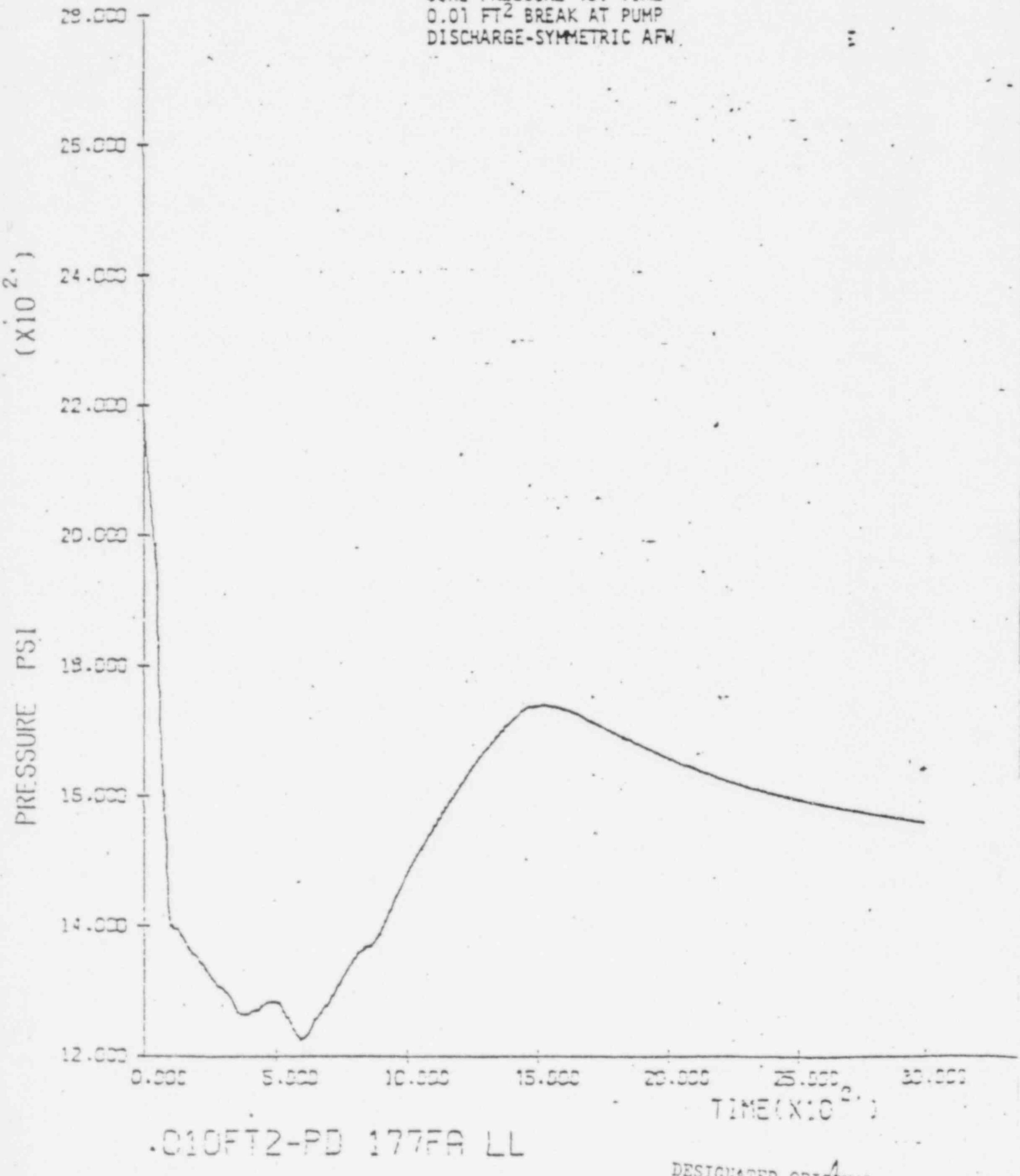

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Figure 2

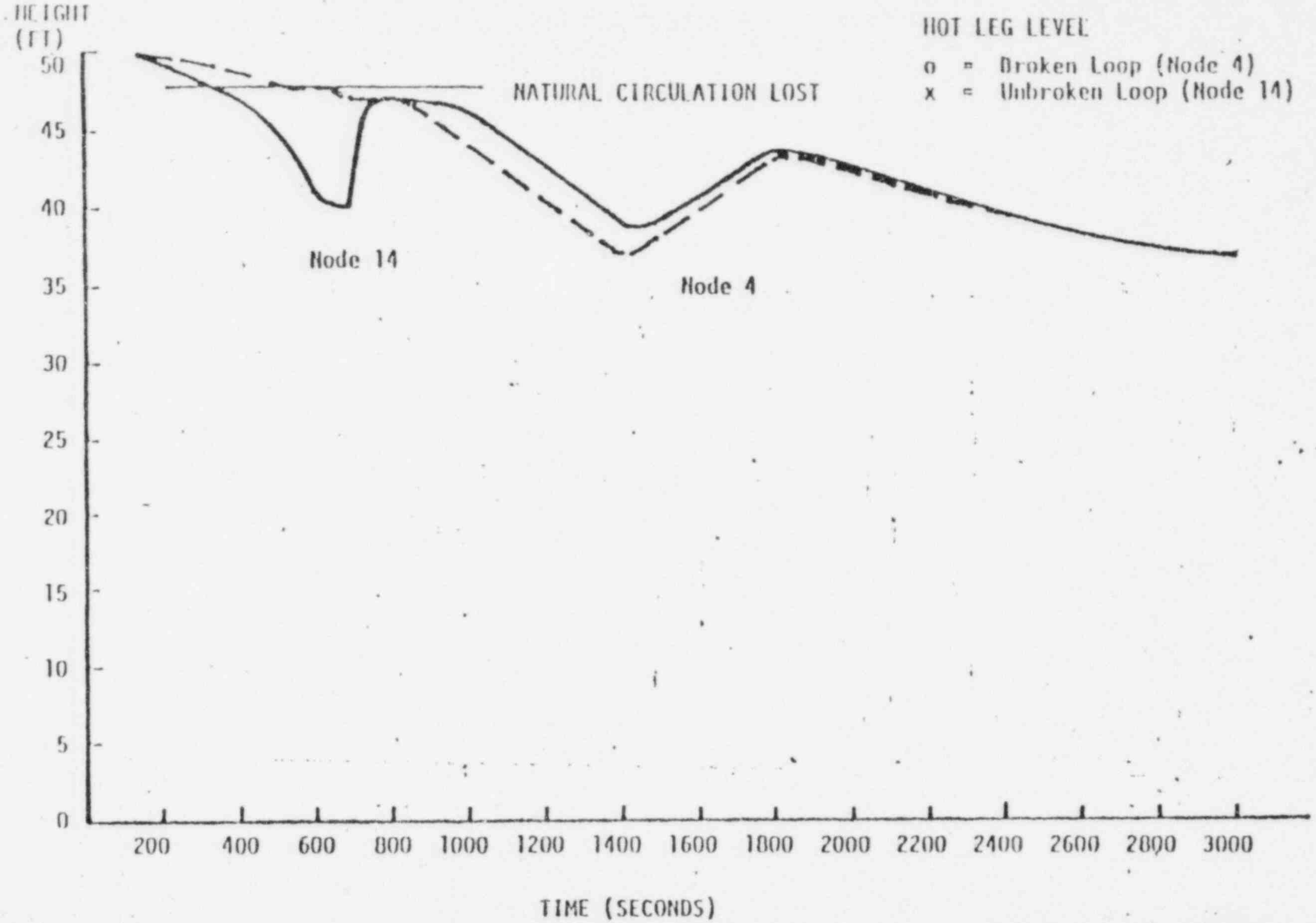

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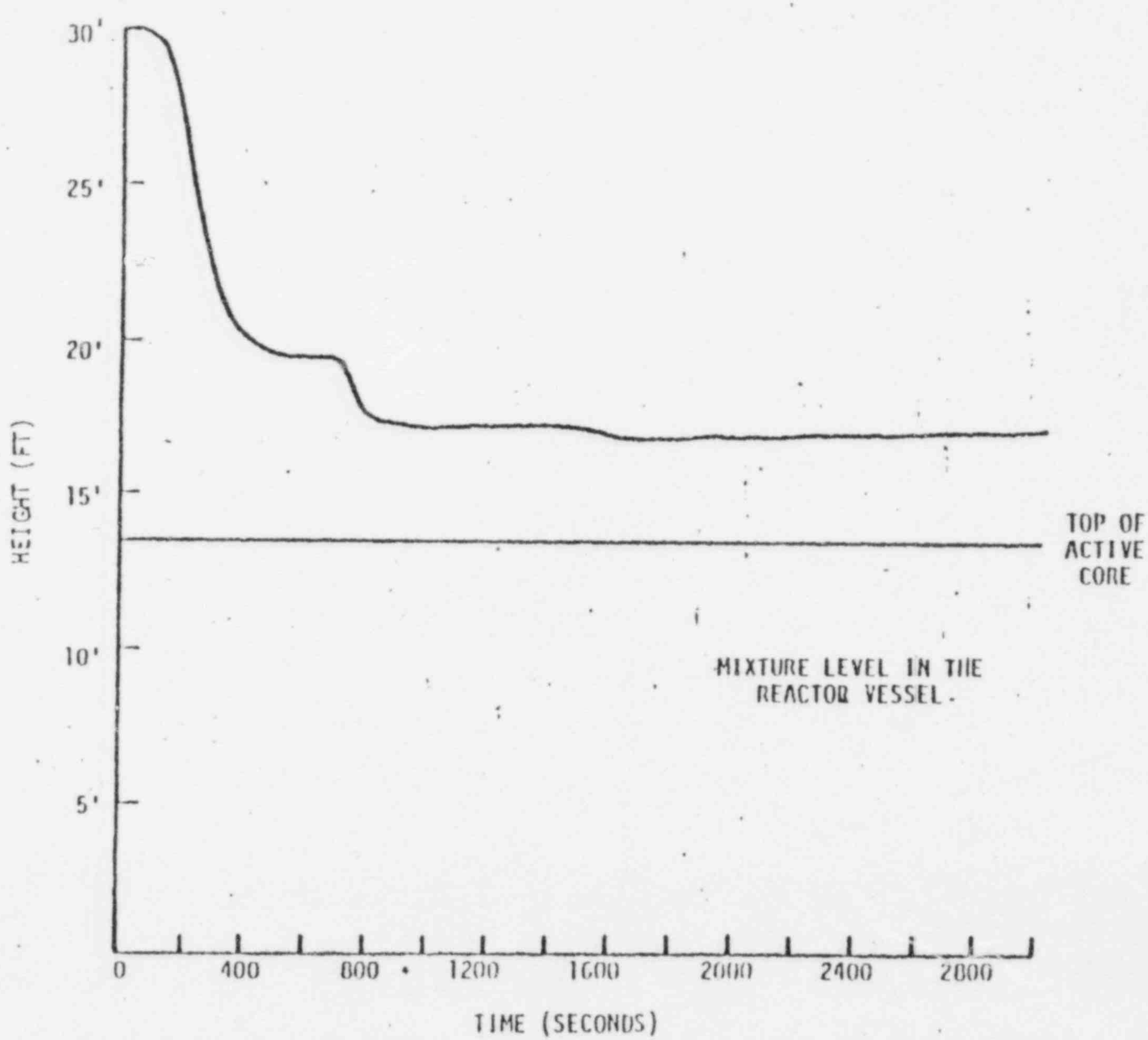
Figure 3


30'
25'
20'
15'
10'
5'
HEIGHT (FT)
TOP OF ACTIVE CORE
MIXTURE LEVEL IN THE REACTOR VESSEL.
0
400
800
1200
1600
2000
2400
2800
TIME (SECONDS)

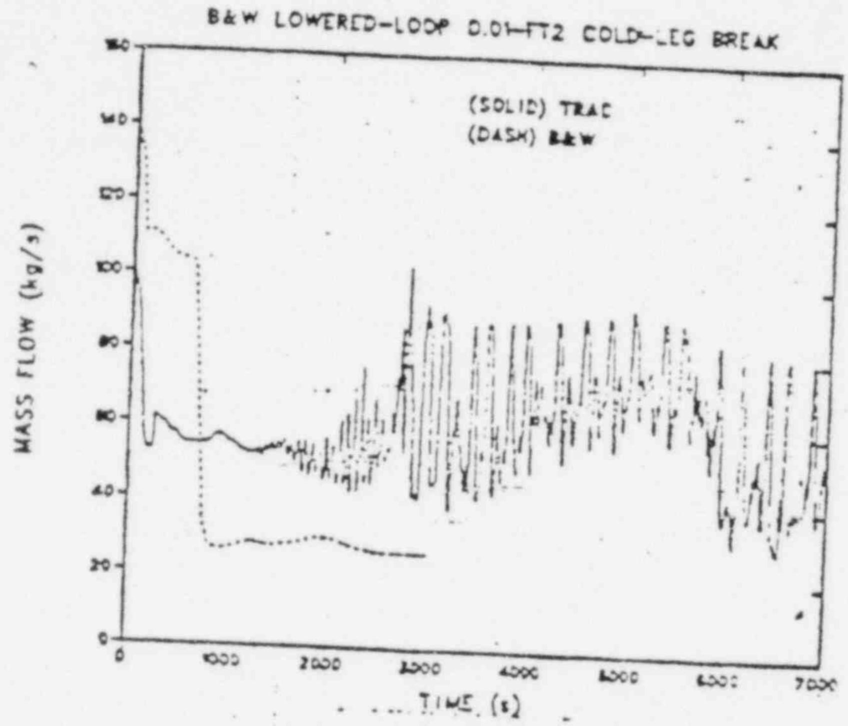


Figure 4
Comparison of TRAC and B&W break flows.

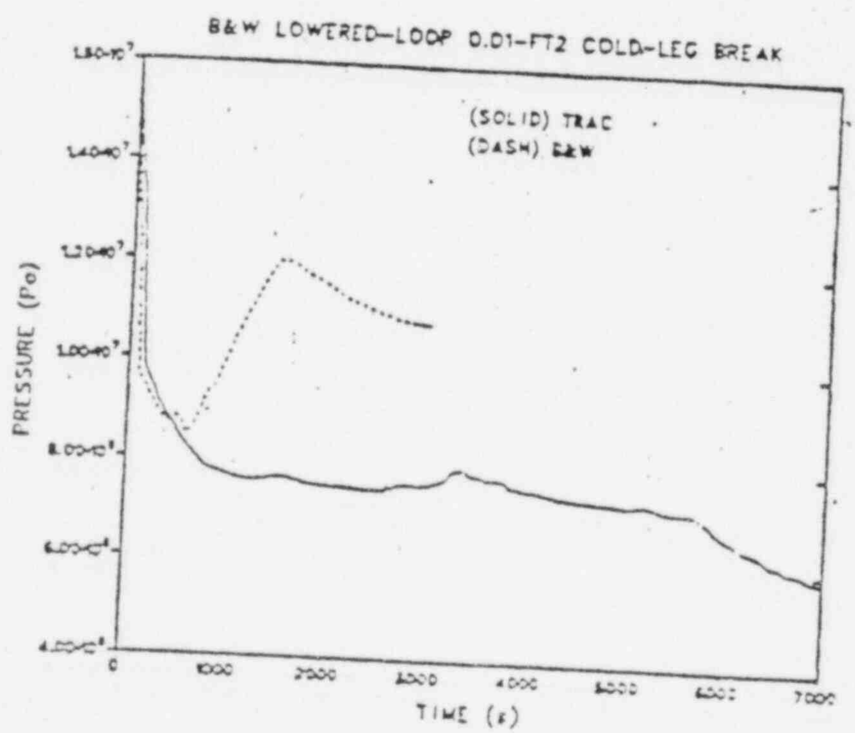


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Figure 5
Comparison of TRAC and B&W primary pressures.

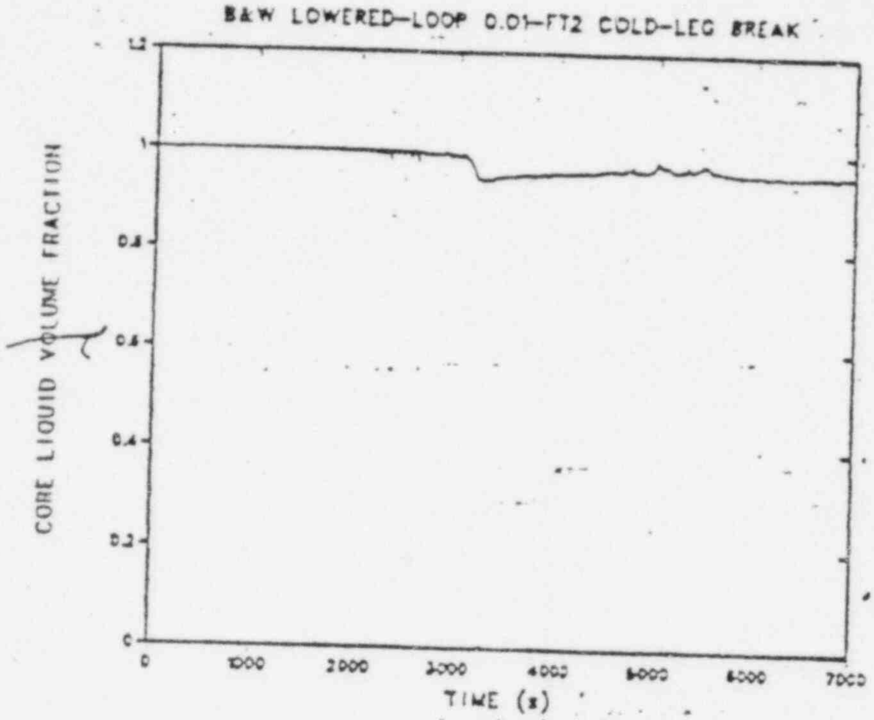

Figure 6
TRAC-calculated core liquid volume fraction.

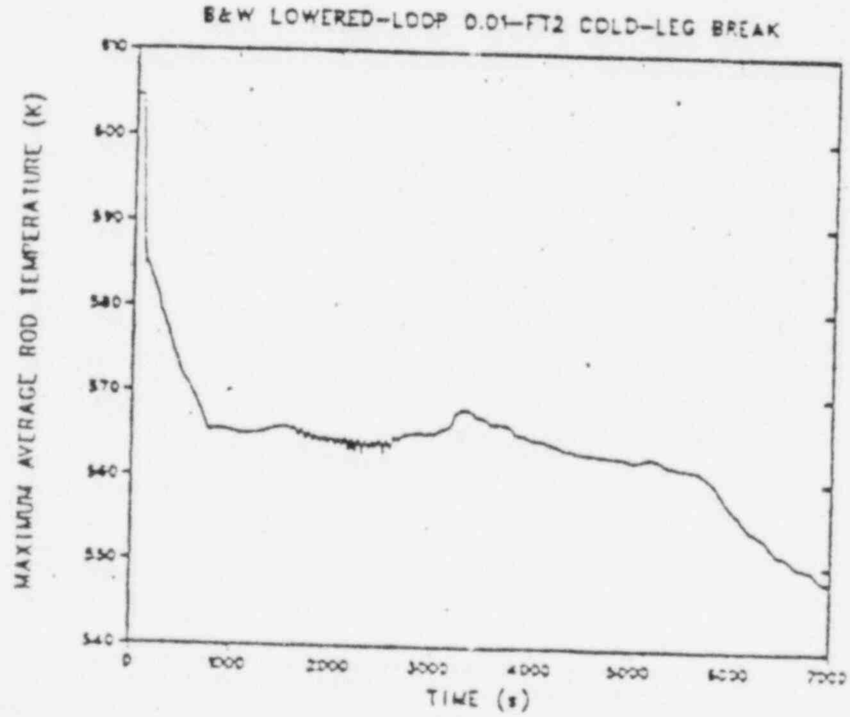

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Figure 7
TRAC-calculated maximum average rod cladding temperature.