UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 8, 1983

DMB 016

Dockets Nos. 50-460, 50-438, 50-439,
50-329, 50-330, 50-302, 50-313,
50-312, 50-269/270/287, 50-289, 50-346

SUBJECT: SUMMARY OF MEETING WITH BABCOCK & WILCOX (B&W) REGULATORY RESPONSE GROUP CONCERNING SMALL BREAK LOCA PROCEDURES AND MAINTAINING PROPER STEAM GENERATOR WATER LEVEL ON FEBRUARY 23, 1983

Introduction

The meeting was held in Bethesda, MD on February 23, 1983 at the request of the NRC staff to discuss potentially significant information identified during the staff's review of the GPU/B&W lawsuit trial transcript. Specifically, two witnesses at the trial expressed concerns related to transients and accidents in which decay heat removal by the steam generators (SG) is relied upon. The specific concerns are (1) the adequacy of emergency operating procedures to assure that a sufficient condensing surface would be established in the SG under all design basis conditions for which decay heat removal by the SG is required and (2) the ability to establish an effective condensing surface at the elevation of the emergency feedwater (EFW) sparger in light of data which shows limited feedwater penetration into the tube bundle from the EFW sparger ring. Enclosure 1 is the meeting attendance list. Enclosure 2 is a staff memorandum describing the concerns. Enclosure 3 is the B&W Regulatory Response Group presentation.

Discussion

To evaluate the significance of the concerns raised by the staff, the B&W Regulatory Response Group (RRG): (1) reevaluated EFW spray effectiveness; (2) reviewed SG level requirements during a small-break loss of coolant accident (SBLOCA); and (3) reviewed SBLOCA-related operating procedures and guidelines.

With respect to EFW spray effectiveness, the RRG provided the results of laboratory and plant tests intended to demonstrate effectiveness even though the EFW spray does not fully wet all tubes in the upper regions of the SG. The RRG also benchmarked code data against observed plant transients and reviewed TMI-2 accident data to support its conclusion that the SG is able to remove core decay heat under SBLOCA conditions via EFW spray.

OFFICE ▶							
SURNAME ▶	8303170136 830308 PDR ADOCK 05000269						
DATE ▶	P PDR						

The RRG reviewed SBLOCA analyses to determine whether 95% on the operating range is the proper SG level in accordance with current emergency procedures to assure plant safety. The review showed that 95% is the proper SG level for lowered loop plants. (Because of a raised loop configuration, 93 inches on the startup range is the proper level at Davis-Besse, to maintain adequate primary to secondary heat transfer). The review also showed that, for isolatable SBLOCAs; (1) if proper SG level is maintained, then one high-pressure injection pump can maintain core cooling; and (2) approximately one hour is available to start raising SG level to 95%.

The RRG reviewed the SBLOCA-related procedures for the plants listed in Enclosure 3 with respect to the staff's concerns. The review confirmed the adequacy of the current emergency procedures, operator training; and that other procedures to which the operator may be directed during the event contain appropriate guidance to assure proper SG cooling.

Conclusion

Based upon the general information presented at the meeting, and upon specific technical information presented to cognizant reviewers at an earlier meeting, it appears that the staff's concerns have been adequately addressed by the RRG. A final decision as to whether further action is required will be made when the staff completes its review of the report that the RRG plans to submit by 3/2/83.

James Van Vliet

James Van Vliet, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
As Stated

cc w/enclosures:
See next page

OFFICE	ORB#4:DL						
SURNAME	JVan Vliet:cf						
DATE	3/9/83						

MEETING SUMMARY DISTRIBUTION

Licensee:

*Copies also sent to those people on service (cc) list for subject plant(s).

Docket File
NRC PDR
L PDR
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Project Manager-JVan Vliet
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ELJordan, IE
JMTaylor, IE
ACRS (10)
NSIC

NRC Meeting Participants:
RPurple
BSheron
GHolahan
WHouston
BYoungblood
DZiemann
BNewlin
RHernan
WJensen
MZeftawy
HOrnstein
ADe Agazio
SBryan
MKeane

ATTENDANCE -NRC/B&W RRG

MEETING REGARDING B&W/GPU TRIAL TESTIMONY

R. Purple	NRR/DL
J. Stolz	NRR/DL
G. Lainas	NRR/DL
B. Sheron	NRR/DSI
G. Holahan	NRR/DL
W. Houston	NRR/DSI
D. H. Roy	B&W
R. Rodriguez	SMUD
N. Rutherford	DPC
D. Howard	AP&L
R. Wilson	GPU
R. Crouse	TECo
B. Youngblood	NRR/DL
D. Ziemann	NRR/DHFS
J. Taylor	B&W
B. Newlin	NRC/PA
R. Hernan	NRR/DL
W. Jensen	NRR/RSB
J. Carlton	B&W
T. Broughton	GPU
N. Trikouros	GPU
M. Ross	GPU
M. Zeftawy	NRC/AEOD
H. Ornstein	NRC/AEOD
G. Wallis	Creale R&D
E. Wallace	GPU
T. Murray	TECo
A. De Agazio	NRR/DL
G. Westafer	FPC
S. Miner	NRR/DL
J. Van Vliet	NRR/DL
S. Bryan	NRR/DHFS
M. Keane	NRR/RSB
B. Short	B&W

Enclosure 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FEB 18 1983

MEMORANDUM FOR: Darrell Eisenhut, Director, Division of Licensing

FROM: Roger Mattson, Director, Division of Systems Integration
Hugh Thompson, Director, Division of Human Factors Safety

SUBJECT: BOARD NOTIFICATION

The purpose of this memorandum is to request that you notify licensing boards associated with reactors designed by Babcock and Wilcox of new and relevant information which has recently come to our attention. A description of this information is provided in the enclosure.

The staff is presently in the process of evaluating this information to determine its safety significance and relevance. In particular, we are evaluating how the new information affects our assessments of the ability of B&W-designed reactors to achieve and maintain natural circulation using the steam generators for transients and accidents for which decay heat removal by the steam generator is required. We anticipate completing our evaluation within a few weeks, depending on whether detailed computer analyses are needed, and, if so, the extent of analysis necessary.

Roger J. Mattson

Roger J. Mattson, Director
Division of Systems Integration

Hugh L. Thompson, Jr.

Hugh L. Thompson, Director
Division of Human Factors Safety

Enclosure:
As stated

cc: W. Dircks
V. Stello
H. Denton
T. Speis
D. Ziemann
J. Stolz
R. Purple
M. Keane
H. Sullivan
G. N. Lauben
W. Jensen
R. Minogue, RES
O. Bassett, RES
D. Ross, RES
S. Bryan

ENCLOSURE

Background

At the direction of the Commission, the staff has undertaken a review of the trial transcript resulting from the GPU-B&W lawsuit. The purpose of this review is to determine if any potentially significant information was identified during the trial. The staff has recently identified one instance of such information. See attachment 1.

Problem

The information identified relates to transients and accidents in which decay heat removal by the steam generators is relied upon. During the trial, testimony by Dr. R. Lahey of Rensselaer Polytechnic Institute (RPI) and Dr. G. Wallis of Dartmouth College identified two concerns. These are (1) the adequacy of emergency operating procedures to assure that a sufficient condensing surface would be established in the steam generators under all design basis conditions for which decay heat removal by the steam generators was required and (2) the ability to establish an effective condensing surface at the elevation of the auxiliary feedwater sparger ring in light of new data which shows limited penetration into the tube bundle of feedwater entering the steam generator from the emergency feedwater sparger ring.

The first concern was raised by Dr. Lahey. It deals with procedures and relates to whether or not the operators have sufficient instructions and training to assure that they will raise the secondary level of the steam generator to 95 percent of the operating level under all conditions necessary to assure natural circulation. Following the TMI-2 accident, it was learned that the then current procedures instructed operators to raise the secondary level to 50 percent of the operating range. Under certain circumstances, it was possible to postulate that natural circulation would not be reestablished with the secondary level at 50 percent. Subsequently, it was determined that raising the level to 95 percent of the operating range would assure natural circulation if the RCS was saturated. However, because of overcooling considerations, it is not desirable to raise the level to 95 percent for all cases of loss of forced circulation. Thus, specific plant circumstances dictate the appropriate steam generator level and the manner to achieve this level. The operating procedures and training to describe the correct actions are, therefore, important to the issue.

A discussion of this issue was presented in NUREG-0565 ("Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock and Wilcox Designed 177-FA Operating Plants," dated January 1980) and is provided in ˙ttachment 2. A copy of the relevant sections of Dr. Lahey's testimony is provided as attachment 3.

The second concern was raised by Dr. Wallis. It involves recent test data from the Alliance Research Center which is reported to show that auxiliary feedwater entering from the sparger ring does not penetrate into the steam generator tube bundle but only contacts a small percentage of the tubes. This has the effect of lowering the elevation of the effective condensing surface in the steam generator. Previous analysis models assume good penetration of auxiliary feedwater spray into the tube bundle but recent B&W models may account for the new data. This data was submitted to the staff by the B&W Owners Group as part of the revised small break ECCS model to meet the requirements of TMI Action Plan Item II.K.3.30.

The staff will meet with representatives of B&W plants and B&W on February 23, 1983, to obtain their views on these two issues. We will review the status of procedures in the operating plants and any analysis that may pertain before reporting back to the boards on our disposition of the two concerns.

ATTACHMENT 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FEB 17 1983

MEMORANDUM FOR: Chairman Palladino
Commissioner Gilinsky
Commissioner Ahearne
Commissioner Roberts
Commissioner Asselstine

FROM: William J. Dircks
Executive Director for Operations

SUBJECT: REVIEW OF B&W-GPU TRIAL RECORD

During the staff's review of the trial record requested by your memorandum of December 29, 1982, potentially significant information has been identified. This information was discussed by Dr. Lahey and Dr. Wallis and concerns the technical adequacy of small break loss of coolant and natural circulation procedures proposed for use following authorization to restart TMI-1 and may be applicable to other B&W reactors. This information has been referred to NRR for evaluation of the technical aspects of this issue. The evaluation of this issue may put a new or different light upon an issue considered in the current restart proceeding and result in Board notifications in accordance with NRR Office Letter No. 19, Rev. 2.

William J. Dircks
Executive Director
for Operations

cc: OGC
OPE
SECY
H. Denton, NRR
V. Stello, Jr., DEDROGR

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 (~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

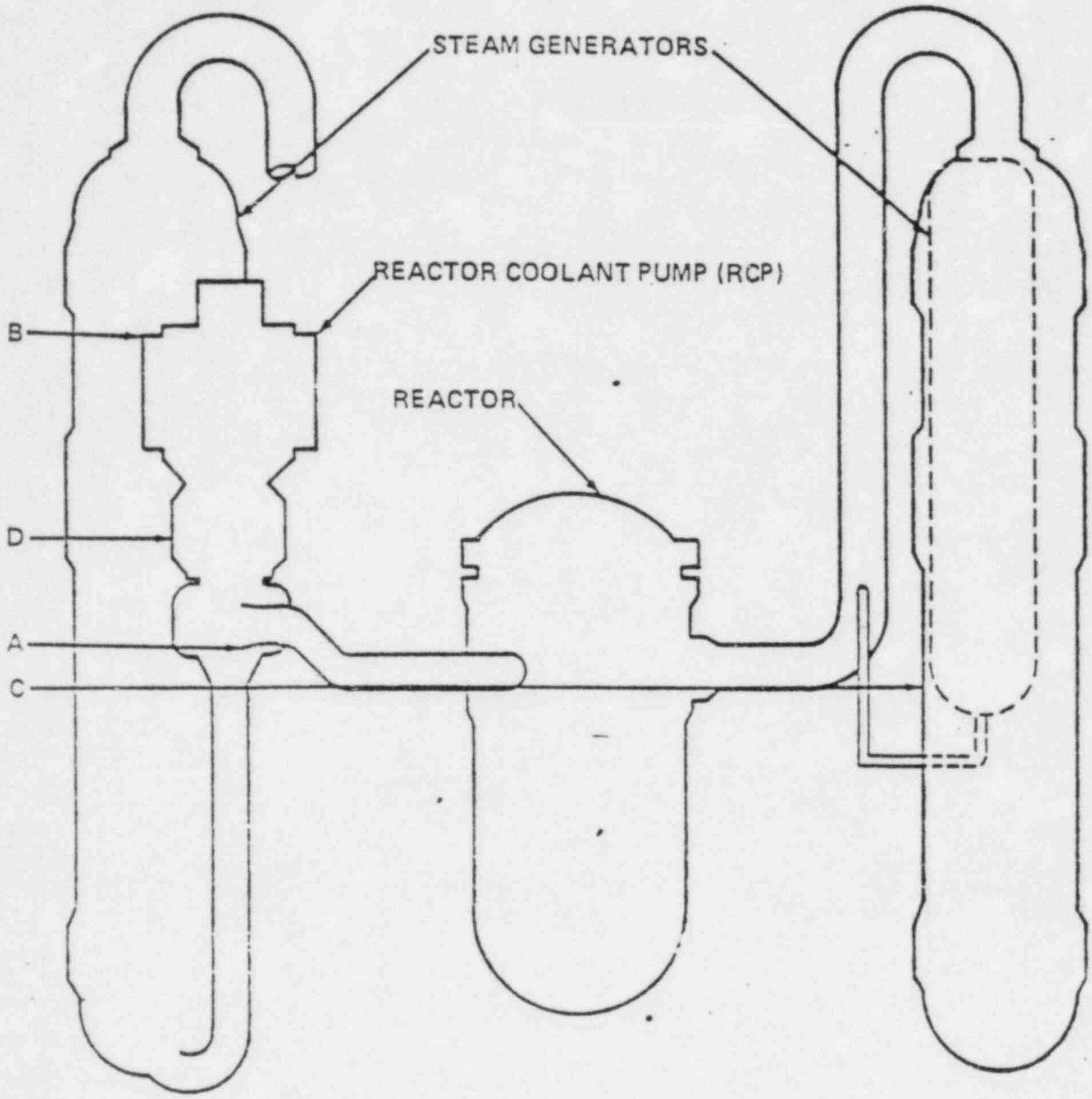
During the course of the accident at TMI-2, the operators tripped the last operating RCP 101 minutes into the accident. Immediately after the RCPs were stopped, reactor coolant temperatures in the hot leg piping were observed to rapidly increase. It was during this period that the majority of the damage 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

Enclosure 3

B&W O.G. RRG MEETING

FEBRUARY 23, 1983

AGENDA

I. INTRODUCTION

II. ACTIONS TAKEN BY B&W O.G.

III. EMERGENCY FEEDWATER EFFECTIVENESS

IV. STEAM GENERATOR COOLING REQUIREMENTS FOR SBLOCA

V. OPERATING PROCEDURE REVIEWS

VI. CONCLUSIONS

NRC CONCERNS

- o STEAM GENERATOR HEAT REMOVAL UNDER SBLOCA TO ASSURE CORE COOLING

- o PROCEDURE AND TRAINING ADEQUACY TO ASSURE REQUIRED SG HEAT REMOVAL

B&W OWNERS GROUP ACTIONS

- RE-EVALUATION OF EFW EFFECTIVENESS
- REVIEW OF STEAM GENERATOR LEVEL REQUIREMENTS
- REVIEW OF SBLOCA-RELATED OPERATING PROCEDURES/GUIDELINES
- DOCUMENT RESULTS TO NRC STAFF (3-2-83)

PERSPECTIVE OF CONCERNS

- RANGE OF BREAK SIZE - 0.005 - 0.04 FT^2
- MUST HAVE SINGLE FAILURE IN HPI (I.E., NO CONCERN WITH 2 HPI)
- OTHER "LICENSING" CONSERVATISMS INCLUDED IN ANALYSIS

SUPPORT FOR EFW SPRAY EFFECTIVENESS USED IN SBLOCA ANALYSES:

- INSTRUMENTED LABORATORY TESTS
- VISUAL LABORATORY TESTS
- SPECIAL INSTRUMENTED TESTS AT OCONEE
- CORRELATION BENCHMARKED AGAINST PLANT TRANSIENTS
- REVIEW OF DATA FROM TMI-2 ACCIDENT

Figure 2-3 OTSG TEMPERATURE SENSOR LOCATIONS

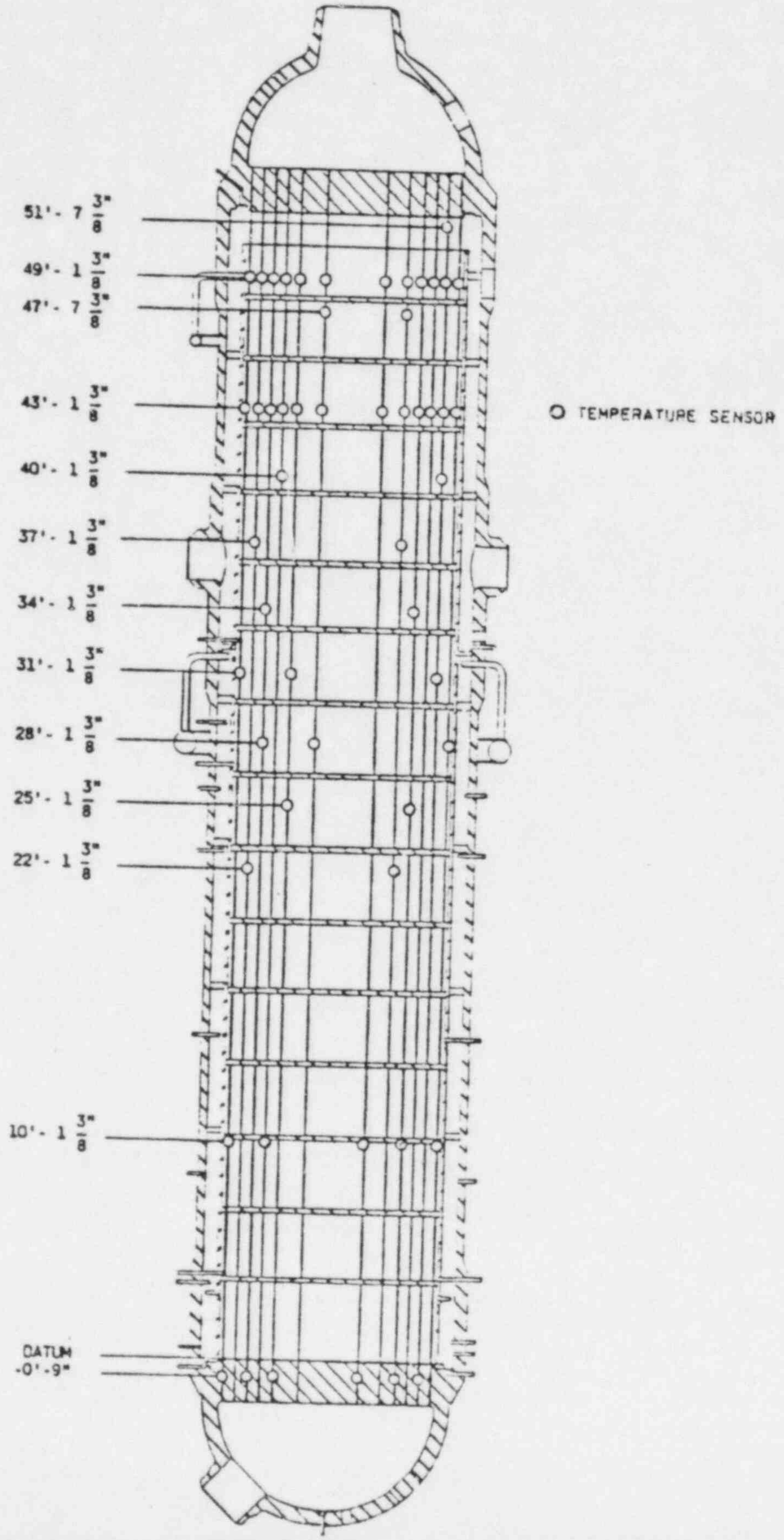


Figure 2-4 IDENTIFICATION OF INSTRUMENTED OTSG TUBES

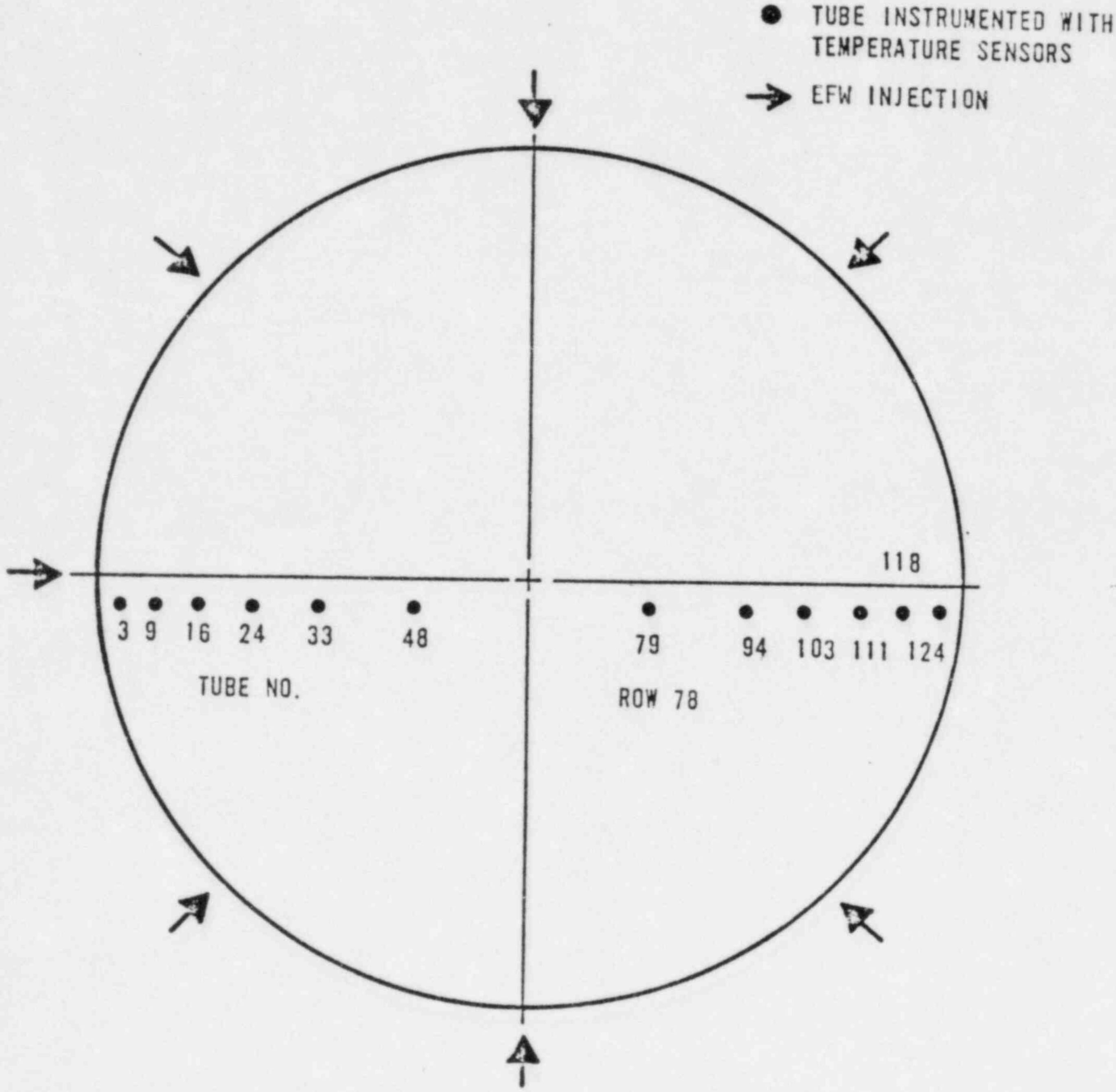


Figure 2-7 EFW AXIAL WETTING PROFILE

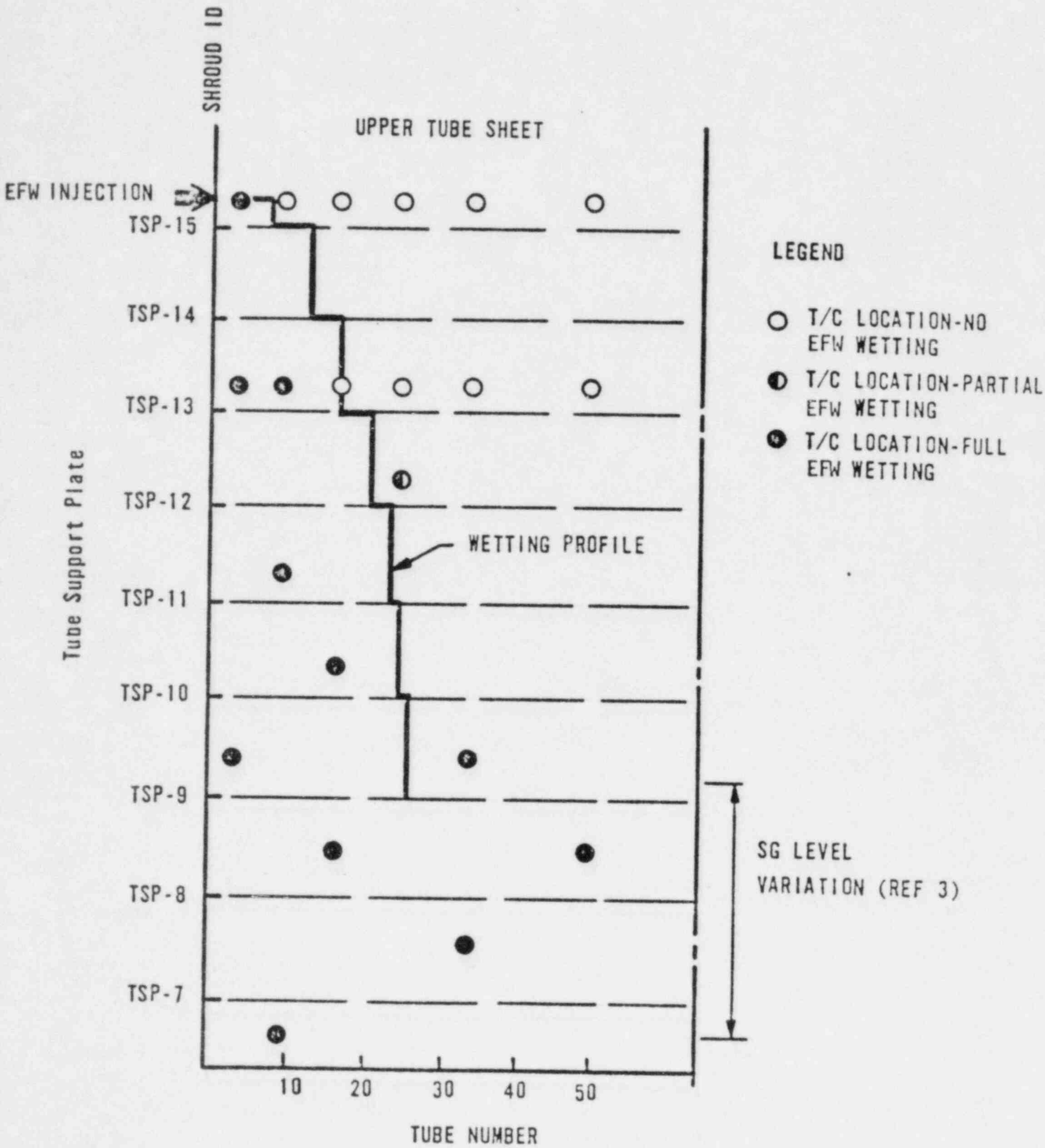


Figure 2-10 EMERGENCY FEEDWATER PENETRATION (EFW) PROFILES

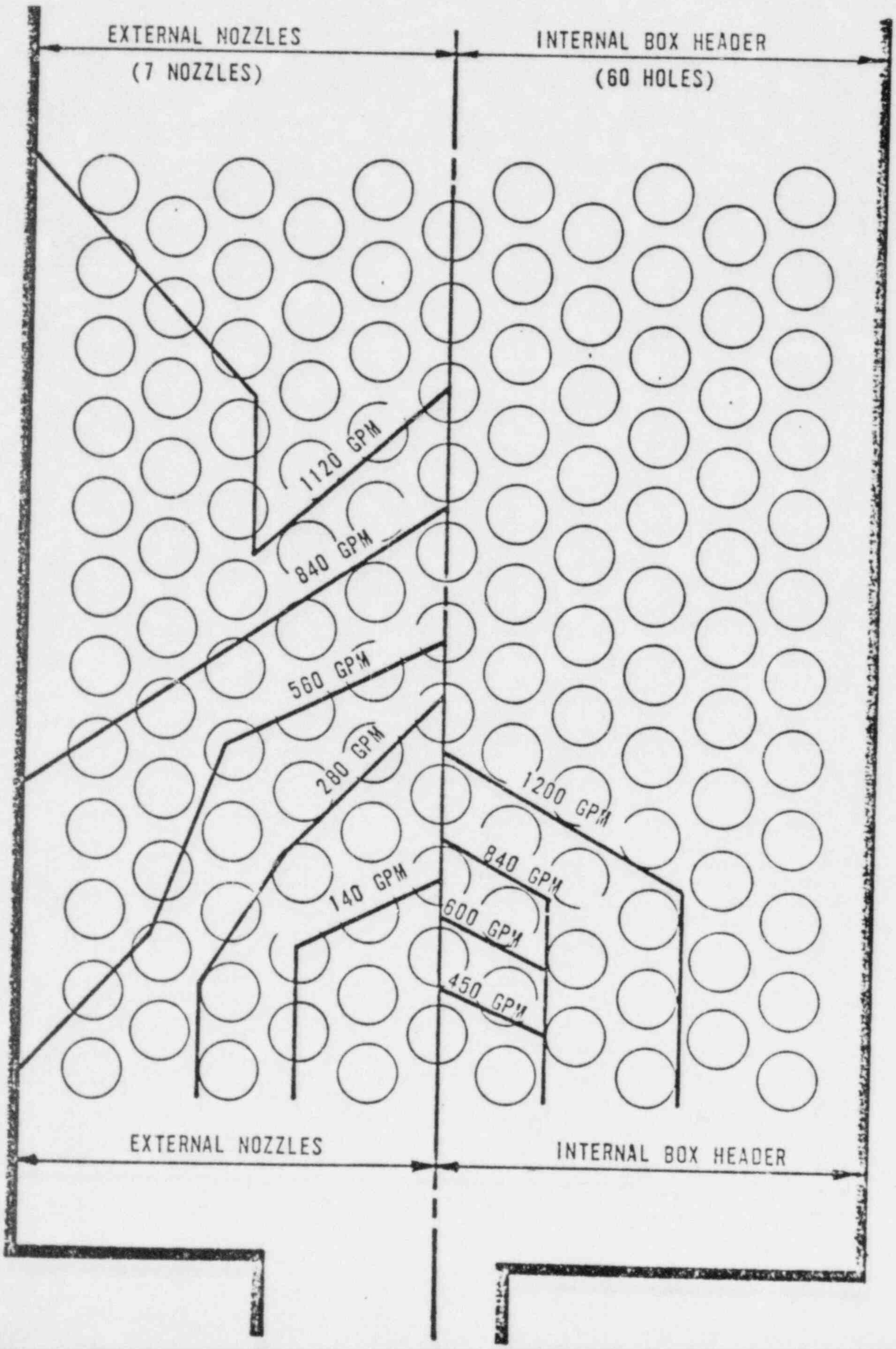


Figure 2-11 EFW WETTING EFFECTIVENESS

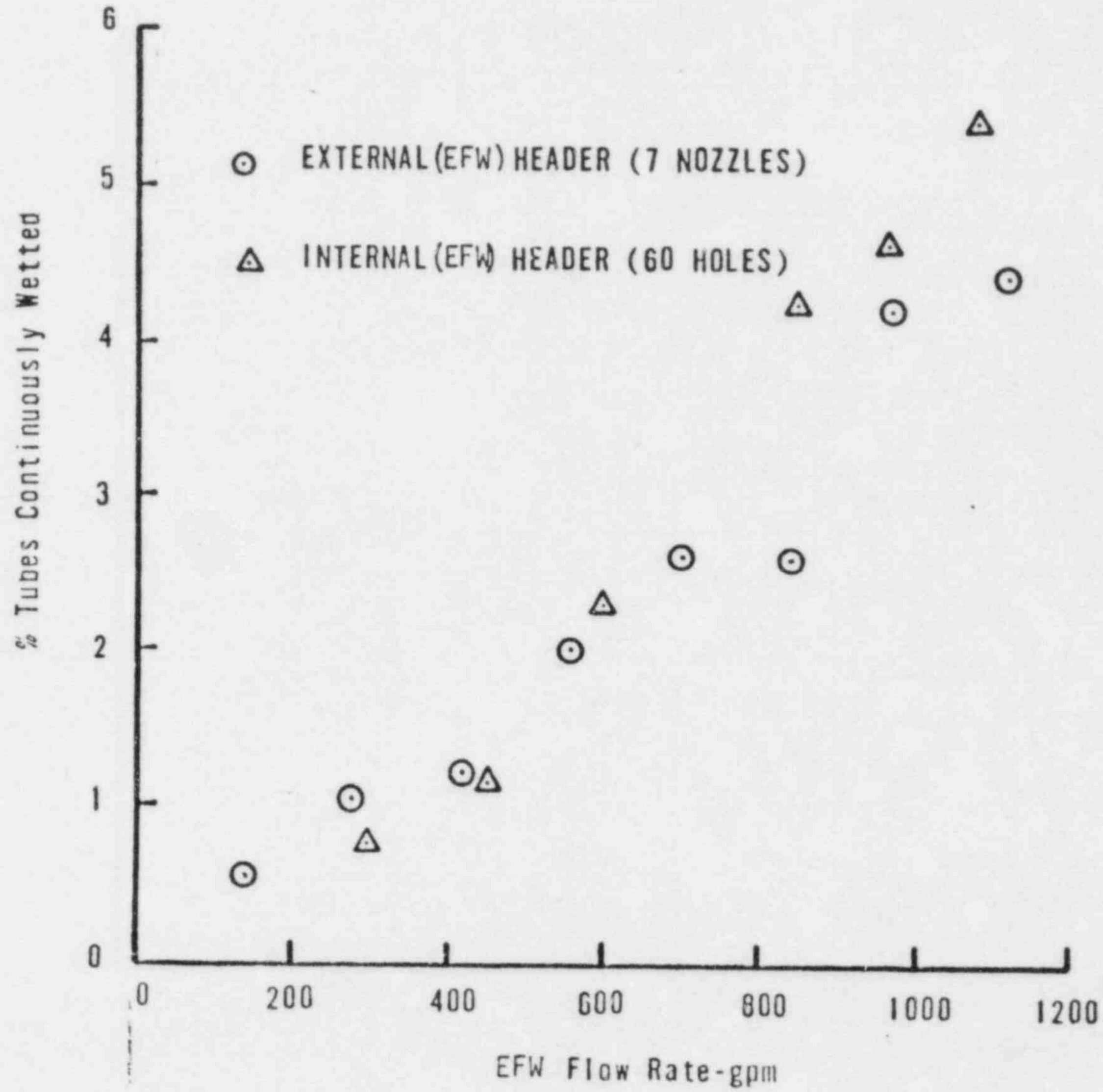


Figure 2-20 RCS Pressure Vs Time Comparison

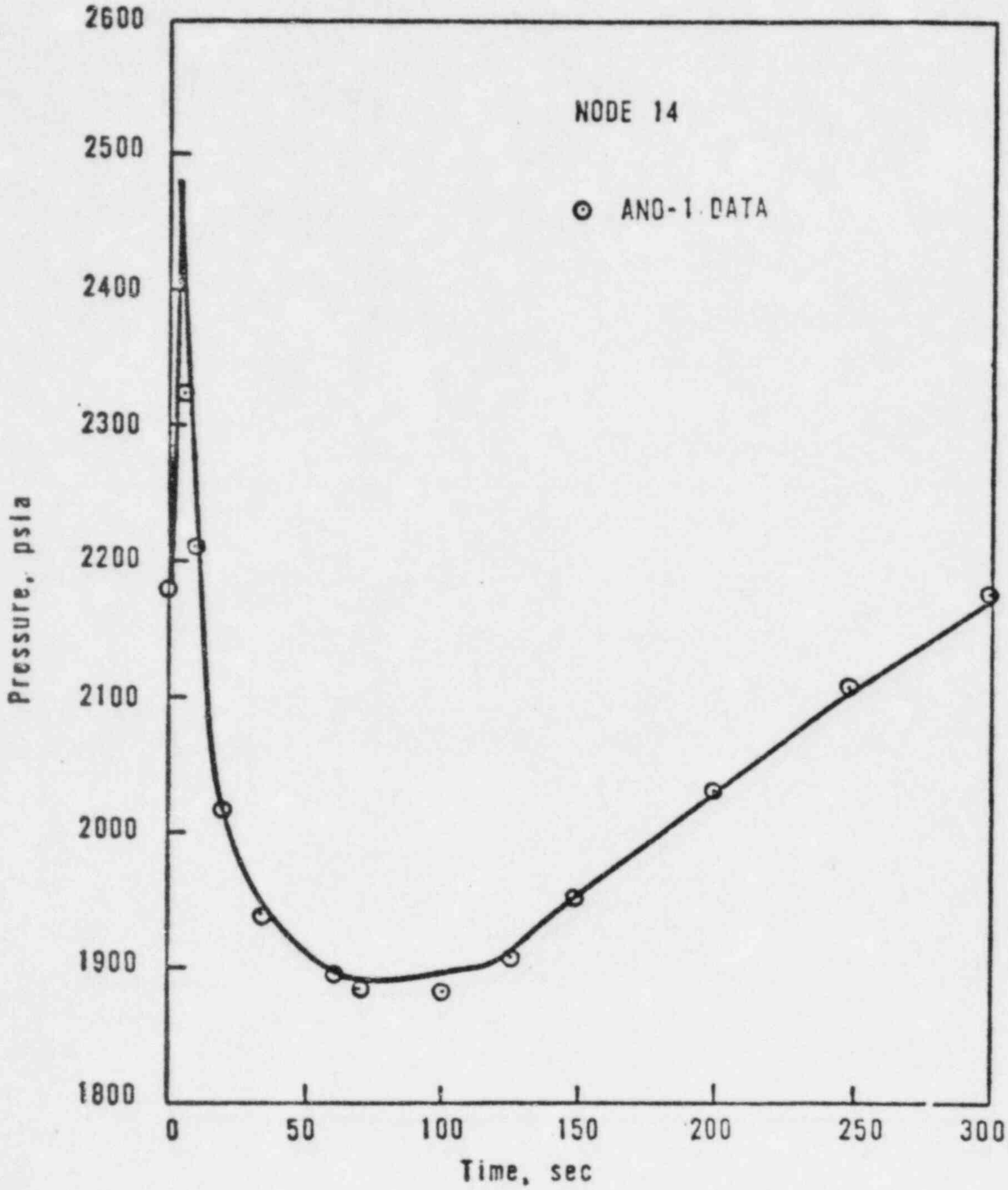


Figure 2-21 Cold Leg Temperature Vs Time Comparison

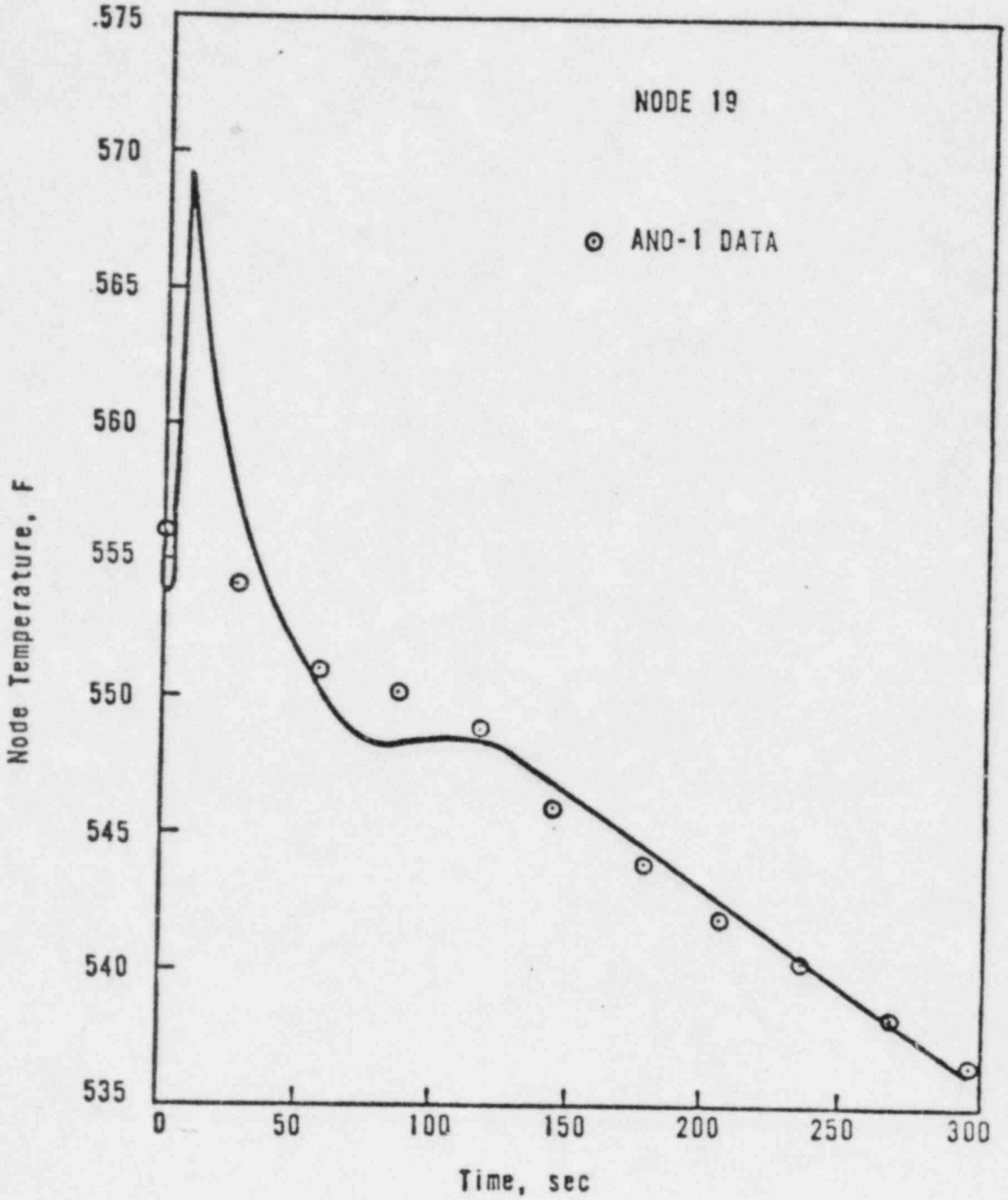


Figure 2-22 Hot Leg Temperature Vs Time Comparison

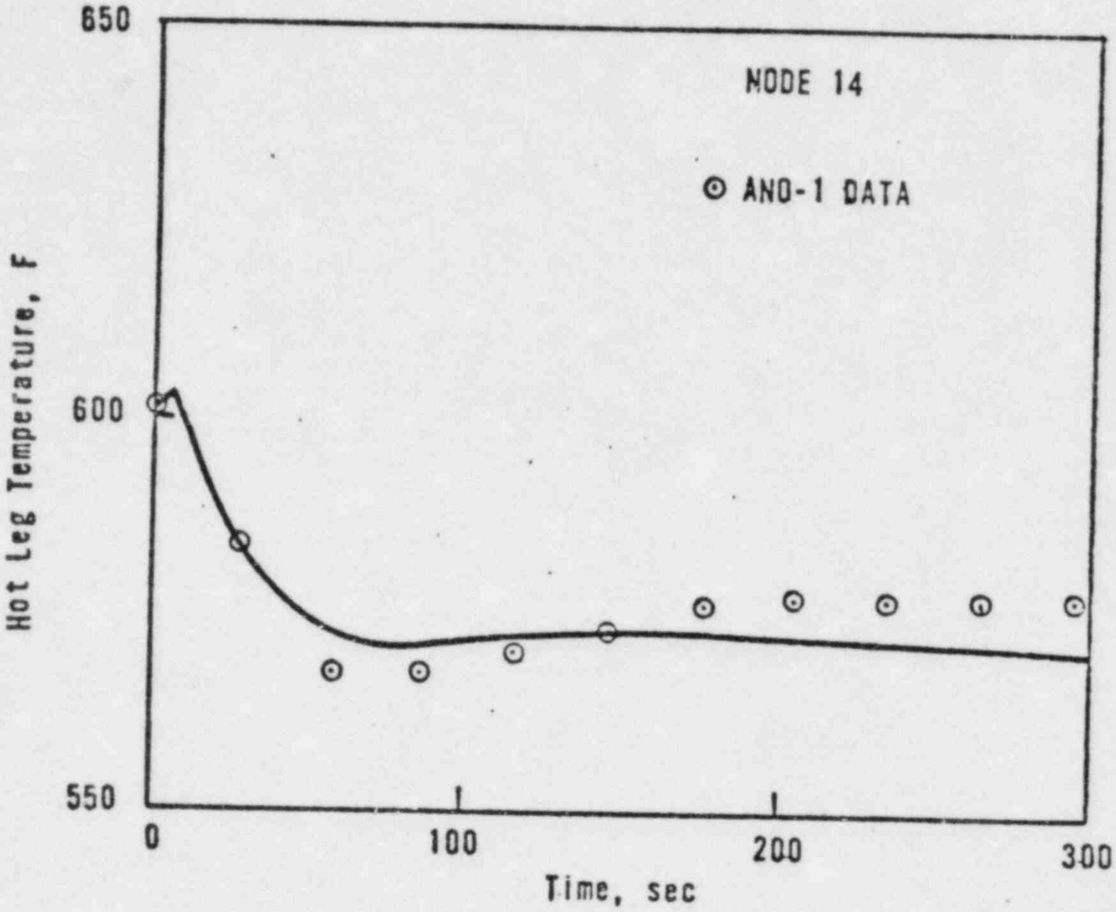


Figure 2-23 Pressurizer Level Vs Time Comparison

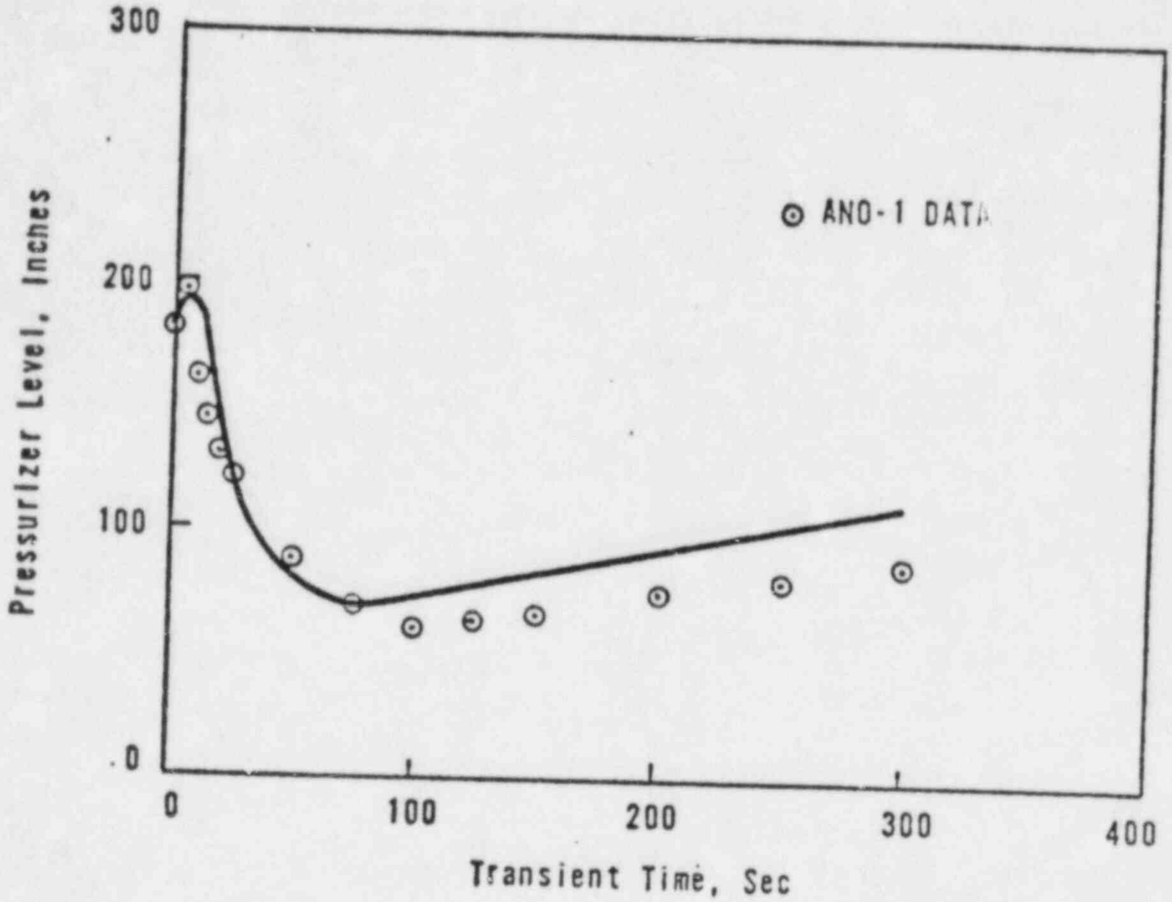
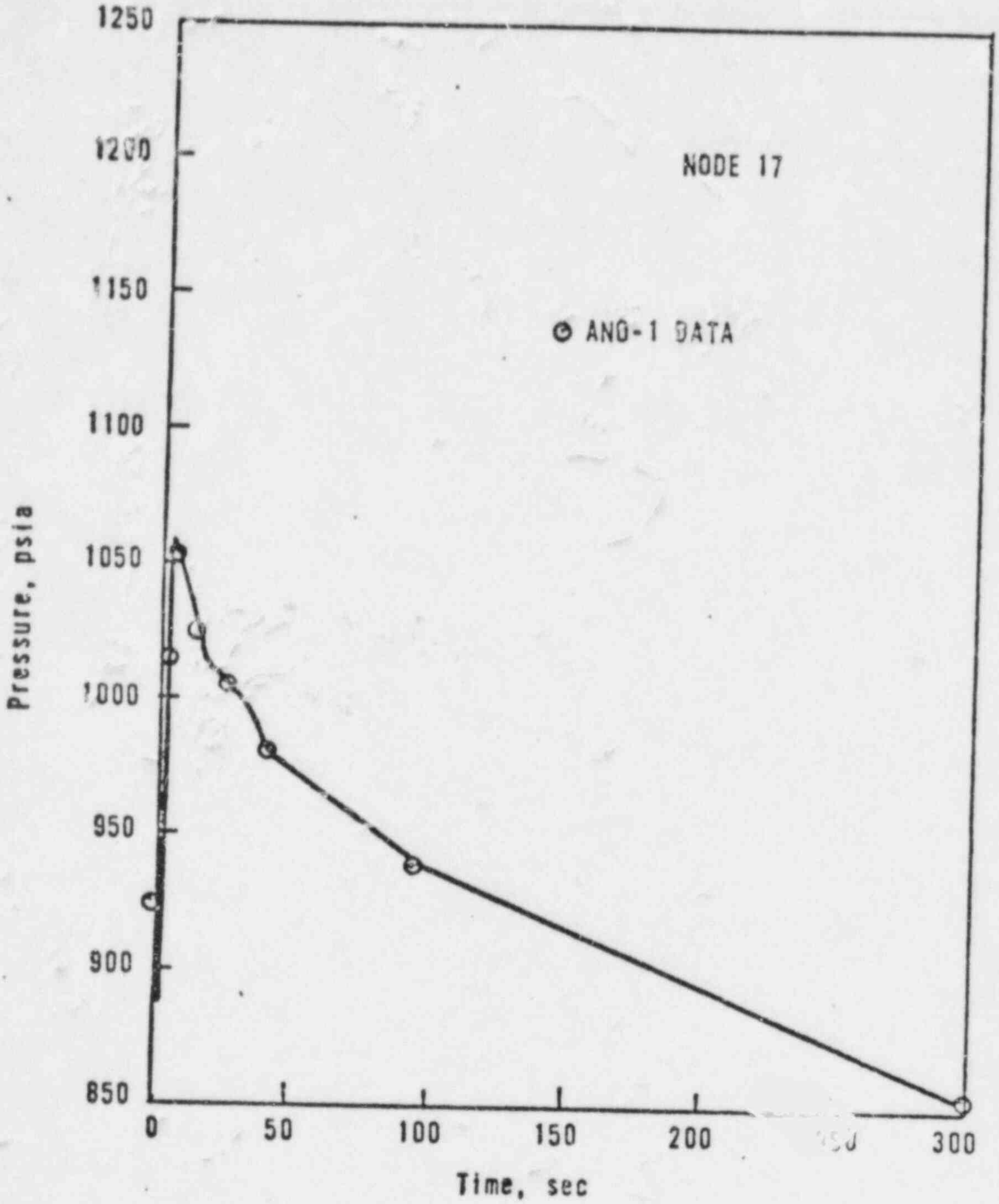


Figure 2-24 Steam Generator Pressure Vs Time Comparison



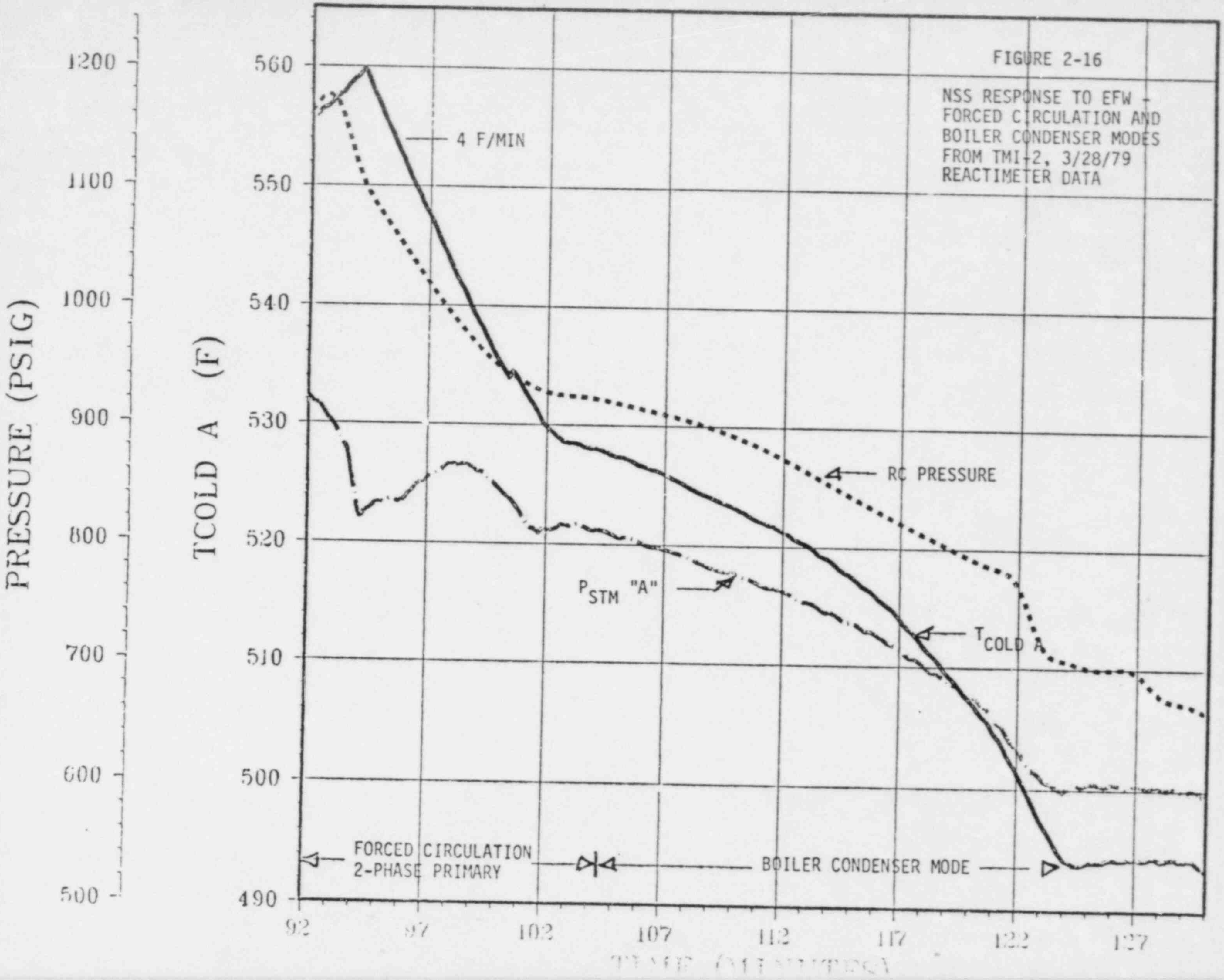


FIGURE 2-16

NSS RESPONSE TO EFW - FORCED CIRCULATION AND BOILER CONDENSER MODES FROM TMI-2, 3/28/79 REACTIMETER DATA

STEAM GENERATOR LEVEL

REQUIREMENTS

DURING SBLOCA

TOPICS

1. SG LEVEL REQUIREMENTS FOR SBLOCA
2. ANALYSIS BASIS FOR SG LEVEL CONTROL
3. ISOLATABLE SBLOCA's

SG LEVELS REQUIRED BY SBLOCA ANALYSES

1. AT 95% ON THE OPERATE RANGE FOR LOWERED LOOP PLANTS.

2. AT 93-INCHES INDICATED ON STARTUP RANGE FOR DAVIS BESSE IF ADEQUATE PRIMARY TO SECONDARY HEAT TRANSFER IS MAINTAINED.

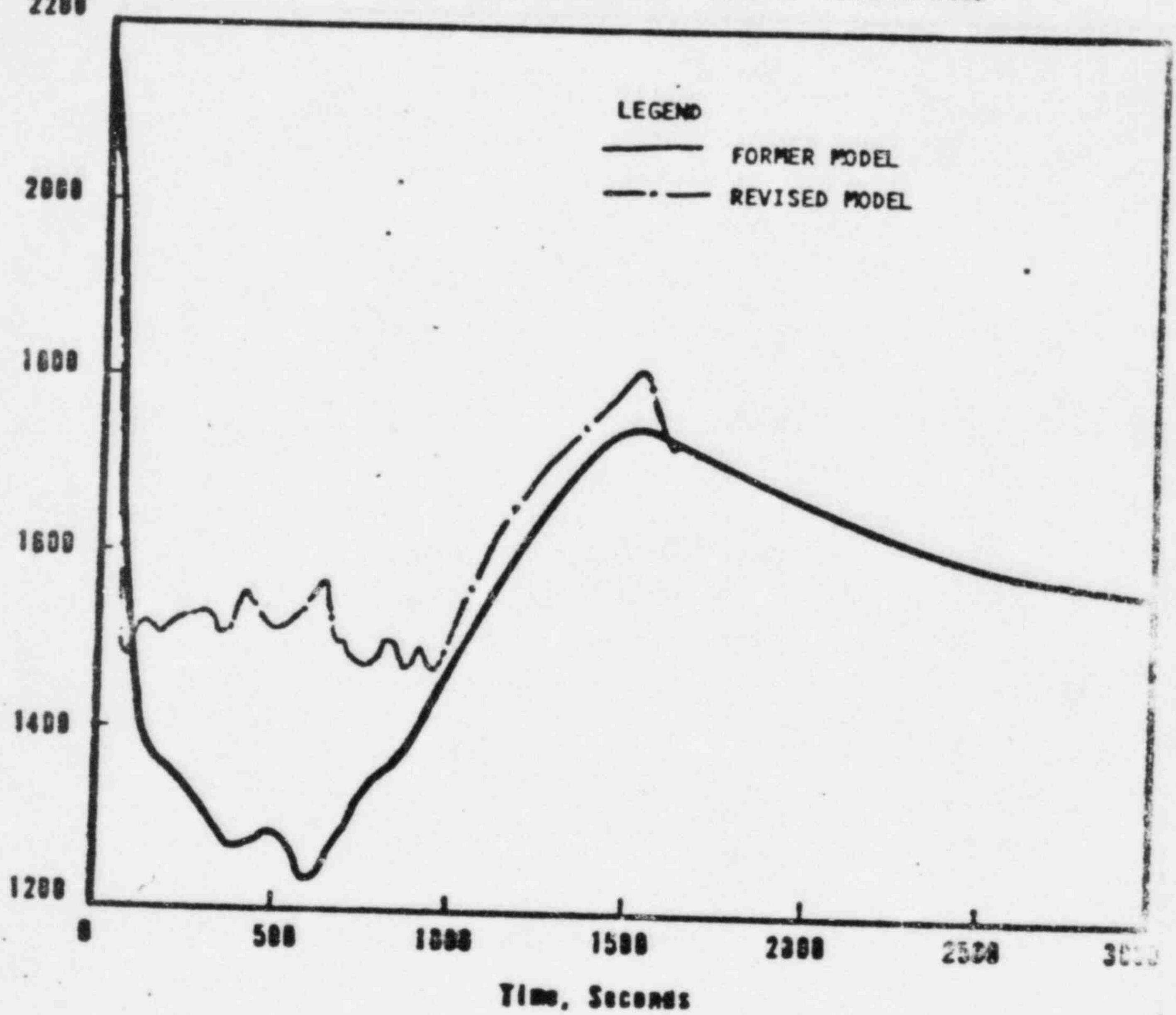
3. AT 95% ON THE OPERATE RANGE FOR DAVIS BESSE IF ADEQUATE PRIMARY TO SECONDAR HEAT TRANSFER IS LOST.

ANALYSIS BASIS FOR SG LEVEL CONTROL

- HEAT TRANSFER ANALYSIS OF SG PERFORMED
  - ADEQUATE HEAT REMOVAL BY EITHER EFW SPRAY OR SG LEVEL
  - NO CORE UNCOVERY PREDICTED
- PREVIOUS SBLOCA ANALYSES REVIEWED
  - EFW PENETRATIONS EFFECTS TIMING AND RESPONSE
  - COMPARISON WITH UPGRADED MODEL SHOWS GOOD AGREEMENT

Pressurizer Pressure vs. Time

Former & Revised Models for 177 Lowered Loop



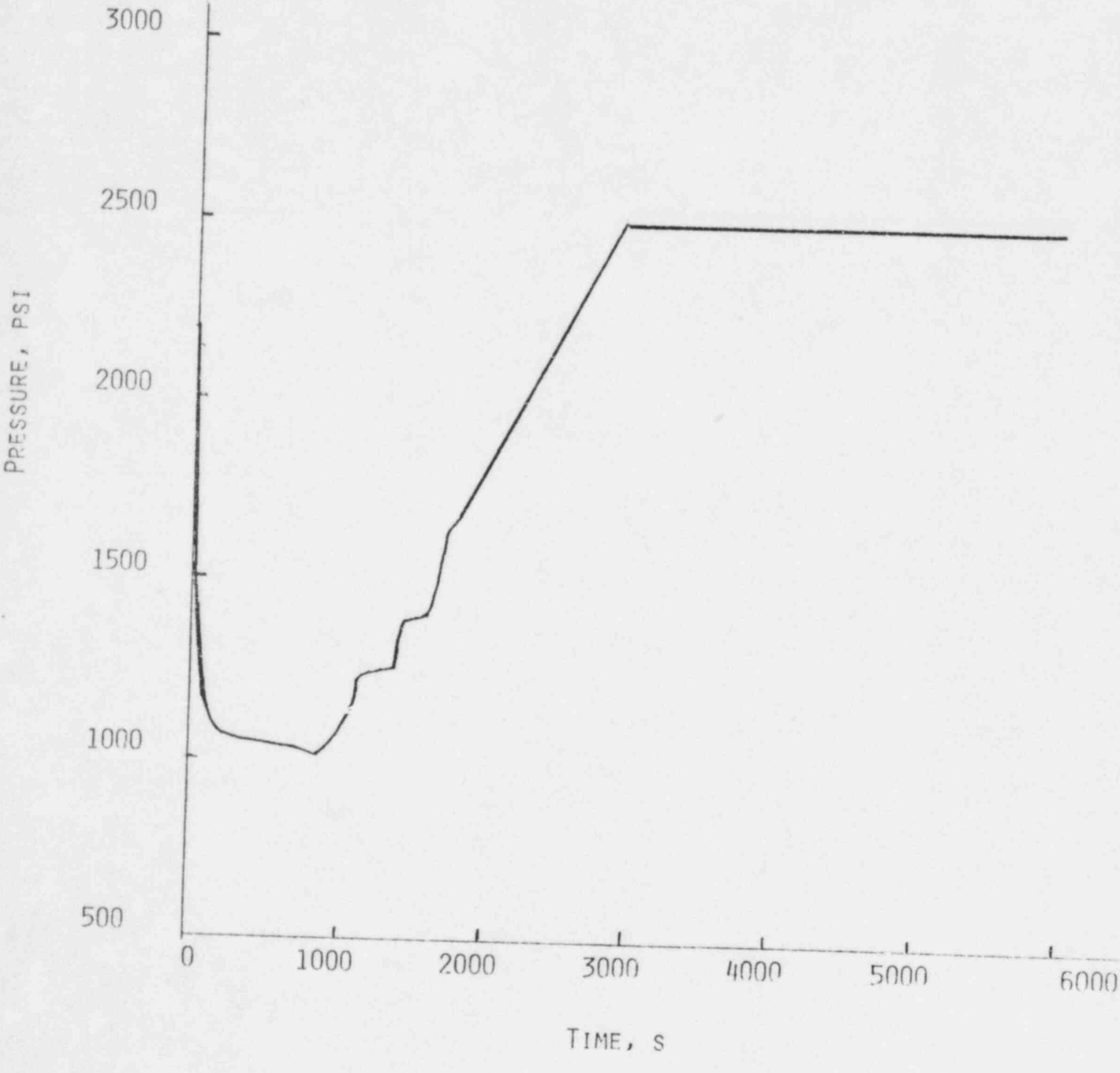
ISOLATABLE SBLOCA's

- WITH MAINTENANCE OF SG LEVEL REQUIREMENTS IN SBLOCA PROCEDURES ONE HPI CAN MAINTAIN CORE COOLING.

- APPROXIMATELY 1 HOUR AVAILABLE TO START RAISING SG LEVEL TO 95%.

- DB-1 HAS ADEQUATE HEAT REMOVAL CAPABILITY WITHOUT RAISING SG LEVEL.

RC PRESSURE vs. TIME FOR
ISOLATED BREAK
(SPRAY LINE)



OPERATING PROCEDURE REVIEW

SBLOCA-RELATED PROCEDURES REVIEWED FOR:

- DAVIS BESSE-1
- OCONEE 1,2,3
- CRYSTAL RIVER 3
- TMI-1
- RANCHO SECO
- ARKANSAS NUCLEAR ONE

OPERATING PROCEDURE REVIEW CRITERIA

1. DOES GUIDANCE EXIST TO ESTABLISH AND MAINTAIN EMERGENCY SG LEVEL WHEN SBLOCA CONDITIONS ARE INDICATED?

2. DOES GUIDANCE EXIST REGARDING HIGH PRESSURE INJECTION REQUIREMENTS WHEN LOSS OF SUBCOOLING MARGIN OCCURS?

3. DOES GUIDANCE EXIST FOR VERIFYING PRIMARY TO SECONDARY HEAT TRANSFER?

4. IF, AFTER A BREAK IS ISOLATED (WITH THE RCS STILL SATURATED), THE OPERATOR BY PROCEDURE, GOES TO THE NATURAL CIRCULATION PROCEDURE OR ANY OTHER PROCEDURE, IS HE STILL REQUIRED TO ESTABLISH AND MAINTAIN EMERGENCY SG LEVEL?

RESULTS OF REVIEW

1. THE REVIEW OF PROCEDURES AND RELATED TRAINING CONFIRMS THAT SG EMERGENCY LEVEL OF 95% ON OPERATE RANGE WILL BE ESTABLISHED FOR INDICATED SBLOCA.

2. RULES AND TRAINING ARE IN PLACE AT ALL UTILITIES TO ASSURE THAT HPI WILL BE MAINTAINED UNTIL ADEQUATE SUBCOOLED MARGIN IS ATTAINED AND SUSTAINED.

3. PROCEDURES AND TRAINING ARE IN PLACE AT ALL UTILITIES TO PERMIT THE OPERATOR TO CONFIRM THAT SG COOLING HAS BEEN ACHIEVED OR TO RECOGNIZE WHEN IT IS LOST AND TAKE APPROPRIATE ACTION.

4. THE REVIEW OF PROCEDURES AND RELATED TRAINING CONFIRMS THAT IF THE OPERATOR IS DIRECTED TO OTHER PROCEDURES, APPROPRIATE GUIDANCE EXISTS TO ASSURE PROPER SG COOLING.

B&W OG CONCLUSIONS AFTER REVIEWING

SBLOCA

NRC STAFF SBLOCA CONCERNS

- THE ABILITY OF THE OTSG TO REMOVE CORE DECAY HEAT UNDER SBLOCA CONDITIONS VIA EFW SPRAY HAS BEEN DEMONSTRATED TO PROVIDE ADEQUATE CORE COOLING.

- SB OPERATING GUIDELINES CONTAIN THE APPROPRIATE REQUIREMENTS REGARDING STEAM GENERATOR LEVEL.

- EXISTING UTILITY OPERATING PROCEDURES AND OPERATOR TRAINING PROGRAMS PROVIDE SUFFICIENT GUIDANCE TO THE OPERATORS TO ASSURE THE SAFE MITIGATION OF SMALL BREAKS (ISOLATABLE AND NON-ISOLATABLE) IN THE PRIMARY SYSTEM.

- THE PROPOSED OPERATING GUIDELINES (ATOG), ALSO PROVIDE SUFFICIENT GUIDANCE TO THE OPERATOR TO ASSURE THE SAFE MITIGATION OF SMALL BREAKS (ISOLATABLE AND NON-ISOLATABLE) IN THE PRIMARY SYSTEM.