NUREG-0565

Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants

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Bulletins and Orders Task Force
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
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ABSTRACT

Slow system depressurization resulting from small break loss-of-coolant accidents (LOCAs) in the reactor coolant system have not, until recently, received detailed analytical study comparable to that devoted to large breaks. Following the TMI-2 accident, the staff had a series of meetings with Babcock & Wilcox (B&W) and the B&W licensees. The staff requested that B&W and the licensees: (1) systematically evaluate plant response for small break loss-of-coolant accidents; (2) address each of the concerns documented in the Michelson report; (3) validate the computer codes used against the TMI-2 accident; (4) extend the break spectrum analysis to very small breaks, giving special consideration to failure of pressurizer valves to close; (5) analyze degraded conditions where AFW is not available; (6) prepare design changes aimed at reducing the probability of loss-of-coolant accidents produced by the failure of a PORV to close; and (7) develop revised emergency procedures for small breaks. This report describes our review of the generic analyses performed by B&W based on the requests stated above.

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1.0 INTRODUCTION

Slow system depressurization resulting from small break loss-of-coolant accidents (LOCAs) in the reactor coolant system have not, until recently, received detailed analytical study comparable to that devoted to large breaks. Typically, the smallest break size analyzed was one that would produce system depressurization without uncovering the core, in accordance with the single failure criterion and other requirements imposed by Appendix K to 10 CFR Part 50. These analyses assumed the capability to remove heat through the steam generators following reactor trip, reactor coolant pump power loss after reactor trip, and the availability of normal plant protective and emergency core cooling systems (ECCS). While the analyses, in general, were sufficient to show compliance with the requirements of 10 CFR 50.46, they failed to provide the necessary information needed for operator action following a small break.[3*]

Recent events in operating plants, including the TMI-2 accident, have shown that: (1) relief or safety valve failures are significantly smaller in size than the smallest break previously analyzed in plant safety evaluations; (2) plant response to valve failures and the associated required operator action are different than for the breaks previously analyzed; (3) auxiliary feedwater systems did not, in some instances, perform as expected; and (4) operator actions in these recent events were not in accordance with the assumptions of the previously approved safety evaluations. For example, in two out of four reported events (Davis-Besse 1, Oconee 3, TMI-2, and a foreign pressurized water reactor built to Westinghouse's design) during which pressurizer pilot-operated relief valves (PORV) were opened by high pressure, and subsequently failed to reclose, the operators terminated high pressure injection (HPI) flow on the basis of indicated high level in the pressurizer during the early part of the transient.

Most of these small break accidents were initiated by main feedwater flow interruption leading to a reactor coolant system pressure rise to the PORV setpoint.

*Indicates the number of the reference listed at the end of this report.

In the Davis-Besse and Oconee incidents, auxiliary feedwater (AFW) was actuated as designed, but did not prevent the reactor coolant pressure rise to the PORV setpoint. Auxiliary feedwater flow at TMI-2 was delayed for eight minutes following loss of main feedwater, because two AFW system valves, being inadvertently closed prior to the loss of main feedwater, were not opened until the error was discovered. Auxiliary feedwater response in the foreign plant was not described in available reports.

In three of the four cases of stuck-open PORVs, reactor coolant pump (RCP) operation was maintained for the duration of the transient. A manual trip of the last RCPs occurred at TMI-2, 101 minutes into the transient. For two of the four cases all steam generators remained operational for the duration of the transient. One steam generator went dry in the Davis-Besse case due to failure of an AFW pump to come up to speed when automatically actuated, and one steam generator was isolated after several hours in the TMI-2 accident due to suspected tube ruptures.

As a result of these events, expecially the TMI-2 accident, reassessment of the failure mechanisms assumed in small break accidents has led the staff and industry to a considerably more detailed study of potential accident scenarios than was conducted previously.

This reassessment has included a realistic evaluation of steam generator heat removal capability, consideration of cases where the conditions for ECCS pump activation were not reached, and cases where RCPs remained operating instead of being tripped. As a basis for this reassessment, the reactor vendors were requested to consider all probable small break scenarios, and their consequences, on a generic basis for their plant designs.

Following the TMI-2 accident, the staff had a series of meetings with B&W and the B&W licensees. The staff requested that B&W and the licensees: (1) systematically evaluate plant response for small break loss-of-coolant accidents; (2) address each of the concerns documented in the Michelson report[117]; (3) validate the computer codes used against the TMI-2 accident; (4) extend the break spectrum analysis to very small breaks, giving special consideration to failure of pressurizer valves to close; (5) analyze degraded conditions where

AFW is not available; (6) prepare design changes aimed at reducing the probability of loss-of-coolant accidents produced by the failure of a PORV to close; and (7) develop revised emergency procedures for small breaks. Responses to these requests by B&W and the licensees are listed in Section 5.0 of this report.

This report describes our review of the generic analyses performed by B&W based on the requests stated above. The work by B&W was referenced by the B&W licensees as partial fulfillment of the requirements of Commission Orders of May 1979.[31-35] In addition, this review supplements the systems analysis for B&W plants that is presented in NUREG-0560.[2] Discussion of our review of IE Bulletins and Commission Orders on B&W Plants is presented in NUREG-0645.[8]

Results of the staff review of this material are summarized in the next section. Further discussion of the findings on each of the principal areas of concern follow the body of this report.

2.0 CONCLUSIONS AND RECOMMENDATIONS

This section of the report contains a summary of the conclusions and recommendations of our evaluation of small break loss-of-coolant accident behavior in the Babcock & Wilcox-designed 177-FA operating plants. Our evaluation was based upon submittals from B&W and the B&W licensees as well as independent staff audit calculations. A listing of the B&W and licensees' submittals is contained in Section 5.0 of this report.

The conclusions and recommendations presented in this section are divided into six major sections: (1) expected frequency of small break LOCAs, (2) small break LOCA analytical model, (3) small break LOCA behavior, (4) staff audit calculations, (5) two-phase natural circulation and accommodation of loss of all feedwater, and (6) Tennessee Valley Authority (TVA) small break concerns. A schedule for implementation of these recommendations is provided in Table 2-1. Table 2-2 serves as a cross-reference listing for the recommendations in this section and the corresponding recommendations found in the body of the text.

2.1 Expected Frequency of Small Break Loss-of-Coolant Accidents

2.1.1 Conclusions

Following the evaluation of the accident at TMI-2, the B&W licensees made changes in the high pressure reactor trip and PORV setpoints and installed anticipatory reactor trips for loss of feedwater and turbine trip. These modifications were made to reduce the likelihood of PORV actuation following anticipated transients which produced an increase in reactor coolant system pressure. With the increase in the PORV lift setpoint, the reduction in the setpoint of the high pressure reactor trip and the addition of the anticipatory reactor trips, the staff concludes that lifting of the PORV is not likely to occur for the loss of feedwater and turbine trip transients. Prior to these modifications, lifting of the PORV was expected in all cases.

Experience gained from the operation of the B&W plants since these modifications were made support this finding. In the first 1.8 reactor-years of operation subsequent to the TMI-2 accident, 12 transients have occurred which would have resulted in PORV openings under the old design. The design changes prevented the PORV from opening in every case. However, the licensees will need to provide analyses which will document that the PORV will not open for all feedwater transients and loss of load transients taking into account the revised setpoints and anticipatory trips. Variations in core physics parameters during the fuel cycle and events which might not actuate the anticipatory reactor trips have not yet been evaluated.

One obvious way to completely eliminate the risk associated with the failure of PORVs is to operate the plants with the block valves closed. However, this could result in an increase in the lift frequency of the pressurizer safety valves. In order to evaluate the desirability or acceptability of this mode of operation, the licensees should provide information on the observed failure rate of safety valves.

2.1.2 Recommendations

a. Provide a system which will assure that the block valve protects against a stuck-open PORV. This system will cause the block valve to close when RCS pressure has decreased to some value below the pressure at which the PORV should have reseated. This system should incorporate an override feature. Each licensee should perform a confirmatory test of the automatic block valve closure system.

b. Most overpressure transients should not result in the PORV opening. Therefore, licensees should document that the PORV will open in less than five percent of all anticipated overpressure transients using the revised setpoints and anticipatory trips for the range of plant conditions which might occur during a fuel cycle.

c. All failures of PORVs to reclose should be reported promptly to the NRC. All challenges should be reported in annual reports.

d. Licensees should submit a report to the NRC which discusses the safety valve failure rate experienced in B&W operating plants.

e. All failures of safety valves to reclose should be reported promptly to the NRC. All challenges should be reported in annual reports.

2.2 Small Break LOCA Analytical Model

2.2.1 Conclusions

The small break analysis methods used by B&W are satisfactory for the purpose of predicting trends in plant behavior following small break LOCAs and for training of reactor operators. However, several concerns regarding the small break model have been identified in Section 4.1.1 of this report. These concerns need to be resolved. In addition, comparison of the total analysis method with available small break integral test data (Semiscale Test S-02-6) has indicated large uncertainties in the calculations. The analysis methods must be revised and verified before they can be considered for NRC approval under 10 CFR 50.46. Predictions of additional small break LOCA tests (Semiscale Test S-07-10B and LOFT Test L3-1) were recently provided to the NRC staff for review and are currently under evaluation (References 89 and 95).

2.2.2 Recommendations

a. The analysis methods used for small break LOCA analysis by B&W should be revised, documented, and submitted for NRC approval.

b. Plant-specific calculations using the NRC approved model for small breaks should be submitted by all licensees to show compliance with 10 CFR 50.46.

c. The effects of core flood tank injection on small break LOCAs should be further investigated to determine the amount of condensation realistically expected and to determine its effect on heatup and core uncovering. The condensation model and modeling procedures (i.e., injection location used in the computer analyses) require further investigation to assure that the effects of CFT injection are biased in a conservative manner. Semiscale and LOFT test data should be used to verify the models.

2.3 Small Break LOCA Behavior

2.3.1 Conclusions

B&W has performed a sufficient spectrum of small break LOCA analyses to identify the anticipated system performance for breaks in this range. These analyses serve as an adequate basis for developing improved operator guidelines for handling small break LOCAs. In addition, these analyses provide an adequate basis for demonstrating that proper operator action, coupled with a combination of heat removal from the primary system through the break, the steam generators, and with the HPI system, assure adequate core cooling. The required operator actions are: (1) tripping the RCPs shortly after initiation of a small break LOCA; (2) termination of HPI in the event of primary system repressurization, provided there is adequate subcooling; and (3) manual restoration of AFW flow to the steam generators in the event of a failure of the AFW system. With regard to tripping of the RCPs, B&W estimates that at least three minutes are available for the operator to perform this action. Reference 6 provides a detailed discussion of the effects of RCP trip during small break LOCA conditions.

In addition to small break LOCAs, if all feedwater flow (both main feedwater and auxiliary feedwater) is lost, heat removal through the steam generators will not occur. In this case, for lowered loop plants, operator action is required to either restore feedwater or to manually initiate HPI flow from both trains. Either action will serve to prevent uncovering the core. Davis-Besse 1 differs from the other B&W 177-FA plants in that it is of the raised loop design and the HPI pumps have a lower shutoff head. At Davis-Besse, HPI operation without steam generator heat removal will not be sufficient to prevent uncovering the core. Therefore, operator action is required to restore feedwater within 20 minutes. The Davis-Besse AFW system is designed to safety-grade criteria.

Calculations for both plant designs indicate that approximately 20 minutes is available for the operators to initiate feedwater flow to prevent uncovering the core. In the case of the lowered loop design, calculations show that operator action to initiate flow from both HPI trains within 20 minutes prevents uncovering the core even if AFW remains unavailable.

Loss of natural circulation would also have the effect of preventing heat removal through the steam generators even if AFW were available. In the event of a continued loss of natural circulation, the plants with the lowered loop design could still provide adequate core cooling in a "feed and bleed" mode which utilized both HPI trains to inject water into the reactor coolant system while bleeding water out the system through the break and/or the PORV. The required operator action at Davis-Besse would be to attempt to restore natural circulation by intermittent operation of a RCP.

2.3.2 Recommendations

a. Tripping of the RCPs in the event of a LOCA is not an ideal solution. The licensees should consider other solutions to the small break problem, for example, an increase in the HPI flow rate. In the interim, until a better solution is found, the RCPs should be tripped automatically in the case of a small break LOCA. The signals designated to initiate the RCP trip should be carefully selected in order to differentiate between a small break LOCA and other events which do not require the RCPs to be tripped.

b. The B&W small break LOCA analyses rely on equipment which has not previously been characterized as part of the reactor protection system or part of the engineered safety features. The equipment used to provide the necessary RCP trip, the pressurizer PORV and PORV block valve, and equipment used to actuate the PORV and PORV block valve fall into this category. The reliability and redundancy of these systems should be reviewed and upgraded, if needed, to comply with the requirement of Section 9 of NUREG-0585,[4] regarding the interaction of non-safety and safety-grade system.

c. Plant simulators used for operator training should offer, as a minimum, the following small break LOCA events:

    (1) continuous depressurization;

    (2) pressure stabilized at a value close to secondary system pressure;

    (3) repressurization;

(4) stuck-open PORV; and

(5) stuck-open letdown valve.

Each of these cases should be simulated with RCPs running as well as tripped. The first three events should be simulated for both cold and hot leg breaks. In addition to the usual assumed single failures in the ECCS and feedwater systems, complete loss of feedwater should also be simulated in conjunction with the above events. It is important that training programs also expose the operators to various kinds of system transients on inadequate core cooling as discussed in Section 2.1.9 of NUREG-0578.[3]

2.4 Staff Audit Calculations

2.4.1 Conclusions

The staff utilized a modified version of RELAP4/MOD7 computer code to audit selected analyses performed by B&W using the CRAFT2 computer code. While some differences exist in the modeling assumptions and the results of the analyses, the staff concludes that the CRAFT2 code can predict the expected plant response to depressurization, pressure stabilization, and repressurization transients. In addition, the CRAFT2 code can predict the loss of natural circulation phenomenon identified in the Michelson Report.[117] Therefore, reasonable assurance is provided that the calculated system response using CRAFT2 may be used as a basis for developing emergency procedures and operator training aimed at detecting and mitigating the consequences of a small break LOCA.

For the range of break sizes evaluated by the staff with RELAP4 (less than 0.07 ft^2, including PORV failures), no uncovering of the core was calculated to occur. The staff analysis of the 0.07 ft^2 break did show some core uncovery, but no significant clad heatup. In this case, CFT injection was calculated to occur and a rapid refill of the reactor vessel prevented any additional heatup. Section 4.1.3 points out the differences between the assumptions used for the B&W analyses compared to the staff analyses.

2.4.2 Recommendations

a. While certain modeling differences and assumptions exist as well as differences in the results of the analyses, it does not alter the staff's primary conclusion of the suitability of the CRAFT2 program to generate the required information upon which operating guidelines are developed. Therefore, it is not recommended, at this time, that the staff perform additional audit analyses to correct these differences.

2.5 Two-Phase Natural Circulation and Accommodation of Loss of All Feedwater

2.5.1 Conclusions

The PWR industry has not provided any data to experimentally verify their analytical predictions of two-phase natural circulation and the transitioning between the various modes of two-phase natural circulation.

The staff finds that the predicted flow through the PORV has a large uncertainty when the flow is two-phase in composition. Section 2.1.2 of NUREG-0578[3] requires that PORV and safety valves be qualified to perform under conditions of both solid water and two-phase flow.

Unless the capability to depressurize the reactor coolant system of Davis-Besse 1, in the case of loss of heat removal through the steam generators, can be demonstrated with due account for input uncertainties and calculational uncertainties, a diverse decay heat removal path, independent of the steam generators, is desirable. Replacement of the high pressure injection pumps with high shutoff head pumps, installation of a high pressure residual heat removal system, or increased PORV capacity are possible solutions to this problem.

2.5.2 Recommendations

a. The NRC TMI-2 Action Plan should consider the need for a diverse decay heat removal path independent of the steam generators for Davis-Besse 1. Consideration of diverse systems should include, for example: (1) increased PORV relieving capacity; (2) higher shutoff head HPI pumps; or (3) installation of a high pressure residual heat removal system. If a system which manually depressurizes the reactor coolant system below the HPI actuation setpoint is selected, the time available to the operator to decide if system depressurization is necessary (i.e., feedwater cannot be restored) should be greater than 20 minutes. The staff believes that times less than 20 minutes do not provide the operator sufficient time in which to fully analyze the situation and could result in incorrect action being taken.

2.6 Tennessee Valley Authority (TVA) Small Break Concerns

2.6.1 Conclusions

A number of concerns related to decay heat removal following a small break LOCA, and other related items, were identified by Mr. C. Michelson of TVA. While these concerns specifically related to the B&W 205-FA design[117] and the Combustion Engineering System 80 design[118], many of the identified TVA concerns have applicability to all PWRs. Therefore, the B&W 177-FA plant design was evaluated with respect to these concerns to assure the design or procedures incorporated resolution of these issues. B&W has reviewed these concerns and provided responses to most of these items. Additional information is still pending on certain items. In particular, postulated modes of two-phase natural circulation play an important role in the B&W analysis. The analysis provides an adequate assessment of these concerns; however, experimental results are not available to support the analytical predictions. Information which has been previously requested by the staff which relates to these concerns, but for which responses have not been received, is included in Table 2-1 (Schedule for Implementing the Bulletins and Orders Task Force Recommendations) for completeness.

2.6.2 Recommendations

a. The various modes of two-phase natural circulation, which are expected to play a significant role in plant response following a small break LOCA, should be demonstrated experimentally. In addition, the staff requires that the licensees provide verification of their analysis models to predict two-phase natural circulation by comparison of the analytical model results to appropriate integral systems tests.

b. Appropriate means, including additional instrumentation, if necessary, should be provided in the control room to facilitate checking whether natural circulation has been established.

c. Licensees should provide an analysis which shows the plant response to a small break which is isolated and the PORV fails-open upon repressurization of the reactor coolant system to the PORV setpoint.

d. Licensees should provide an analysis which shows the plant response to a break in the pressurizer spray line with a failure of the spray isolation valve to close.

e. Licensees should provide confirmatory information to show that HPI and CFT flows during small breaks are insufficient to form water slugs, or if they do, to show that the structural design bases of the primary system includes loads due to:

(1) water slug intertial motion;

(2) water slug impact; and

(3) pressure oscillation due to steam condensation.

*f. Licensees should provide an analysis of the possibility and impact of RCP seal damage and leakage due to loss of seal cooling on loss of offsite power. If damage cannot be precluded, licensees should provide an analysis of the limiting small break LOCA with subsequent RCP seal failure.

g. Licensees shall provide pretest predictions of LOFT Test L3-6 (Reactor Coolant Pumps Running).

*B&W licensees were requested to provide this information previously in a letter to all B&W operating plants from R. W. Reid dated November 21, 1979.

h. With regard to the effects of noncondensible gases during a small break LOCA, the licensees should provide the following information:

(1) The technical justification for omitting the radiolytic decomposition of injected ECC water as a source of noncondensible gas; and

(2) Confirmatory information to verify the predicted condensation heat transfer degradation in the presence of noncondensible gases.

*i. By use of analysis and/or experiment, address the mechanical effects of induced slug flow on steam generator tubes.

*B&W licensees were requested to provide this information previously in a letter to all B&W operating plants from R. W. Reid dated November 21, 1979.

TABLE 2-1

SCHEDULE FOR IMPLEMENTING THE
BULLETINS AND ORDERS TASK FORCE RECOMMENDATIONS

Recommendation Number	Required Action	Schedule Date
Expected Frequency of Small Break LOCAs		
2.1.2.a	Install automatic block valve closure system	07/01/80
	operational test of automatic block valve closure system	NOTE 1
2.1.2.b	Evaluation of PORV opening probability during overpressure transients	05/01/80
2.1.2.c	Reporting of failures and challenges to the PORV	NOTE 2
2.1.2.d	Evaluation of safety valve reliability	06/01/80
2.1.2.e	Reporting of failures and challenges to safety valves	NOTE 2
Small Break LOCA Analytical Model		
2.2.2.a	Analysis methods for SBLOCA (including evaluation of noding)	07/01/80
2.2.2.b	Plant-specific calculations to show compliance with 10 CFR 50.46	01/01/81
2.2.2.c	Evaluation of effects of core flood tank injection on SBLOCAs	07/01/80
Small Break LOCA Behavior		
2.3.2.a	Automatic trip for RCPs during SBLOCA - installed and operational	01/01/81
2.3.2.b	Review and upgrade reliability and redundancy of nonsafety-grade equipment upon which SBLOCA mitigation relies	TMI-2 ACTION PLAN
2.3.2.c	Minimum simulator training requirements for SBLOCAs	01/01/81
Staff Audit Calculations		
2.4.2.a	Additional staff audit calculations of B&W's SBLOCA analyses	NRC ACTION

TABLE 2-1 (Continued)

Recommendation Number	Required Action	Schedule Date
Two-Phase Natural Circulation and Accommodation of Loss of All Feedwater		
2.5.2.a	Consideration of diverse decay heat removal path for Davis-Besse Unit 1	TMI-2 ACTION PLAN
Tennessee Valley Authority (TVA) Small Break Concerns		
2.6.2.a	Experimental verification of two-phase natural circulation	01/01/81
2.6.2.b	Instrumentation to verify natural circulation - installed and operational	04/01/81
2.6.2.c	Analysis of plant response to a small break which is isolated, causing RCS repressurization and subsequent stuck-open PORV	06/01/80
2.6.2.d	Analysis of plant response to a small break in the pressurizer spray line with a stuck-open spray line isolation valve	05/01/80
2.6.2.e	Evaluation of effects of water slugs in piping caused by HPI and CFT flows	05/01/80
*2.6.2.f	Evaluation of RCP seal damage and leakage during a SBLOCA	03/01/80
2.6.2.g	Submit predictions for LOFT Test L3-6 (RCPs running) (schedule for performing test not finalized)	PRE-TEST
2.6.2.h	Submit requested information on the effects of noncondensible gases: (1) justification for omission of radiolytic decomposition as a source of noncondensible gases, and (2) verification of predicted condensation heat transfer degradation	05/01/80
*2.6.2.i	Evaluation of mechanical effects of slug-flow on steam generator tubes	03/01/80

NOTES:

1 - Confirmatory test of automatic block valve closure system shall be performed during the first refueling outage after installation of the system.

2 - PORV and safety valve failures and challenges: failures shall be reported to the NRC promptly by licensees; challenges shall be reported in licensees' annual reports.

3 - TMI-2 Action Plan refers to NUREG-0660 entitled, "Action Plans for Implementing Recommendations of the President's Commission and Other Studies of the TMI-2 Accident," Draft dated 12/10/79.

* - Information requested in recommendations 2.6.2.f and 2.6.2.i were previously requested from the B&W licensees in our letter from R. W. Reid, dated 11/21/79.

TABLE 2-2

CROSS-REFERENCE LISTING OF RECOMMENDATIONS

Recommendation Number From Section 2.0	Recommendation Number In The Body of the Report	Page No.
Expected Frequency of Small Break LOCAs		
2.1.2.a	3.6.a	3-8
2.1.2.b	3.6.b	3-9
2.1.2.c	3.6.c	3-9
2.1.2.d	3.6.d	3-9
2.1.2.e	3.6.e	3-9
Small Break LOCA Analytical Model		
2.2.2.a	4.1.1.5.a	4-11
2.2.2.b	4.1.1.5.b	4-11
2.2.2.c	4.1.1.5.c	4-11
Small Break LOCA Behavior		
2.3.2.a	4.1.2.6.a	4-26
2.3.2.b	4.1.2.6.b	4-27
2.3.2.c	4.1.2.6.c	4-27
Staff Audit Calculations		
2.4.2.a	4.1.3.6.a	4-44
Two-Phase Natural Circulation and Accommodation of Loss of All Feedwater		
2.5.2.a	4.1.4.3.a	4-46
Tennessee Valley Authority (TVA) Small Break Concerns		
2.6.2.a	4.2.2.5.a	4-55
2.6.2.b	4.2.2.5.b	4-55
2.6.2.c	4.2.5.3.a	4-59
2.6.2.d	4.2.5.3.b	4-59
2.6.2.e	4.2.6.5.a	4-61
2.6.2.f	4.2.10.4.a	4-66
2.6.2.g	4.2.10.4.b	4-66
2.6.2.h	4.2.11.4.a	4-72
2.6.2.i	4.2.14.4.a	4-77

3.0 EXPECTED FREQUENCY OF SMALL BREAK LOSS-OF-COOLANT ACCIDENTS

3.1 Introduction

The total expected frequency of a small break LOCA is the sum of the probability of a reactor coolant system (RCS) piping rupture and the failure (to close) of a valve which isolates the RCS. The most likely of these events has been the failure of a PORV.

The median frequency of small RCS ruptures based on pipe breaks is stated to be 10^{-3} per reactor year in WASH-1400.[10] For breaks between 1/2 and 2 inches equivalent diameter, the spread in this probability is from 10^{-2} to 10^{-4} per reactor year.[10*]

The probability of a small LOCA specifically from relief and safety valve failure was not addressed in WASH-1400; however, the frequency of valve failure was stated to be 10^{-2} per challenge.[11] We have evaluated the expected frequency of a stuck-open PORV or safety valve in two ways: (1) based on operating experience, and (2) based on predictions of plant transient response.

We have also reviewed the operating experience of B&W reactors subsequent to the TMI-2 accident, to determine the effect of plant modifications designed to limit the PORV opening frequency. These evaluations are based on a summary of operational data provided by the B&W licensees,[106] B&W analyses of anticipated transients,[62] and a history of PORV failures at B&W plants.[93]

3.2 Data from Operating Reactors

A tabulation of the 10 PORV failures at B&W plants,[93] including six failures prior to initial operation, is given in Table 3-1. This table is based on 31 reactor-years of operation. Based on this data, the resulting frequency of

*Steam generator tubes, whose diameters are also in this range have failed several times in a number of ways. To a first approximation, the tube failure rate appears indistinguishable, perhaps higher, than that for a stuck-open, unisolated PORV.

TABLE 3-1

INSTANCES OF PORV FAILURE TO CLOSE AT B&W PLANTS

Plant	Date	Power Level	Cause of Failure
Oconee 2	08/15/73	Preoperation	Wiring error
Oconee 2	11/16/73	Preoperation	Leaking pilot valve
Oconee 3	06/04/74	Preoperation	Jammed solenoid shaft
*Oconee 3	06/13/75	12%	Corrosion of pilot valve lever pin
Crystal River 3	11/75	Preoperation	Stuck solenoid
ANO-1	1974	Preoperation	Improper venting
*Davis-Besse 1	09/24/77	9%	Bent pilot stem
Davis-Besse 1	10/13/77	Hot standby	Binding pilot stem
TMI-2	03/29/78	Preoperation	Loss of power to solenoid operator
*TMI-2	03/28/79	97%	Unknown

*PORV opening events resulting from transients.

a LOCA brought about by PORV failure is 0.3 events per reactor-year. Most of the recorded PORV failures were corrected during preoperational testing. Four of the events occurred following initial reactor operation, when fission product heating may have caused core damage had the core become uncovered. In the TMI-2 event of March 28, 1979, the core was uncovered and fuel damage occurred. The frequency of PORV failures for B&W plants after startup is 0.13 events per reactor-year. Thus, the probability of a small break LOCA produced by valve failure in the 31 reactor-years of B&W plant operation was considerably higher than the probability of a small break LOCA caused by pipe rupture.

Three of the four operational PORV failures were the result of openings during anticipated transients. One way to reduce LOCA frequency is to reduce the number of transients that cause the PORV to open. This method is discussed below. Another means is to improve the reliability of the PORV and block valve. The staff's recommendations on improving PORV reliability are detailed in NUREG-0578[3] and NUREG-0585.[4]

In 31 reactor-years of plant operation, 162 instances of PORV openings have been recorded for transients which produced reactor trips.[106] WASH-1400[11] states that the probability of a valve not closing if challenged is 10^{-2} per event; consequently, the probability of a small break LOCA would be 0.05 events per reactor year. Since B&W plants operated during this period with the PORV pressure setpoint below the reactor trip setpoints, transients occurred which opened the PORV but did not trip the reactor. Reactors trips are routinely recorded and reported to the NRC; however, a record of all PORV openings could be obtained only by a detailed examination of RCS pressure strip charts. This examination was not performed for any of the B&W plants.

3.3 Evaluation of Anticipated Transients

This section describes analyses by B&W to evaluate the reactor system sensitivity to anticipated transients, to determine if the PORV challenge rate could be reduced.[62] Based on these analyses, the high pressure reactor trip and PORV lift setpoints were modified (see Table 3-2) and anticipatory reactor trips were added for loss of feedwater and turbine trip.

TABLE 3-2

TYPICAL SETPOINTS FOR PORV, SAFETY VALVES, AND OVERPRESSURE TRIP

Setpoints (psig)	Prior to 03/28/79	Present
Operating pressure	2155	2155
PORV actuation	2255	2450
High pressure reactor trip	2355	2300
Safety valve actuation	2500	2500

The CADDS digital computer code is described in B&W's Topical Report BAW-10098P, Rev. 1.[51] The CADDS code describes the primary and secondary system of a B&W PWR. Time-dependent neutron kinetics equations are solved for the fuel pins to provide the heat source for the thermal-hydraulic compution of the RCS behavior.

The CADDS computer program contains several models that facilitate the evaluation of loss of feedwater and turbine trip transients. These include a non-equilibrium pressurizer model and a detailed primary-to-secondary heat demand model derived using TMI-2 accident information. The program is limited, however, in that it represents the primary system as a single loop and can consider only single-phase conditions.

The CADDS computer program has been reviewed and approved by the staff for analysis of anticipated transients without scram (ATWS)[1] and is currently under review for the analysis of other transients. B&W has benchmarked the code by providing a comparison with the RCS pressure, temperature, and pressurizer water level recorded during the TMI-2 accident. These comparisons demonstrate a close agreement with the measured results, in particular, for the initial period when an in-surge into the pressurizer occurred. The comparison for this period indicates that the nonequilibrium pressurizer model in the code is programmed correctly for the prediction of RCS pressure.

Anticipated transients which produce an increase in RCS pressure and might cause the PORV to open include loss of feedwater, loss of external electrical load, turbine trip, loss of condenser, inadvertent closure of main steam isolation valves, and inadvertent moderator boron dilution. For the above anticipated transients, the B&W plant Final Safety Analysis Reports (FSARs) indicate that loss of feedwater and turbine trip (prior to the addition of the anticipatory reactor trips) produce the bounding pressures for the plant. These transients were selected as the basis for a set of sensitivity studies using the CADDS code to evaluate challenges to the PORV.

The CADDS code was used to calculate the maximum reactor system pressure as a function of the high pressure reactor trip setpoint. The analyses were performed two ways: (1) assuming the expected values of core reactivity coefficients, instrument error, pressurizer spray, and heater performance; and (2) assuming conservative values that were used for transient analyses in the plant FSARs. Using the expected values and an assumed high pressure reactor trip setpoint of 2300 psig, the calculated maximum reactor system pressure was always less than 2450 psig. Using the conservative assumptions of the FSAR analyses, the RCS pressure was limited to less than 2500 psig. For this latter case, the PORV opened but the safety valves did not.

Additional CADDS analyses using expected values demonstrated that, by adding an anticipatory reactor trip signal upon loss of feedwater, the maximum system pressure and pressurizer in-surge could be further reduced. The addition of this trip signal prevents the system pressure from exceeding the initial operating pressure of 2155 psig following the loss of feedwater.

The calculations performed by B&W assumed core physics parameters that were typical for the plants' fuel cycle (i.e., best estimate). These calculations demonstrated that the PORV will not open during anticipated transients. For the beginning and end of the plant cycle, the selected values of core physics parameters may no longer be appropriate. In addition, certain transients might occur which would not actuate the anticipatory trip incorporated on operating plants following the TMI-2 accident. Since one of the current post-TMI-2 anticipatory trips is initiated on a loss of the main feedwater pumps, inadvertent closure of an isolation valve in the feedwater lines would not cause an anticipatory reactor trip even though there was an actual loss of feedwater.

Based on the best estimate calculations performed by B&W, we believe that the frequency of PORV challenges has been reduced using the revised PORV and high pressure trip setpoints and assuming the anticipatory trips function as designed. We cannot, however, make a quantitative judgment of the expected frequency at this time. We therefore recommend that additional analyses be performed for anticipated transients which indicate the sensitivity of PORV challenges to: (1) the variation in core physics parameters which may occur in the plant cycle; (2) single failures in mitigating systems; and (3) transients which do not actuate the anticipatory trips. Analytical assumptions should include those in the plant FSARs. The expected frequency of PORV openings based on these more detailed and extensive studies should be determined and reported to the NRC.

3.4 Plant Modifications

Subsequent to the TMI-2 accident, the PORV and high pressure reactor trip settings were modified as given in Table 3-2. Anticipatory reactor trips were also installed for loss of main feedwater and turbine trip. Experience gained in the first 1.8 reactor-years of post-TMI-2 operation indicates that lifting the PORV is now an unlikely event. During this period, 20 reactor trips have been recorded. Twelve of these trips resulted from transients which have previously caused the PORV to open. The design changes prevented the PORV from opening in every case.

We recommend, based on our review, that in order to reduce the frequency of small break LOCAs still further, licensees should design and install a control system which provides an automatic closure of the PORV block valve to prevent a small break LOCA caused by the failure of the PORV to close. One such design would cause the block valve to close after the PORV opens and the RCS pressure decreases to a specific value below the PORV reset pressure. This system would be provided with an override so that pressure relief could be accommodated at lower pressures as necessary. Justification would be required to assure the staff that failure of this system would not decrease overall safety by acting to intensify plant transients and accidents. Licensees should provide a conceptual design for staff review.

3.5 Conclusions

PORVs in PWRs, which fail in the open position, contribute significantly to the probability of a small break LOCA.

With the revised PORV setpoint, high pressure reactor trip setting, and the added anticipatory reactor trip signals, lifting of the PORV is not likely to occur for the loss of feedwater or turbine trip transients. Without these revisions, lifting of the PORV was expected in all cases.

Experience gained on operating plants since the design changes were introduced support this finding. In the first 1.8 reactor-years of operation subsequent to TMI-2, 12 transients occurred which would have resulted in PORV openings under the old design. The design changes prevented the PORV from opening in every case.

Using the FSAR or best-estimate assumptions, lifting of the safety valves will not occur for either the loss of feedwater or turbine trip transients, even if the PORV fails to open at the new setpoint.

As demonstrated by analyses in the FSARs, the PORV and safety valves might open during transients produced by a rod group withdrawal. This transient has never occurred in a B&W operating plant and is considered very unlikely.

B&W has not demonstrated that the PORV will not open for all feedwater and loss of load transients using the revised setpoints and anticipatory trips. Variations in core physics parameters during the fuel cycle and events which might not actuate the anticipatory reactor trips were not evaluated.

An obvious way to completely eliminate the risk associated with the failure of PORVs is to operate the plants with the block valves closed and to build new plants without PORVs. However, this could result in an increase in the lift frequency of safety valves. In order to evaluate the desirability or acceptability of this mode of operation, the licensees should provide information on the observed failure rate of safety valves.

3.6 Recommendations

a. Provide a system which will assure that the block valve protects against a stuck-open PORV. This system will cause the block valve to close when RCS pressure has decreased to some value below the pressure at which the PORV should have reseated. This system should incorporate an override feature. Each licensee should perform a confirmatory test of the automatic block valve closure system.

b. In order to minimize the opening of the PORV, most overpressure transients should not result in the PORV opening. Licensees of B&W plants should document that the PORV will only open in less than five percent of all anticipated overpressure transients using the revised setpoints and anticipatory trips for the range of plant conditions which might occur during a fuel cycle.

c. All failures of PORVs to reclose should be reported promptly to the NRC. All challenges should be reported in annual reports.

d. Licensees should submit a report to the NRC which discusses the safety valve failure rate experienced in B&W operating plants.

e. All failures of safety valves to reclose should be reported promptly to the NRC. All challenges should be reported in annual reports.

4.0 EXPECTED CONSEQUENCES OF SMALL BREAK LOSS-OF-COOLANT ACCIDENTS

4.1 Small Break LOCA Analysis

4.1.1 Analytical Model

4.1.1.1 Introduction and Staff Concerns

B&W has performed a series of analyses for small breaks in the cold legs and in the pressurizer of a lowered loop plant, and in the cold legs of a raised loop plant using their approved small break ECCS model.[50,52] These calculations are beyond the scope normally considered in small break analyses, and the staff has a number of concerns about the applicability of the current models. The staff concerns are as follows:

(1) Following postulated small break loss-of-coolant accidents, a primary mechanism for heat removal is natural circulation. The staff is concerned about the ability of the computer programs to correctly predict the various modes of natural circulation and the interruption of natural circulation, if it occurs. Experimental data for the verification of methods for two-phase natural circulation are currently not available.

(2) The experimental verification of small break analysis methods with systems data is currently limited. The available small break data from the Semiscale Test S-02-6, although containing a number of deficiencies, is the best information now available. The analytical methods used to predict the results of this test do not correctly predict the overall system depressurization rate, and the depressurization rate following core flood tank injection. These are significant parameters in that they affect the injection rate of the core flood tank fluid. Analyses by B&W of Semiscale Test S-07-10B and LOFT Test L3-1 have been submitted by B&W in References 89 and 95 and are currently being evaluated by the staff.

(3) The appropriateness of the pressurizer model for analyses of small breaks at various locations is a potential concern. The equilibrium pressurizer model assumed in the B&W analyses gives somewhat different results from hand calculations assuming non-equilibrium conditions. These modeling

differences may be significant for various postulated breaks. Also, the representation of flooding in the surge line could affect draining of the pressurizer. A flooding check is not made for the surge line in the computer program.

(4) The calculation of core level and core heat transfer are important features of the small break model. Limited experimental data is currently available to justify these models. Although the current comparisons have been satisfactory, the experiments are not challenging to the codes. More experimental data must be obtained for further code verification.

(5) The number of nodes used to represent the primary system for small break LOCA analyses should be sufficiently detailed to model the flashing of hot fluid in various locations. This modeling detail is necessary since the calculated system pressure during the decompression process is controlled by the flashing of the hottest fluid existing at any time in the model. The assumption of thermal equilibrium requires that the fluid combined in a single node be represented by the average fluid properties. If fluid from several adjacent regions is combined in one node, the calculated system process during a portion of the transient may be lower than could occur if the smaller regions of hot fluid flashed and maintained the system at the corresponding saturation pressure. Thus, the modeling detail could have a significant effect on the calculated times for various events such as ECCS actuation.

(6) During the recovery period from a small break LOCA, the thermodynamic equilibrium assumed in fluid control volumes could result in errors in the predicted system pressure. This could, in turn, introduce errors in both the break discharge and safety injection flow. The rate at which the water is refilling the system can effect steam condensation. If the condensation efficiency is less than 100 percent, system pressure would be higher than that predicted.

(7) The discharge rate of two-phase fluid through the PORV and safety valves is an important consideration for some transients. These include postulated stuck-open PORV or safety valves, and primary system depressurization for

very small or zero break LOCAs by opening the PORV if all feedwater is lost. There is a lack of discharge rate data for two-phase fluid at high pressure; most experimental data is for steam at low pressure. If the actual valve flow is lower than assumed in the calculations, primary system depressurization to the high pressure injection setpoint might not occur within the calculated times.

(8) The reduction in the primary system pressure determines the rate and amount of core flood tank water injected. The reflooding of the core is dependent on this flow. As discussed in NUREG-0611,[5] the sensitivity analyses performed demonstrates the influence of core flood tank injection. The amount of steam present at the injection location is the predominant factor which determines the core flood tank mass delivery. The results of an analysis will be influenced by the model and the modeling assumptions used to calculate the core flood tank flow. Additional studies will be required to obtain the necessary information to perform an Appendix K analysis. Additional work in this area is under way at EG&G-Idaho, since more recent experimental data, including LOFT Test L3-1, indicate less depressurization than the Semiscale Test S-02-6.

The B&W small break LOCA analyses were performed shortly after the TMI-2 accident with the models normally used for evaluation model (EM) calculations. These calculations were performed to meet the requirements of the Commission confirmatory orders issued in May 1979 for the operating B&W plants.[31-35] Due to the need to quickly evaluate small break LOCAs and provide improved guidelines for operator training, the staff did not have the opportunity for detailed discussions with B&W regarding these models. Nevertheless, certain features of the B&W model (e.g., system noding) were considered during the staff review of the small break LOCA calculations. These items and a general discussion of the B&W small break model are discussed in the following paragraphs.

4.1.1.2 B&W Small Break LOCA Model

The B&W small break evaluation model incorporates the CRAFT2 code which uses a phase separation model to describe coolant dynamics in the various plant components for small break depressurization transients. The break range analyzed, using the phase separation model, extends from breaks for which the makeup pumps cannot provide adequate coolant inventory to prevent depressurization up to breaks as large as 0.5 ft^2. Above 0.5 ft^2, the large break model using a homogeneous coolant model is used. Comparison calculations between the heterogeneous small break model and the homogeneous large break model at the 0.5 ft^2 transition break size have shown acceptable agreement from staff reviews of such comparisons.

Licensing analyses for small breaks have generally considered a few break sizes in the small break range starting from the larger breaks and progressing to smaller breaks to the point where uncovering the core was not computed to occur. Recent analyses[53] by B&W on lowered loop plants, covering the range from 0.15 ft^2 down to 0.04 ft^2 breaks, indicate that only one break (0.07 ft^2) has an appreciable core uncovery (approximately 1.5 ft) for just over a three-minute interval, resulting in peak clad temperature (PCT) of under 1100°F in the uncovered region. Minor core uncovery (less than one foot) was computed for break sizes on either side of the maximum uncovery break size with correspondingly lower PCTs, thus assuring identification of the peak core uncovery and PCT break size.

Smaller breaks, less than those showing the maintenance of a fully covered core with HPI action alone, have traditionally not been analyzed because they were not expected to demonstrate core uncovery with the combined effects of removing decay heat through the break and through the steam generators (assuming auxiliary feedwater flow to the secondary side).

While performing the analyses presented in Reference 62, B&W found that additional nodal detail was required to more accurately model some components when certain types of transients were imposed. In performing the TMI-2 transient comparison, B&W found that a four-node pressurizer model was needed in place of a single-node model to properly compute the effects on PORV flow of incoming cooler water during the initial primary coolant swelling phase of a loss of feedwater (LOFW) accident. It was also found that for the small break transients involving natural circulation that the development of a steam bubble in the upper 180° bend of the hot leg leading into the steam generator (candy cane) could not be adequately described because of the coarse noding detail for the pipe and steam generator primary side in the existing nodal representation. As a result, a small node was added to each steam generator to more accurately consider the accumulation of steam at these points in the system and to better predict the interruption of natural circulation when the appropriate conditions existed.

The small break nodal model used for some of the B&W analyses was identical to that used for Appendix K applications, and is identified in Table 4-1 by a "K" in the "Model" column, whereas calculations using the revised hot leg model are identified in Table 4-1 by an "S."

As was noted above for the pressurizer and hot leg nodal models, simplified nodal representation of plant components can result in inadequate modeling for transient response. Therefore, B&W should confirm through nodal sensitivity studies the effects of the nodal detail used in other parts of the system model; in particular, the hot regions of the system such as the upper head and upper plenum should be reviewed to properly account for flashing of hot fluid during decompression.

The B&W small break model does not account for the effect of noncondensible gases being released to the primary system. During a small break LOCA, if significant amounts of noncondensible gases accumulate in the primary system, they could pose a problem to decay heat removal by: (1) reducing the heat transfer in the steam generators; and (2) collecting in system high points and interrupting natural circulation flow. In addition, they could alter the expected thermal-hydraulic behavior of the system.

Staff estimates on the effect of noncondensible gases is given in Section 4.2.11 of this report. From these estimates, the staff does not believe that enough noncondensible gases would enter the system to disrupt natural circulation flow or significantly degrade steam generator heat transfer. Evaluations by B&W of the sources and effect of noncondensible gases were submitted to the NRC in Reference 98.

Although the acceptance of B&W's small break model in 1974 did not include the ability of the model to predict natural circulation, we are reasonably confident that the CRAFT2 code can model this phenomenon. In the event that the RCPs are lost and the reactor is shut down, natural circulation provides a means of removing decay heat. Natural circulation results from the heating of the water in the core, which decreases the coolant density causing it to rise. The cooling of the water in the steam generators, which increases the coolant density, causes it to flow downward. The driving head for natural circulation is the difference in static head between the upward flow path through the core and the downward flow path through the steam generators. The natural circulation flow that results is a balance between the flow losses and the driving head. The CRAFT2 code models both coolant density changes and flow losses and should therefore model both single and two-phase natural circulation.

Five natural circulation tests have been conducted in B&W plants with the lowered loop arrangement and two tests were conducted at Davis-Besse, which has the raised loop arrangement. Some of these tests approximated station blackout conditions with the reactor generating decay heat. In addition, an unscheduled loss of load transient occurred at one lowered loop plant and at Davis-Besse. In each case, sufficient natural circulation flow was established so that the reactor coolant was maintained in a subcooled state and core heat was removed through the steam generators. These tests and system transients demonstrated that the operating B&W reactors work well in the natural circulation mode when the coolant system is subcooled. Since two-phase conditions are predicted for some small break LOCAs, the various modes of two-phase natural circulation should be demonstrated experimentally and should be verified analytically. In addition, appropriate control room indication for determining the existence of natural circulation should be provided.

Even though the models used in performing these analyses have produced predictions of small break transients that are considered by the staff to be in reasonable agreement with data obtained from the accident at TMI-2 and the NRC audit calculations, the model changes have not been fully reviewed and assessed, and evaluation of the small break LOCA model for EM calculations is required. Review of TMI-2 accident data has provided experimental verification for the pressurizer noding change; however, sensitivity of the parameters and assumptions used are still needed. Verification of the hot leg nodal model for behavior under natural convection conditions as well as an assessment of parametric uncertainties for this model will also be required.

4.1.1.3 Experimental Verification

In order to fully understand plant response to small reactor coolant system breaks, it is necessary to verify the calculational model used to predict the small break response. Many of the individual models within the overall B&W evaluation model have previously undergone comparisons against experimental data as well as other methods of verification. However, the accident at TMI-2 has emphasized the importance of certain phenomena which are expected to occur during a small break LOCA. From this, the staff has identified certain models, methods, or features of the evaluation codes which require more extensive verification.

In addition to verification of individual models, it is also necessary to assure the proper interaction of these models within the overall systems evaluation model. This is accomplished through verification by comparison to integral systems tests. In the following subsections, both past and planned integral systems tests designed for small break code verification are discussed.

4.1.1.3.1 Semiscale Small Break Test S-02-6

As part of both the United States Standard Problem Program and the International Standard Problem Program,* a test was conducted in November 1975 in the Semiscale Facility to simulate the behavior of a six percent small break in the cold leg. This test was designated S-02-6. Comparisons of pretest predictions by PWR vendors to the measured data showed poor agreement. However, certain test measurements were questionable, including the measured break flow and vessel inventory. A more complete description of test S-02-6 and the comparisons of vendor predictions to the measured data can be found in Reference 120.

As a result of this test, the staff concluded that additional data on small break behavior was needed. Available information indicated that large uncertainties

*The U.S. and International Standard Problem Programs are voluntary participation programs in which participants predict the thermal-hydraulic behavior of agreed upon experimental tests. The tests selected are designed to challenge certain methods, features, and/or models of loss-of-coolant accident analysis computer codes used in reactor design and safety evaluation.

in the calculational models could exist and better and more extensive comparisons of calculational models to experimental data were needed.

4.1.1.3.2 Semiscale Small Break Test S-7-10B

As a result of the possible data inaccuracies in test S-02-6, as well as the inaccurate predictions of the measured data, a second test, designed to provide integral system thermal-hydraulic behavior of a cold leg small break, was performed in the Semiscale Facility by EG&G Idaho, Inc., on January 19, 1979.

B&W, along with other reactor vendors, performed a "blind"* prediction of this small break test. The data from this test were withheld from public disclosure until all of the vendor predictions were received by the staff. Only the initial test conditions were provided for the calculations. B&W submitted their pretest prediction of test S-07-10B on October 9, 1979.[89] EG&G, Idaho will evaluate how well each of the vendor's predictions compare with the test data. From these comparisons, as well as those performed by the staff, the need for each vendor to improve certain models or certain aspects of the models will be determined.

4.1.1.3.3 LOFT Small Break Tests

In addition to verifying vendor small break analysis models with data from Semiscale Test S-07-10B, a specified small break simulation test (LOFT Test L3-1) has been performed in the LOFT Facility. The licensees were requested to prepare pretest predictions of this test's thermal-hydraulic behavior on the same way it was done for Semiscale Test S-07-10B.

The specific conditions and characteristics for the LOFT test were as follows:

(1) Power = 50 MWth;
(2) Heat rate = 16 KW/ft;

*"Blind" predictions are calculations made to predict the expected test behavior. Typically, the test is run and the test results are not released until all of the predictions have been made. The reason the test is run in advance of the predictions is so that the actual initial conditions of the test can be used for the predictions. Once the prédictions have been made, then the test data is released.

(3) Break size = approximately 2.5 percent; and
(4) Break location = cold leg.

The test was conducted on November 20, 1979. Results of B&W's predictions of L3-1 were forwarded to the NRC on December 13, 1979.[95] In addition to Test L3-1, the staff has required in Reference 6 that all PWR vendors and fuel suppliers provide pretest predictions of LOFT Test L3-6, scheduled to be run in the spring or summer of 1980. LOFT Test L3-6 will be a small break loss-of-coolant test in which the RCPs will remain running throughout the test. A more complete description of the basis for this test is provided in Section 4.0 of Reference 6.

4.1.1.3.4 Testing Basis

As part of the evaluation of the extent of additional small break model verification required in light of TMI-2, the staff concluded that model comparisons to integral systems tests in both the Semiscale and LOFT test facilities is necessary.* The primary reason for requiring model comparison to tests in both facilities is the need to facilitate extrapolation to full scale.

At present, there are no test data available on small break behavior in a large scale PWR. The data from TMI-2, while extremely valuable, are not complete for the purpose of integral model verification. For example, the actual secondary heat load was not known, as well as the system inventory or PORV discharge mass flow. In addition, the system behavior was primarily applicable to B&W lowered loop design plants and not Westinghouse or Combustion Engineering plants.

Because of this, integral systems verification of analytical models and methods must depend on scaled tests. In order to extrapolate the scaled data to full size plants, at least two different size scaled tests are needed. This has been the underlying basis for the NRC's integral systems research program and is accomplished with the Semiscale and LOFT facilities.

*The staff did not foreclose the option for the industry to propose alternative integral systems tests for model verification. However, to date, no alternative tests have been proposed by the industry.

The thermal-hydraulic phenomena that are predicted to occur during a small break LOCA are complex. Because of size and design constraints, neither Semiscale nor LOFT can satisfy all of the scaling requirements needed to directly extrapolate test results to large scale PWRs.

The Semiscale facility has some atypicalities associated with it. For example, the Semiscale facility is highly one-dimensional and because of this has a much larger surface-to-volume ratio than a large PWR. Thus, heat losses from the Semiscale system are expected to be greater.

The LOFT facility, on the other hand, has a much shorter core (5.5 ft) than a large PWR (12 ft), although the downcomer height is approximately the same as a PWR. The upper plenum is also disproportionately large.

Thus, while each system has unique atypicalities, they are in many respects complementary and combined, provide a substantially improved data base for model verification than either facility would on its own.

4.1.1.4 Conclusions

The small break analysis methods used by B&W are satisfactory for the purpose of predicting trends in plant behavior following a small LOCA. The results of these analyses have been used to develop improved emergency procedure guidelines and for training of reactor operators. However, several concerns regarding the small break model have been identified in the previous discussion. These concerns should be evaluated before the B&W methods can be considered for NRC approval under the requirements of 10 CFR 50.46.

The comparison of the total analysis method with available small break integral data (Semiscale Test S-02-6) has indicated large uncertainties in the calculations. Accordingly, integral verification of the methods should be included as part of the request for approval under 10 CFR 50.46.

4.1.1.5 Recommendations

a. The analysis methods used for small break LOCA analyses should be revised and documented, and submitted for NRC approval.

b. Plant-specific calculations using the NRC approved model for small breaks should be submitted by licensees to show compliance with 10 CFR 50.46.

c. The effects of core flood tank (CFT) injection on small break LOCAs should be further investigated to determine the amount of condensation realistically expected and to determine its effect on heatup and core uncovering. The condensation model and modeling procedures (i.e., injection location used in the computer analyses) require further investigation to assure that the effects of CFT injection are biased in a conservative manner. Semiscale and LOFT test data should be used to verify the models.

4.1.2 Small Break Behavior

To assess small break response at the lower end of the small break spectrum in B&W plants, a series of analyses was performed to evaluate: the role of the steam generator in small break transients; breaks in the pressurizer; and breaks too small to remove the decay heat input energy to the system coolant inventory.

4.1.2.1 Small Breaks Without Auxiliary Feedwater

All B&W plants use a once-through steam generator (OTSG) design for producing steam for the main turbine-generator. The OTSG design provides a fundamentally different approach to steam generation from that used in other PWRs which utilize recirculation type steam generators. In the OTSG, the heating of feedwater through vaporization and superheat are accomplished in a single vertical pass. The water inventory in a OTSG is considerably less than in a recirculation type steam generator for a comparably sized plant. Thus the heat removal capacity following loss of feedwater is corresponding less and greater reliance on early auxiliary feedwater response is required for adequate decay heat removal. To assess the importance of decay heat removal through the steam generators, B&W

recomputed one (0.07 ft^2) of the blowdown transients analyzed in Reference 53 and analyzed two additional breaks (0.02 ft^2 and 0.01 ft^2) at the low end of the small break spectrum. In these calculations, normal steam generator cooling by the auxiliary feedwater (AFW) system was not available. The high pressure injection (HPI) system was conservatively assumed to be initiated at the lower end of the pressure measurement error band setpoint (1365 psia). RCPs were assumed tripped due to concurrent loss of offsite power with reactor trip. Both HPI trains were assumed to be operable in these calculations (as compared to the single-train assumptions used in previous calculations required by Appendix K of 10 CFR 50.46). The systems, components, and pump trip sequence for these analyses are given as Runs 1 through 3 on Table 4-1.

4.1.2.1.1 <u>0.07 ft^2 Break (Lowered Loop Plant)</u>

The calculations performed for the largest (0.07 ft^2) cold leg break, considered in this study showed that the system would depressurize to the CFT actuation pressure level in 20 minutes. The minimum reactor vessel mixture level would remain just over five feet above the top of the core so that the PCT would remain below the initial 720°F level throughout the transient.

Results of this analysis have shown a considerable improvement in the consequences for this break over that shown in the analysis of Reference 53 (resulting from the action of two HPI trains instead of the single train assumed in the previous analysis). Because of the ability for breaks in the upper end of the small break spectrum to remove coolant energy at a greater rate than decay power input, rapid depressurization to the HPI actuation setpoint level occurred for the break with consequent core cooldown and reactor vessel refill under control of the ECC systems.

4.1.2.1.2 <u>0.02 ft^2 Break (Lowered Loop Plant)</u>

A reduction in the break size to 0.02 ft^2, to assess the effect of a longer depressurization interval to reach the HPI actuation setpoint, resulted in depressurization to the conservative HPI trip level (1365 psia) at 80 seconds. Assuming a 35-second delay for HPI pump activation, HPI flow was started at 115 seconds. The energy loss from the break in this case results in cessation

TABLE 4-1

SUMMARY OF B&W SMALL BREAK ANALYSES

Run #	Break Size	Time of Break	Reactor Trip	Turbine Trip	RCP Trip	HPI Flow	AFW Flow	Model Used	Remarks
1	0.07 ft^2	0	08.0 (-)	08.0	08.0	70	None	K	OTSG Heat Removal Not Required
2	0.02 ft^2	0	29.0 (-)	29.0	29.0	115	None	K	OTSG Heat Removal Not Required
3	0.01 ft^2	0	54.5 (-)	54.5	54.5	1730	1250 (1)	K	Largest Break Requiring OTSG
4	0.01 ft^2	0	50.0 (-)	50.0	50.0	1865	1200 (1)	NRC	
5	0.01 ft^2	0	54.5 (-)	54.5	54.5	1200	None	K	Manual Initiation of HPI
6	0.0073 ft^2	6	12.0 (+)	00.0	None	155 (1)	40	K	Loss of Feedwater - initiated
7	0.0073 ft^2	0	60.0 (-)	60.0	60.0	260 (1)	100	K	
8	0.0073 ft^2	0	60.0 (-)	60.0	60.0	250 (1)	None	S	
9	0.01 ft^2	0	50.0 (-)	50.0	50.0	190	90	S	
10	0.01 ft^2	0	50.0 (-)	50.0	50.0	190	90	S	Asymmetric AFW Flow (1 OTSG)
11	0.005 ft^2	0	100.0 (-)	100.0	100.0	360	140	S	
12	0.01 ft^2	0	54.0 (-)	54.0	54.0	150	94	S	Raised Loop Design

Legend

Break size of 0.0073 ft^2 is the area of the PORV
All analyses done for a break in the cold leg except runs 6, 7 and 8 (PORV)
All times in seconds
RCP - reactor coolant pump
HPI - high pressure injection
AFW - auxiliary feedwater
Model used: K-Analysis performed using previously approved B&W analytical model
S-Analysis performed with revised hot leg nodalization
NRC-NRC staff calculation
(1) - Single train only assumed operational

Initiation Signals and Time Delays

Low Pressure Reactor Trip - 1900 psia with 0.5 second delay (-)
High Pressure Reactor Trip - 2300 psia with 0.5 second delay (+)
HPI Actuation - 1365 psia with 35 second delay
AFW Actuation - coincident with turbine trip with a 40 second delay

of the system pressure decrease at about 1350 psia where an equilibrium energy balance was established between decay heat input, HPI mass flow input, and mass and energy loss out of the break. As a result, level in the reactor vessel stabilized about eight feet above the top of the core with PCT held near the coolant saturation temperature or below the initial 720°F starting temperature. Long-term operation could continue in this state without heat removal through the steam generators and without uncovering the core. Eventual reduction in core energy input would gradually result in a very slow system depressurization to the CFT actuation pressure level with consequent reactor vessel refill.

4.1.2.1.3 0.01 ft^2 (Lowered Loop Plant)

To assess small break behavior when depressurization to the HPI actuation setpoint does not occur and steam generator heat removal is not present, B&W analyzed the 0.01 ft^2 break. This break results in a maximum system coolant inventory loss without depressurization to the HPI conservative pressure trip level. As in previous analyses, RCPs were also assumed to be tripped, so that the only means of heat removal occurred through the break alone.

Left in this state, the system would repressurize to a level at which an equilibrium energy balance between decay heat input and energy removal through the break would occur. For the smaller breaks, this would occur at the pressurizer PORV setpoint. Without coolant inventory replacement, core boiloff down to and below the top of the core will occur. Because this should be a relatively slow process in terms of plant operator response capability, B&W assumed that in such a case operator action to manually initiate AFW would be accomplished within 20 minutes after the break occurred.

Actuation of the AFW system 20 minutes after the 0.01 ft^2 break will terminate the system pressure rise by steam condensation in the steam generator primary side at a pressure just below the pressurizer safety valve setpoint, and will subsequently decrease to the conservative HPI activation setpoint about eight minutes later. The hot legs and the reactor vessel upper plenum regions develop high-quality steam. Natural circulation is interrupted about 19 minutes into the transient, or about one minute before AFW system actuation. Up to this time, some heat removal to the steam generator secondary side has taken place by boiloff of residual secondary water. This energy is then removed through the secondary system safety valves.

At the time of AFW system actuation, the core mixture level is down to five feet above the top of the core (approximately the hot leg nozzle elevation) so that PCT will remain near the coolant saturation temperature (660°F). Following AFW system actuation, vessel level is reduced about one foot by the additional steam separation resulting from depressurization, but stabilizes at about four feet above the core. The subsequent HPI actuation raises this level eight minutes later back to the level of the hot leg nozzles.

Because of variations in AFW system delivery rates between operating B&W plant designs, a conservatively low delivery rate (2/3 of the rate used above) was assumed in an additional analysis for the same 0.01 ft^2 break with AFW system actuation at 20 minutes. The reduced delivery rate resulted in a slower depressurization rate after AFW started and an increase of six minutes in the time required to depressurize the system to the HPI trip level. Upon AFW delivery at 20 minutes, the slow reduction in the reactor vessel mixture level, that occurred between eight to 20 minutes, is arrested near the four-foot elevation above the core. As a result, the six-minute delay to HPI injection began with essentially the same level of coolant covering the core as computed for the higher delivery rate in the original analysis.

4.1.2.1.4 <u>Raised Loop Plant</u>

The analyses of the three breaks discussed above were performed for a lowered loop plant. Breaks in the upper portion of the small break spectrum in the raised loop design will exhibit the same characteristic capability of being able to relieve more primary coolant energy than the core can input by decay heat generation, resulting in rapid depressurization of the system and HPI injection without the need for heat removal through the steam generators. For small breaks that would not result in depressurization without AFW system actuation, the reactor coolant inventory in the loops available for drainage to the reactor vessel when the RCPs are tripped is greater in the raised loop plants. As a result, a longer interval for core coolant boiloff down to the top of the core is available in the raised loop plants so that AFW system operation at 20 minutes followed by HPI would start with a greater vessel coolant inventory.

4.1.2.2 Small Breaks in the Pressurizer

B&W has analyzed: (1) the case of a loss of feedwater flow accident with a subsequent stuck-open PORV without loss of offsite power and one HPI train operational; (2) the case of a stuck-open PORV with loss of offsite power and one HPI train operational; and (3) the case of a stuck-open PORV with loss of offsite power and one HPI train operational, but with no AFW for the duration of the transient. The trip sequence for these cases are given in Table 4-1 as Runs 6 to 8. Discussion of these cases is provided below.

4.1.2.2.1 Loss of Feedwater and Stuck-Open PORV

The LOFW transient postulated for this analysis assumes retention of offsite power and normal automatic AFW actuation 40 seconds after turbine trip. Reactor coolant expansion, from the LOFW, produces a pressurizer level and system pressure increase resulting in the PORV opening in six seconds followed by a high-pressure reactor trip at 12 seconds approximating the initial TMI-2 sequence of events. Steam discharge through the stuck-open PORV results in a depressurization to the HPI activation setpoint in $\sim$ 2.5 minutes, at which point pressurizer inventory loss ceases and reactor coolant from the hot leg enters the pressurizer, completely filling the pressurizer with a two-phase mixture approximately 7.5 minutes into the transient. By this time, feedwater flow coastdown is complete and AFW flow has been established and is being controlled at the 30-inch level in the steam generators. Steam generator secondary pressure stabilized at the secondary safety valve setpoint. In this mode of operation, long-term core decay heat removal can be maintained near the secondary safety valve pressure setpoint. A balance is established between the energy relieved through the stuck-open PORV and through the steam generator secondary and the energy input from the core decay heat. The primary loop ΔT is maintained at 3°F to 4°F by RCP operation, and no core uncovery occurs because the reactor vessel remains filled with a subcooled reactor coolant.

RCPs were assumed to be operative for this analysis to approximate TMI-2 conditions; however, such operations would be terminated following reactor trip for future analysis in accordance with recently established pump trip requirements which have been established on the basis of analysis reported in Reference 6.

4.1.2.2.2 <u>Stuck-Open PORV and Loss of Offsite Power</u>

The transient assumed in this analysis is initiated by a stuck-open PORV. Initial system response is to depressurize to the low-pressure reactor trip setpoint (1900 psia) in one minute due to the steam being relieved through the PORV. Concurrent turbine trip, main feedwater trip, and RCP trip are assumed upon reactor trip. AFW flow is initiated 40 seconds after reactor trip and the 50% control level in the secondary side of the steam generators is not reached until five minutes into the transient. At a little over three minutes into the transient, hot leg pressure reaches saturation, and a two-phase mixture begins flowing under natural convection to the steam generator. Steam separation in the reactor vessel forms a large steam bubble in the reactor vessel upper head. HPI injection is initiated by one HPI train at just over four minutes into the transient, and continued in-flow of two-phase mixture from the hot leg into the pressurizer results in filling of the pressurizer in four minutes. The reduced mass loss through the break, when two-phase flow begins, halts pressure reduction taking place up to that time, and system pressure stabilizes near 1200 psia with core decay heat removal being shared by energy loss through the break and primary hot leg mixture condensing in the steam generator. The condensation action produces a natural circulation action resulting in a loop ΔT of approximately 10°F with a loop mass flow rate on the order of 10%, indicating that the major portion of decay heat removal ($\sim$ three percent rated power) is taking place through the steam generators.

Reactor coolant inventory loss for this accident reduces the vessel mixture level to about 5.5 feet above the top of the core so that the PCT experienced for the transient was also at its greatest level prior to reactor trip.

4.1.2.2.3 Stuck-Open PORV, Loss of Offsite Power, and One HPI Pump (No Auxiliary Feedwater)

The accident assumed for this case is identical to the stuck-open PORV described in the preceding section with the only difference being that no auxiliary feedwater is available at any time in the transient, leaving the single HPI pump and the break as the only means of accomplishing decay heat removal. Calculations for this case out to 30 minutes indicate that the core remained covered; however, continuation of the trends established at that point by hand calculation shows that uncovering of the core would occur at about 40 minutes if no operator action was taken. Actuation of an additional HPI pump, or one AFW pump before that time, would prevent uncovering the core.

All analyses performed in the B&W study included the use of the American Nuclear Society (ANS) decay heat curve times 1.20, required by Appendix K, and an initial power level of 1.02 times rated full power. An additional calculation for this case was performed by B&W that assessed the margin of conservatism introduced by the use of the 1.2 multiplier on the ANS decay heat curve. Using a multiplier of 1.0 for decay heat resulted in the maintenance of the reactor vessel mixture height above the core until an equilibrium between HPI injection, leak rate, and core boiloff was established. At this point in the transient (4700 seconds), the system pressure rise stabilizes near 1600 psia, and continued HPI begins to overtake core boiloff. From this point, system pressure decreases in proportion to decay heat reduction, and system cooldown is established with the core remaining covered during the entire cooldown to the CFT actuation pressure level.

4.1.2.2.4 Stuck-Open Pressurizer Safety Valve

To assess the plant response to a stuck-open pressurizer safety valve, B&W provided a discussion of this transient based on results obtained from the stuck-open PORV analysis and the cold leg break analyses. B&W did not perform an analysis for this specific failure as they considered existing analyses adequate for qualitatively describing the outcome.

For B&W plants, the internal vent valves tend to equalize the reactor coolant pressure following postulated breaks. Consequently, the break flow and system inventory loss would be comparable for the same break areas at various locations. For a break area similar to the safety valve throat area, the primary system would depressurize faster than for a stuck-open PORV, and ultimately be maintained at approximately 1200 psia. However, all of the HPI flow would be delivered to the reactor coolant system, and the resulting consequences would be bounded by the results of a comparable break in a cold leg.

4.1.2.3 Small Breaks with Repressurization

To address the concerns relative to the potential for uncovering the core raised in the Michelson Report,[117] B&W modified their small break model to allow more detailed consideration of the top of the hot leg piping 180° bend entering the steam generator vessel. Because this is the highest point in the system, collection of steam in a large bubble would probably occur at this point if system depressurization to hot leg saturation pressure occured accompanied by loss of power to the RCPs. The natural convection flow that would be established for a system pressurized above the saturation level would be interrupted by a steam bubble in this portion of the loop.

To substantiate results obtained with the original model and to provide a quantitative assessment of plant reponse to very small breaks, B&W analyzed three small break conditions: (1) a 0.01 ft^2 break at the pump discharge with steam generator heat removal; (2) a 0.01 ft^2 break with heat removal through only one steam generator (in the event of loss of one steam generator for heat removal through loss of AFW or steam generator tube rupture); and (3) a 0.005 ft^2 break, to complete coverage of the lower end of the small break spectrum. The system components and pump trip sequence are listed in runs 9 through 11 in Table 4-2. A discussion of system response for these three cases is provided below.

4.1.2.3.1 0.01 ft^2 Break (Lowered Loop Plant)

The primary difference between this analysis and the previous analysis performed with the original model is the nodal change for the hot leg and actuation of AFW 40 seconds after reactor trip, as designed, instead of 20 minutes later. Because of the AFW actuation, HPI actuation occurs in just over three minutes and, combined with the heat removal already taking place through the steam generator, the repressurization computed in the earlier analysis at this point in the transient does not occur, so that depressurization continues toward long-term cooldown. However, at just over five minutes in the intact loop, and just under 11 minutes in the broken loop, natural convection is interrupted by vapor accumulation in the top of the hot legs so that repressurization of the system begins, due to essentially having isolated the steam generators from the heat source. Heat removal by coolant loss through the break is less than the decay heat input so that system pressure rises until primary side steam condensation in the steam generator removes sufficient heat to halt the pressure rise at 1750 psia, about 25 minutes into the transient. Pressure decay from this point will be controlled by decay heat generation. Coolant inventory makeup by the HPI system provides adequate coolant injection to maintain reactor vessel mixture level at least five feet above the active core elevation, so that fuel clad PCT experienced for such an accident remains near saturation following reactor trip. Core cooling out to 81 minutes was computed manually due to the quasi-steady-state heat balance conditions established after the pressure rise was halted. These calculations demonstrated that adequate HPI injection would occur to supply core boiloff and to keep the core covered with a two-phase mixture until long-term cooldown was initiated.

4.1.2.3.2 0.01 ft^2 Break With One Steam Generator Isolated (Lowered Loop Plant)

Consideration of asymmetric steam generator operation following a small break accident was included to address the possibility of failures in the AFW supply or tube ruptures in one steam generator requiring isolation of the unit. The analysis has included the heat removal capacity of the failed unit over the first three minutes of a small break transient to the point of steam generator dryout, at which point heat removal stops.

The initial system pressure reduction over the first three minutes is identical to that obtained with two steam generators. After this time, the reduced heat removal capability with the single steam generator results in a slower depressurization and a delay to about 18 minutes before natural circulation is lost by steam bubble formation in the top of the operating loop hot leg. With both steam generators operational, this phenomena occured at 11 minutes because of the greater system heat removal capability. Primary system repressurization begins at this point and continues as before until a balance is reached between the core decay heat input and heat removal through the break and through primary condensation in the single operating steam generator. This balance was computed to occur at 1600 psia or 43 minutes in the transient. Repressurization to the peak pressure level is slower in this instance, relative to the balanced case with both steam generators, because natural circulation is interrupted about seven minutes later and repressurization occurs later on the core decay heat curve. Following stabilization of the pressure rise by condensation in the operating steam generator, cooling proceeds as in the balanced case. Hand calculations have also shown that the core remains covered throughout the cooldown out to initiation of long-term cooldown.

4.1.2.3.3 0.005 ft^2 Break (Lowered Loop Plant)

This break was analyzed to assess system response to the break considered as the smallest that could result in loss of natural circulation and possibly lead to system repressurization to the PORV setpoint level.

The slower depressurization obtained for this break resulted in a considerable delay in loss of natural circulation after AFW and HPI actuation. AFW was started at 2.3 minutes, and HPI was actuated at six minutes, while natural circulation loss was delayed to just under 44 minutes for this accident. Subsequent repressurization occurred at a very low rate due to reduced decay power generation rate combined with heat removal through the break and through the steam generators. At about 82 minutes into the transient, equilibrium energy removal was established near 1700 psia, and the core remained covered throughout the remainder of the transient to long-term cooldown conditions.

For breaks smaller than 0.005 ft^2, including those breaks within the capacity of the makeup pumps, should the makeup pumps fail, system pressure will decrease to the reactor trip and HPI actuation setpoints prior to the formation of the steam bubble in the hot leg so that the system remains solid prior to HPI actuation. In the analysis for the 0.005 ft^2 break, no flashing occurred in the system until the primary system pressure decreased to 1400 psia. For smaller breaks, the system will depressurize slower yet with flashing occuring at a lower temperature due to reduced decay heat generation rate with time after reactor trip. Since the nominal HPI actuation setpoints on the 177-FA lowered and raised loop plants are all greater than 1500 psig, and the realistic instrument errors are only 50 psi, the ESFAS system will actuate HPI prior to the formation of a bubble in the hot leg that could interrupt natural circulation.

4.1.2.3.4 0.01 ft^2 Break (Raised Loop Plant)

Analysis of the same small break for the raised loop design, to assess natural convection characteristics and decay heat removal mechanisms, was based on a model nodalized in the same manner as the lowered loop plants. The only model differences consisted of component elevations, and minor equipment design differences between plants. The trip sequence for this analysis is shown for run 12 on Table 4-1.

System depressurization to reactor trip, RCP trip, AFW initiation and HPI initiation all occur at very similar times relative to the lowered loop system. Because of the raised hot legs, steam accumulation in the upper part of the hot leg results in loss of natural circulation in about eight minutes after the break, or about three minutes earlier than in the lowered loop system. Repressurization begins earlier from a slightly higher initial pressure, and at a slightly higher rate as a result of the elevated steam generators. At 11.7 minutes, steam generator heat removal from the two-phase mixture has progressed to the point where natural circulation is reestablished in the intact loop, and is lost again in the broken loop after a short recovery. Heat removal by the steam generator rapidly reduces primary pressure to about 1100 psia where equilibrium heat removal conditions are established by the secondary safety valve setpoint.

At about 23 minutes, a second interruption of natural circulation in the intact loop occurs due to a very slow depressurization. Repressurization is reinitiated at a reduced rate due to a lower decay heat generation rate at this stage of the post-reactor trip power decay. The pressure rise continues up to about 1300 psia where steam condensation in the steam generators again provides enough cold leg condensation to reestablish an oscillatory natural circulation, which terminates in three cycles with permanent loss of natural circulation. The primary pressure is once again rapidly reduced to about 1100 psia after the first oscillation where equilibrium heat removal by steam condensation is reestablished under control of the secondary safety valve setpoint.

Reactor vessel two-phase mixture was maintained at the hot leg elevation, or about five feet above the core throughout the transient; therefore, keeping the core covered and assuring PCT was never greater than at initial full-power operation.

Subsequent to the raised loop plant analyses discussed above, some parametric differences were noted between assumptions used in the generic analysis and actual operating conditions in the raised loop plant. These differences consisted of a greater HPI flow (10%-20%) for the head flow curve[115] and a greater PORV relieving capacity (∿ 12%) for steam in the actual plant compared to the assumptions used in the generic analysis. Both differences make the generic analysis assumptions more conservative in that a larger PORV relieving rate results in a more rapid depressurization to the HPI actuation setpoint. The higher HPI flow will result in a more rapid vessel coolant inventory makeup and an earlier long-term cooldown following actuation by system depressurization in a LOCA.

Because of the lower shutoff head of the HPI pumps (1630 psig) at Davis-Besse 1 compared to that available in the lowered loop plants (> 2600 psig), small breaks resulting in system repressurization above the HPI setpoint with permanent loss of AFW is of concern because of the inability to maintain vessel coolant inventory at high pressure with HPI injection. Such a condition in the lowered loop plants was shown to be mitigated, without uncovering the core, by manual initiation of HPI. The HPI systems in lowered loop plants are capable of injecting coolant at pressures higher than the pressurizer safety valve settings.

Analyses by B&W indicate that the operator has 20 minutes to manually initiate AFW for Davis-Besse following a small break LOCA or loss of feedwater event for which the AFW system fails to automatically start. An additional analysis[115A] indicates that for a complete loss of feedwater event without a break, operator action to manually open the PORV, initiate feedwater to the steam generators using the startup feedwater pump and inject water into the reactor coolant system through the use of the makeup system will prevent uncovering the core.

The Davis-Besse small break emergency procedures have been modified to incorporate these actions.

4.1.2.4 B&W Conclusions on Small Break Behavior

The following B&W conclusions appear in Section 6.3 of Reference 62:

"The analyses which have been performed and are documented in Section 6.2 [Reference 62] demonstrate that the B&W 177-FA raised and lowered loop plant ECCS systems will control small breaks and satisfy the criteria of 10 CFR 50.46. Specific conclusions from the analyses documented in Section 6.2 are:

1. In the event of a loss of all feedwater accident and two HPI pumps are available for both the raised and lowered loop 177-FA plants, operator action at 20 minutes to establish auxiliary feedwater assures that the core will not be uncovered for small breaks.

2. Analyses of the PORV stuck-open event show that one HPI pump is sufficient to assure that the core remains covered during the transient.

3. The consequences of breaks in the hot legs of the pressurizer were demonstrated to be bounded by small break analyses performed for breaks in the cold leg pump discharge piping.

4. For very small breaks on either the raised or lowered loop configuration which require the steam generator as a heat removal system, it was shown that system repressurization will occur. However, the establishment of steam condensation by the steam generator as a heat removal mechanism controls the repressurization and assures core covery. This answers the concerns raised by the small break paper written by C. Michelson [Reference 117].

5. It was demonstrated that asymmetric auxiliary feedwater injection provides adequate heat removal to assure that core uncovery does not occur."

4.1.2.5 Staff Conclusions for Small Break Behavior

B&W has performed a sufficient spectrum of small break LOCA analyses to identify the anticipated system performance for breaks in this range. These analyses serve as an adequate basis for developing improved operator guidelines for handling small break LOCAs. In addition, these analyses provide an adequate basis for demonstrating that proper operator action coupled with a combination of heat removal from the primary system through the break, the steam generators and with the HPI system, assure adequate core cooling. The required operator actions are: (1) tripping of the RCPs shortly after initiation of a small break LOCA; (2) termination of HPI in the event of primary system repressurization, provided there is adequate subcooling; and (3) manual restoration of AFW flow to the steam generators in the event of a failure of the AFW system. With regard to tripping of the RCPs, B&W estimates at least three minutes are available for the operator to perform this action. NUREG-0623[7] provides additional information on the staff's assessment of RCP trip during small break LOCAs.

The range of small breaks and the small break accident cooling modes, presented in the analyses, are considered appropriate for demonstrating plant response to potential small break LOCAs which may be encountered during operation of the B&W 177-FA plants. The retention of a two-phase mixture above the top of the core (generally at the hot leg elevation five feet above the top of the core in these analyses) demonstrates that there is an acceptable combination of available cooling modes, automatic system operation and operator action to assure adequate core cooling in the event of breaks at the lower end of the small break spectrum.

In addition to small break LOCAs, if all feedwater flow (both main feedwater and auxiliary feedwater) is lost, there will be no heat removal through the steam generators. In this case, operator action is required to either restore feedwater, or to manually initiate HPI flow from both trains for the lowered loop plants. Either action will serve to prevent uncovering the core. Davis-Besse 1 differs from the other B&W 177-FA plants in that it is of the raised loop design and its HPI pumps have a lower shutoff head. At Davis-Besse, HPI operation without steam generator heat removal will not be sufficient to prevent uncovering the core. Therefore, operator action is required to restore feedwater within 20 minutes. The Davis-Besse AFW system is designed to safety-grade criteria.

Loss of natural circulation due to the formation of a large steam bubble in the volume between the top of the 180° bend in the hot leg piping and the AFW injection level in the steam generators was not predicted using B&W's approved small break model. However, B&W improved its limited nodal detail in this area and added a phase separation node to its model for each steam generator. Using the revised model, B&W could predict the phenomena described in the Michelson Report[117] regarding cyclic recovery and loss of natural circulation. For the lowered loop plant design, B&W was able to predict loss of natural circulation, which was not regained. However, heat removal from the core occurred by means of steam flow to the steam generators where condensation took place in the tubes. For the raised loop design, temporary reestablishment of natural circulation was computed to occur as a result of the longer run of hot leg piping above the core heat source, but the cyclic flow conditions that were seen for the one case analyzed rapidly damped after three cycles due to system heat removal dynamics that could not be readily included in steady-state calculations, such as performed in the Michelson Report.

Loss of natural circulation would also have the effect of preventing heat removal through the steam generators even if AFW were available. In the event of a continued loss of natural circulation, the plants with the lowered loop design could still provide adequate core cooling in a "feed and bleed" mode which utilizes both HPI trains to inject water into the reactor coolant system while bleeding water out the system through the break and/or the PORV. The required operator action at Davis-Besse would be to attempt to restore natural circulation be intermittent operation of a RCP.

4.1.2.6 Staff Recommendations for Small Break Behavior

a. Tripping of the RCPs in the event of a LOCA is not an ideal solution. The licensees should consider other solutions to the small break problem (for example, an increase in the safety injection flow rate). In the meantime, until a better solution is found, the RCPs should be tripped automatically in the event of a small break LOCA. The signals designated to initiate the RCP trip should be carefully selected in order to differentiate between a small break LOCA and other events which do not require RCP trip. (By letters dated December 17 and 18, 1979,[28] the B&W licensees were given preliminary design approval for their proposed automatic RCP trip system.) As specified in NUREG-0623[6], we require that the automatic RCP system be installed and made operational by January 1, 1981.

b. The B&W small break LOCA analyses relied on equipment which has not previously been characterized as part of the reactor protection system or part of the engineered safety features. The equipment used to trip the RCPs, the PORV, the PORV block valve, and the equipment used to automatically actuate the PORV and block valve fall into this category. The reliability and redundancy of these systems should be reviewed and upgraded, if necessary, to comply with the requirements of NUREG-0585,[4] Section 9, regarding the interaction of nonsafety and safety-grade systems. The schedule for NRC action on this matter is being developed under the NRC TMI-2 Action Plan.

c. Plant simulators should offer, as a minimum, the following small break LOCA scenarios:

(1) continuous depressurization;
(2) pressure stabilized at a value close to secondary system pressure;
(3) repressurization;
(4) stuck-open PORV; and
(5) stuck-open letdown valve.

Each of these cases should be simulated with the RCPs running and with the RCPs not running. The first three events listed above should be simulated for breaks in the hot leg and in the cold leg. In addition to the usual single failures assumed in the ECCS, complete loss of feedwater should also be simulated in conjunction with the above events. It is important that training programs also expose the operators to various kinds of system transients on inadequate core cooling as discussed in Section 2.1.9 of NUREG-0578.[3]

4.1.3 Staff Audit Calculations

4.1.3.1 Introduction

Increased attention has been focused on small break LOCA behavior and overall system response to such breaks as a result of the TMI-2 accident. The primary purpose of the audit calculations is to provide reasonable assurance that the calculated system response obtained from the CRAFT2 computer program can be used as a basis for the development of operator guidelines and plant emergency procedures to be used to detect and to mitigate the consequences of a small break LOCA.

Three types of system transients are studied: (1) a depressurization transient; (2) a pressure hangup transient; and (3) a repressurization transient. The corresponding break sizes are a 0.07 ft^2 break, a 0.01 ft^2 break with auxiliary feedwater, and a 0.01 ft^2 break with delayed auxiliary feedwater, respectively. The break location is assumed to be in the pump discharge leg.

The model and assumptions used in the staff audit analyses are described in Section 4.1.3.2. The differences between the RELAP4 and the CRAFT2 analyses which have a significant effect on the analyses are discussed in Section 4.1.3.3, and the results of the analyses are presented in Section 4.1.3.4.

4.1.3.2 Staff Audit Model and Assumptions

The computer code used for the analyses is a preliminary version of the RELAP4/MOD7 computer code (internally designated as RELAP4/MOD7, Version G92). Modifications made to the code for these analyses are:

(1) Addition of a mixture level dependent fill model to allow control of steam generator auxiliary feedwater flow as a function of secondary side mixture level.

(2) Modification to the heat transfer logic to set the local heat slab surface qualities representing the outside surface of the steam generator tubes

below the AFW nozzle to zero during AFW flow. This allows for approximating the heat transfer coefficient of a falling film of water.

(3) Deletion of the time step control due to zero flow crossings at a junction. This change improved the problem running time by a factor of four without adversely affecting the calculation stability.

The RELAP4 nodalization used for these analyses was that developed for the TMI-2 calculations. A nodalization diagram is shown in Figure 4-1. Unlike the TMI-2 analyses, however, a single volume pressurizer with associated surge line volume and a three-volume core are used. Bubble rise was used in all vessel volumes except the upper plenum exit annulus (volume 28), the cold legs between the steam generators and pumps, steam generator secondaries, pressurizer, and accumulator. Slip is used in the downcomer, lower plenum to core, through the core, core to upper plenum, vertical hot leg junctions, steam generator inlet annulus to active tubes, and pressurizer surge line to hot leg. The break is assumed to occur in the "B" loop pump discharge by (node 17) and the critical flow is obtained with the Henry-Fauske/HEM model. A discharge multiplier of 1.0 is assumed and a 0.02 quality transition is used from Henry-Fauske to HEM.

The following boundary and initial conditions are assumed:

(1) Initial power = 2689.0 MWt;

(2) ANS + 20% decay heat;

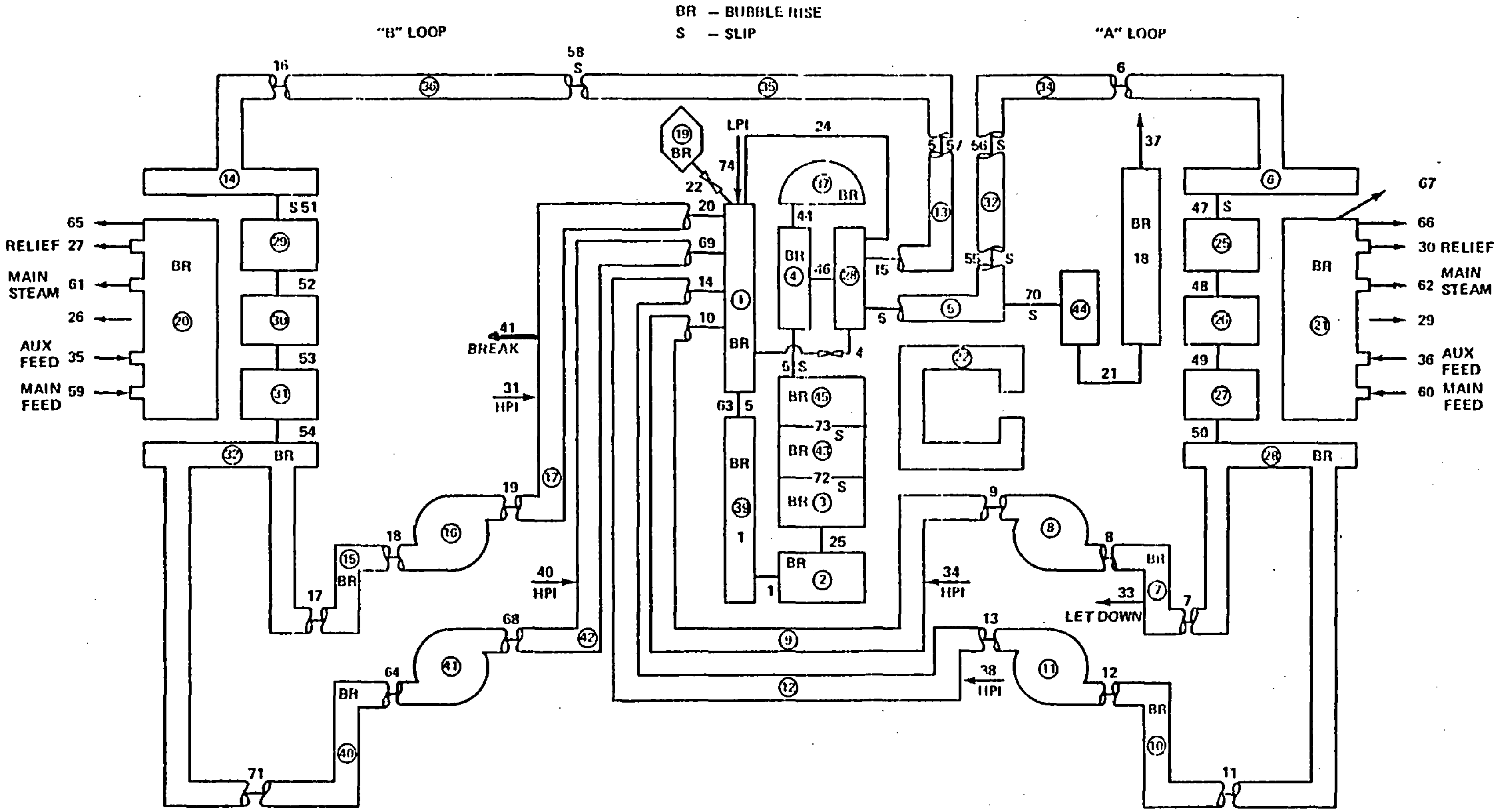
(3) Reactor trip on 1900 psia pressure in the hot leg plus a 0.5 second delay;

(4) Turbine trip and reactor coolant pump trip concurrent with reactor trip;

(5) Steam generator AFW flow initiated 36 seconds after reactor trip;

(6) HPI begins when the hot leg pressure reaches 1365 psia with a 35 second delay. Fifty percent of the flow is injected in the broken cold leg with

RELAP4/MOD7 MODEL FOR A BABCOCK & WILCOX 177 FA LOWER LOOP PLANT



NODALIZATION DIAGRAM FOR STAFF AUDIT CALCULATIONS

FIGURE 4-1

the remainder split evenly to the other loops. Flow is based on a single operating HPI pump; and

(7) AFW flow is controlled to maintain secondary side at 95% on the operating range level indication.

4.1.3.3 Model and Modeling Differences

The analyses performed by B&W which were audited by the staff were for a 177-FA lowered loop plant. While the intent of the audit was to study the same events and model the system and input parameters as closely as possible, some differences between the two sets of analyses do exist.

One difference which has an effect on the small break LOCA analyses is the critical flow model used to obtain the break flow. The staff analyses used the Henry-Fauske/HEM model. B&W uses the Bernoulli equation for subcooled flow and the Moody model for saturated and two-phase flow. The B&W model will result in higher mass flow rates out the break during the subcooled portion of the blowdown, leading to a somewhat faster system depressurization.

The other difference which has a significant influence on the analyses is the modeling of the steam generator heat transfer. As described in Section 4.1.3.2, a number of code modifications were made to RELAP4 in an attempt to provide a better OTSG model and to provide a realistic model of the expected heat transfer during AFW injection. The model used by B&W is a simple model. The heat transfer to the secondary is a mixture height weighted multiplier on a constant heat transfer coefficient, which is determined by steady-state initial conditions. No credit is taken for any other enhanced transfer as a result of the AFW injection.

One other difference which influences the transient response occurred in the 0.07 ft^2 break analysis. In the staff analysis, the AFW was assumed to actuate, while in the B&W analysis no AFW was assumed. However, it will be shown that the depressurization response of the system is similar and that the primary intent of the audit analysis is demonstrated.

4.1.3.4 Analyses Results

The results of the analyses for the 0.07 ft^2 break are shown in Figures 4-2 through 4-5. It is noted that AFW is taken into account for the staff analysis, while no AFW was assumed for the B&W analysis.

As expected, the B&W analysis shows a more rapid depressurization early in the transient (Figure 4-2), a result of the critical flow model used, as shown in Figure 4-3. In the staff analysis depressurization occurs until about 650 seconds when the AFW has refilled the steam generator secondaries to the control level of 20.2 feet. At approximately 950 seconds, the primary and secondary side equilibriate and the heat transfer switches from a secondary to primary mode for the remainder of the transient. This can be seen in Figure 4-4.

At approximately 1730 seconds, CFT actuation occurs. A rapid refilling of the system is expected. The core mixture level is shown in Figure 4-5. While some uncovering of the core is predicted in the staff analysis, no clad heatup occurs, and the peak clad temperature does not exceed the initial steady-state value.

The overall system response to this break demonstrates the ability of CRAFT2 computer program to qualitatively predict a depressurization transient.

The analysis of the 0.01 ft^2 break with AFW is shown in Figures 4-6 and 4-8. The reactor coolant system pressure response is shown in Figure 4-6. The break flow is shown in Figure 4-7, and the secondary side pressure response is shown in Figure 4-8.

The influence of the critical flow model and the OTSG model is seen in Figures 4-7 and 4-8. The interesting comparison is found in the reactor coolant system pressure response, Figure 4-6. In the B&W analyis, the loss of natural circulation is quite pronounced, resulting in the repressurization to 1750 psia at about 1500 seconds. In the staff analysis, natural circulation is lost for a much shorter period, and the pressure tends to hang up while the steam generator pressure is near the safety valve setpoint.

BABCOCK & WILCOX 0.07 FT**2 COLD LEG BREAK

NOTE STAFF WITH AUX FEED – B&W W/O AUX FEED

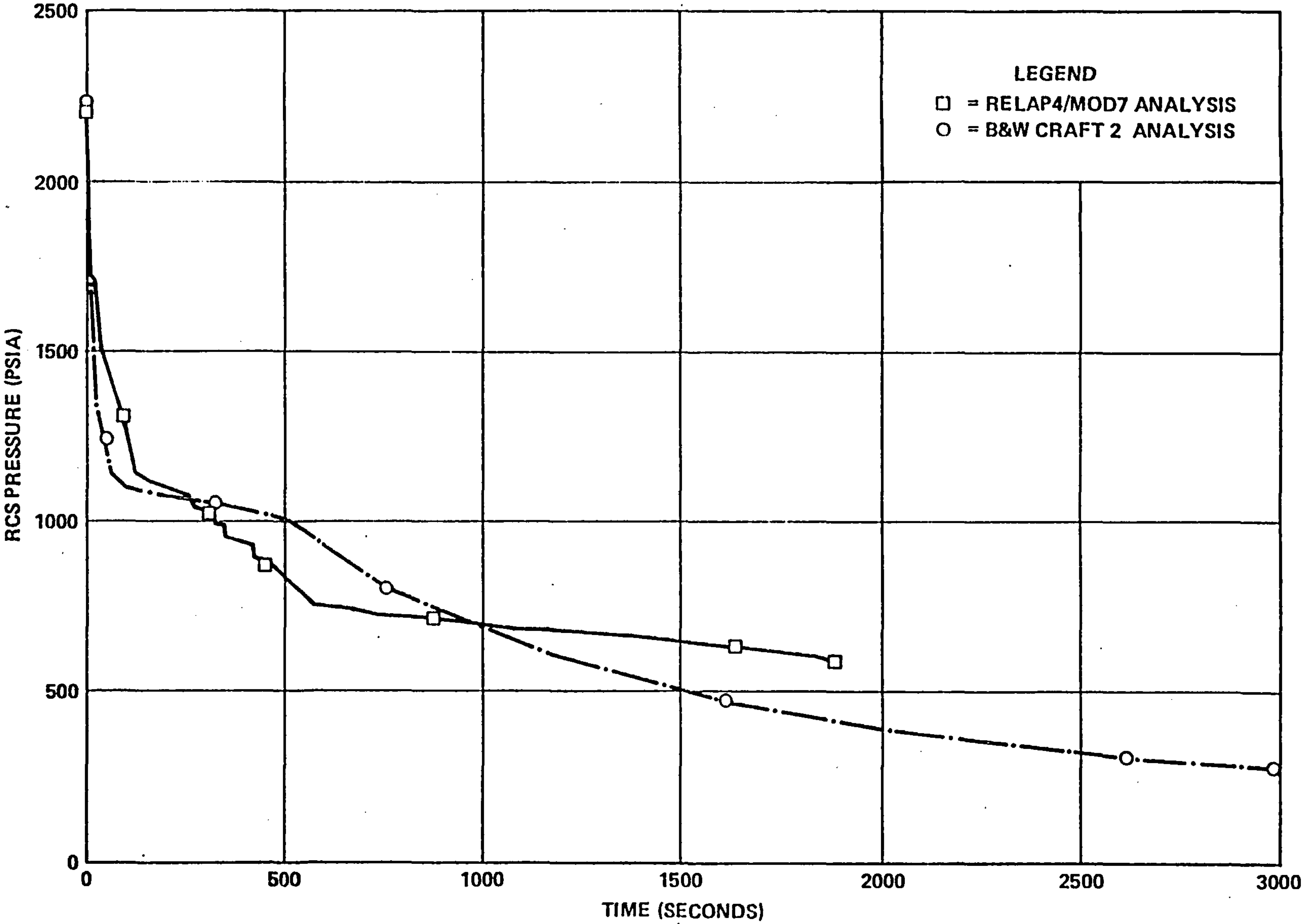


FIGURE 4-2

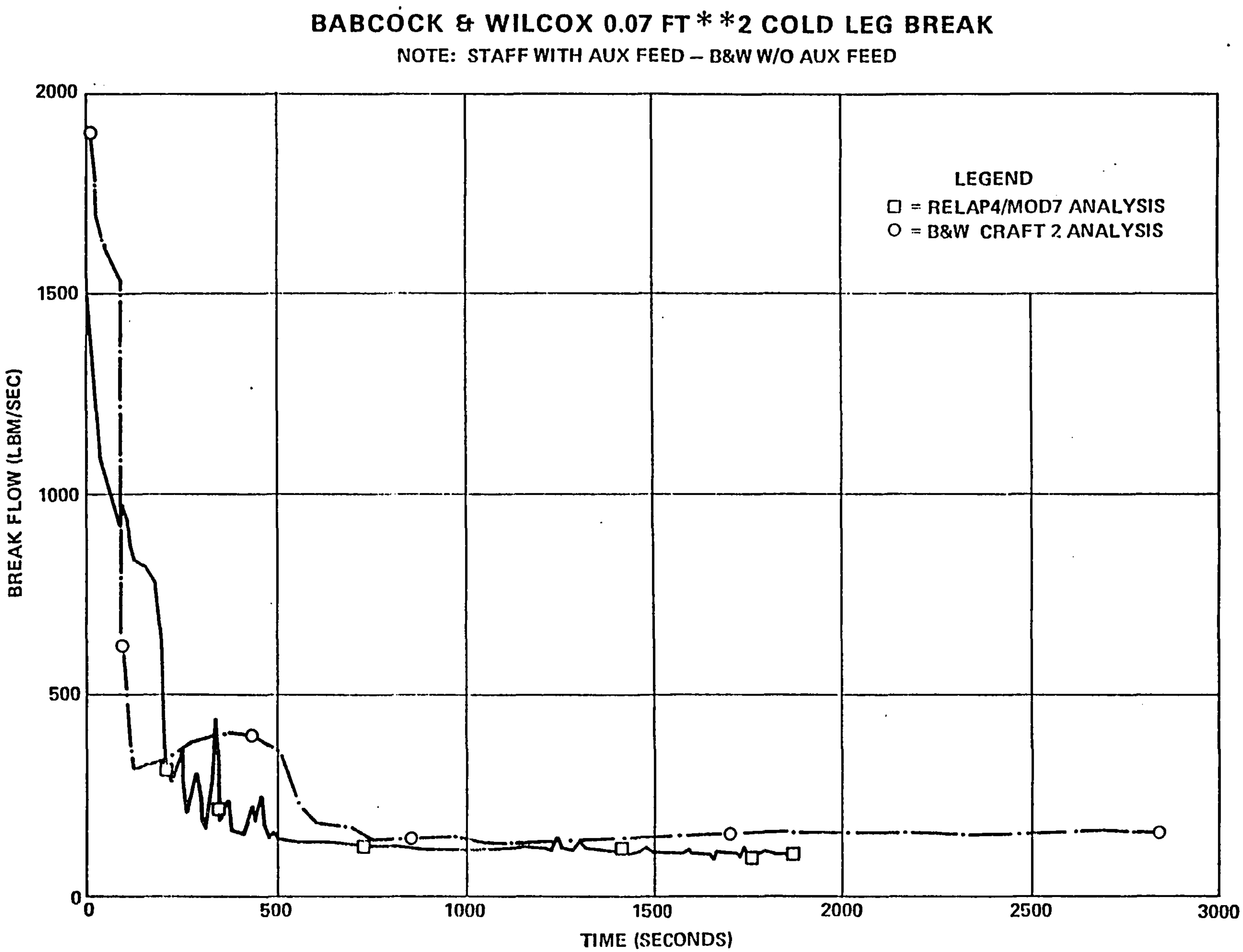

BABCOCK & WILCOX 0.07 FT**2 COLD LEG BREAK
NOTE: STAFF WITH AUX FEED – B&W W/O AUX FEED
LEGEND
□ = RELAP4/MOD7 ANALYSIS
○ = B&W CRAFT 2 ANALYSIS
BREAK FLOW (LBM/SEC)
2000
1500
1000
500
0
0
500
1000
1500
2000
2500
3000
TIME (SECONDS)

FIGURE 4-3

BABCOCK & WILCOX 0.07 FT2 COLD LEG BREAK**

NOTE: STAFF WITH AUX FEED – B&W W/O AUX FEED

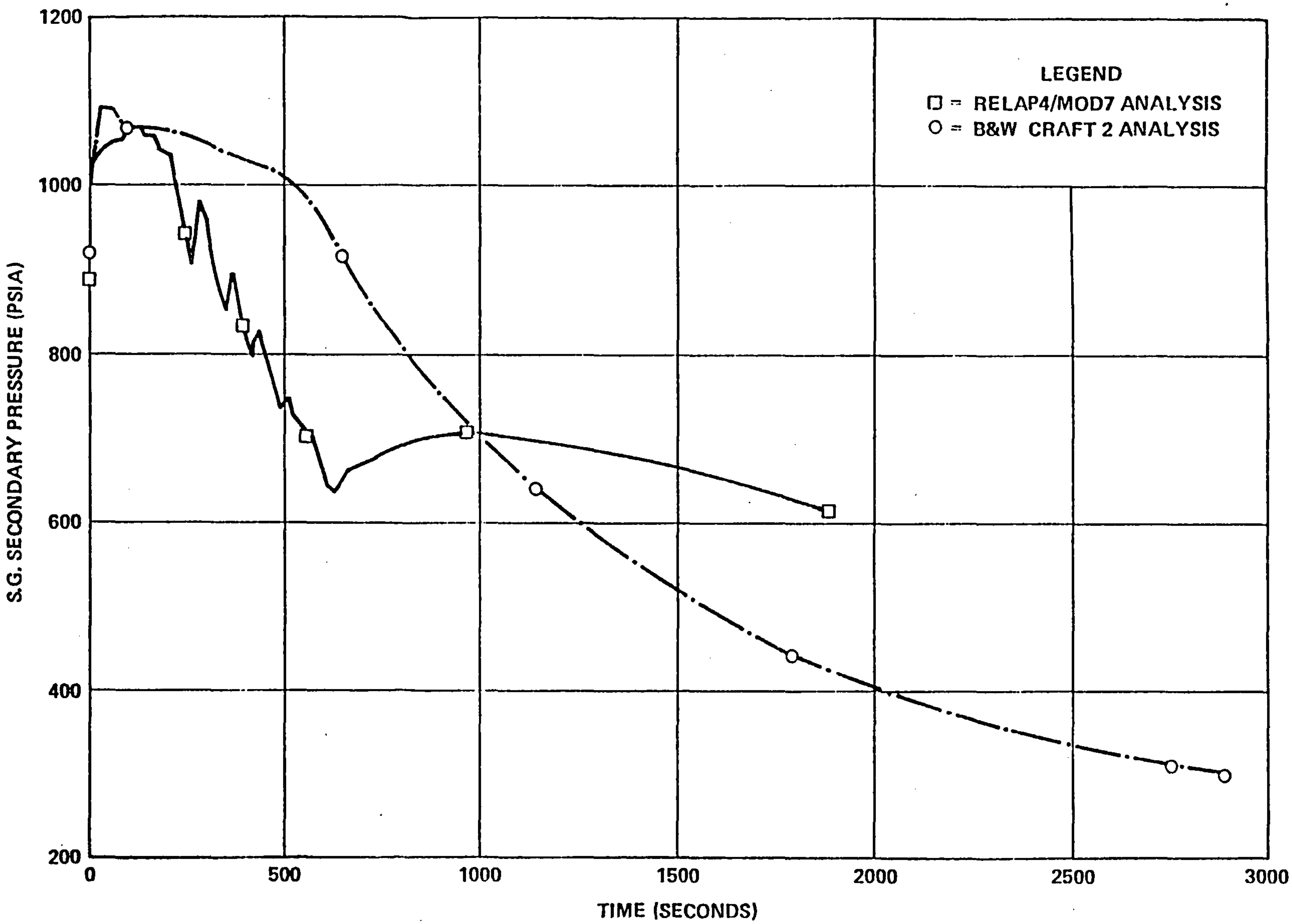


FIGURE 4-4

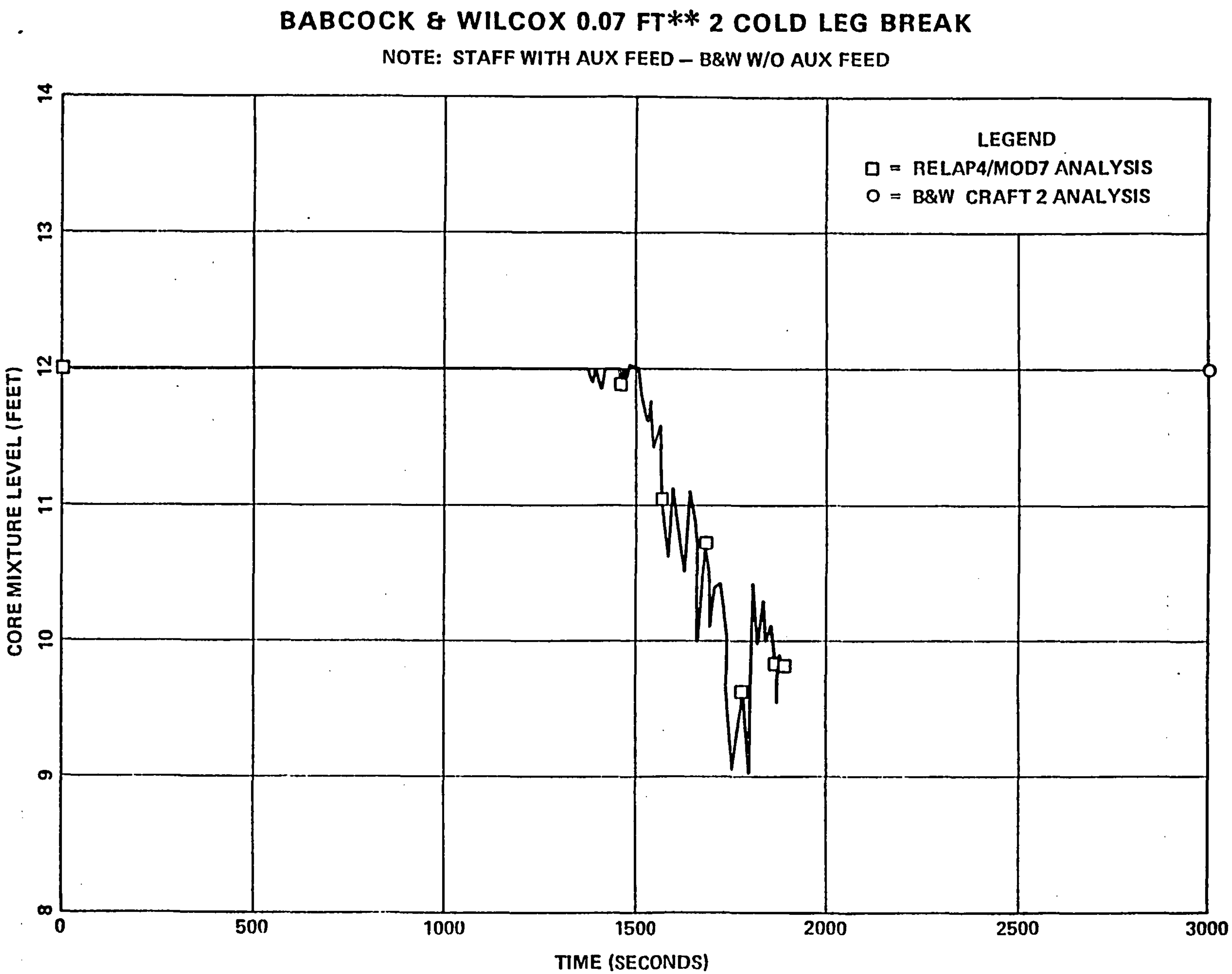

BABCOCK & WILCOX 0.07 FT** 2 COLD LEG BREAK
NOTE: STAFF WITH AUX FEED – B&W W/O AUX FEED
LEGEND
□ = RELAP4/MOD7 ANALYSIS
O = B&W CRAFT 2 ANALYSIS
CORE MIXTURE LEVEL (FEET)
14
13
12
11
10
9
8
0
500
1000
1500
2000
2500
3000
TIME (SECONDS)

FIGURE 4-5

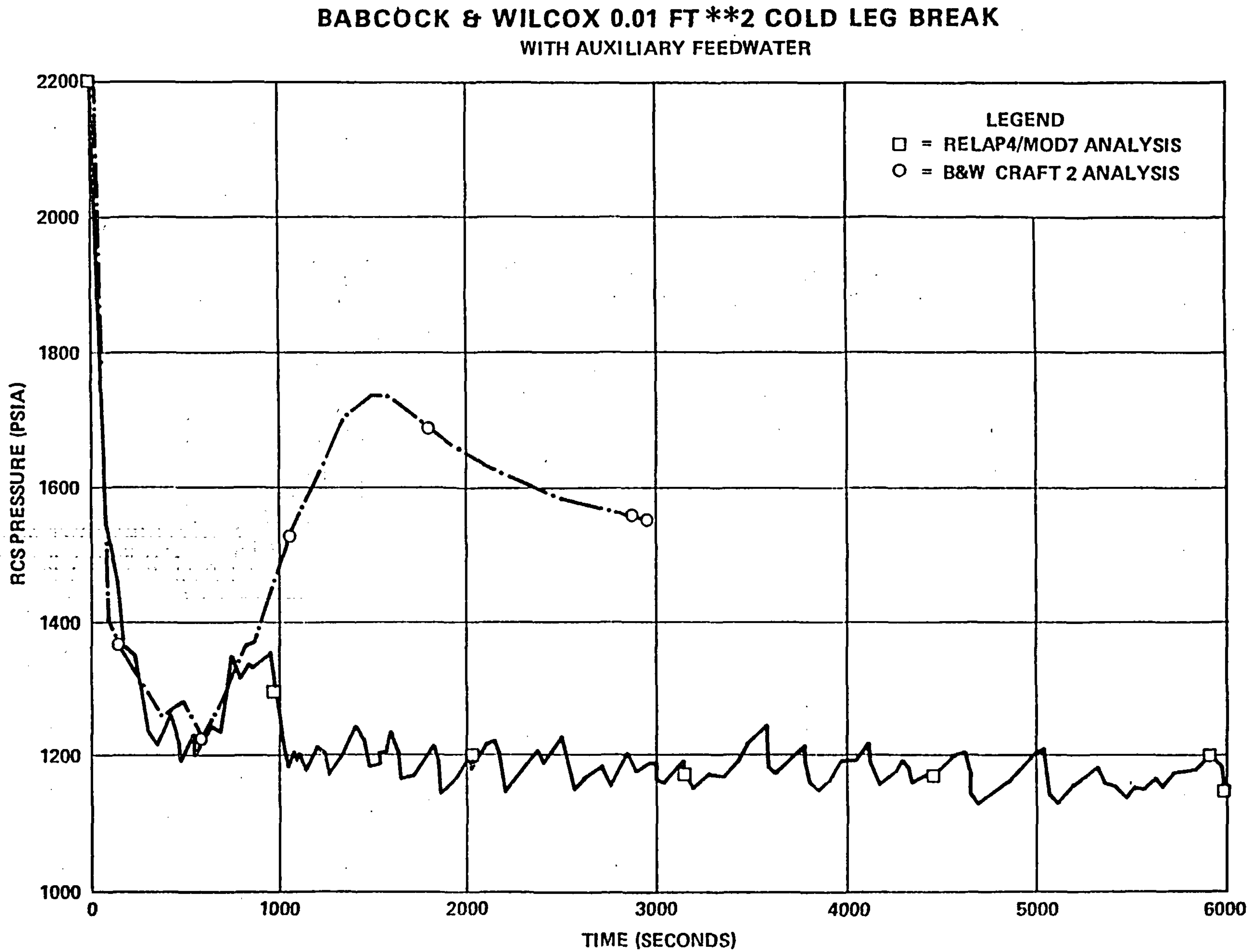

BABCOCK & WILCOX 0.01 FT **2 COLD LEG BREAK
WITH AUXILIARY FEEDWATER
LEGEND
□ = RELAP4/MOD7 ANALYSIS
○ = B&W CRAFT 2 ANALYSIS
RCS PRESSURE (PSIA)
TIME (SECONDS)
2200
2000
1800
1600
1400
1200
1000
0
1000
2000
3000
4000
5000
6000

FIGURE 4-6

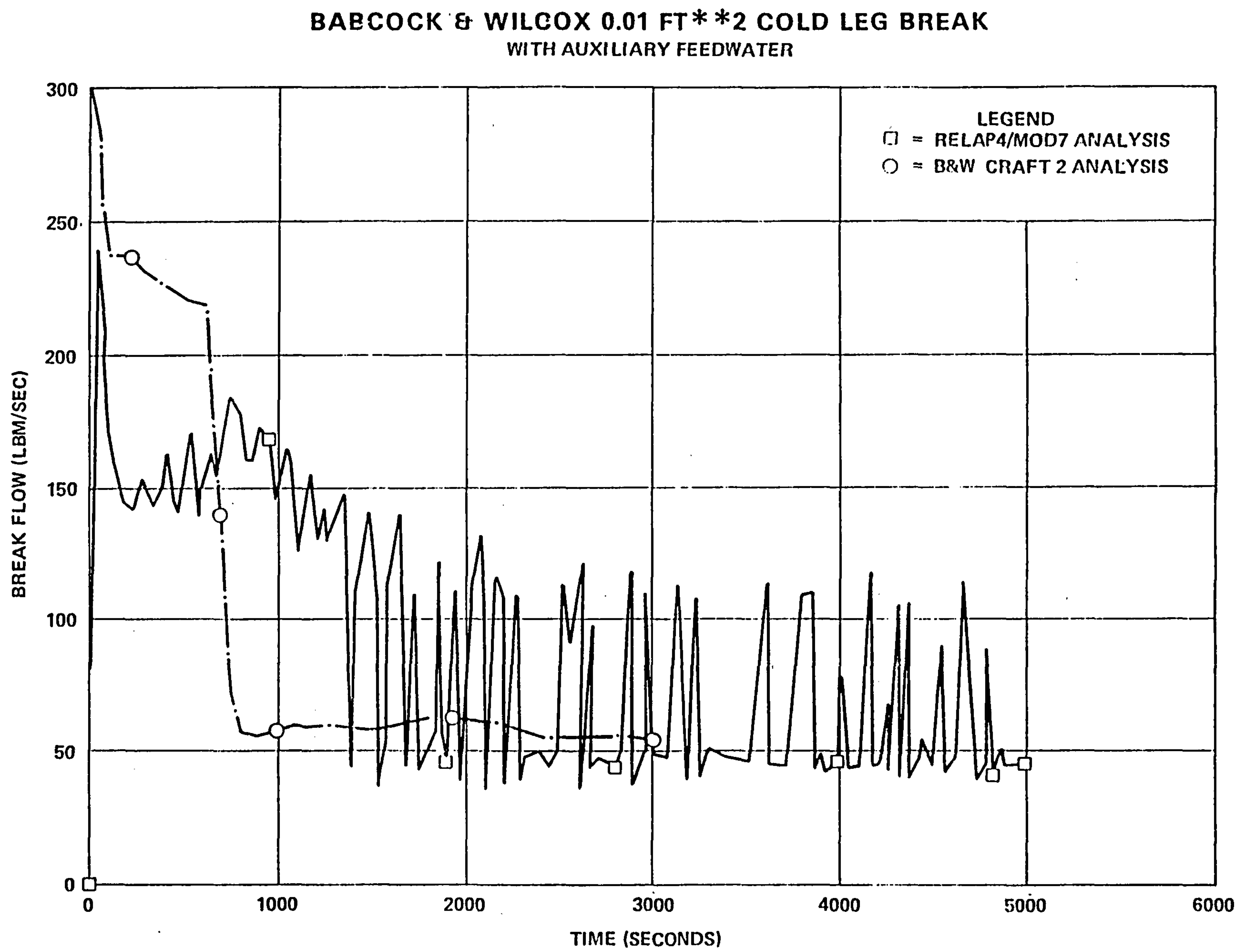

BABCOCK & WILCOX 0.01 FT**2 COLD LEG BREAK
WITH AUXILIARY FEEDWATER
LEGEND
□ = RELAP4/MOD7 ANALYSIS
○ = B&W CRAFT 2 ANALYSIS
BREAK FLOW (LBM/SEC)
TIME (SECONDS)
300
250
200
150
100
50
0
0
1000
2000
3000
4000
5000
6000

FIGURE 4-7

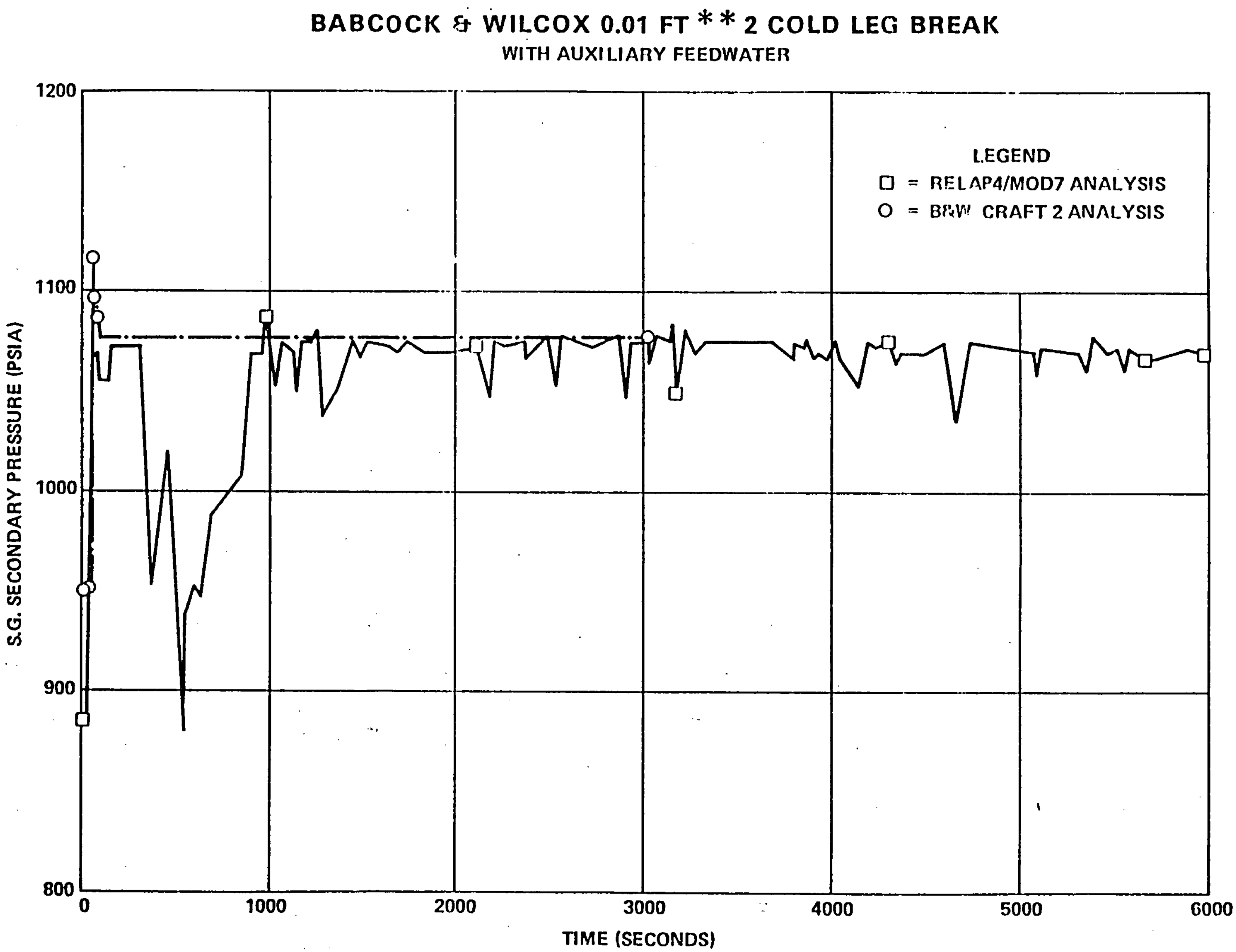

BABCOCK & WILCOX 0.01 FT ** 2 COLD LEG BREAK
WITH AUXILIARY FEEDWATER
LEGEND
□ = RELAP4/MOD7 ANALYSIS
O = B&W CRAFT 2 ANALYSIS
S.G. SECONDARY PRESSURE (PSIA)
TIME (SECONDS)
1200
1100
1000
900
800
0
1000
2000
3000
4000
5000
6000

FIGURE 4-8

The differences in the transient responses are attributed to the differences in two models, RELAP4 versus CRAFT2. While this analysis, as performed by B&W, does not show the pressure hangup response, the B&W analysis of a 0.02 ft^2 break with no AFW does show this response.

The analyses performed by B&W for intermediate size small breaks, 0.01 ft^2 and 0.02 ft^2, demonstrate the capability of the CRAFT2 computer program to qualitatively predict a pressure hangup response, and to predict loss of natural circulation, the phenomenon identified by Michelson.[117]

The results of the analysis of the 0.01 ft^2 break, with auxiliary feedwater delayed for 20 minutes, is shown in Figures 4-9 through 4-11. The staff analysis differs from that described in Section 4.1.3.2. The nodal model employed is somewhat different than shown in Figure 4-1 and both HPI trains are assumed to be available. The delayed AFW assumption reduces the influence of the secondary heat transfer model. Again, the differences between the critical flow models is seen in Figure 4-10. The comparison of the staff and the B&W analyses demonstrates the ability of the CRAFT2 computer program to qualitatively predict the repressurization response.

4.1.3.5 Conclusions on Staff Audit Calculations

The calculated system response to the break sizes analyzed demonstrate the ability of the CRAFT2 computer program to predict the expected behavior of depressurization, pressure hangup, and repressurization transients. In addition, CRAFT2 can predict the loss of natural circulation phenomenon identified by Michelson.[117] Therefore, reasonable assurance is provided that the calculated system response using CRAFT2 may be used as a base in the development of operator training and plant emergency procedures to be used to detect and to mitigate the consequences of a small break LOCA.

For the range of break sizes evaluated by the staff with RELAP4 (less than 0.07 ft^2, including PORV failures), no uncovering of the core was calculated to occur. The staff analysis of the 0.07 ft^2 break did show some uncovery, but no significant clad heatup. In this case, CFT injection was calculated to occur and a rapid refill of the reactor vessel prevented any additional heatup.

BABCOCK & WILCOX 0.01 FT**2 COLD LEG BREAK

WITHOUT AUXILIARY FEEDWATER

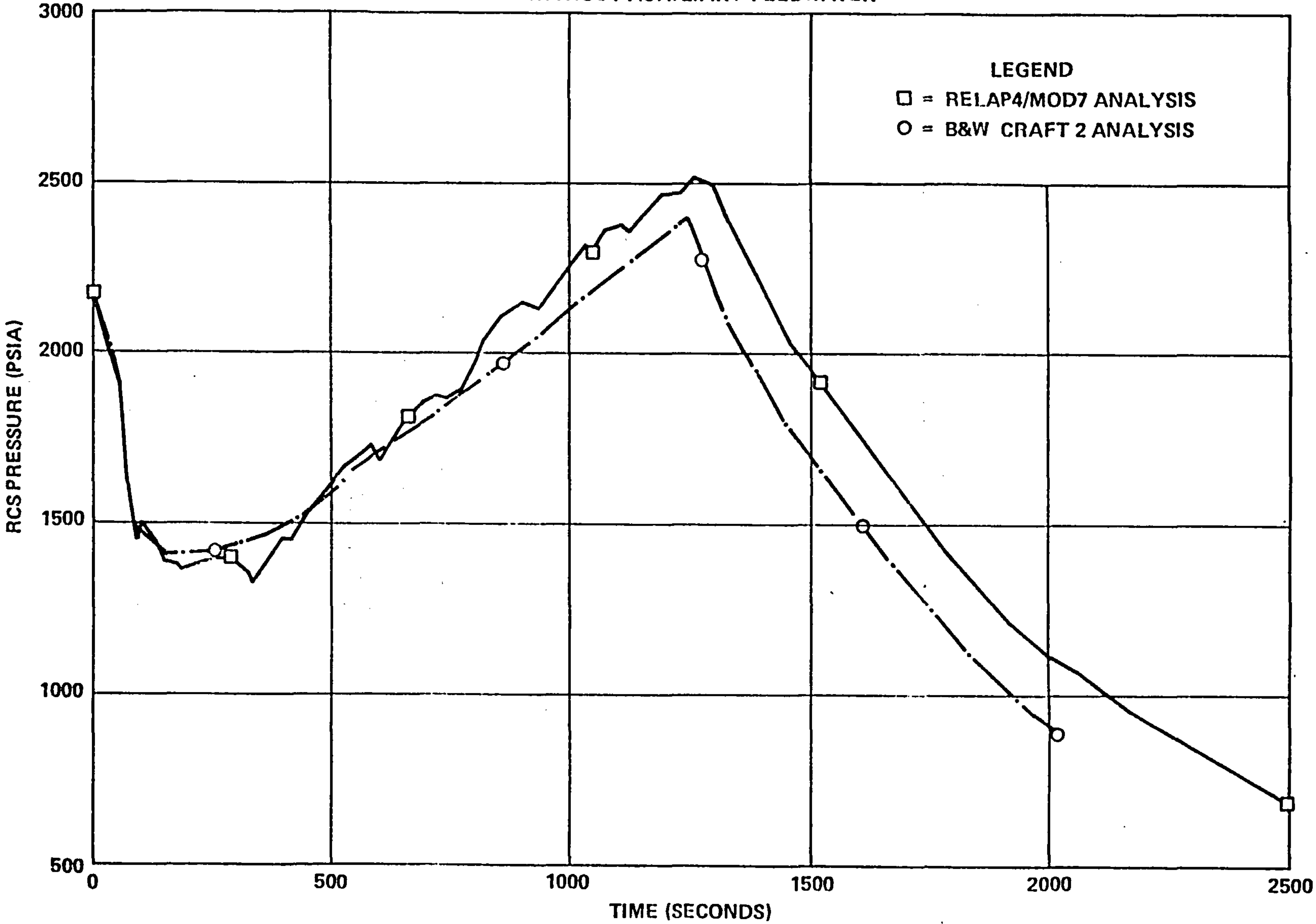


FIGURE 4-9

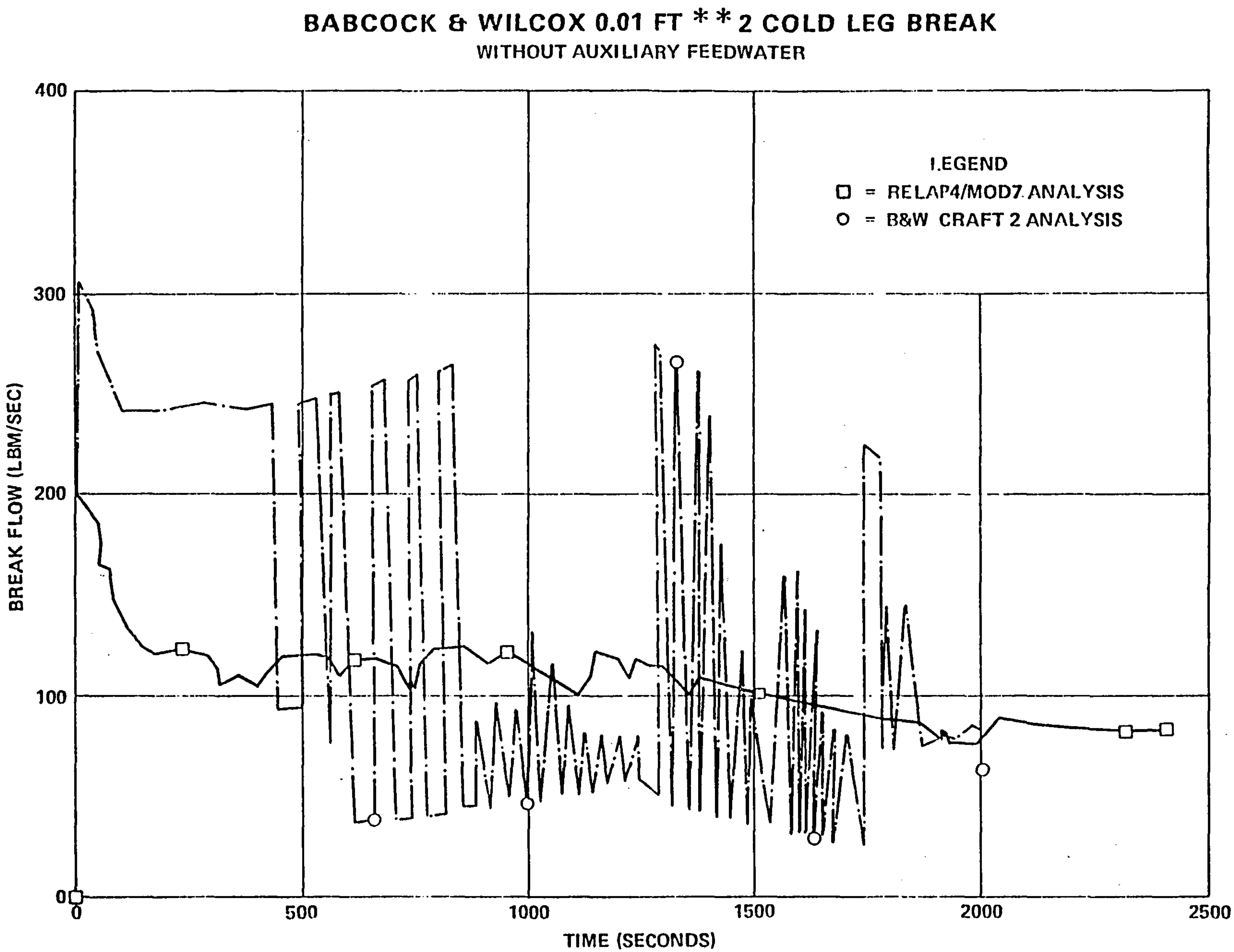

BABCOCK & WILCOX 0.01 FT * * 2 COLD LEG BREAK
WITHOUT AUXILIARY FEEDWATER
LEGEND
□ = RELAP4/MOD7 ANALYSIS
○ = B&W CRAFT 2 ANALYSIS
BREAK FLOW (LBM/SEC)
400
300
200
100
0
0
500
1000
1500
2000
2500
TIME (SECONDS)

FIGURE 4-10

BABCOCK & WILCOX 0.01 FT**2 COLD LEG BREAK
WITHOUT AUXILIARY FEEDWATER

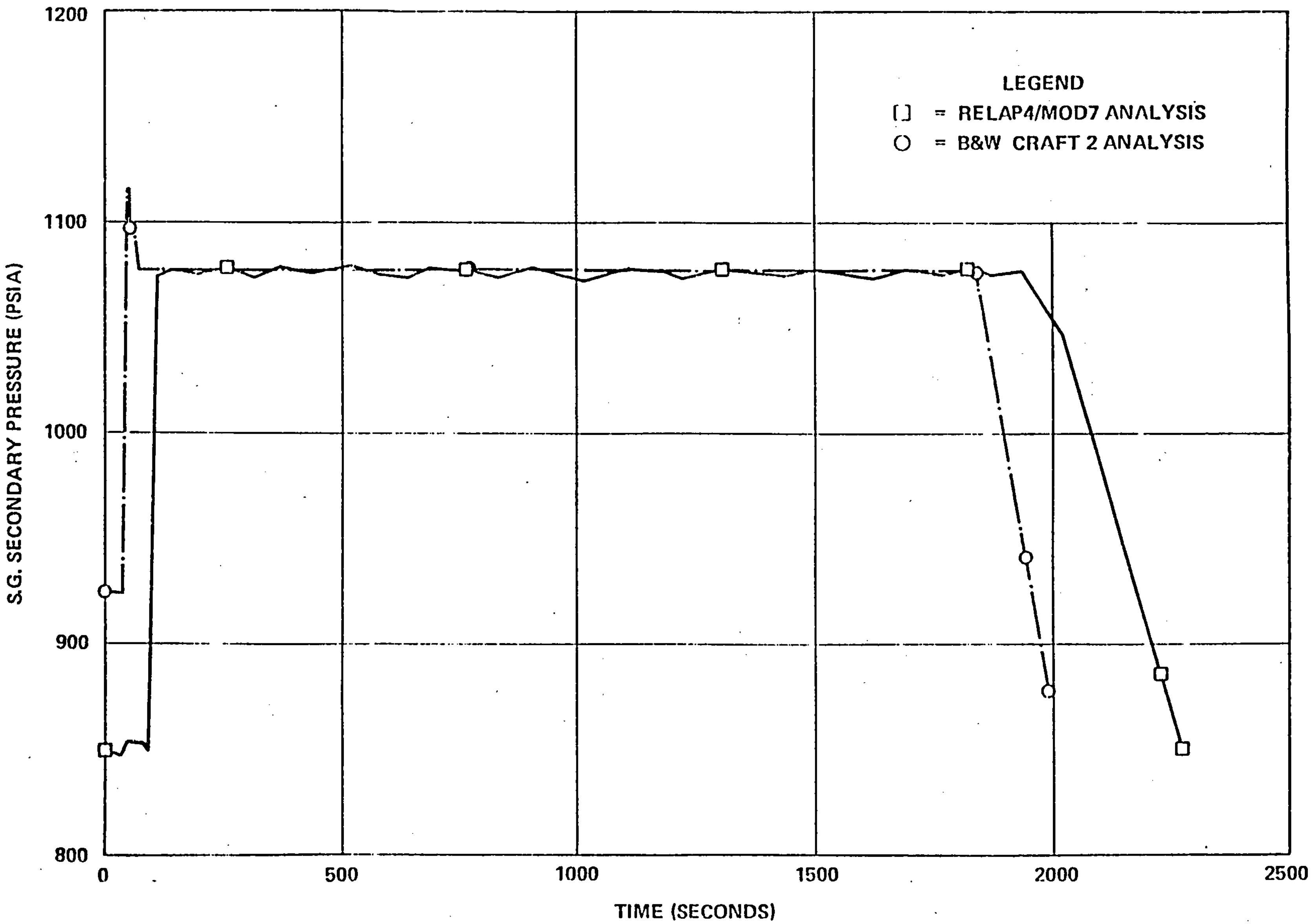


FIGURE 4-11

The AFW availability assumptions used by the staff and by B&W are different for the 0.07 ft^2 break.

4.1.3.6 Recommendations on Staff Audit Calculations

a. While certain modeling differences and assumptions exist between the staff and B&W, such as the critical flow models used, the AFW availability assumptions used for the 0.07 ft^2 break, and the OTSG model, as well as differences in the results of the analyses, it does not alter the staff's primary conclusion of the suitability of the CRAFT2 program to generate the required information upon which operating guidelines are developed. Therefore, it is recommended that the staff not perform additional audit analyses to correct these differences at this time.

4.1.4 Two-Phase Natural Circulation and Accommodation of Loss of All Feedwater

4.1.4.1 Discussion

For small break LOCAs which cannot remove all of the decay heat through the break, secondary heat removal is required or the reactor coolant system will repressurize. For the Davis-Besse 1 facility, which has low shutoff head HPI pumps (1630 psig), repressurization of the reactor coolant system above the shutoff head of the HPI pumps makes them ineffective for supplying makeup water to the primary system. Unless action is taken to reduce the reactor coolant system pressure to below the HPI shutoff head, uncovering of the core and fuel damage could result.

One method to depressurize the reactor coolant system is to open the PORV. However, the B&W design only includes one PORV, thus limiting the PORV's usefulness to perform this function. In addition, there is a high degree of uncertainty as to the functioning of the PORV under conditions of two-phase and solid water flow. The need to provide additional relieving capacity for the reactor coolant system has been identified in the past as part of the ATWS (anticipated transients without scram) report (NUREG-0460)[1] and more recently as part of the staff's post-TMI-2 requirements to provide venting capability at the primary system high points.[3]

As noted above, in order to mitigate the consequences of this type of small break at Davis-Besse, secondary heat removal must be maintained. Loss of secondary heat removal can occur in two ways: loss of feedwater, or failure to establish or maintain natural circulation.

In this regard, the staff has reviewed the proposed limits and assumptions associated with the intermittent operation of the reactor coolant pumps (RCPs) to start or help maintain natural circulation with two-phase fluid in the primary system.[62,64] The first B&W submittal,[62] discussed the restart and operating capability of Westinghouse- and Bingham- supplied RCPs during highly voided system conditions. The second B&W submittal[64] contains additional information, requested by the NRC, regarding the basic assumptions and limiting conditions of RCP operation under the highly voided conditions addressed in Reference 62. B&W has stated that the RCPs can be momentarily started (i.e., bumped), but one hour is required between bumps to permit the stator to cool. The B&W operating guidelines[103] require a 15 minute interval between bumps and that the four RCPs be bumped alternately to provide a one hour cooling period for the individual RCPs. Based upon our review of the B&W submittals, restart of the RCPs is acceptable provided that the component cooling water (CCW) system cannot be overpressurized and that other active components will not be starved of CCW due to operating the RCPs under the conditions of a small break LOCA.

To date, the PWR industry has not provided any data to experimentally verify their analytical predictions of two-phase natural circulation. Notwithstanding the ability of the PWR industry to verify their prediction of natural circulation, the staff believes that a diverse heat removal path, independent of the secondary system, is desirable. To this end, the NRC is considering in the TMI-2 Action Plan[9] this and other system changes designed to improve overall plant safety and reliability. Specific Commission policy and schedule will be established subsequent to Commission adoption of this plan.

4.1.4.2 Conclusions

The staff finds that the predicted flow through the PORV has a large uncertainty when the flow is two-phase in composition. Because of this high uncertainty, we cannot conclude that for Davis-Besse, with its low shutoff head HPI pumps, the reactor coolant system can be depressurized to initiate HPI flow in sufficient time to preclude unacceptable core uncovery should the capability to remove heat from steam generators be lost. Section 2.1.2 of NUREG-0578[3] requires that both PORV and safety valves be qualified to function under conditions of solid water and two-phase flow as well as saturated steam flow.

The various modes of two-phase natural circulation and the transitioning between modes, predicted by the vendors' analytical models, needs to be verified against appropriate experimental data. Section 4.2.2 of this report provides additional discussion on two-phase natural circulation.

A diverse decay heat removal path, independent of the steam generators, is desirable. This desirability stems from the fact that although the probability of losing all feedwater or natural circulation is considered low, it is nevertheless unquantified and finite.

4.1.4.3 Recommendations

a. The NRC TMI-2 Action Plan should consider the need for a diverse decay heat removal path independent of the steam generators for Davis-Besse 1. Consideration for diverse systems should include, for example: (1) increased PORV relieving capacity; (2) higher shutoff head HPI pumps; or (3) installation of a high pressure residual heat removal system. If a system which manually depressurizes the reactor coolant system below the HPI actuation setpoint is selected, the time available to the operator to decide if system depressurization is necessary (i.e., feedwater cannot be restored) should be greater than 20 minutes. The staff believes that times less than 20 minutes do not provide the operator sufficient time in which to fully analyze the situation and could result in incorrect action being taken.

4.2 Tennessee Valley Authority (TVA) Small Break Concerns

4.2.1 Introduction

By letter to B&W dated April 26, 1978,[117] TVA documented concerns regarding the ability of the B&W designed 205 fuel assembly (205-FA) plants to adequately remove decay heat from very small breaks which are not explicitly analyzed in Safety Analysis Report (SAR) submittals. These concerns were based on a report by Mr. C. Michelson of TVA, which has been commonly referred to as the "Michelson Report." (Mr. Michelson is also a consultant for the NRC's Advisory Committee on Reactor Safeguards (ACRS).) The conclusions of the report were based on hand calculations approximating small break behavior that assumed thermodynamic equilibrium and conserved mass and energy in the steady state.

The TVA concerns focused on six areas:

(1) The possibility and acceptability of intermittent natural circulation due to the alternating formation and then condensation of voids in the top of the hot leg piping;

(2) The time delay involved with lowering the primary side liquid level in the steam generators so that steam condensation can commence when the transition occurs from liquid natural circulation to two-phase natural circulation in the core;

(3) Pressurizer level not providing a correct indication of water level in the reactor core during small break LOCAs;

(4) The isolation of small breaks causing repressurization and exposure of the PORV and safety valves to two-phase flow;

(5) Pressure boundary damage resulting from the injection of cold water into steam and the collapse of steam bubbles in subcooled water; and

(6) The high pressure injection water may bypass the reactor core and exit the reactor coolant system directly via the break, thus not providing core cooling.

In a meeting between Combustion Engineering (CE) and TVA on May 27-28, 1975 on auxiliary feedwater, TVA expressed concerns to CE regarding a class of small break LOCAs whose depressurization rates were slower than those analyzed in the CE System 80 Standard Safety Analysis Report (CESSAR). The basis for the concerns was that for this class of small breaks, the makeup rates from the HPI pumps may not be adequate, leading to uncovering the core and resulting in unacceptable peak cladding temperatures. Subsequent to this meeting, Mr. Michelson drafted a report[118] expressing his concerns regarding problems associated with decay heat removal during recovery from a very small break LOCA for the CE System 80 design.

The CE System 80 concerns focused primarily on the following eight areas:

(1) The use of the HPI pumps with the shutdown cooling system;

(2) The use of the recirculation mode of HPI operation at high pressures not being an established design requirement including minimum flow protection for HPI pumps during this mode of operation;

(3) Reactor coolant pump or pump seal damage caused by the continuous operation of the reactor coolant pumps during a small break LOCA;

(4) The effect of core flood tank discharge with respect to noncondensible gas accumulation in the reactor coolant system;

(5) Initiation of containment spray during a small break LOCA;

(6) Long-term sources of auxiliary feedwater;

(7) The mechanical effects of slug flow on steam generator tubes; and

(8) The fact that during a small break LOCA the reactor vessel coolant level may be unknown to the operators and the adequacy of emergency procedures to cover this situation.

While the concerns expressed in References 117 and 118 were addressed to the B&W 205-FA plants and the CE System 80 design, respectively, most concerns have direct applicability to all PWR designs including the B&W 177-FA design.

By letter dated January 23, 1979,[96] B&W responded to the TVA concerns expressed in Reference 117. In the letter, B&W concluded that it had performed sufficient analyses to "ensure the ability of the B&W 205 plant's ECCS system to control small breaks in the RCS." These analyses were documented in BAW-10074A, Revision 1.[49] In April 1979, the staff met with B&W to discuss in detail the concerns expressed in Reference 117. As a result of these meetings, B&W submitted a comprehensive report[62] regarding the response of the 177-FA plant to small break LOCAs. The information contained in Appendix 5 of Reference 62 included information on the concerns expressed in Reference 117.

The staff has reviewed each of the TVA concerns presented in References 117 and 118. We have reviewed the B&W responses to the Reference 117 concerns and we have also examined available information in order to address the concerns presented in Reference 118. Where information was not available to the staff, it was requested from B&W or the B&W licensees. As pointed out later in this section, certain responses to the staff requests have not been received to date. This information is considered to remain outstanding.

A detailed discussion of each of the TVA concerns is provided below.

4.2.2 Intermittent Natural Circulation

4.2.2.1 Background and Analysis Results

This mode of decay heat removal was characterized by TVA as steam bubbles being generated in the core and accumulating at the top of the hot leg U-bend. If sufficient vapor accumulated to fill the U-bend, natural circulation would be lost. The loss of natural circulation and subsequent loss of the steam generator as a heat sink would cause the system to repressurize, provided the break could not remove all of the decay heat. Repressurization would then cause the steam bubble in the hot leg U-bend to condense and natural circulation would be reestablished. This, in turn, would lower the pressure and the steam bubble would form again.

According to TVA, a steam bubble would also accumulate in the upper part of the reactor vessel. This bubble would not completely condense during repressurization and would become larger during each natural circulation/repressurization cycle due to the net decrease in mass flow through the break. The ability to alternately stop and subsequently reestablish natural circulation as described by TVA was questioned as an unstable mode of operation. With regard to the growth of the bubble in the vessel head, B&W stated that "because of the internal vent valves, no extensive steam bubble will form within the reactor vessel while any significant liquid inventory remains in the loop."[96]

The staff agrees that liquid levels around the system would be in equilibrium with the vessel level before the vessel level would drop below the hot leg piping and into the active core region. The pressure in the vessel dome, necessary to sustain a significantly higher head of liquid in the steam generators, would be sufficient to open the vent valves and allow equalization.

B&W also stated in Reference 96 that intermittent natural circulation as described in the TVA report would not occur "due to the slow nature of the small break transient." Specifically, B&W stated that once natural circulation was lost, some repressurization would occur, but only until the liquid level on the primary side of the steam generators dropped below the liquid level of the secondary side. Once this occurred, decay heat removal through the steam generators would begin and the system would then depressurize. The basic question is whether the steam generator primary liquid level would drop fast enough from the discharge flow to establish decay heat removal by condensation heat transfer before repressurization condensed the steam bubble in the top of the hot legs and refilled the steam generators.

In Section 6.2.5 of Reference 62, B&W presented analyses of three small break events that showed repressurization. These were 0.01 ft^2 and 0.005 ft^2 breaks in a 177-FA lowered loop plant and a 0.01 ft^2 break in a 177-FA raised loop plant. These analyses were performed with the CRAFT2 code for simulations out to 3000 seconds.

The analyses for the lowered loop design showed that for the 0.01 ft^2 and the 0.005 ft^2 breaks, no cyclic repressurization occurred. Liquid natural circulation continued until enough mass was lost from the system through the break to cause the hot leg U-bend to start draining. Once the hot leg U-bend commenced draining, liquid natural circulation stopped. Eventually, enough mass was lost from the system to expose a condensing surface in the steam generators, causing decay heat removal to be reestablished via two-phase natural circulation. The analysis for the raised loop plant showed that for the 0.01 ft^2 break, cyclic repressurization did occur, as predicted by Michelson. However, the peak pressure reached was significantly less on each successive cycle and died out completely after three cycles. Once the cyclic repressurization phenomenon ceased, two-phase natural circulation commenced. For both the raised and lowered loop designs, the core remained covered throughout the entire period of these transients, thus assuring acceptable peak cladding temperatures.

A significant factor in the establishment of some mode of natural circulation (i.e., all liquid or two-phase) is that a steam-condensing surface must exist in the steam generators before the core could begin to uncover. For raised loop plants, this occurs from relative elevation differences. For lowered loop plants, this occurs because the AFW enters the steam generator from the top. For all B&W lowered loop plants, the small break emergency procedures require that the levels in the secondary side of the steam generators be raised to 95% on the operating range level indicators if the RCPs are not running. Auxiliary feedwater is automatically fed to the steam generators when the level reaches the low level limits ($\sim$30 inches on the startup range indication) if the RCPs are running. Analyses by B&W show that auxiliary feedwater will be initiated before the vessel water level drops below the top of the core.

4.2.2.2 Relationship of Concern to Events at TMI-2

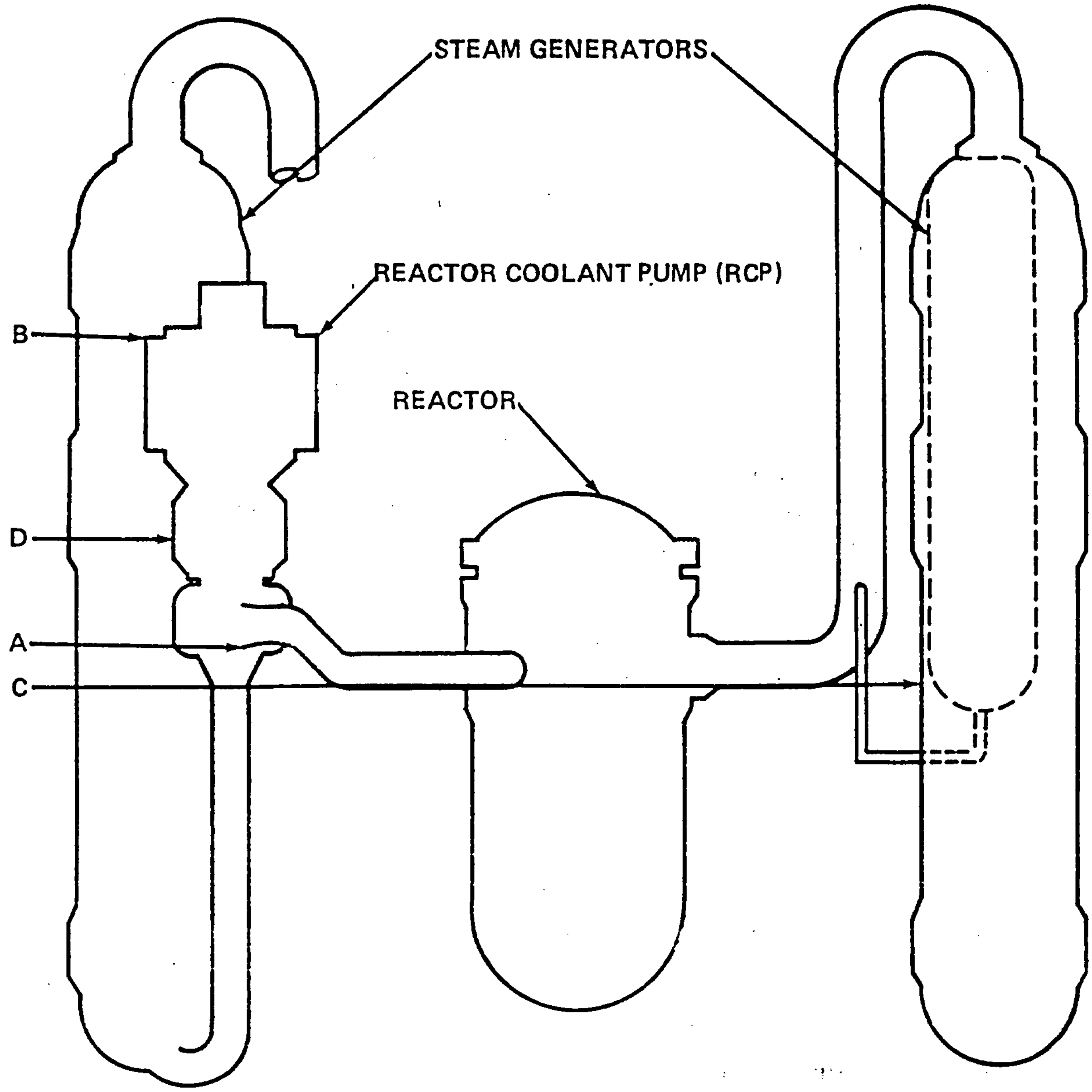
During the course of the accident at TMI-2, the operators tripped the last operating RCP 101 minutes into the accident. Immediately after the RCPs were stopped, reactor coolant temperatures in the hot leg piping were observed to rapidly increase. It was during this period that the majority of the damage to the reactor core was postulated to occur. Because of this occurrence, concern was raised as to why natural circulation was not established after the RCPs were tripped.

Based on examination of the component elevations in the plant, the steam generator secondary level setpoints, and the 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 RCPs were operating, the primary system evolved to a high system void fraction due to the continuous loss of inventory through the stuck-open PORV and the limited make-up due to manual throttling of the HPI flow. Despite the high system voids, operation of the RCPs circulated the steam and water as a two-phase saturated mixture throughout the system and provided ample cooling of the fuel rods.

When the RCPs 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 generators, the RCP suction piping, and the bottom of the reactor vessel, as can be seen on Figure 4-12. Also shown on Figure 4-12 is the elevation (elevation C) of the automatic feedwater control level setpoint, which was set to control level at 50 percent of the operating range when the RCPs were tripped. Since the TMI-2 accident, B&W has recommended that this level setpoint be increased to 95 percent of the operating range whenever the RCPs are tripped.

After the RCP trip, it is postulated that liquid existed in the bottom of the reactor vessel, the bottom of the steam generators and the RCP suction piping. Steam existed in the hot leg piping; upper portions of the core, the reactor vessel, and the steam generators, and also in the RCP and the RCP discharge piping. In order to initiate natural circulation, the liquid level in the RCP suction piping would have to increase such that liquid could flow through the RCP, into the discharge piping and into the reactor vessel downcomer. In order to raise the liquid level in the RCP suction piping, the liquid level in the steam generator tubes must be raised to an elevation above that of the bottom of the RCP discharge nozzle. This, in turn, can only be accomplished by establishing a condensing surface in the steam generators above this elevation



ELEVATION A – BOTTOM OF RCP DISCHARGE NOZZLE
ELEVATION B – APPROXIMATE ELEVATION OF AFW SPARGER
ELEVATION C – 50% OF OPERATING RANGE
ELEVATION D – 95% OF OPERATING RANGE

FIGURE 4-12 REACTOR COOLANT SYSTEM ARRANGEMENT FOR THREE MILE ISLAND UNIT 2 (SELECTED ELEVATIONS) (elevation A in Figure 4-12). Feedwater enters the steam generators through a sparger at elevation B (see Figure 4-12), and would normally produce a condensing surface well above that needed to force the water in the RCP suction piping up through the RCP and into the discharge piping; however, feedwater will only be supplied if it is replenishing liquid lost through boiling. Without the initial flow of liquid out of the steam generators and into the reactor vessel, the stagnant primary coolant in the lower portion of the steam generator tubes will eventually reach equilibrium with the secondary water which will be held at the 50 percent level on the operating range (elevation C in Figure 4-12). 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 RCP discharge nozzle was lost. The only condensing surface left was the secondary water level, and it was below the elevation necessary to allow water to flow through the RCP and into the reactor vessel. Thus, liquid could not flow from the steam generators to the vessel, and the steam produced in the core could not condense in the steam generators.

4.2.2.3 Corrective Action By The B&W Licensees

Subsequent to the accident at TMI-2, B&W has included in its operating guidelines for small breaks the requirement for the operators at the lowered loop plants to manually raise the steam generator secondary water level to 95 percent on the operating range in the event that the RCPs are tripped. This is shown as elevation D in Figure 4.12.

This level assures that a steam condensing surface will exist at elevations above the bottom of the RCP 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 TMI-2. 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 inadequate liquid inventory in the reactor coolant system.

Due to the uniqueness of the B&W raised loop design, the inadequacies described in Section 4.2.2.2 and the corrective action discussed in the section are not applicable to the raised loop design.

4.2.2.4 Conclusions

The potential for disrupting natural circulation during a small break LOCA via the cyclic repressurization phenomenon described by Michelson has been analyzed and evaluated by B&W. For the raised loop design, this phenomenon was shown to exist temporarily but died out after three cycles. The disruption in natural circulation did not lead to uncovering of the core and peak cladding temperatures remained acceptable. For the lowered loop design, the cyclic repressurization phenomenon was not exibited.

4.2.2.5 Recommendations

a. The various modes of two-phase natural circulation, which are expected to play a significant role in plant response following a small break LOCA, should be demonstrated experimentally. In addition, the staff requires that the licensees provide verification of their analysis models to predict two-phase natural circulation by comparison of the analytical model results to appropriate integral systems tests.

b. Appropriate means, including additional instrumentation, if necessary, should be provided in the control room to facilitate checking whether natural circulation has been established.

4.2.3 Time Delay Associated with Transitioning Between Modes of Natural Circulation

4.2.3.1 Discussion

TVA expressed concern that once liquid natural circulation was lost, the time required for the primary side steam generator level to drop level the secondary side level (exposing a condensing surface and thus commencing two-phase natural circulation) might be of sufficient length to allow the reactor coolant system to repressurize (with a subsequent increase in flow rate through the break) to deplete the system of enough mass to cause uncovering of the core. In particular, as the break size is decreased, the repressurization rate is increased, leading to higher mass discharges and faster inventory reductions which could lead to uncovering the core for very small breaks.

B&W stated that TVA's concern was not valid since "the volume relief out the break increases with increasing system pressure and break size. The volume of steam being generated in the core decreases with increasing pressure. As the break decreases in size, the reactor coolant system (RCS) will repressurize to a higher value; thus the volume relief out the break necessary to match the volume of steam being created decreases."[96]

B&W submitted two analyses (0.01 ft^2 and 0.005 ft^2) for 177-FA lowered loop plants which exhibit repressurization. Both analyses confirm B&W's statement. For both cases, the system repressurized to about 1800 psia and in both cases the vessel level dropped to approximately 17 ft (about four ft above the top of the core).

4.2.3.2 Conclusion

Based on these analyses and the previous discussion that the core cannot uncover without establishing a condensing surface in the steam generators, the staff agrees in principle that for very small breaks the time delay for transition from solid liquid to two-phase natural circulation will not lead to uncovering the core. However, we will require the licensees to provide adequate verification of their analytical model to confirm this conclusion. (See the recommendation in Section 4.2.2 of this report.)

4.2.4 Adequacy of Pressurizer Level Indication During a Small Break LOCA

4.2.4.1 Discussion

TVA expressed concern that, during a small break, the loop seal in the pressurizer surge line would prevent the pressurizer from draining and could maintain a pressurizer level even with the core partially uncovered. In their response, B&W acknowledged that during a small break, pressurizer level was not a reliable indication of water level in the RCS. The accident at TMI-2 has substantiated the validity of this concern.

The importance of a more reliable indication of primary system fluid inventory has been recognized. The revised plant operating procedures alert the operators of this problem and direct them to check other system parameters to maintain system inventory. In Section 2.1.3 of NUREG-0578[3] more direct and more easily interpreted indicators for both water inventory in the primary system and inadequate cooling are recommended to make operator actions more reliable.

4.2.4.2 Conclusion

Based on the revisions made to emergency procedures and the recommendations in Section 2.1.3 of NUREG-0578, we conclude that this concern has been adequately addressed.

4.2.5 Small Break LOCA Isolation

4.2.5.1 Discussion

The isolation of small breaks and the subsequent repressurization of the RCS was expressed as a concern by TVA. The concern involved postulating a small break occurring in a location that could be remotely isolated by the operator (e.g., the letdown line) at some time after the break initiation. This isolation would cause the system to repressurize (assuming natural circulation had been lost) up to the PORV or safety valve setpoints. Because these valves are not designed to relieve two-phase flow, valve damage could occur and the valves could remain open, thus creating a scenario that has not been explicitly analyzed in the 177-FA Safety Analysis Reports.

B&W agreed that the described scenario was reasonable and that the repressurization with or without PORV or safety valve failure was credible. B&W concluded that the valve failure was of little consequence since a failed valve would appear as a slightly larger break. If the valve did not fail, then the system would remain at the valve setpoint until the primary liquid level in the steam generators was reduced to below the secondary side level and two-phase natural circulation could start. The heat removal by the secondary system would then cause the reactor coolant system pressure to decrease until the setpoint for HPI actuation was reached. Although not addressed by B&W, the progression of the accident would then involve a refilling of the reactor coolant system until the water level in the primary side of the steam generators exceeded the level in the secondary side. Two-phase natural circulation would be lost and the system would begin to repressurize. As the pressure increased, HPI flow would decrease and a new equilibrium pressure would be reached. The acceptability of this accident progression is discussed in more detail in Sections 4.1.2 and 4.1.3 of this report.

The staff agrees in principle that the failure of either the PORV or the safety valves, after repressurization due to isolation of the small break, would appear as another small break. No new and/or unusual behavior is expected to occur and the slightly larger break size would be bounded within the small break spectrum presently required to be analyzed for licensing. However, the staff will require that the licensees perform an analysis of this scenario in order to confirm B&W's assessment.

With regard to the ability to predict the effect of two-phase flow through the PORV and safety valves, very little data are available at high pressure. The NRC had previously recognized this and requested that a program to obtain the data be initiated by the Office of Nuclear Regulatory Research (RES).[41] In support of this request, RES performed a literature review to determine available data. The results of this review[42] concluded that there was a lack of meaningful data in this area. As a result, Section 2.1.2 of NUREG-0578[3] recommends that programs be promptly initiated to test the performance of PORV and safety valves under normal, transient, and accident conditions.

The staff has also identified additional concerns associated with break isolation. In particular, it is assumed in B&W's small break analyses that the pressurizer spray isolation valve closes on a low pressure signal. B&W takes credit for this isolation although the valve and the associated control system are neither safety-grade nor single failure proof. Therefore, we will require that the licensees analyze the case of a break in the pressurizer spray line with a failure of the spray isolation valve to close.

4.2.5.2 Conclusions

Safe recovery from a small break can most reliably be achieved by isolating the break, if possible.

In the event that main or auxiliary feedwater are not available when a break is isolated, operator action may be necessary to manually open the PORV and initiate HPI. Emergency procedures are required to instruct the operators of this necessary action.

4.2.5.3 Recommendations

a. Licensees should perform an analysis in which a small break is isolated and the PORV fails-open upon repressurization of the RCS to the PORV setpoint.

b. Licensees should provide an analysis which shows the plant response to a break in the pressurizer spray line with a failure of the spray isolation valve to close.

4.2.6 Pressure Boundary Damage Due to Bubble Collapse

4.2.6.1 Introduction and Background

The Michelson Report[117] discussed the possibility of damage to system components due to water hammer effects from condensing steam. In particular, the TVA concern focused on the injection of cold HPI makeup water into a steam-filled cold leg, as well as on the bubbling of steam through subcooled liquid.

B&W responded that any instabilities arising from water hammer effects would be less than design conditions. Moreover, any pressure pulses due to condensing steam would produce loads less than the large break LOCA plus seismic loads that the system is designed to withstand.

The staff has assessed the impact of both steam bubbles in subcooled liquid as well as cold water injection into a steam-filled pipe. These topics are discussed in the following sections.

4.2.6.2 Steam Bubbles in Subcooled Liquid

Because present analytical models assume thermodynamic equilibrium, a condition of steam bubbles in subcooled liquid is not predicted to occur. However, the staff cannot rule out the possibility of such a situation existing during a small break LOCA. If the bubbling of saturated steam through subcooled water did occur, pressure pulses would be non-directional, and a system containing steam bubbles would be hydraulically "soft" and would attenuate pressure pulses prior to contact with structural boundaries. It is expected that these loadings would be bounded by the more severe case of cold water injection into steam as discussed in Section 4.2.6.3. It is the staff's judgment that these loadings will not exceed those associated with the large break LOCA which the system is designed to accommodate.

4.2.6.3 Cold Water Injection into a Steam-Filled Pipe

The staff has previously examined the possibility of hydraulic loads associated with injecting a cold fluid (core flood tank water) into steam-filled pipes. Two sources of experimental data, the Semiscale and LOFT facilities, were examined by EG&G Idaho.

The results of their examination[116] indicated that fluid oscillations could occur, but that pressure oscillations were "small (less than 10 psig) to nonexistent."

Of significant concern during the injection of cold water into steam is the potential for "slugging," in which the liquid flow entering the pipe is sufficient to fill the cross-sectional area of the pipe. Steam condensing on the surfaces of this water slug could set up pressure gradients of sufficient magnitude to initiate oscillations of this water slug in the pipe. This oscillating liquid slug may not only give rise to pressure oscillations, as observed in the tests, but can also produce inertial loads at pipe bends, and impact loads should the slug impact on surfaces such as the core barrel.

Calculational models designed to predict these pressure oscillations due to water slugs indicate that the amplitude of pressure oscillations should not increase as the pipe diameter increases as long as all other factors, such as system pressure, length-to-diameter ratio, etc., do not change. However, the amplitude of these pressure oscillations is predicted to increase as system pressure increases.

The previous discussion applies to cases when the pipe is filled with a water slug. If the cold water injection flow is insufficient to produce a water slug that fills the cross-sectional area of the pipe, then the above phenomena are not expected to occur.

4.2.6.4 Conclusions

If bubbling of saturated steam through subcooled liquid did occur, it is the staff's judgment that the pressure pulses and the resultant loadings on structural boundaries and components would not exceed those associated with the large break LOCA which the system is designed to accommodate.

With regard to the injection of cold water into a steam-filled pipe, the staff concludes that there is not enough information to support B&W's conclusions that the pressure oscillations are accommodated in the structural design. It is the staff's jugment that these loadings will not exceed those associated with the large break LOCA. However, we will require the licensees to provide confirmatory information on this subject.

4.2.6.5 Recommendations

a. Licensees should provide confirmatory information to show that HPI and CFT flows during small breaks are insufficient to form water slugs, or if they do, to show that the structural design basis of the primary system includes loads due to:

    (1) water slug intertial motion;
    (2) water slug impact; and
    (3) pressure oscillation due to steam condensation.

Any test data cited must be shown to be applicable to the actual system design.

4.2.7 Possible Bypassing of the Core by Injected HPI Water

4.2.7.1 Discussion

This concern regards the possible bypassing of the core by injected HPI water, and the discharge of HPI water out of the break. If the break flow energy is not representative of the core exit energy, then not all of the decay heat load would be removed by the break.

In the B&W small break evaluational model, fluid enthalpy distribution and transport within the primary system are accounted for. In considering the possibility of injected HPI water bypassing the core and discharging out the break, the B&W model assumes that 30% of the HPI fluid injected into the broken leg (for cold leg breaks) is lost out of the break. For cold leg breaks, the amount of HPI water injected into the cold legs of the intact loops, that travels around the downcomer annulus and exits the broken loop break without entering the core, is also accounted for. The B&W analytical model is discussed further in Section 4.1.1 of this report.

Based on this concern, as well as previous vendor system pressure predictions for Semiscale Test S-02-06, the staff is performing an ongoing evaluation of small break system behavior during ECC injection. The scope of this evaluation includes both injection location modeling and system pressure behavior during injection.

It is expected that the adequacy of the present models will be determined by comparison to the Semiscale Test S-07-10B and LOFT Small Break Test L3-1.

4.2.7.2 Conclusion

The staff concludes that while the B&W evaluation model accounts for the fluid enthalpy distribution and transport in the primary system, as well as injected HPI fluid bypassing the core, the overall adequacy of the predicted system behavior during ECC injecton should be confirmed by comparison to experimental data. The B&W analytical model is discussed in Section 4.1.1 of this report.

4.2.8 Use of HPI Pumps with Long-Term Cooling Systems

4.2.8.1 Discussion

For both CE and B&W plants, the long-term cooling after recovery from the small break LOCA is performed by a low head, high volume cooling system (termed shut-down cooling system (SCS) for CE plants and residual heat removal (RHR) system for B&W plants). For CE plants, this system circulates primary coolant through heat exchangers, cools it, then returns it to the primary system. For CE plants, inventory in the primary system during this period must still be maintained by the HPI pumps. TVA expressed a concern on the simultaneous use of these two systems (HPI and SCS), "since it involves some common piping including a common suction pipe." Moreover, TVA was concerned that "this mode of operation has also not been a design requirement."

For all B&W plants, there is a basic design difference from the CE System 80 design which does not make this concern directly applicable. For CE System 80 plants, the SCS and the HPI pumps share a common suction from the refueling water storage tank during the injection mode. During the post-accident cooling modes (initial recirculation, simultaneous injection, and shutdown cooling), the HPI and SCS pumps switch suction from the refueling water tank to other sources. The HPI switches to the containment sump while the SCS switches to the hot legs. Any cross-connects or common piping shared during the injection mode must be isolated from the HPI suction path after the injection mode for CE plants to prevent loss of NPSH due to common suction.

In B&W plants, the HPI pumps take suction from the RHR pumps and the RHR pumps take suction from the containment sump in the recirculation mode for long-term decay heat removal.

4.2.8.2 Conclusion

Due to basic design differences between the CE System 80 and the B&W 177-FA plants, this concern is not applicable to B&W operating plants.

4.2.9 Recirculation Mode of HPI Operation at High Pressures

4.2.9.1 Discussion

This concern was based on the fact that for some CE plants, the HPI pumps have a shutoff head around 1300 psig. If decay heat removal could not be accomplished due to disruption of natural circulation (i.e., loss of heat sink), the system would repressurize. If the system then remained at high pressure for an extended period, the recirculation mode for HPI operation would eventually be activated. TVA's concern was the ability of a HPI sytem design to operate in the recirculation mode with the reactor at high pressure, and the evaluation of such operation for feasibility. In addition, because of this low shutoff head, TVA was concerned that repressurizaton during the recirculation mode of operation when the pumps do not have minimum flow protection could deadhead the pumps and damage them.

For all of the B&W plants except Davis-Besse 1, the operation of the HPI system in the recirculation mode at high pressure is an established design requirement. All of these plants have HPI pumps whose shutoff head is above that of the PORV and safety valve setpoints. Deadheading of these pumps would not occur prior to lifting the PORV and safety valves, thus providing minimum flow protection.

At Davis-Besse, when the reactor coolant system pressure increases above the shutoff head of the HPI pumps (1630 psig), the emergency procedures instruct the operator to secure the HPI pumps and operate the makeup system to inject water into the reactor coolant system at these higher pressures. The recirculation mode of operation at high system pressures is not an established design requirement at Davis-Besse. The only minimum flow protection, in this mode, is operator action to stop the HPI pumps.

4.2.9.2 Conclusion

For all of the B&W plants except Davis-Besse, the operation of the HPI system in the recirculation mode is an established design requirement and there is minimum flow protection for the HPI pumps at high system pressure. At Davis-Besse, the HPI pumps are not utilized for system pressures above their shutoff head. The emergency procedures instruct the operator to use the makeup system to inject water into the reactor coolant system whenever system pressure is above the shutoff head of the HPI pumps.

4.2.10 Effects of Small Break LOCAs on Reactor Coolant Pumps

4.2.10.1 RCP Seal Damage

During a small break LOCA with a loss of offsite power, there is a possibility for RCP seal damage and leakage due to loss of seal cooling. Most B&W plants have either Byron-Jackson or Bingham pumps; these pumps have canned seals and would probably not experience substantial leakage. Oconee 1 and TMI-1 have Westinghouse pumps; these pumps have a seal design which is more prone to leakage if seal cooling is lost.

The licensees have been asked to evaluate the possibility and impact of seal damage and leakage.[26] If seal damage cannot be precluded, the licensees should provide an analysis of the limiting small break LOCA with subsequent pump seal failure. Therefore, this potential problem is currently unresolved.

4.2.10.2 RCP Operation

Almost all large break LOCA analyses to date have concluded that tripping of the RCP at the initiation of the accident leads to the highest calculated peak cladding temperatures. Assuming RCP trip at the start of the accident was also consistent with the assumption that offsite power was lost. Because this typically proved to be the most limiting condition for the RCPs during the large break accident, it was generally assumed that it produced the limiting conditions for the small break LOCA as well. Subsequent to the TMI-2 accident, B&W along with the other two PWR vendors performed analyses of small break LOCAs in which the RCPs were assumed to remain running during the accident. They concluded that the major effect of pump operation during a small break LOCA was to redistribute liquid in the primary system. This redistribution resulted in the liquid being made available to the break for a longer period of time than for the same case with the RCPs tripped. They concluded that for a given range of break sizes, there existed a range of times in which, if the pumps were tripped for any reason (mechanical failure, operator action, etc.), the depth and duration of core uncovery would result in calculated cladding temperatures in excess of the 2200°F licensing limit.

The staff has performed a generic assessment of the effect of delayed RCP trip. This assessment and the conclusions are documented in Reference 6. The key conclusions of Reference 6 are: (1) the uncertainty in small break analysis models with RCPs running is large, and cannot at this time be quantitatively relied upon; (2) the RCPs need to be tripped early in the accident and that this trip should be performed automatically; and (3) that model verification by pretest prediction of a forthcoming LOFT small break test with the RCPs running will be required.

4.2.10.3 Conclusions

The possibility and impact of RCP seal damage and leakage during a small break LOCA is currently unresolved. The B&W licensees have been requested by our letter dated November 21, 1979[26] to evaluate this matter.

NUREG-0623[6] provides a detailed assessment of RCP trip during a small break LOCA in pressurizer water reactors. In addition, NUREG-0623 requires that the RCP trip be performed automatically and specifies that this automatic trip system should be installed and operational by January 1, 1981.

4.2.10.4 Recommendations

a. Licensees shall submit (as requested in our letter dated November 21, 1979) an analysis of the possibility and impact of RCP seal damage and leakage due to loss of seal cooling on loss of offsite power. If damage cannot be precluded, licensees should provide an analysis of the limiting small break LOCA with subsequent RCP seal failure.

b. Licensees shall provide pretest predictions of LOFT Test L3-6 (RCPs running).

4.2.11 Effect of Core Flood Tank (CFT) Discharge and Accumulation of Noncondensible Gases

4.2.11.1 Effect of Core Flood Tank Discharge

This concern involves the effect of the noncondensible gas in the CFT discharging into the reactor coolant system once the primary system pressure drops below approximately 600 psig. Since these tanks are pressurized with nitrogen gas, there is the potential for injecting this gas into the system. Introduction of noncondensible gases into the reactor coolant system has the potential to disrupt natural circulation and reduce steam generator condensation heat transfer.

4.2.11.1.1 CFT Discharge During Small Breaks

For steam generators to remove decay heat during natural circulation, the reactor coolant system pressure is slightly higher than the secondary system pressure in order to establish the necessary temperature gradient from primary to secondary. For any small breaks in which the primary system depressurizes to a value below the secondary pressure (about 1050 psi), sufficient heat is being removed from the primary such that the steam generators are no longer needed to remove decay heat and in fact become a heat source to the primary system. Thus, CFT discharge and possible nitrogen injection during small break LOCAs is not a concern since, for all small breaks in which the CFT are calculated to inject, the steam generators are not relied upon for decay heat removal. The CFTs are calculated to inject for breaks of 0.07 ft^2 and larger, whereas the steam generators are only required to remove decay heat for breaks of 0.01 ft^2 and smaller.

4.2.11.2.1 CFT Discharge During Steam Line Breaks

A severe secondary overcooling event could conceivably drop the reactor coolant system pressure to below the CFT actuation pressure such that nitrogen gas could be injected. In Reference 104, the B&W licensees provided an analysis of a 12.2 ft^2 double-ended steam line break. This analysis showed that the primary system was calculated to depressurize to below the pressure necessary to empty the CFTs of water and allow gas to enter the system. However, this calculation was based on the RCPs running throughout the accident. Present requirements in NUREG-0623[6] require all RCPs for B&W plants to be tripped upon reactor trip and ESFAS actuation on low reactor coolant system pressure. Thus, this scenario is not applicable as the RCPs would have been tripped. Analysis of the same event with the RCPs tripped showed that the calculated pressure does not drop below about 275 psig, which is about 75 to 100 psi above the pressure where gas could enter the system.

4.2.11.1.3 Procedures for Isolating CFTs

Once the plant has recovered from a small break and the operator has established the required pressure and level control, procedures instruct the operator to bring the plant to a shutdown condition. This includes depressurization. These procedures require the CFT to be isolated while the system is at a pressure above the actuation pressure. This precludes inadvertent injection during normal cooldown.

Even if the tanks did begin to discharge as the pressure dropped below the 600 psig actuation pressure, they would increase the system inventory and compress the pressurizer steam bubble such as to hold up the system pressure. System pressure would have to drop to about 200 psig before the tanks were completely discharged and could allow gas to enter the system. Prior to this, the effect of tank injection would alter the system behavior such that the operators would recognize that the tanks were injecting and could take the necessary action to isolate them.

4.2.11.2 Effect of Noncondensible Gases

In the event of a small break LOCA, noncondensible gases could be introduced into the primary system from a number of sources. These noncondensible gases can affect the system behavior in a number of ways: condensation heat transfer in the steam generators can be degraded, noncondensible gas accumulation in system high points can degrade or potentially stop natural circulation flow, and significant amounts of noncondensible gas could introduce errors in analysis models based on single fluid assumptions.

In Reference 98, the B&W licensees provided an evaluation of the effect of noncondensible gases on a small break loss-of-coolant accident. The staff review of this evaluation and conclusions is provided in the following sections.

4.2.11.2.1 Sources of Noncondensible Gas

In a PWR, there are nine sources of noncondensible gas which are already in, or could potentially be introduced into, the primary system. These are:

(1) dissolved hydrogen in the primary coolant;
(2) dissolved nitrogen in the CFT water;
(3) dissolved air in the borated water storage tank;
(4) hydrogen releases from zirconium-water reaction;
(5) free nitrogen used to pressurize core flood tanks;
(6) hydrogen released from radiolytic decomposition of injected water;
(7) fission and fill gas in reactor fuel;
(8) hydrogen gas (free and dissolved in the makeup tank); and
(9) pressurizer steam space gas.

With the exception of the source due to radiolytic decomposition, B&W has accounted for each of these sources in their analyses. Because the CFT actuation pressure is approximately 450 psig below the secondary system relief valve setpoint, the steam generators will be heat sources rather than sinks for any breaks which depressurize to the CFT setpoints and natural circulation would not be a requirement for decay heat removal. Therefore, gas sources from the CFT have not been included in the analyses. The licensees have also concluded that for all small breaks considered in the design bases of 177-FA plants, peak cladding temperatures are low enough that fission gas sources due to cladding rupture or oxidation sources are negligible. Therefore, they have concluded that gas from sources identified as items (1), (3), (8), and (9), along with fission and fill gases assuming one percent failed fuel in the core, are available to the primary system.

Assuming that all of the gas comes out of solution, no noncondensible gas is lost through the break, and the amount of water injected by the HPI system from the borated water storage tank is 6.4×10^4 lbm (corresponding to 1500 seconds of HPI). The licensees estimated that a total of 780 SCF* of noncondensible gas would be in the primary system. At a system pressure of 1050 psig, this would occupy a volume of 22.4 ft^3.

4.2.11.2.2 Effect on Natural Circulation Flow

In order to inhibit natural circulation at pressures representative of small breaks requiring secondary system heat removal, the gas would have to fill the U-bends at the top of the hot legs. These bends have a volume of 125 ft^3. Thus, the conclusion drawn by the licensees is that the maximum amount of noncondensible gas calculated to be available is approximately a factor of five less than the amount needed to inhibit natural circulation. This analysis did not assume that any gas accumulated in the upper head or plenum of the reactor vessel, which is considered the more likely location for gas accumulation. Thus, no reduction in natural circulation flow is predicted by the licensees due to noncondensible gas accumulation. However, as pointed out previously, B&W has neglected any gas source due to radiolytic decomposition of the water. The staff requires that licensees provide justification why this source can be omitted from the analyses.

4.2.11.2.3 Effect on Steam Generator Condensation Heat Transfer

The licensees have evaluated the effect of noncondensible gases on steam generator condensation heat transfer. If a noncondensible gas layer accumulates on the condensing surface, the vapor must diffuse through this layer in order to condense. This diffusion process represents a reduction in the condensation heat transfer rate and could affect the system thermal-hydraulic behavior.

The licensees evaluated the effects of noncondensible gases on condensation heat transfer using the model of Colburn and Houger and neglected the heat transfer across the vapor layer (this assumed condensation occurs by energy transfer due to mass diffusion only).

*SCF is Standard Cubic Feet, and corresponds to the volume the gas would occupy at 32°F and 1 atmosphere pressure.

The effects of the reduced condensation heat transfer rate were evaluated for 0.04 ft^2 and 0.01 ft^2 cold leg breaks, since these breaks are calculated to rely on decay heat removal by the steam generator for a significant portion of the accident.

The conclusions reached were: (1) the effect of noncondensible gas on steam generator heat transfer is negligible, and (2) the maximum increase in system pressure was 25 psi for the 0.04 ft^2 break and 40 psi for the 0.01 ft^2 break. Of the 25 psi increase for the 0.04 ft^2 break, 24 psi was attributed to the partial pressure of the noncondensible gases. Similarly, 34 psi was attributed to the partial pressure of the noncondensible gas for the 0.01 ft^2 break. These increases in system pressure would have a negligible effect on the break flow rates and on the minimum system inventory.

The licensees consider the above evaluation to be conservative since they assume all of the noncondensible gases are located in the steam generators and that no gas is accumulated in the reactor vessel or has escaped out of the break.

The staff will require that licensees provide confirmatory information to verify the predicted condensation heat transfer degradation in the presence of noncondensible gases.

4.2.11.3 Conclusions

The staff has reviewed the licensees' analyses of the effects of CFT discharge on small break LOCAs and non-LOCA transients. We have also reviewed their analyses of the effects of noncondensible gases on: (1) natural circulation flow; (2) steam generator condensation heat transfer; and (3) system hydraulics. Based on this review, we have concluded the following:

1. During long-term recovery from a small break LOCA, after pressure and level are regained, cooldown procedures instruct the operators to isolate the CFTs. The staff feels that this provides adequate assurance that the CFTs will not inadvertently discharge.

2. For small break LOCAs in which the CFT discharge is predicted to occur, the steam generators are not relied upon for decay heat removal and in fact become a heat source to the primary system. However, steam generator heat removal may be required eventually if the RHR system is not operational.

3. The steam line break is considered to be the non-LOCA accident which produces the most severe overcooling and depressurization of the reactor coolant system. For this event, CFTs might inject into the primary system, but if so, this injection stabilizes primary system pressure just below the injection setpoint, and no gas injection is calculated to occur, provided all RCPs are tripped on reactor trip and ESFAS actuation occurs on low reactor coolant system pressure.

4. The expected amounts of noncondensible gases that could accumulate in the primary system during a small break LOCA are small compared to the amounts needed to: (1) block natural circulation flow; (2) significantly degrade steam generator condensation heat transfer; and (3) invalidate single-fluid analysis models.

5. Present small break analyses show that for all breaks which rely on the steam generators for decay heat removal, core uncovery is not predicted to occur. Therefore, omission of noncondensible gas sources due to cladding rupture and/or oxidation is considered justified and present analytical models need not implicitly consider the effects of noncondensible gases on the calculated results.

4.2.11.4 Recommendations

a. With regard to the effects of noncondensible gases during a small break LOCA, the licensees should provide the following information:

   (1) The technical justification for omitting the radiolytic decomposition of injected ECC water as a source of noncondensible gas; and

   (2) Confirmatory information to verify the predicted condensation heat transfer degradation in the presence of noncondensible gases.

4.2.12 The Effect of Containment Spray on Nonsafety-Grade Equipment

In the event of a very small break, low pressure would appear early in the accident and initiate containment isolation. Subsequent high containment pressure could then initiate containment spray. In particular, a prolonged isolation due to delay in cooldown would increase the likelihood of initiating containment spray. The containment spray environment might damage control-grade equipment within the containment which may be relied upon to mitigate the consequences of a small break LOCA.

Recently, the staff has identified the need[4] for owners of operating plants and all plants under construction to "...reevaluate the interaction of nonsafety- and safety-grade systems during normal operation, transients, and design basis accidents...." This reevaluation is expected to identify any additional equipment inside of containment that is required to operate or whose failure could jeopardize plant safety under conditions of containment spray. For that equipment identified, appropriate corrective action will be required.

4.2.13 Long-Term Sources of Auxiliary Feedwater

The assurance of "a continuous long-term source of clean auxiliary feedwater from the steam generators...," was raised by TVA as a potential concern if the recovery should be delayed due to prolonged cooldown.

There are presently nine plants designed by B&W which have been previously granted operating licenses by the NRC. These are: Oconee 1, 2, and 3; Davis-Besse 1; Three Mile Island 1 and 2; Rancho Seco; Crystal River 3; and Arkansas Nuclear One 1. Table 4-2 lists the sources of auxiliary feedwater available to each plant. Note that for Davis Besse, Rancho Seco, ANO-1, and TMI the service water system is a feedwater source and takes its supply from the ultimate heat sink or an unlimited source of water. For Crystal River 3, there are about 850,000 gallons available. While this is considered an ample supply which provides a large margin over the nominal inventory needed, the staff is studying this further as part of the auxiliary feedwater reliability study. For Oconee there are 244,000 gallons for each unit immediately available and each unit's tanks can be used to serve any other unit. The auxiliary service

TABLE 4-2

SOURCES OF AUXILIARY FEEDWATER IN B&W PLANTS

Oconee	Davis-Besse	TMI	Rancho Seco	Crystal River	ANO-1
1. Upper Surge Tanks A&B (72,000 gal/unit)	1. Condensate Storage Tank(s) (250,000 gal/each)	1. Condensate Storage Tanks (2/unit = 500,000 total)	1. Condensate Storage Tank (250,000 gal)	1. Condensate Storage Tank (200,000 gal)	1. Condensate Storage Tank
2. Hotwell (142,000 gal/unit)	2. Fire Protection System		2. Folsom Canal via pumps (supplied by Folsom River)	2. Hotwell (200,000 gal)	2. Service Water System - Ultimate Heat Sink
3. Condensate Storage Tank (30,000 gal per minute - station total)	3. Service Water System - Ultimate Heat Sink	3. Service Water System - Ultimate Heat Sink	3. Reservoir on Site - Essentially Unlimited	3. Demineralized Water Storage Tank (450,000 gal)	
4. Makeup Demineralizers (900 gal - station total)					
5. Auxiliary Service Water System					

water supply at Oconee can only be used following steam generator depressurization. The staff is also studying this further as part of the B&W auxiliary feedwater reliability study. If it is shown that an unlimited or larger source of feedwater is necessary for Oconee and/or Crystal River, then appropriate modifications will be required.

4.2.14 Mechanical Effects of Small Break LOCAs on Steam Generator Tubes

4.2.14.1 Mechanical Effects of Slug Flow on Steam Generator Tubes

Once the HPI pumps begin to add mass to the reactor coolant system at a rate greater than that of the flow out the break, the system inventory will stop decreasing and begin to increase. This is what TVA termed the recovery period in Reference 117. If the liquid level had dropped below the hot leg outlets, decay heat removal via two-phase natural circulation would have been established. As the vessel level increases during the recovery period, the hot legs would become blocked with liquid, disrupting steam flow from the core to the steam generator. Vapor generated in the core would accumulate in the vessel upper plenum and upper head with a subsequent rise in pressure. This pressure rise would force the vessel liquid level down until a venting path for the steam from the vessel to the steam generator or downcomer annulus was established. TVA's concern was that this condition could produce slug flow in the hot leg piping, resulting in mechanical loadings on the steam generator tubes.

In our letter to the B&W licensees dated November 21, 1979,[26] we requested that they evaluate the impact of slug flow on steam generator tubes with respect to assuring that the tubes can withstand any mechanical loadings that could result from slug flow. While the staff believes that the potential for slug flow is not great in B&W plants because of the venting path provided by the internal vent valves, we require an evaluation of this concern by the licensees.

4.2.14.2 Steam Generator Tube Thermal Stress Evaluation

In addition to the mechanical effects of slug flow on steam generator tubes, the staff was concerned about the thermal stresses which might exist in the steam generator tubes during a small break LOCA.

Following a small break LOCA, a condition was postulated to occur when the primary system has significant voiding. The secondary side of the once-through steam generator (OTSG) is assumed to be completely depressurized and is boiled dry. Cold auxiliary feedwater is initiated, which produces rapid cooling of the tubes resulting in significant tensile loads in the tubes' axial direction. If nonuniform cooling of the tubes is also assumed, a condition which is considered unlikely, significant bending stresses would be produced. The maximum stress occurs at the tube wall outside diameter. The stress results from pressure loads, loads resulting from the tube-to-shell temperature difference, and thermal bending loads due to nonuniform cooling.

Since there is a possibility for a defective tube in the steam generator, the load required to fail a defective steam generator tube under the above loads was also investigated. An analysis, performed by B&W in Reference 65, indicates that, under the combined pressure and axial loading developed during this transient, plastic deformation of the tube would result. However, primary to secondary leakage in tubes even with 70% through-wall defects is not likely to occur. This is substantiated by test data obtained from artifically defected tubes subjected to similar pressure and axial loads. If the nonuniform cooling phenomenon is also assumed, large plastic deformations are likely to occur, and the analyses indicate the tube would fail in fatigue after about ten cycles of the assumed transients. The analysis performed by B&W verifies the integrity of both defective and non-defective tubing under severe thermal stress conditions. Using a number of worst-case system parameters, ten cycles of the transient would be allowable for a tube with a flaw that is oriented so that the maximum stress concentration is achieved. It is believed that the severity of the transient considered is such that ten cycles of the event with the assumed parameters could not occur. The combination of conservative analyses and test results provides assurance that the structural integrity of the primary pressure boundary will be maintained.

4.2.14.3 Conclusions

While the staff believes that the potential for slug flow is not great in the B&W plants, because of the venting path provided by their internal vent valves, the staff will require that an evaluation of the effects of slug flow on steam generator tubes be performed by the licensees to assure that the tubes can withstand any mechanical loading which could result from slug flow.

The staff has also considered the effects of thermal stress on the steam generator tubes following a small break LOCA. We agree with the results of B&W's analysis which verifies the integrity of both defective and non-defective tubing under severe thermal stress conditions. The combination of conservative analyses and test results provides assurance that the structural integrity of the primary coolant pressure boundary will be maintained.

4.2.14.4 Recommendations

a. Licensees shall address (as previously requested in our letter dated November 21, 1979)[26] by use of analysis and/or experiment, the mechanical effects of slug flow on steam generator tubes.

4.2.15 Reactor Vessel Coolant Level Unknown to Operators/Adequacy of Emergency Operating Procedures

During the course of a small break LOCA, the mode of decay heat removal will switch from single phase liquid natural circulation to two-phase natural circulation and then back to solid liquid natural circulation. As was discussed in Section 4.2.14 the system hydraulic behavior could become erratic during the transitioning due to slug flow, etc., and the operators could observe fluctuations in system pressure. TVA's concern was that during a period of potentially unstable system hydraulic behavior, the operators would not have accurate vessel level indication, and without clear emergency procedures to guide them, could take the wrong or inappropriate actions.

The staff agrees that system behavior could become erratic during this period. This does not pose a problem, however, provided that the potential erratic behavior does not precipitate incorrect operation actions. In particular, the operator should not terminate or degrade HPI or otherwise jeopardize the normal sequence of actions of plant safety features. The staff has reviewed the B&W operator emergency guidelines for small break LOCAs and it is our conclusion that if these precautions are properly incorporated in plant emergency procedures, no operator action will be needed during this phase of the small break LOCA, and wrong or inappropriate operator action will not be precipitated.

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98. Noncondensible Gases: Sources, Effects, Operator Actions

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Letter from W. O. Parker (Duke) to H. R. Denton (NRC) dated January 8, 1980.

103. Small Break Operating Guidelines (Incorporating Changes Made to Meet Requirements of IE Bulletin 79-05C, "Tripping of RCPs," and Addressing Inadequate Core Cooling

Letter from W. O. Parker (Duke) to J. P. O'Reilly (NRC) dated November 9, 1979.

Letter from D. C. Trimble (AP&L) to R. W. Reid (NRC) dated December 13, 1979.

Letter from J. J. Mattimoe (SMUD) to R. W. Reid (NRC) dated November 26, 1979.

Letter from J. A. Hancock (FPC) to J. P. O'Reilly (NRC) dated November 14, 1979.

Letter from R. P. Crouse (TECO) to R. W. Reid (NRC) dated December 27, 1979.

104. Responses to IE Bulletin 79-05C, "Tripping of RCPs During a Small Break LOCA"

Letters from W. O. Parker (Duke) to J. P. O'Reilly (NRC) dated August 24, 1979, September 14, 1979, and November 9, 1979.

Letters from D. C. Trimble (AP&L) to K. V. Seyfrit (NRC) dated August 29, 1979 and September 14, 1979.

Letter from D. C. Trimble (AP&L) to R. W. Reid (NRC) dated December 13, 1979.

Letters from J. J. Mattimoe (SMUD) to R. H. Engelken (NRC) dated August 27, 1979, September 19, 1979, and October 24, 1979.

Letter from J. J. Mattimoe (SMUD) to R. W. Reid (NRC) dated November 26, 1979.

Letters from W. P. Stewart (FPC) to J. P. O'Reilly (NRC) dated August 24, 1979 and September 14, 1979.

Letter from J. A. Handcock (FPC) to J. P. O'Reilly (NRC) dated November 14, 1979.

Letters from R. P. Crouse (TECO) to J. G. Keppler (NRC) dated August 29, 1979, September 20, 1979, and November 1, 1979.

Letter from R. P. Crouse (TECO) to R. W. Reid (NRC) dated December 27, 1979.

105. Anticipatory Reactor Trip for Loss of Feedwater and Turbine Trip

Letters from W. O. Parker (Duke) to H. R. Denton (NRC) dated May 21, 1979 and October 5, 1979.

Letter from D. C. Trimble (AP&L) to K. V. Seyfrit (NRC) dated May 21, 1979.

Letter from D. C. Trimble (AP&L) to R. W. Reid (NRC) dated October 8, 1979.

Letter from W. C. Walbridge (SMUD) to R. H. Engelken (NRC) dated May 21, 1979.

Letter from J. J. Mattimoe (SMUD) to R. W. Reid (NRC) dated October 5, 1979.

Letter from W. P. Stewart (FPC) to J. P. O'Reilly (NRC) dated May 21, 1979.

Letter from W. P. Stewart (FPC) to R. W. Reid (NRC) dated October 2, 1979.

Letter from J. S. Grant (TECO) to J. G. Keppler (NRC) dated May 21, 1979.

Letter from L. E. Roe (TECO) to R. W. Reid (NRC) dated October 3, 1979.

106. PORV Actuation Data and Reactor Trip Frequency

Letter from W. O. Parker (Duke) to H. R. Denton (NRC) dated November 26, 1979.

Letter from D. C. Trimble (AP&L) to R. W. Reid (NRC) dated November 15, 1979.

Letter from J. J. Mattimoe (SMUD) to R. W. Reid (NRC) dated November 16, 1979.

Letter from G. C. Moore (FPC) to R. W. Reid (NRC) dated November 15, 1979.

Letter from L. E. Roe (TECO) to R. W. Reid (NRC) dated November 20, 1979.

107. Impact of RCP Seal Damage and Leakage During Small Break With Loss of Offsite Power

Letter from W. O. Parker (Duke) to H. R. Denton (NRC) dated December 28, 1979.

Letter from J. J. Mattimoe (SMUD) to R. W. Reid (NRC) dated December 19, 1979.

Letter from J. A. Handcock (FPC) to R. W. Reid (NRC) dated December 10, 1979.

Letter from R. P. Crouse (TECO) to R. W. Reid (NRC) dated December 28, 1979.

Correspondence from Licensees (not categorized by subject)

108. Letter from W. O. Parker (Duke) to H. R. Denton (NRC), "Forwards Report Entitled 'MODEL DOCUMENTATION' Which Describes the Modeling Techniques Used in Small Break LOCA Analyses to Verify ECCS Capability," dated September 13, 1979.

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110. Letter from W. O. Parker (Duke) to H. R. Denton (NRC), "Discusses Possible Revision to Requirements for Tripping RCPs During a Small Break LOCA," dated December 6, 1979.

111. Letter from J. J. Mattimoe (SMUD) to R. W. Reid (NRC), "Forwards Guidelines for Inadequate Core Cooling Decay Heat Removal System Mode of Operation," dated January 7, 1980.

112. Letter from J. J. Mattimoe (SMUD) to D. G. Eisenhut (NRC), "Responds to NRC Request to Review Unnecessary Challenges to Reactor Trip and Safeguards Systems," dated September 28, 1979.

113. Letter from J. J. Mattimoe (SMUD) to R. H. Engelken (NRC), "Followup Report on RO 79-6. Analysis by B&W Indicates that B&W Analysis Shows that Loss of RCPs Sometime After Two Minutes Into An Event Involving a Certain Spectrum of SBLOCAs Could Exceed 10 CFR 50, Appendix K, Criteria," dated August 2, 1979.

114. Letter from W. P. Stewart (FPC) to R. W. Reid (NRC), "Comparison of Key Parameters Used in B&W Small Break Analyses and the Actual Equipment Parameters Used at Crystal River 3," dated June 15, 1979.

115. Letter from L. E. Roe (TECO) to R. W. Reid (NRC), "Forwards Report Entitled, 'Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177-Fuel Assembly Plant - May 16, 1979 - Volume 3 - Raised Loop Plant (Davis-Besse 1),'" dated May 22, 1979.

115A. Letter from L. E. Roe (TECO) to H. R. Denton (NRC), "Transmits Results of B&W Analysis on Davis-Besse 1 for Loss of Feedwater Concurrent With Loss of Offsite Power," dated June 15, 1979.

Miscellaneous Correspondence

116. Letter from L. P. Leach (EG&G) to R. E. Tiller (NRC), "Pressure Oscillations Resulting from ECC Injection - LPL-47-77," dated April 20, 1977.

117. Letter from D. R. Patterson (TVA) to J. McFarland (B&W), "Transmitting Report by C. Michelson entitled, 'Decay Heat Removal During a Very Small Break LOCA for a B&W 205-Fuel Assembly PWR,'" April 1978.

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119. Byron Jackson Topical Report, "Performance and Safety Features of the Byron Jackson Primary Reactor Coolant Pump for Light Water Reactors," BJNPQ-101, First Draft, dated March 1978.

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16. ABSTRACT (200 words or less)

Slow system depressurization resulting from small break loss-of-coolant accidents (LOCAs) in the reactor coolant system have not, until recently, received detailed analytical study comparable to that devoted to large breaks. Following the TMI-2 accident, the staff had a series of meetings with Babcock & Wilcox (B&W) and the B&W licensees. The staff requested that B&W and the licensees: (1) systematically evaluate plant response for small break loss-of-coolant accidents; (2) address each of the concerns documented in the Michelson Report; (3) validate the computer codes used against the TMI-2 accident; (4) extend the break spectrum analysis to very small breaks, giving special consideration to failure of pressurizer valves to close; (5) analyze degraded conditions where AFW is not available; (6) prepare design changes aimed at reducing the probability of loss-of-coolant accidents produced by the failure of a PORV to close; and (7) develop revised emergency procedures for small breaks. This report describes our review of the generic analyses performed by B&W based on the requests stated above.

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