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In the Matter of
Metropolitan Edison Company
(Three Mile Island Nuclear Station, Unit No. 1)
Docket No. 50-289 (Restart)

Administrative Judges Edles, Buck and Gotchy:

Board Notification 83-21, February 18, 1983, which the Staff has provided to the Commission and to the Appeal Board in this proceeding, addresses two concerns: (1) the adequacy of emergency operating procedures to assure that a significant 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.

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Please find enclosed, for your information, a copy of B&W 77-1141270-00, "Evaluation of SBLOCA Operating Procedures and Effectiveness of Emergency Feedwater Spray for B&W-Designed Operating NSSs" (February 1983). This report, which responds to the concerns raised in BN 83-21, has just been submitted to the NRC Staff.

The second concern, identified above, is relevant to the testimony of Licensee witness Jones in response to ALAB-708 Issue No. 7. The enclosed report addresses the concern in Section 2, "Emergency Feedwater (EFW) Spray Effectiveness," and provides information which supplements the Jones testimony. Consequently, Licensee may offer that portion of the report in order to complete the record. Section 3 of the enclosed report, on "Steam Generator Level Requirements During a LOCA," relates to, but largely repeats, information already provided in the Jones testimony.

Sincerely,

Thomas A. Baxter

Thomas A. Baxter
Counsel for Licensee

TAB:jah

Enclosure

cc (w/enc.): Service List attached

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY	)	Docket No. 50-289
	)	(Restart)
(Three Mile Island Nuclear	)	
Station, Unit No. 1)	)	

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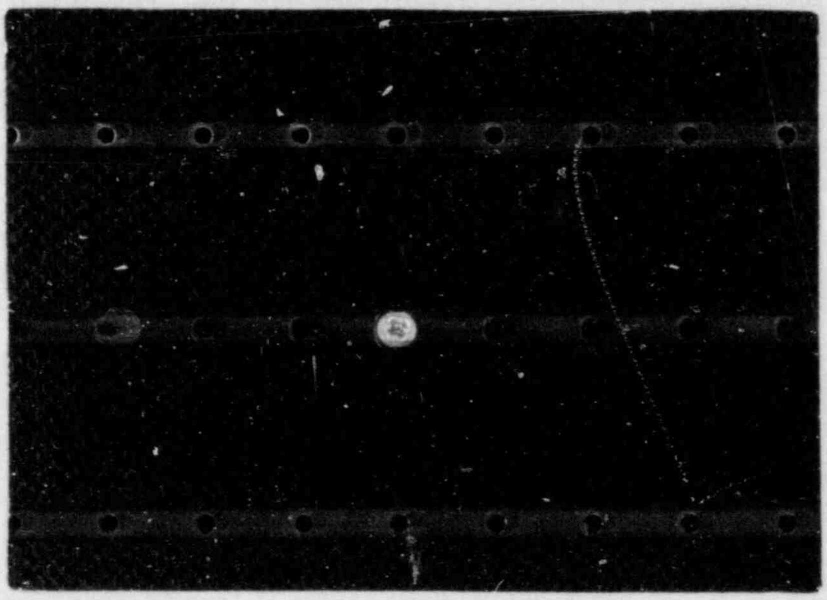

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B&W DOC. ID. 77-1141270-00

EVALUATION OF SBLOCA OPERATING PROCEDURES AND EFFECTIVENESS OF EMERGENCY FEEDWATER SPRAY FOR B&W-DESIGNED OPERATING NSSS

FEBRUARY 1983

Utilities with B&W-Designed Operating NSSS

- ARKANSAS POWER & LIGHT
- DUKE POWER COMPANY
- FLORIDA POWER CORPORATION
- GPU NUCLEAR CORPORATION
- SACRAMENTO MUNICIPAL UTILITY DISTRICT
- TOLEDO EDISON COMPANY

EXECUTIVE SUMMARY

This report provides the results of a re-evaluation of two concerns which the NRC communicated to the Chairman of the B&W Regulatory Response Group (RRG) in mid-February 1983.

1. The ability of emergency feedwater (EFW) spray to provide adequate heat transfer for a range of small break LOCAs which result in the boiler/condenser mode of operation.

2. The adequacy of existing Utility operating procedures to provide sufficient guidance to the operators to assure the safe mitigation of small isolatable breaks in the primary system.

These concerns were addressed for the following plants:

ANO-1	Arkansas Power & Light Company
Oconee	Duke Power Company
CR-3	Florida Power Corporation
TMI-1	GPU Nuclear Corporation
Rancho Seco	Sacramento Municipal Utility District
Davis-Besse	Toledo Edison Company

This re-evaluation considered engineering calculations, test results, actual plant transient data, small break operating guidelines, in-place operating procedures, and operator training. The following conclusions were presented to the NRC staff on February 23, 1983:

1. The EFW spray is effective under small break LOCA conditions and adequate core cooling is provided.

2. 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.

Other transients (in addition to the small break LOCA) were re-considered to determine if similar concerns could exist - none were found. The only event for which 95% level on the operate range (93 inches on the startup range for Davis-Besse) is required is a SBLOCA. The other transients considered were:

- Large break LOCA (design basis event)
- Loss of off-site power
- Steam generator tube rupture
- Steam line breaks
- Excessive feedwater flow during post reactor trip conditions
- Natural circulation cooldown

Also for completeness, the proposed abnormal transient operating guidelines (ATOG) were re-reviewed and it was concluded that they also provide sufficient guidance to the operator to assure the safe mitigation of small breaks (isolatable and non-isolatable) in the primary system.

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1.0 INTRODUCTION

1.1 Background

The NRC has recently reviewed the transcripts of the GPUN/B&W litigation testimony to determine if there is any new information of which the Staff should be aware. In reviewing the testimony of Professors Lahey and Wallis, the NRC identified two possible concerns. The Commissioners were advised of these concerns in a letter from Mr. Victor Stello dated February 17, 1983. The NRC also contacted Mr. R. P. Crouse, Chairman of the B&W Regulatory Response Group, to request that these concerns be addressed and responded to in a meeting with the NRC on February 23, 1983.

The NRC has identified two specific concerns:

a. The ability of emergency feedwater (EFW) spray to provide sufficient heat transfer for the range of break sizes in the primary system which result in the boiler/condenser mode of operation and thereby to assure that adequate core cooling is attained.

b. The extent to which existing utility operating procedures provide sufficient guidance to the operators to assure the safe mitigation of small isolatable breaks in the primary system.

In response to the Staff's request the B&W Owners Group, RRG, initiated three actions:

- o The effectiveness of EFW was re-evaluated.
- o The technical bases for steam generator (SG) level requirements during small break LOCA conditions were re-reviewed.
- o Appropriate operating guidelines and utility operating procedures were re-reviewed.

This report has been prepared to provide the NRC the results of these actions and to aid in their overall assessment of the specific concerns.

It is important to note that the small break LOCA sizes which require heat removal by the steam generators comprise a narrow size range even when based upon the conservative assumptions used in these and previous evaluations. This is further illustrated by considering that when two HPI pumps are available there are no small break LOCAs which require steam generator heat removal.

Although not identified as a specific concern, other events (in addition to the small break LOCA) which require RCS natural circulation and steam generator cooling have been evaluated for completeness. The additional events include:

- Loss of off-site power
- Steam generator tube rupture
- Steam line breaks
- Uncontrolled, excessive feedwater flow during post reactor trip conditions
- Natural circulation cooldown.

In addition, the large break LOCA (the design basis event) was also reviewed.

1.2 Conclusions

Based on the results of these reviews, the following general conclusions have been reached:

o 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.

- Adequate core cooling is provided for the design basis event (large LOCA) and for the other transients considered, which include overcooling transients. The only event for which 95% level on the operate range (93-inches on the startup range for Davis-Besse) is required is a SBLOCA.

2.0 EMERGENCY FEEDWATER (EFW) SPRAY EFFECTIVENESS

2.1 Introduction

The B&W NSS provides Emergency Feedwater (EFW) to the steam generator to provide for core and NSS cooling when main feedwater is not available, or under conditions of natural circulation.

The concern raised by the NRC staff regarding the effectiveness of EFW spray in removing heat from the steam generator has been re-reviewed. This section of the report summarizes the results of this review.

The effectiveness of EFW spray as employed in the SBLOCA analyses is supported by the following:

- Instrumented Laboratory Tests
- Visual Laboratory Tests
- Specifically Instrumented In-Plant Tests (Oconee 1)
- Plant Transient Benchmarked Correlation
- Review of Data from the TMI-2 Accident

Operating plant data and test data indicate that the EFW is an effective heat removal system for the NSS.

The remaining sections of this report consolidate information contained in the following B&W reports:

"Benchmarks for AFW (EFW) Models"
B&W Document #12-1132555-00, April 1982

"Supporting Data for AFW (EFW) Models
- Auxiliary Feedwater Axial Flow Distribution"
B&W Document #12-1132543-00, April 1982

"Supporting Data for AFW (EFW) Models
- Auxiliary Feedwater Penetrations"
B&W Document #12-1132513-00, April 1982

2.2 Discussion of EFW Effectiveness

In the once-through steam generator (OTSG) used on all 177 FA B&W plants, emergency feedwater (EFW) is injected between the upper most tube support plate (TSP) and the upper tube sheet (UTS), (see Figure 2-1, Item 20).

All 177 FA B&W plants now use the external EFW header configuration shown typically in Figure 2-2. Individual feedwater nozzles inject feedwater into the tube bundle.

When the EFW is operating, there are several heat transfer regimes in the steam generator depending on RC system conditions. On the primary side (tube I.D.), heat transfer is by: forced convection, or natural convection, or a combination of condensation and natural convection.

On the secondary side, (tube O.D.), the heat transfer is by: direct contact (steam heats cold feedwater), film boiling, and nucleate boiling.

The heat transfer process is initiated as cold water sprays on and into the steam generator tube bundle at the injection points. The principle heat transfer mechanism at and near the EFW injection point is direct contact heating of incoming feedwater to or near to saturation conditions at the existing steam pressure. The mixture of saturated or near

saturated water (feedwater plus condensed steam) collects on the wetted tubes and falls to the first tube support plate. The tube support plates provide a means for collecting and then spreading the water to locations deeper into the tube bundle. The quantity of water and the amount of tube surface covered by the feed condensate are a function of the EFW flow rate, the temperature and pressure difference conditions between the primary and secondary system. Above the waterpool, the heat transfer surface is a function of: (1) the EFW flow rate and (2) the number of tube support plates crossed before reaching the pool.

In the pool, the heat transfer surface is a function of the depth of the pool, i.e., the available heat transfer surface is that covered by saturated water extending from the lower tube sheet to the froth height.

Test data from the instrumented Oconee 1B steam generator and flow visualization tests at B&W's Alliance Research Center show that the amount of tube surface fully wetted above the pool level is a function of the number of tube rows crossed by the feedwater in the down flow direction.

2.3 Summary of Testing

2.3.1 Oconee 1 - (1B Steam Generator)

In order to fully monitor the thermal-hydraulic performance of the first full full size OTSG's to go on line, the Oconee 1B steam generator was specially instrumented.

To measure axial and radial internal temperature distributions within the steam generator, 54 individual thermocouples were installed at the locations shown on Figures 2-3 and 2-4. The thermocouples were

ungrounded, 1/16 inch diameter chromel-alumel with an Inconel sheath. They were vertically inserted inside 12 selected steam generator tubes located across the tube bundle diameter. Each of the twelve instrumented tubes was plugged at its lower end to provide a stagnant column of reactor coolant which, during steady state operation, attained thermal equilibrium with the surrounding secondary fluid. The thermocouples were inserted from the upper tubesheet ends of the tubes to pre-measured depths for proper axial positioning of the thermocouple junctions. The thermocouple leads were routed to a special OTSG upper head penetration plate via a housing conduit. Outside the OTSG, the thermocouple leads were routed to an automatic multipoint millivolt recorder.

The temperature measuring scheme (i.e. thermocouples immersed in stagnant water-filled steam generator tubes) was analyzed for possible measurement errors prior to data gathering. Analysis indicated that axial heat conduction downward along the water-filled plugged tubes from the higher-temperature (upper) regions of the OTSG toward the lower-temperature regions of the tube bundle would not induce any measureable error in the indicated internal secondary temperatures. It was concluded also that natural convection currents in the water columns would not be a concern since the operating secondary fluid axial temperature gradient promotes a naturally stagnating density gradient in the water-filled tubes.

a. Natural Circulation Tests at Oconee 1

During hot functional testing (prior to initial core loading) at Oconee Unit 1, two natural circulation tests were performed primarily to provide information for developing test procedures for natural circulation with the core in place. The RC pumps were utilized to raise the primary fluid temperature to the saturation temperature corresponding to the secondary side pressure. All RC pumps were then

stopped, and approximately five minutes was allowed for RC system flow to stop prior to initiating either of the natural circulation tests.

The first test was initiated by lowering the secondary steam pressure approximately 150 psi to 730 psig by opening the turbine bypass valves. This resulted in a decrease in secondary side saturation temperature of approximately 25°F which started primary side natural circulation flow and caused a drop in secondary water level of approximately 100 inches in 3 to 4 minutes. At this point the operator initiated emergency feedwater for about 3.5 minutes at about 930 gpm per steam generator to restore the water level back to 373 inches on the operate range instrumentation. During the injection of 100°F emergency feedwater, a further reduction of steam pressure of approximately 50 psi occurred due to steam condensation.

Prior to the start of the second natural circulation test, the water level on the secondary side of the steam generator was reduced to approximately 301 inches on the operate range instrumentation. Natural circulation was then initiated by raising the level in the steam generator to 370 inches using emergency feedwater. The EFW flow was approximately 600 to 700 gpm/SG during the first three minutes and was throttled back over the last minute as the final desired SG level was approached.

b. Results of Oconee Tests

One set of thermocouples was located approximately one inch below the centerline of the EFW injection elevation. The response of this set of thermocouples is shown in Figure 2-5. The temperature of tube 3 is significantly different from the rest, indicating EFW wetting. Note, however, that although the EFW temperature is approximately 100°F, the

measured thermocouple temperature of this elevation is close to the saturation temperature (saturation temperature was calculated from the recorded steam pressure in the steam line downstream of the OTSG).

Similar responses were seen at other instrumented elevations. The behavior of the thermocouples nine feet below the upper tubesheet is shown in Figure 2-6. At this elevation the thermocouple in tube 9 also shows a distinct response to the EFW injection.

Figure 2-7 is a summary sketch of the thermocouple responses for tubes 3, 9, 16, 24 and 33. Indicated are those thermocouples showing a response due to tube wetting. The response shows the effects of flow spreading due to pooling of the EFW water as it passes through the OTSG tube support plates.

2.3.2 Alliance Research Center Tests

In 1978, tests were conducted at the Babcock & Wilcox Alliance Research Center (ARC) to characterize the EFW stream as it enters the OTSG bundle. Two tests called Flow Trajectory and Flow Visualization Studies were conducted. They are described below.

a. Flow Trajectory Study

Tests were conducted without tubes in order to characterize the EFW nozzles. A conceptual sketch of these tests is shown in Figure 2-8. The model consists of a 72" x 59" x 8-1/2" box constructed of 1/4" thick carbon steel plate. Four large transparent Lexan windows (two in front and two in back) as well as two small windows (one in each narrow wall below the EFW connections) allowed lighting, visual observation and video recording of the EFW stream. A two-dimensional grid background provided a means of defining the jet trajectory.

Tests were performed in air and steam environments at atmospheric pressure. A range of flowrates from 20 gpm to 160 gpm (corresponding to total EFW of 140-1120 gpm for seven nozzles) for the arrangement were used for this phase of the EFW injection tests.

In order to investigate the effect of a steam environment on the EFW flow stream, a movable impact plate instrumented with strain gauges was placed in the path of the jet. The presence of steam was not found to affect the nozzle discharge characteristics. The effect of steam flow through the OTSG is not expected to degrade the wetting characteristics of the steam generator tubes because the low steaming rates occurring during EFW injection will not entrain the auxiliary feedwater. Although the tests were conducted at atmospheric pressure using unheated tubes, OTSG tube temperatures will not reach the elevated Leidenfrost or sputtering temperatures during EFW injection at which the rewetting characteristics might be altered. The subsequent Flow Visualization Study was performed in an air environment.

b. Flow Visualization Study

The objective of the Flow Visualization Study was to visually observe the distribution of the EFW stream after impacting the unheated tubes. Photographic and video data were obtained which show the radial and circumferential dispersion of the stream as it penetrates the tube bundle. Data was recorded for the following flow range and flow increments:

Arrangement	Per Nozzle Flow Range (gpm)	Flow Increments (gpm)
External Header	20-160	20

c. Description of Model

The model was a full scale section of the steam generator eleven (11) tube rows wide and approximately sixty-five (65) rows deep between the 15th tube support plate (TSP) and the upper tube sheet (UTS). The model arrangement is shown in Figure 2-9. The elevations of the EFW inlet center lines were at 10-5/8" above the 15th TSP. Internally, an actual TSP segment was used. There were approximately 646 tubes of which approximately 350 were clear cast acrylic 5/8" OD rods with the remainder being 0.625 inch OD x 0.555 inch ID carbon steel tubes. The upper support consisted of a simulated UTS made of clear Lexan. The acrylic rods and the Lexan UTS provided visibility and light transmission for the photographing of flow patterns.

d. Results of Flow Visualization Study

Experimental results of the EFW Flow Visualization Study were reviewed and profile maps sketched identifying those tubes that are continuously wetted, those tubes that are intermittently spattered, and those tubes which are not wetted at the injection elevation. Figure 2-10 is a summary sketch showing the continuously wetted tubes as a function of EFW flowrate and EFW header configuration (at high EFW flows, the results for the spray tests are adjusted to account for the effects of the walls of the test section on the total number of tubes which are wetted). Figure 2-11 presents the data which has been reduced in terms of percent of OTSG tube wetted at the level of injection.

2.3.3 EFW Modeling and Benchmarks

A digital computer code, AUX, was developed to quantify EFW system performance during natural circulation conditions. During development the code was exercised assuming that steam generator heat removal occurred only between the primary coolant and the secondary liquid pool. This model indicated that additional heat transfer above the pool was necessary to match operating plant natural circulation data. Implementation of an EFW flow model based on operating plant natural circulation data resulted in reasonable agreement between the analytic results and available plant data.

a. EFW Flow Model

Experimental results indicate that EFW wets the tubes on the periphery of the tube bundle at the injection plane in proportion to the EFW flow (Figure 2-10).

EFW, nominally at 100°F, enters near the top of the OTSG (Figure 2-1). The Oconee 1B OTSG data shows that this EFW is heated to saturation by direct condensation of steam in the vicinity of the injection plane. In addition, the Oconee 1B OTSG data indicates the pooling of the EFW plus condensate occurs at the tube support plates, and that the pooling of the fluid at the tube support plates causes inward spreading of the fluid to cause significant tube wetting below the injection plane.

The EFW tube surface wetting model is based upon the results of the natural circulation test data from the Oconee 1B OTSG and the B&W Alliance Research Center (ARC) test data described in paragraphs 2.3.1 and 2.3.2. The Oconee 1B and the ARC data indicate that tube wetting

at the plane of injection is flow dependent. The results of the data, for the EFW injection plane are given in Figure 2-11. These data are used to describe the surface wetted at the injection plane as follows.

$$\frac{A(0)}{A(T)} = 3.33 \times 10^{-5} \times \text{EFW (GPM)} \qquad (1)$$

where:

$A(0)$ = wetted surface, ft^2, at the injection plane per OTSG

$A(T)$ = total OTSG surface, ft^2, per OTSG

EFW (GPM) = EFW flow rate in GPM

As the total EFW flow (EFW plus condensed steam from direct contact heating of EFW at or near the injection plane) falls from the injection plane to the tube support plates, spreading of the fluid occurs and causes increased wetting of tubes as tube support plates are crossed. The spreading effect is shown in Figure 2-7. The data indicates that normalized wetting of tube surface is:

Point	Normalized Number of Tubes Wetted	Normalized Surface Wetted
Injection Plane	1	1
1st Tube Support Crossed	1.7	2.9
2nd Tube Support Crossed	2.2	4.8
3rd Tube Support Crossed	2.6	6.7
4th Tube Support Crossed	2.9	8.4
5th Tube Support Crossed	3.15	9.7
6th Tube Support Crossed	3.25	10.5
Nth Tube Support Crossed	3.25	10.5

This is also shown in Figure 2-7.

The tube wetting caused by the fluid crossing tube support plates is represented in the EFW flow model as a function of distance from the injection plane.

$$\frac{A(Z)}{A(0)} = (1 + 0.545Z), \qquad (2)$$

where:

$A(Z)$ = total wetted surface, ft^2, down to distance Z below the EFW injection plane,

$A(0)$ = wetted surface, ft^2, at the injection plane,

Z = distance, ft, below the injection plane

The maximum value of $A(Z)/A(0)$ is limited to a maximum tube support plate spreading factor of 10 to account for uncertainties in the test data. Combining equations 1 and 2 yields the fraction of SG tubes wetted as the feedwater cascades through the tube sheets. For example, an EFW flow rate of 300 gpm (equation 1) will continuously wet approximately 1% of the SG at the injection elevation. The percentage of wetted tubes increases linearly until, at 16.5 feet below the injection point, equation 2 indicates 10% of the tubes are wetted. The wetted area is assumed to remain constant at 10% until the feedwater enters the pool.

The following benchmarks illustrate reasonable agreement of the EFW flow and heat transfer model with available 177 FA plant data.

b. Benchmarks

The benchmarks chosen for AUX EFW model verification were selected to represent a range of conditions under which the EFW system will function.

Benchmark #1 - Low Initial Power, Low Decay Heat
Benchmark #2 - Raised Loop configuration
Benchmark #3 - High Initial Power, High Decay Heat (Overheating)
Benchmark #4 - High Initial Power, High Decay heat (Nominal EFW Operation)

The following assumptions were made:

(1) The reactor is assumed to be scrammed at time zero with coincident closing of the turbine bypass and stop valves.
(2) The initial conditions correspond to the nominal power level.
(3) Power to the RC pumps is lost and a nominal pump coastdown curve is followed until the onset of natural circulation occurs.
(4) The inception of natural circulation is predicted by simultaneously calculating RC natural circulation flows while using the input coastdown flow rates.
(5) Once the input flow rates are exceeded, inertial flow rates are calculated independently for each loop.
(6) Nominal 177 FA plant parameters were chosen and were unchanged from case to case.

1) Benchmark #1

LOSS OF OFF-SITE POWER TEST FROM 15% POWER LEVEL AT TMI-2 (April 22, 1978)

The data are from a LOOP test conducted at Three Mile Island, Unit 2 on April 22, 1978 from an initial power level of 15%. Review of the previous power history showed that the core had only 0.5 effective full power days of operation at the time of the test.

Both diesel generators were off initially but started automatically with 17 seconds after reactor trip. The emergency feedwater system began to deliver water to the steam generators within one minute. EFW was terminated at about 12 minutes when the startup level reached 165 inches (about 40% on the operate range).

Comparison of measured and calculated parameters is shown in Figures 2-12A and 2-12B. An EFW flow rate of 500 gpm initiated at 40 seconds results in excellent correspondence between the measured and calculated startup levels. (The EFW flow rate of 500 gpm per generator is consistent with expected EFW performance.)

Emergency feedwater flow was excessive for the core power history, resulting in continuous cooldown of the primary system and loss of steam pressure until EFW was terminated at twelve minutes.

Despite the overcooling of the primary system, pressurizer level remained on-scale. As part of this test, provisions were made for high pressure injection. A makeup flow of 350 gpm initiated at 180 seconds results in excellent correspondence between the measured and calculated pressurizer levels. (This makeup flow model is reasonably consistent with plant capability and post trip response).

2) <u>Benchmark #2</u>

LOSS OF OFF-SITE POWER EVENT FROM 40% POWER LEVEL AT DB-1
(November 29, 1977)

The data are from a post trip review for a loss of power event which occurred at Davis Besse Unit 1 on November 29, 1977. The unit, operating at 40% power, experienced a complete loss of power when the reactor tripped on overpower at 50%. The reactor trip initiated a turbine trip; the operator opened the main circuit breakers and the station experienced a blackout.

The steam generator rapidly filled to 120 inches; resulting in loss of indicated pressure level at about 250 seconds. When EFW flow was terminated, the primary system began to reheat and indicated pressurizer level was regained at about 630 seconds.

Comparison of the data and the calculations are presented in Figures 2-13A and 2-13B. Correspondence of measured and calculated RC temperatures is very good. Secondary pressure is good until 180 seconds when the code overpredicts the secondary pressure. This may be attributed to the uncertainty of the amount of steam bled to the EFW turbine driven pump.

The initial drop in pressurizer level is underpredicted. This may be attributed to the fact that the operator tripped the main circuit breakers about 9 seconds after turbine trip; power was supplied to the RC pumps for approximately 30 seconds following reactor trip.

3) Benchmark #3

LOSS OF OFF-SITE POWER EVENT FROM 100% POWER LEVEL AT ANO-1 (February 22, 1975)

The data are from the post trip review for a loss of off-site power event occurring at Arkansas Nuclear One, Unit 1 on February 22 1975. The unit, operating at 100% power, experienced a complete loss of electrical power. The reactor tripped, the RC pumps shut down and the main feedwater pumps shut down. About thirty seconds after the reactor trip, the makeup pump was switched to the borated water storage tank and water was supplied to the RC system to maintain pressurizer level. Five minutes after reactor trip, one RC pump in each loop was restarted.

Comparison of the data and the calculations are presented in Figures 2-14A and 2-14B. It is assumed that no emergency feedwater was actually delivered to the steam generators. This is a reasonable assumption since the corresondence between measured and calculated generator levels and cold leg temperature are excellent.

A makeup flow of 350 gpm initiated at 30 seconds results in good correspondence between measured and calculated pressurizer levels. (This makeup flow of 350 gpm is consistent with post trip procedure and makeup system capability.)

4) Benchmark #4

LOSS OF OFF-SITE POWER EVENT FROM 100% POWER LEVEL AT ANO-1 (June 24, 1980)

The data are from the post-trip review for a loss of off-site power event occurring at Arkansas Nuclear One, Unit 1 on June 24, 1980. The unit, operating at 100% power, experienced a complete loss of power when the 500 KV breakers and 161 KV breakers opened, separating the generator from the transmission grid.

The plant was completely without AC power until the diesel generator units auto started, restoring AC power to the ES busses. The steam driven emergency feedwater pump tripped on overspeed upon initial actuation. The electric motor driven EFW pump was manually started by the operator rather than wait for a 100 second auto-start time delay.

Comparison of the data and the calculations are presented in Figures 2-15A and 2-15B. It is assumed that a constant emergency feedwater flow rate of 450 gpm (consistent with expected performance) was delivered to the steam generators after 90 seconds. Only the strip charts for the OTSG operate levels are available which provide no comparative data until the steam generator level rises above 102 inches (lower operate tap) at about six minutes.

The benchmarks presented provide an excellent demonstration of the adequacy of EFW models.

The EFW model is a simple analytic representation of the phenomena occurring during EFW injection. Nevertheless, the success of the model in simulating the important characteristics of EFW injection is

very satisfactory. The model is based on a physical interpretation of data which is directly applicable to the phenomenon of interest. This obviates the need to scale or to adopt models developed for atypical geometries or conditions.

A good understanding of the role of EFW in promoting natural circulation during the establishment of a secondary SG level is necessary. During natural circulation the effective driving head for the RC loop is provided by heat transfer caused by the EFW spray and the secondary pool water level.

The ability to predict plant response following loss of off-site power for the conditions under which the EFW system will function has been demonstrated by the foregoing comparison to operating plant data.

2.4 Supporting TMI-2 EFW Data

During the TMI-2 accident on 3/28/79, the EFW was used a number of times. Of major interest to EFW effectiveness is the RC system cooling response to EFW in the time period from 90 to 125 minutes. During this period EFW was supplied to the "A" steam generator at rates up to about 600 gpm. During the first part of the initiation (94 to 100 minutes) the "A" RC pumps were running, pumping a two phase mixture, providing forced circulation in the A loop. At about 100 minutes, the "A" pumps were tripped, resulting in a transient Boiler-Condenser mode of heat transfer for about 30 minutes.

Figure 2-16 shows the response of RC T_c, RC Pressure, and Steam Pressure during this EFW actuation.

The relatively rapid cooling rates of about:

4°F/min for RC forced circulation, and

1.6°F/min for the boiler-condenser mode

demonstrate the very effective EFW cooling.

2.5 Tube Plugging Considerations

Steam Generator Tube plugging has occurred in many of the operating 177 NSS Units. To date, the most extensive tube plugging has occurred at TMI, Unit 1. Figures 2.17 and 2.18 show the plugged tube locations. As the figures indicate, the plugged tubes tend to be concentrated in the periphery of the tube bundle. The effect of the tube plugging is to reduce the available surface for heat transfer above the pool. This reduction in surface is accounted for in the analysis by subtracting the plugged tube surface from the total wetted surface.

2.6 CRAFT2 SBLOCA Steam Generator Modeling

This Section provides a summary of the steam generator modeling features employed in the CRAFT2 SBLOCA Version. Details of the CRAFT2 SBLOCA modeling are presented in "CRAFT2, Fortran Program for Digital Simulation of a Multimode Reactor During Loss of Coolant", BAW-10092, Rev. 3, October, 1982.

2.6.1 Summary Description: Steam Generator Model

a. General

In the CRAFT2 SBLOCA version, the steam generator is described by up to 10 axial regions each of which contain one secondary volume and one or two primary volumes. Heat transfer between volumes and the corresponding secondary volume is calculated based on a regime-dependent correlation set and the results of an implicit tube calculation. This model contains all of the features comprising the standard steam generator model and such special features as level rate dependent auxiliary feedwater control, aspirator model (OTSG), recirculation model (UTSG), and a model that accounts for condensation of steam in the presence of noncondensibles on the primary side of the steam generator.

b. Primary Side Heat Transfer

In addition to a tube wall heat transfer model, the primary side heat transfer includes:

Forced Convection - Dittus-Boelter correlation
Saturated Nucleate Boiling - Chen correlation
Subcooled Nucleate Boiling - Modified Chen
Natural Convection (Laminar flow) - McAdams
Natural Convection (Turbulent flow) - McAdams
Condensation - Kreith

c. Secondary Heat Transfer

The EFW enters the secondary side of the steam generator subcooled. If the feedwater enters at an elevation below the mixture level, the mass and energy are uniformly deposited within the EFW inlet control volume. EFW injection into a steam environment results in enhanced heat transfer due to the EFW wetting of the steam generator tubes.

The fraction of the total number of tubes wetted is determined from the injection site penetration and subsequent spreading of the injected fluid as it falls downward through the steam generator tube support plates. This fraction is assumed to be a linear function of distance from the injection elevation to the elevation at which the maximum fraction of tubes are wetted.

The EFW entering a superheated control volume becomes saturated within the first several feet of fall. This user specified saturation point should fall within the confines of the inlet control volume. The subsequent calculations of downward liquid flow induced by the EFW injection is based on sensible heat, condensation, and tube wall heat transfer components.

The heat transfer coefficient for the EFW is calculated from Drew's correlation for a falling film of saturated liquid.

$$h = 0.01 \left(\frac{k^3 \rho g}{\mu_f^2} \right)^{1/3} (Pr \cdot Re)^{1/3}$$

Pr = Prandtl number, $(c_\rho \mu / k)$

Re = Reynolds number, $(4\Gamma / \mu_f)$

Γ = Flow/wetted perimeter, $(w_{aux} / n \pi D_o)$

k = Thermal conductivity, (Btu/s - ft - F)

ρ = Density of the film (lbm/ft^3)

g = Gravitational Constant, (ft/s^2)

μ_f = Film viscosity, (lbm/ft - s)

n = Number of wetted tubes

D_o = Outside tube diameter, (ft)

T_{sat} = Saturated film temperature, (F)

W_{aux} = Average EFW flow rate in the control volume, (lbm/s)

q = $h (T_w - T_{sat})$

2.6.2 CRAFT2 SBLOCA Version Benchmark

The CRAFT2 SBLOCA model information presented in this section is condensed from Appendix H of CRAFT2 - Loss of Off-Site Power Plant Transient Prediction in BAW-10154P, B&W's Small Break LOCA ECCS Evaluation Model, November, 1982.

2.6.2.1 Summary

A benchmark of the upgraded CRAFT2 computer model to a B&W plant transient has been completed. The selected transient was a loss of off-site power (LOOP) event that occurred at Arkansas Nuclear One, Unit 1, on June 24, 1980. Trees falling on the power lines caused the loss of power. This created a ground fault separating the plant from the grid. The turbine governor and intercept valves partially closed, creating a mismatch between steam production and steam demand. The reactor tripped shortly thereafter on the RPS high pressure signal. This transient was significant because the plant was cooled by natural circulation for one hour and 40 minutes.

The results of the benchmark between the calculated results and the ANO-1 data show that the upgraded CRAFT2 code data compared extremely well with the transient data. Both the steam generator and the pressurizer models correctly predicted their response to the LOOP event. This analysis demonstrated that the upgraded CRAFT2 code was capable of predicting the natural circulation mode observed during the small-break transients.

2.6.2.2 Model Description

The CRAFT2 noding scheme used in this analysis is shown in Figure 2-19. The CRAFT2 model comprises 57 nodes and 79 flow paths. The model that was used in this analysis was the same as the one developed for the 177 FA lowered loop plants.

The steam generator (OTSG) model used in this analysis was the new, upgraded model described in the CRAFT2 topical report. Both steam generators were characterized by two radial and six axial regions on the primary side and six axial regions on the secondary side with a separate node for the downcomer. Also, both generators were treated alike with regard to OTSG pressure, main feedwater flow, and emergency feedwater (EFW) flow.

At the initiation of the ANO-1 transient, the turbine governor and intercept valves partially closed, causing a mismatch between the energy produced and the energy demanded. As a result, the RCS, along with the steam generator, repressurized. This caused the reactor to trip on high pressure. In the analysis, the reactor was tripped at 4.5 seconds.

During the transient, the steam driven EFW pump tripped on overspeed and was restarted manually before the 100-second automatic time delay. The EFW pumps were started at 90 seconds after reactor trip. The EFW flow was held constant throughout the transient at 450 gpm to each OTSG.

2.6.2.3 Analytical Results

This section describes the comparisons of the results of the upgraded CRAFT2 code model and the LOOP event at ANO-1 on June 24, 1980. The RC

system pressure, RC temperature, and pressurizer responses are discussed separately.

a) RCS Pressure Response

A comparison of the calculated primary system pressure to the ANO-1 data is shown in Figure 2-20A. As can be seen from the figure, the updated CRAFT2 code model results compare very well with the actual RC pressure response. The only difference between the ANO-1 test data and the CRAFT2 prediction was the over-estimation of the peak pressure, which was due to the following: During the ANO-1 transient, the partial closing of the turbine intercept and governor valves was responsible for the repressurization and for the eventual reactor trip at 4.5 seconds. To simulate this with CRAFT2, the RCS had to be pressurized. This was achieved by completely closing the valves to the turbine (since partial closing of the valves is not possible with the CRAFT2 code prior to the reactor trip). Because of the termination of the total steam flow to the turbine, the RCS is expected to pressurize to a higher value than for the ANO-1 transient. Steam generator pressure response is shown in Figure 2-20E and agrees very well with the plant data.

The subsequent depressurization in RC pressure was caused by the decrease in fission power when the reactor trips. At roughly 20 seconds the steam generators start drying out, causing the RCS pressure to level out at 1900 psia. The RCS repressurization at 100 seconds was due to manual actuation of the two HPI trains. The mass added by the HPI system began filling the pressurizer, compressing the steam bubble in the top of the pressurizer and thus increasing the pressure. The pressure is expected to remain at roughly 2175 psia (hot standby pressure) for the remainder of the transient.

b) Temperature Response

The cold and hot leg temperatures calculated by the upgraded CRAFT2 code are shown in Figures 2-20B and 2-20C respectively. The cold leg temperature increased sharply at the beginning of the transient because of the pressurization of the OTSG. As the pressure increases, so does the steam temperature. The resulting temperature difference between the primary and secondary side was significantly smaller than that at the beginning of the transient. This caused the reduction in heat transfer, which resulted in a higher cold leg temperature. After the reactor trip, the cold leg temperature drops to 548.5F in approximately 80 seconds. Shortly after EFW actuation, the cold leg temperature began to decrease. This was due to the EFW. The cold leg temperature decreased thereafter because of the decreasing decay heat. As can be seen in Figure 2-20B, the predicted cold leg temperature compared reasonably well with the transient data.

The hot leg temperature (Figure 2-20C), increased slightly at the beginning of the transient. The hot leg temperature dropped to approximately 570°F in 80 seconds after the reactor trip. The hot leg temperature then increased slightly as natural circulation was established. As can be seen in Figure 2-20C, the CRAFT2 prediced hot leg temperature compared reasonably well with the ANO-1 data.

The pressurizer level response calculated by the upgraded CRAFT2 code is shown in Figure 2-20D. As can be seen in the figure, CRAFT2 predicted the level very well through the first 75 seconds of the transient. At this time the hot leg temperature, as predicted by CRAFT2 and shown in Figure 2-20C was approximately 4°F higher than the ANO-1 data. The 4°F temperature difference accounts for the

14-inch higher level and was due solely to the expansion of the liquid. As the steam generators began to dry out and the hot leg temperature began rising, the pressurizer level started to increase. At 100 seconds, two HPI trains were started. The additional mass entering the system continued filling the pressurizer, as can be seen in Figure 2-20D.

Overall, the CRAFT2 non-equilibrium pressurizer model compared well with the transient data by showing the same trends and by calculating essentially the same level.

Figure 2-1 NUCLEAR ONCE-THROUGH STEAM GENERATOR (OTSG)

1. PRIMARY INLET NOZZLE
2. PRIMARY OUTLET NOZZLE (2)
3. FEEDWATER HEADER
4. FEEDWATER SPRAY NOZZLES (32)
5. FEEDWATER HEATING CHAMBER
6. "BLEED" STEAM PORT
7. SATURATED FEEDWATER
8. PORTS
9. GENERATING TUBES (15,500)
10. DEPARTURE FROM NUCLEATE BOILING
11. 100% QUALITY
12. SUPERHEATED STEAM
13. STEAM ANNULUS
14. STEAM OUTLET NOZZLES (2)
15. LOWER SHELL
16. UPPER SHELL
17. LOWER TUBE SHEET
18. UPPER TUBE SHEET
19. ADJUSTABLE ORIFICE
20. EMERGENCY FEEDWATER INLET
21. TUBE SUPPORT PLATES (15)
22. CYLINDRICAL BAFFLE
23. STARTUP RANGE LEVEL
24. OPERATING RANGE LEVEL
25. WIDE RANGE LEVEL

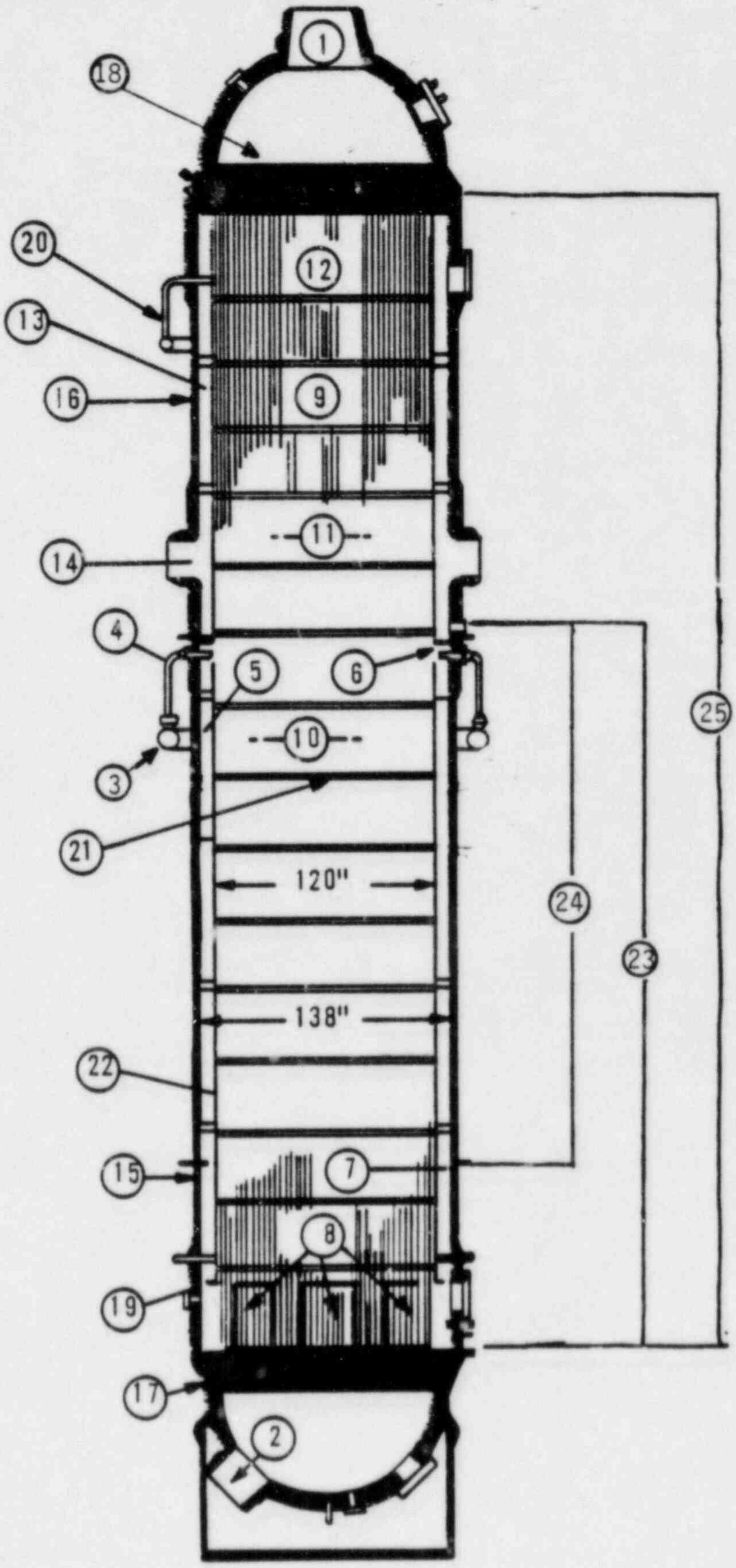


FIGURE 2-2 TYPICAL EMERGENCY FEEDWATER ARRANGEMENT

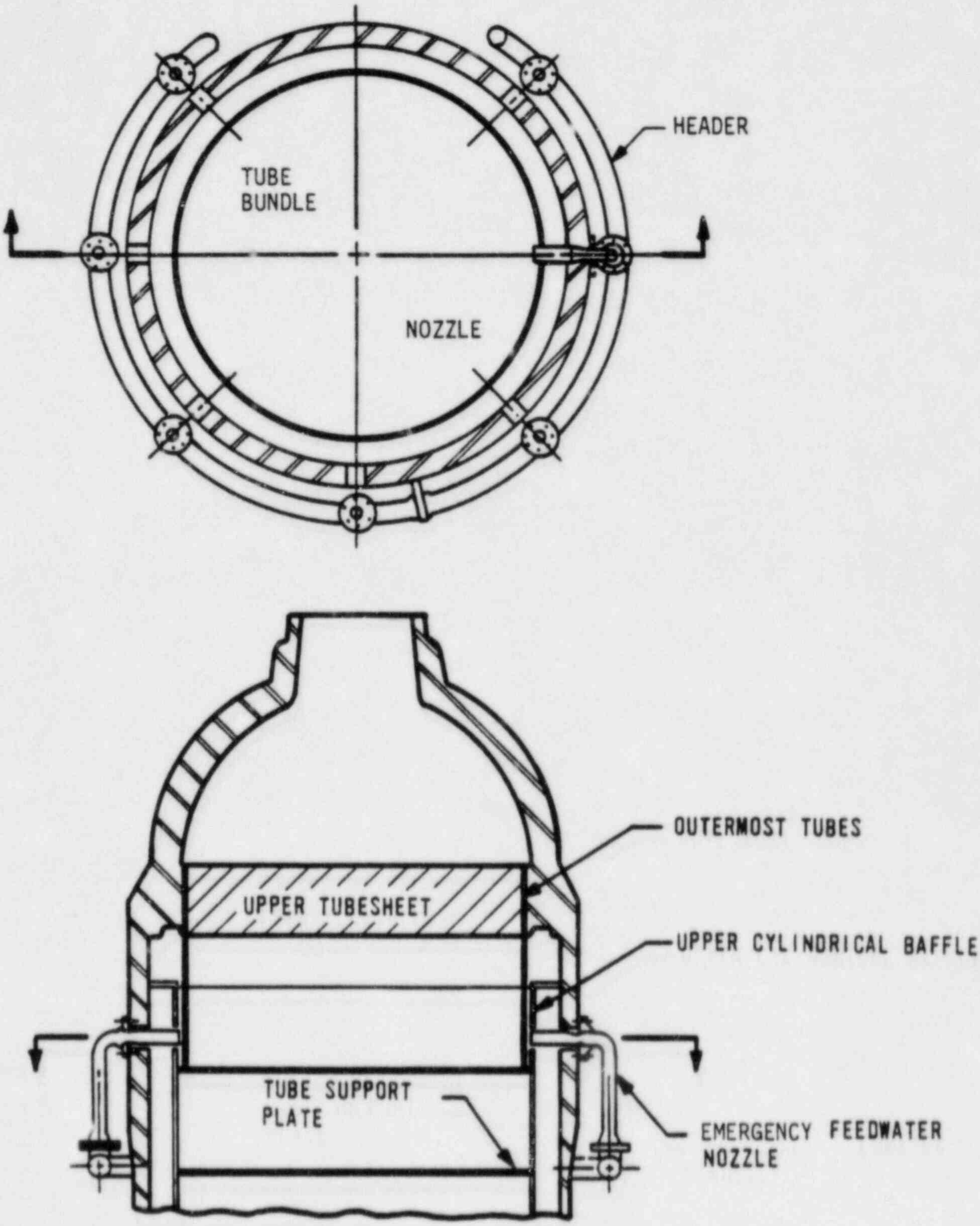


Figure 2-3 OTSG TEMPERATURE SENSOR LOCATIONS
(OCONEE 1 - 1B OTSG)

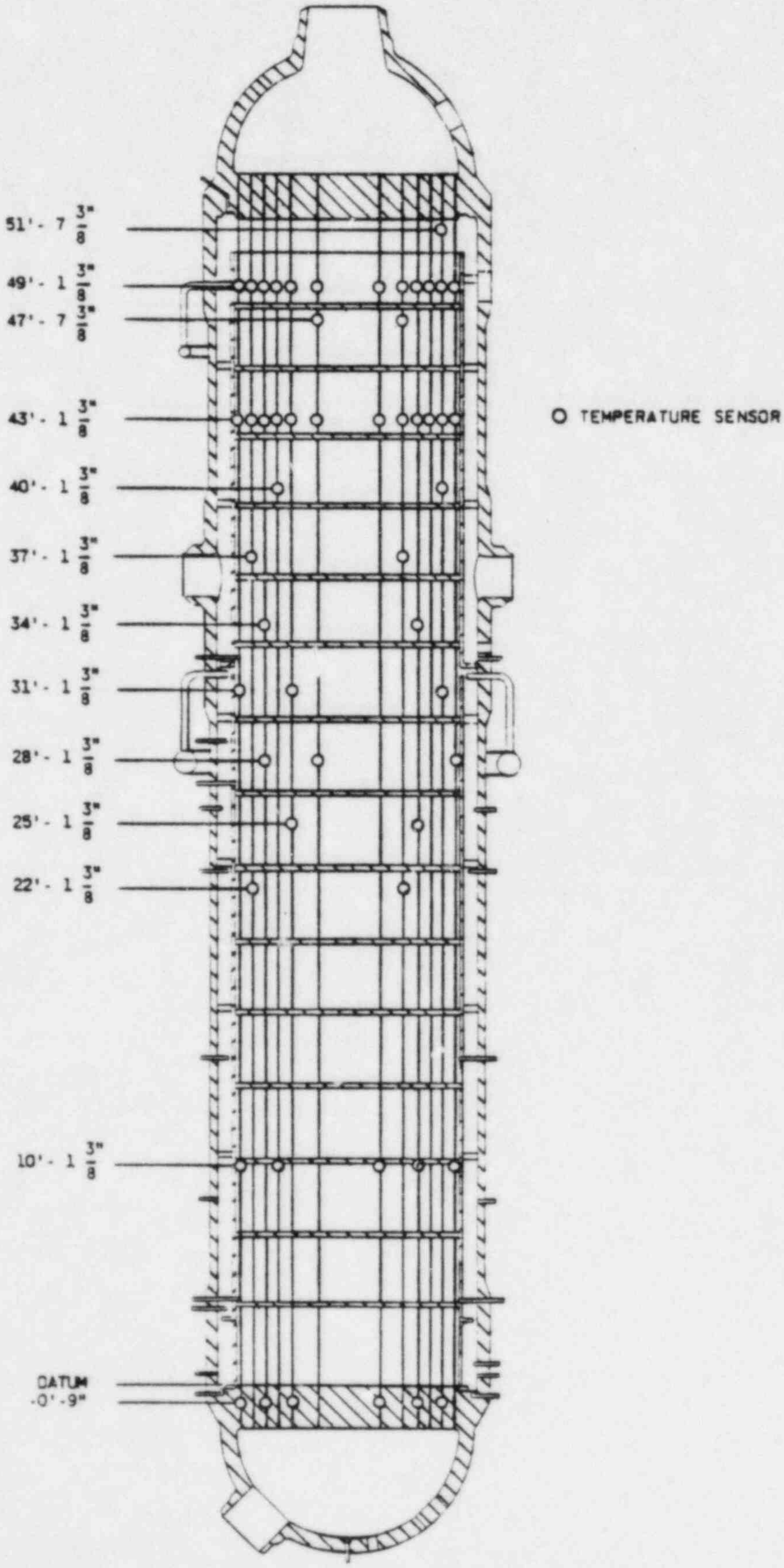


Figure 2-4 IDENTIFICATION OF INSTRUMENTED OTSG TUBES
(OCONEE-1, 1-B OTSG)

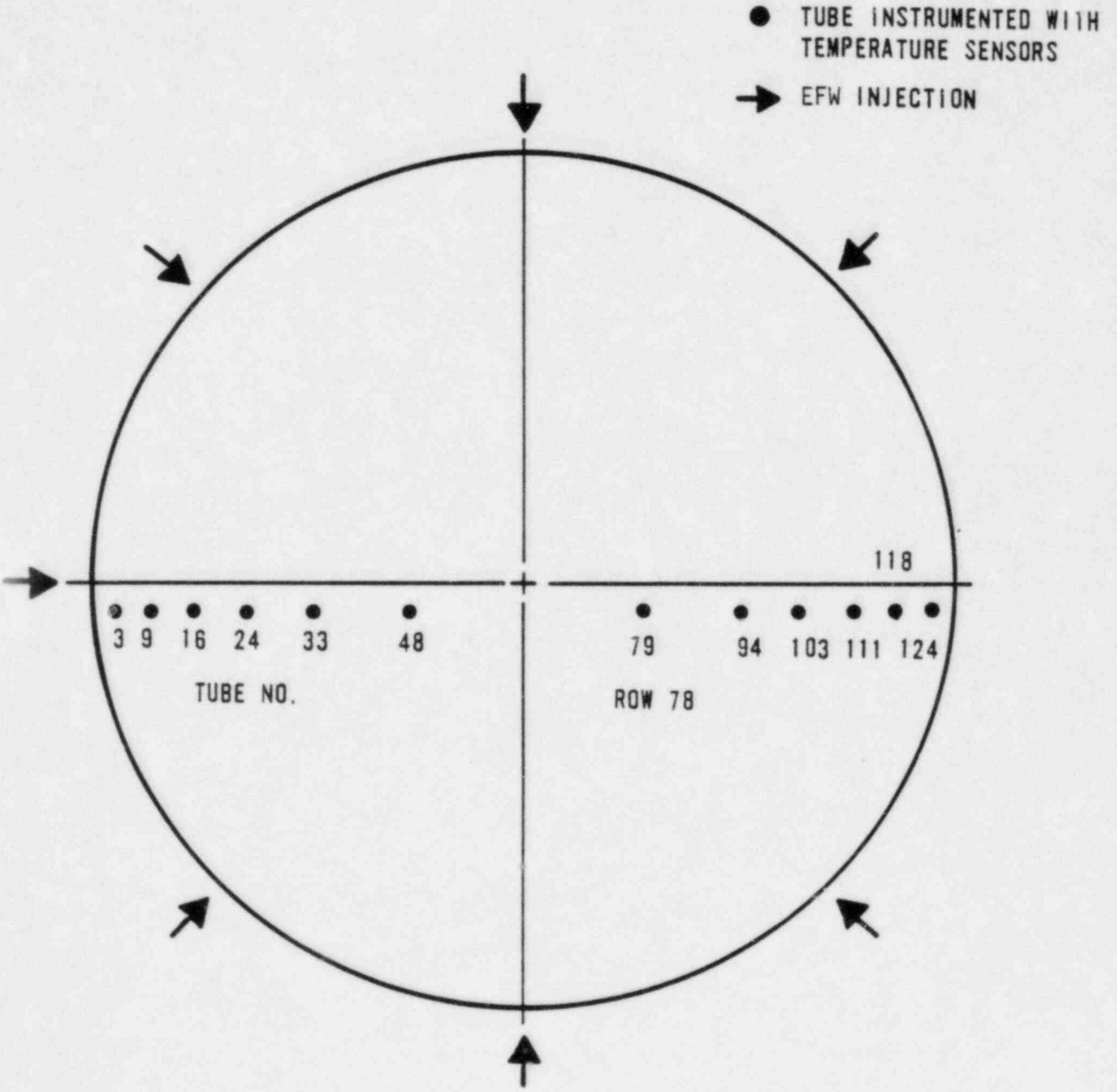


Figure 2-5 THERMOCOUPLE RESPONSE AT EFW INJECTION ELEVATION
(OCONEE 1 - IBOTSG)

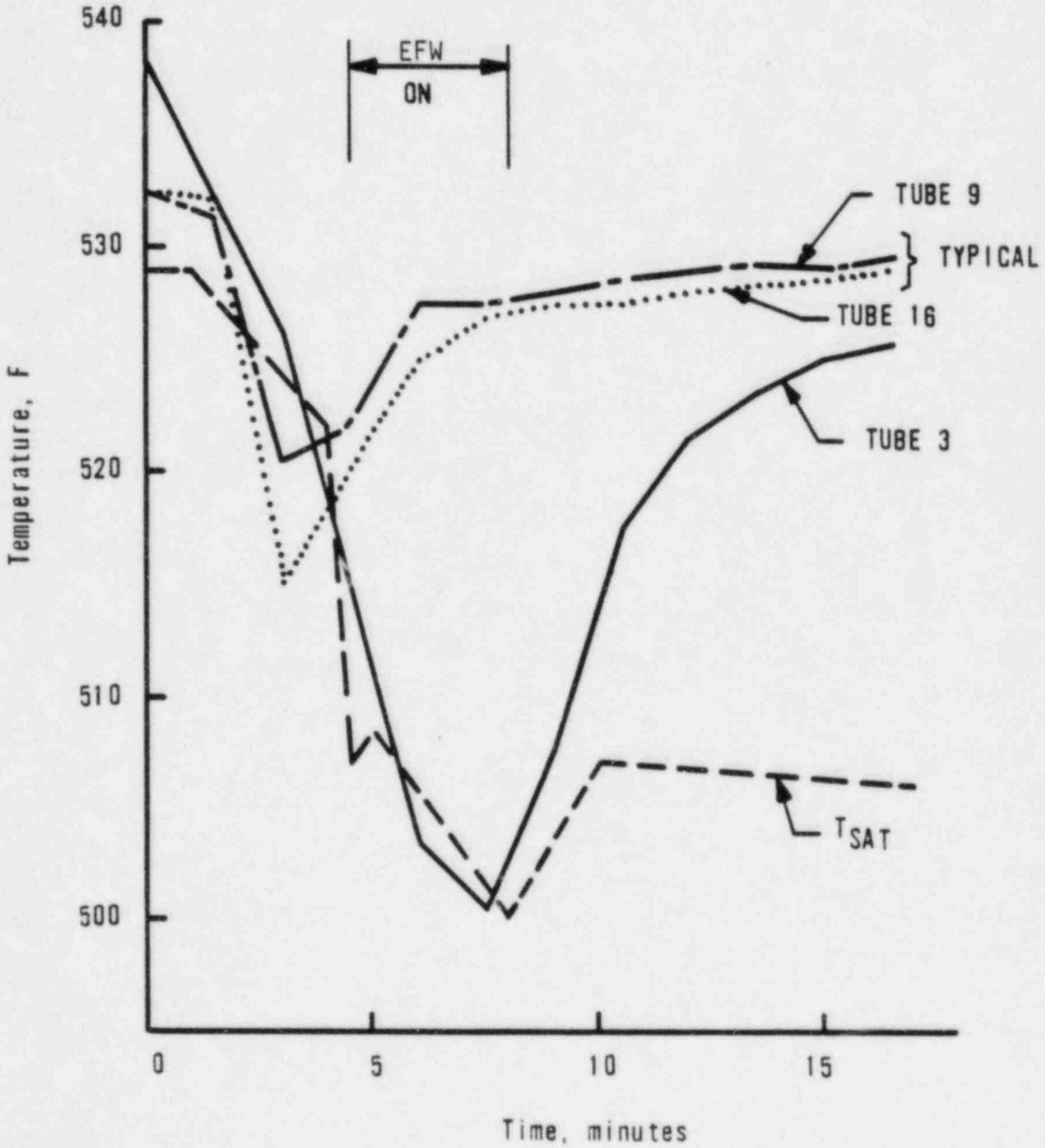


Figure 2-6 THERMOCOUPLE RESPONSE AT 9 FT B'.LOW UPPER TUBESHEET (OCONEE 1 - 1B OTSG)

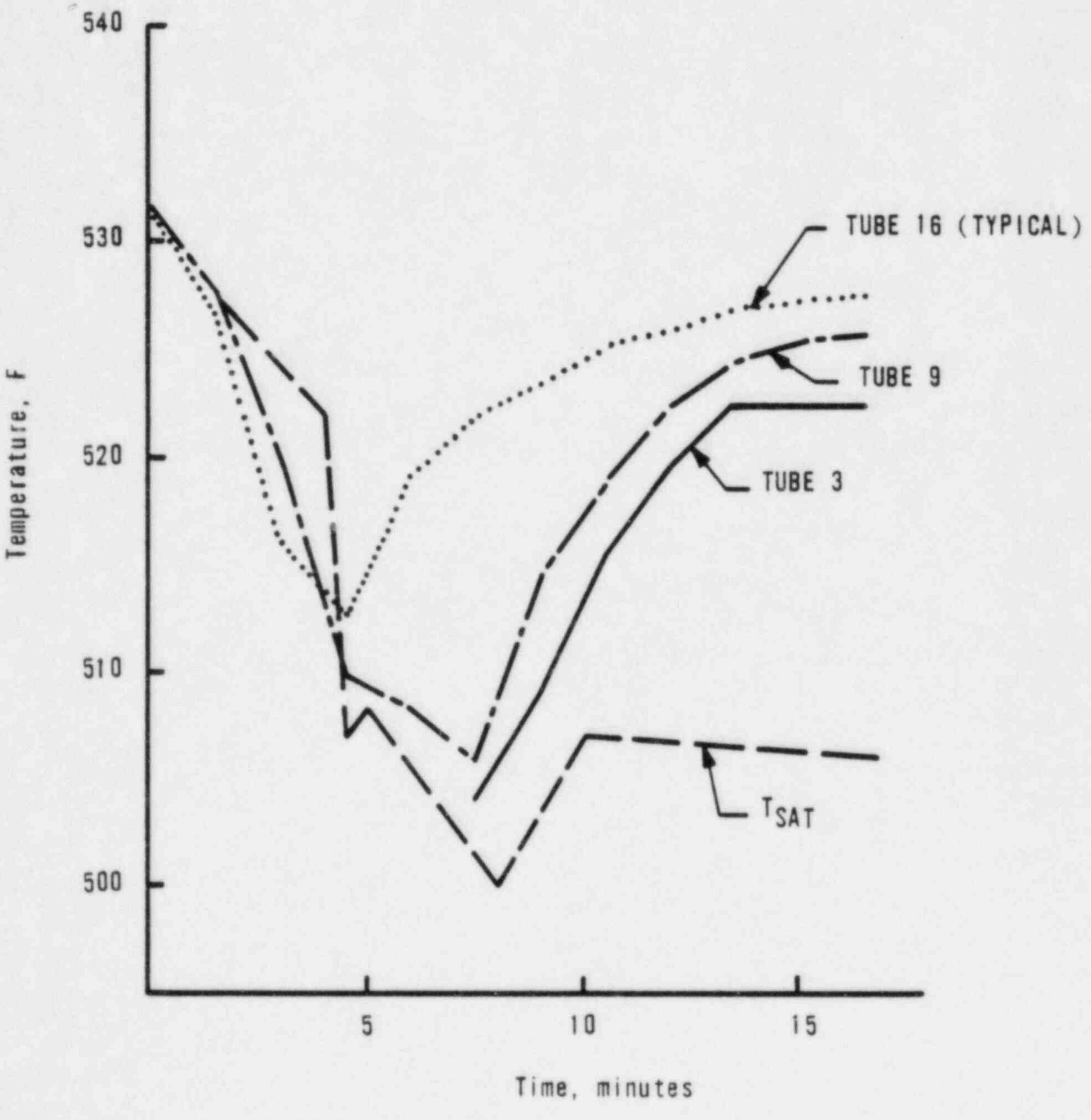


Figure 2-7 EFW AXIAL WETTING PROFILE
(OCONEE 1 - 1B OTSG)

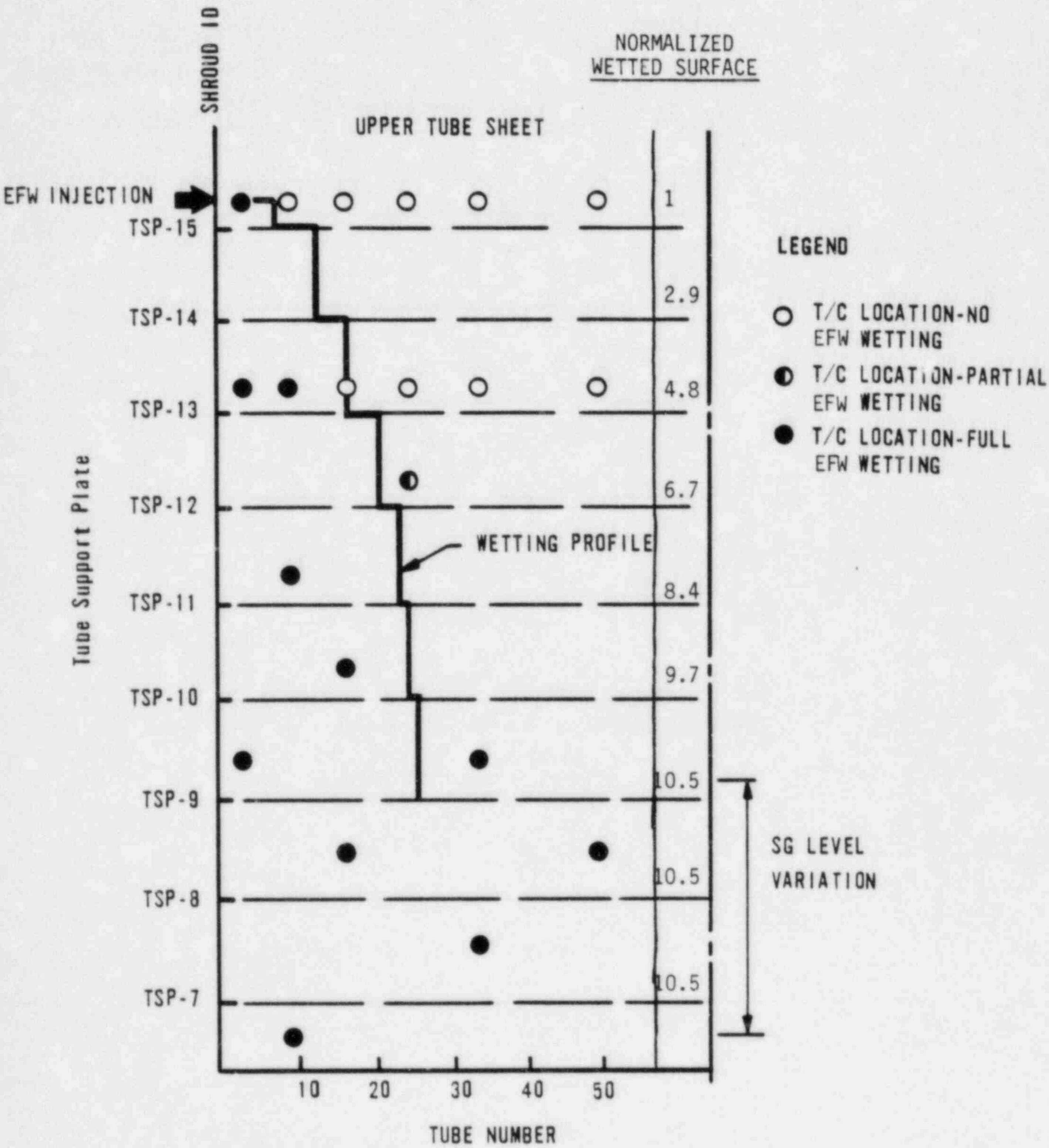


Figure 2-8 EFW NOZZLE CHARACTERIZATION TEST

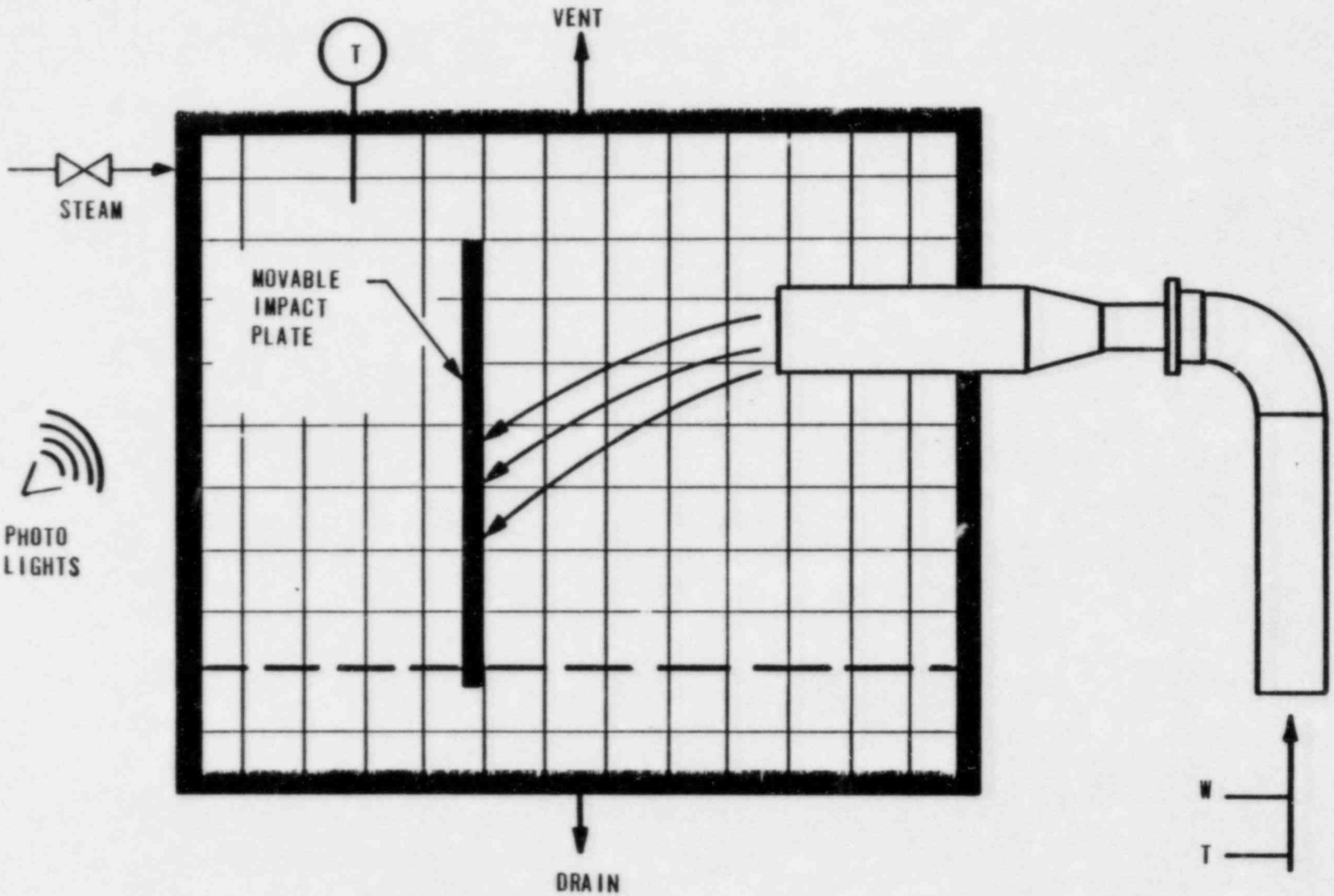


Figure 2-9 ARC - EFW MODEL ARRANGEMENT

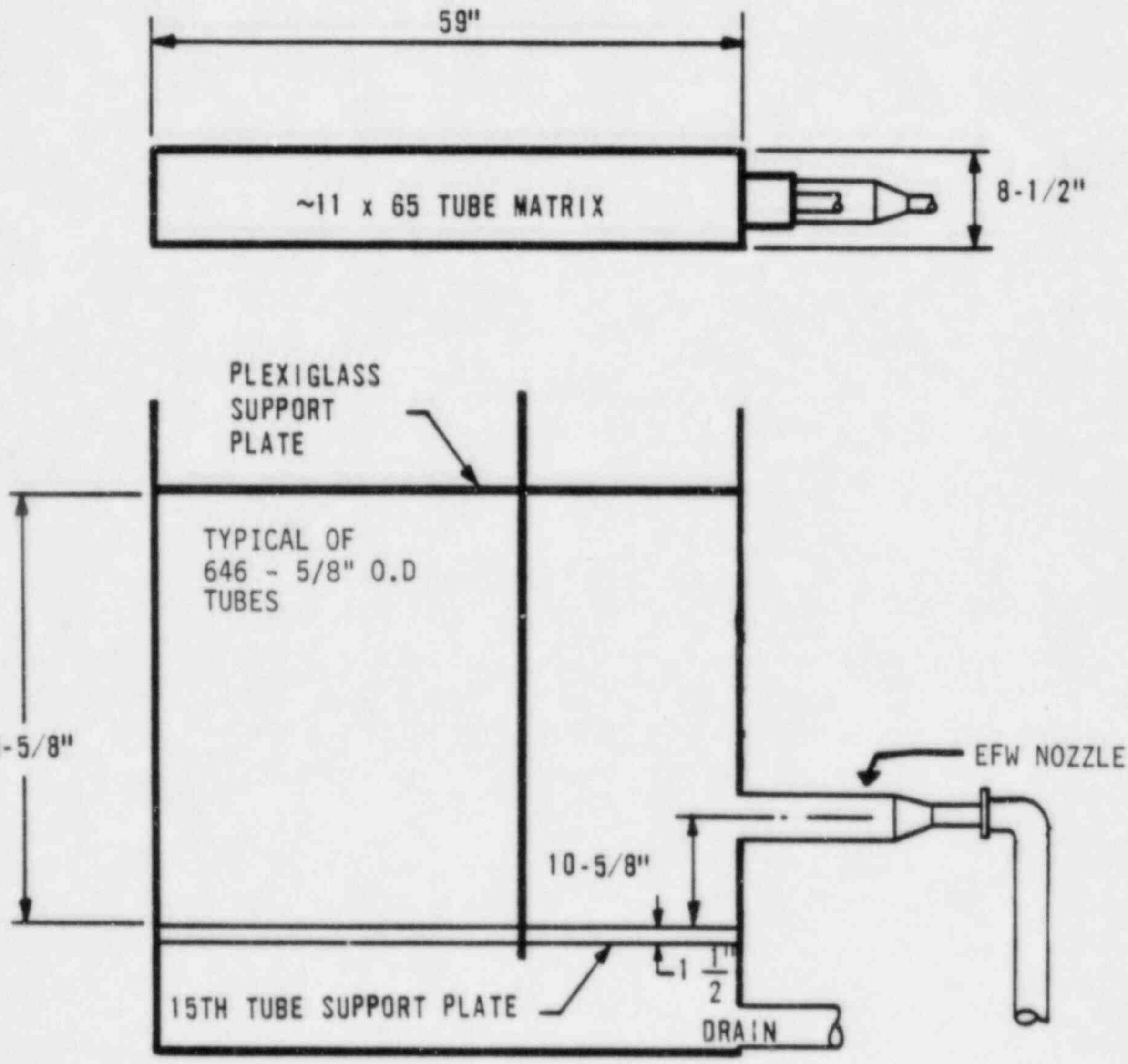


Figure 2-10 ARC TEST EMERGENCY FEEDWATER PENETRATION PROFILES

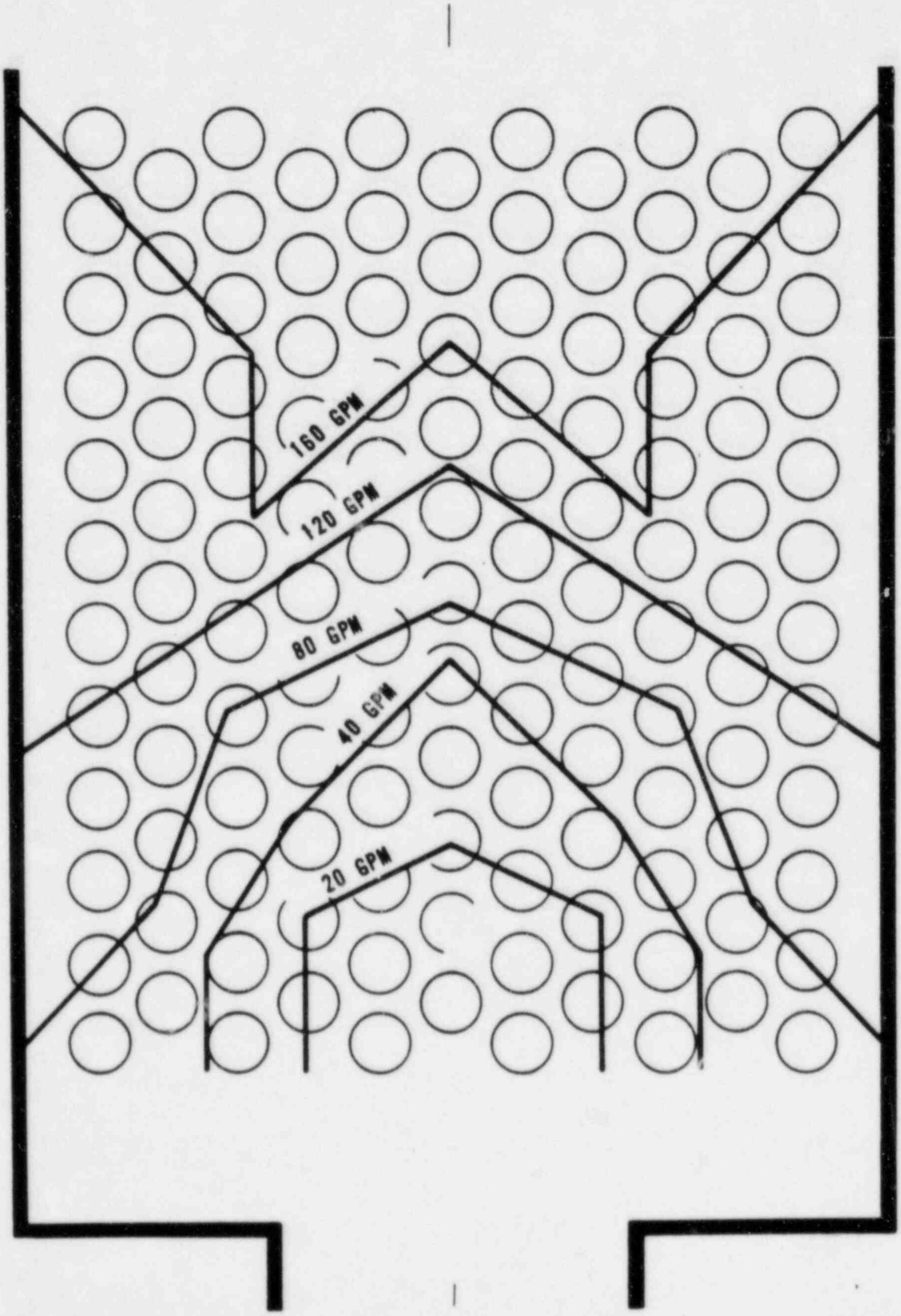


Figure 2-11 EFW WETTING EFFECTIVENESS - TUBE WETTED PER NOZZLE AT THE INJECTION PLANE

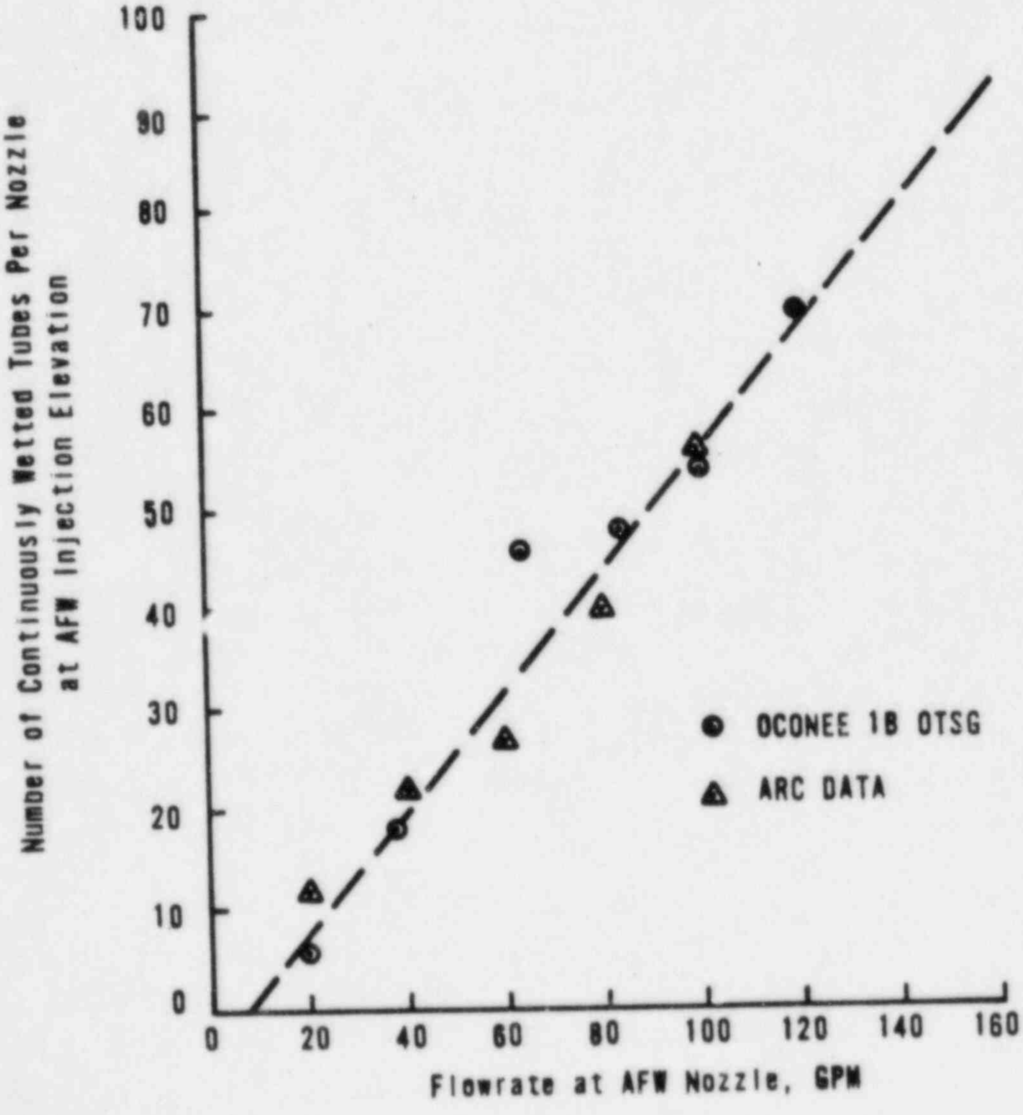
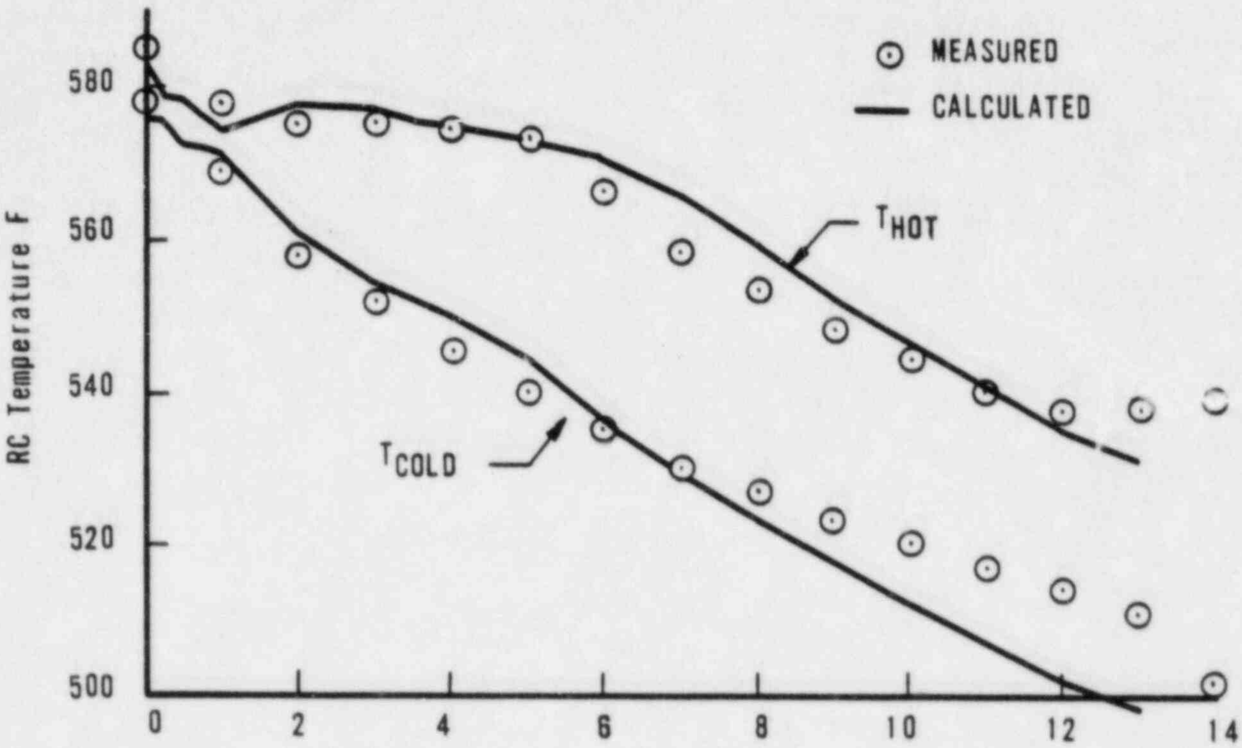


Figure 2-12A AUX BENCHMARK - CASE 1

LOOP TEST AT TMI-2 FROM 15% POWER
(22 APRIL 1978)



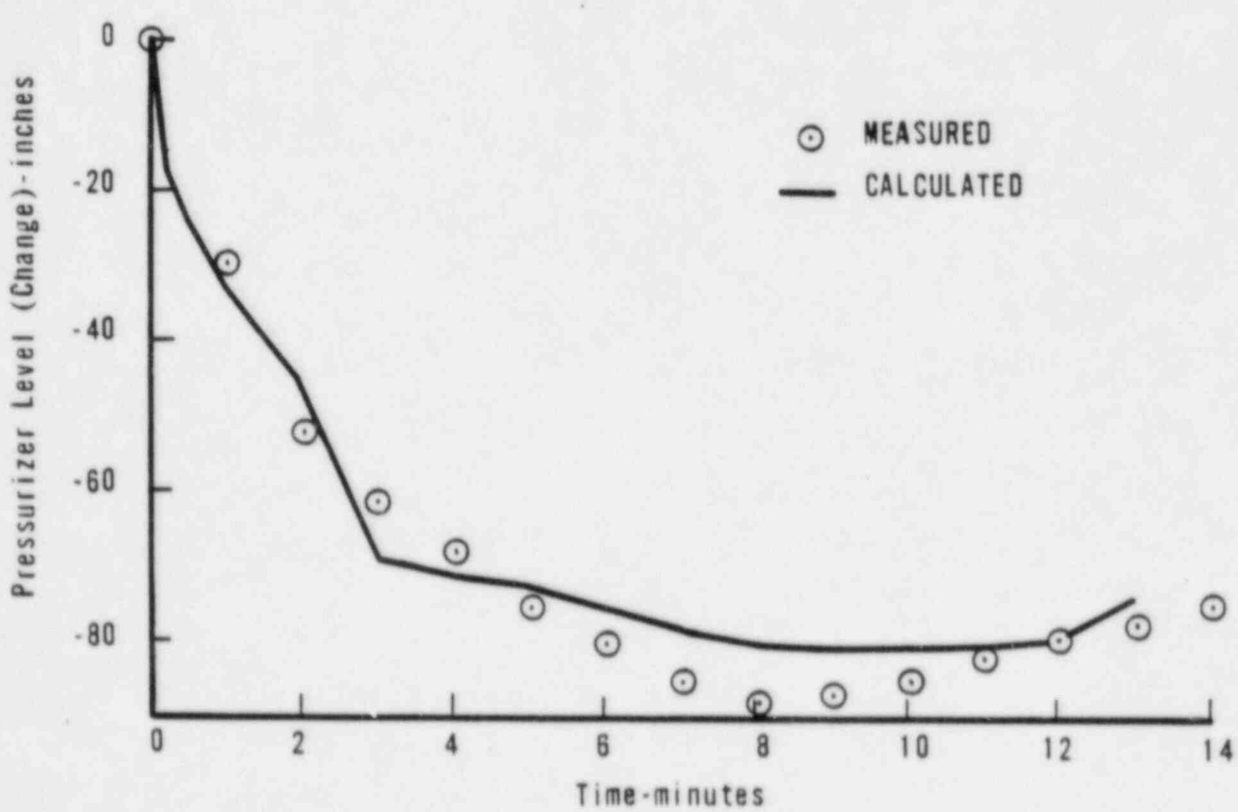
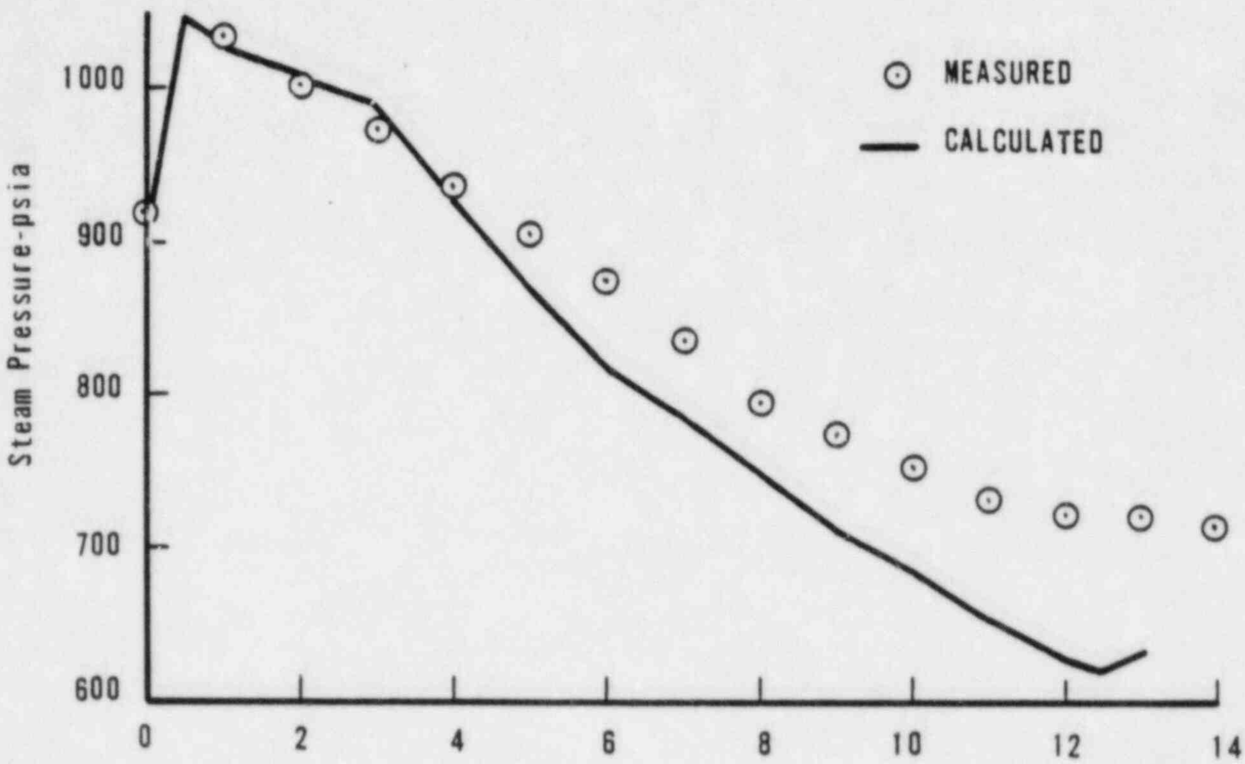


Figure 2-12B AUX BENCHMARK - CASE 1
LOOP TEST AT TMI-2 FROM 15% POWER
(22 APRIL 1978)



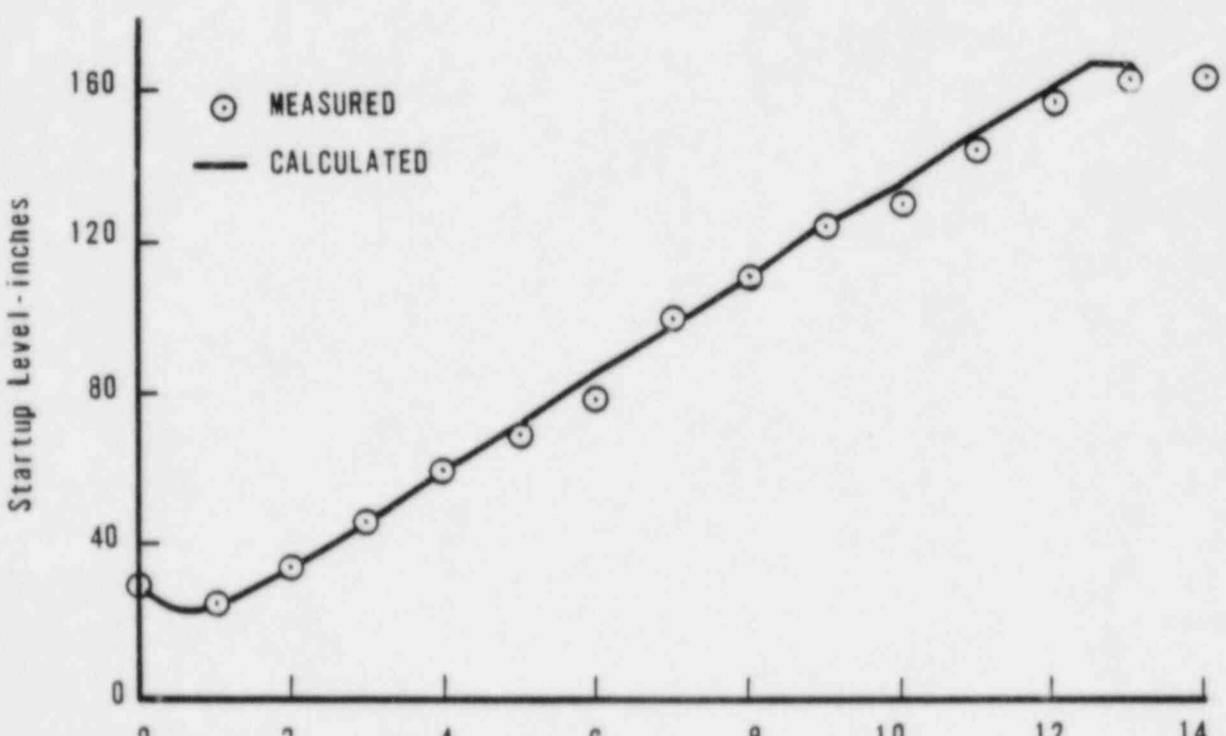


Figure 2-13A AUX BENCHMARK - CASE 2
LOOP EVENT AT DB-1 FROM 40% POWER
(29 NOV 1977)

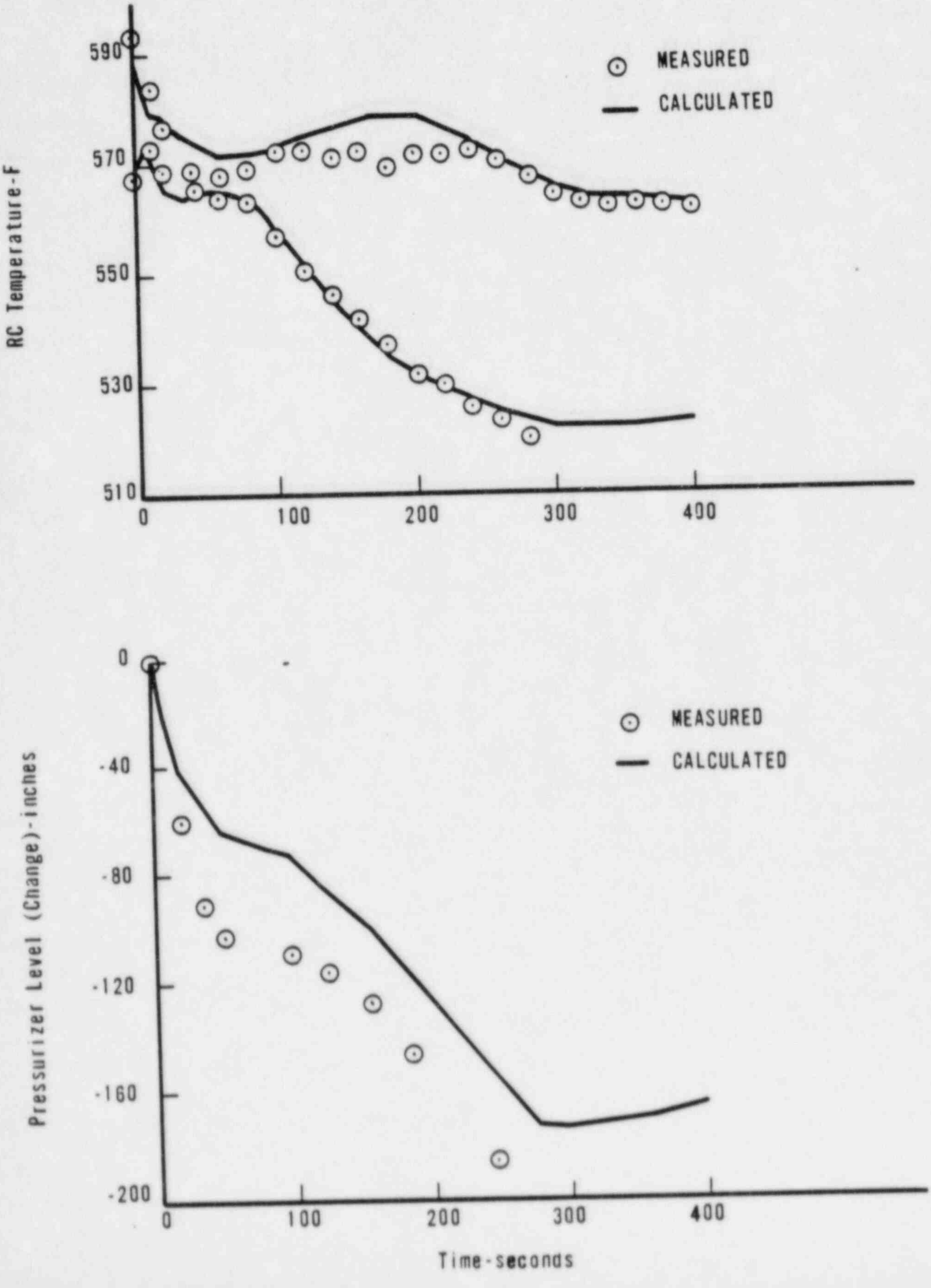


Figure 2-13B AUX BENCHMARK - CASE 2
LOOP EVENT AT DB-1 FROM 40% POWER
(29 NOV 1977)

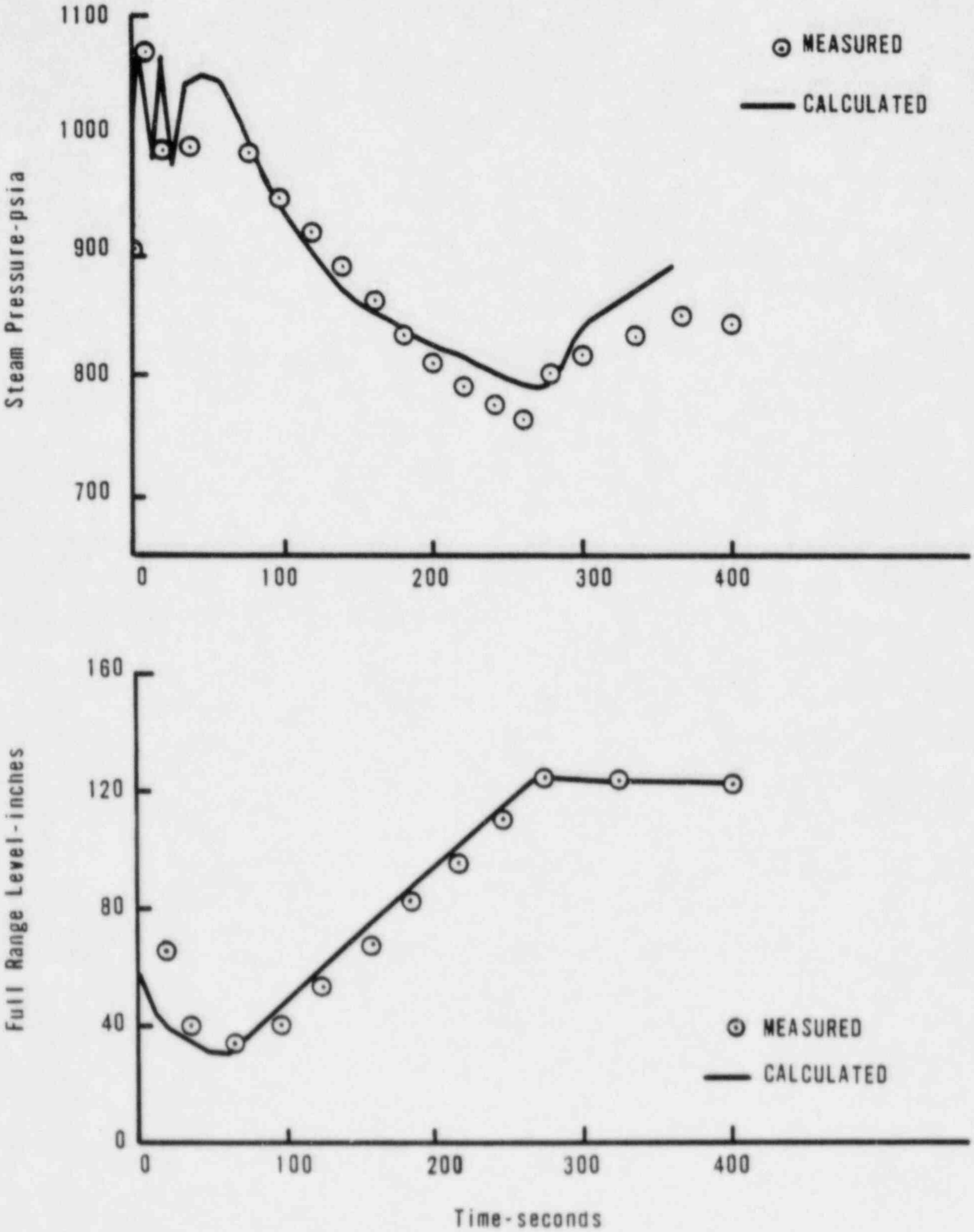


Figure 2-14A AUX BENCHMARK - CASE 3
LOOP EVENT AT ANO-1 FROM 100% POWER
(22 FEB. 1975)

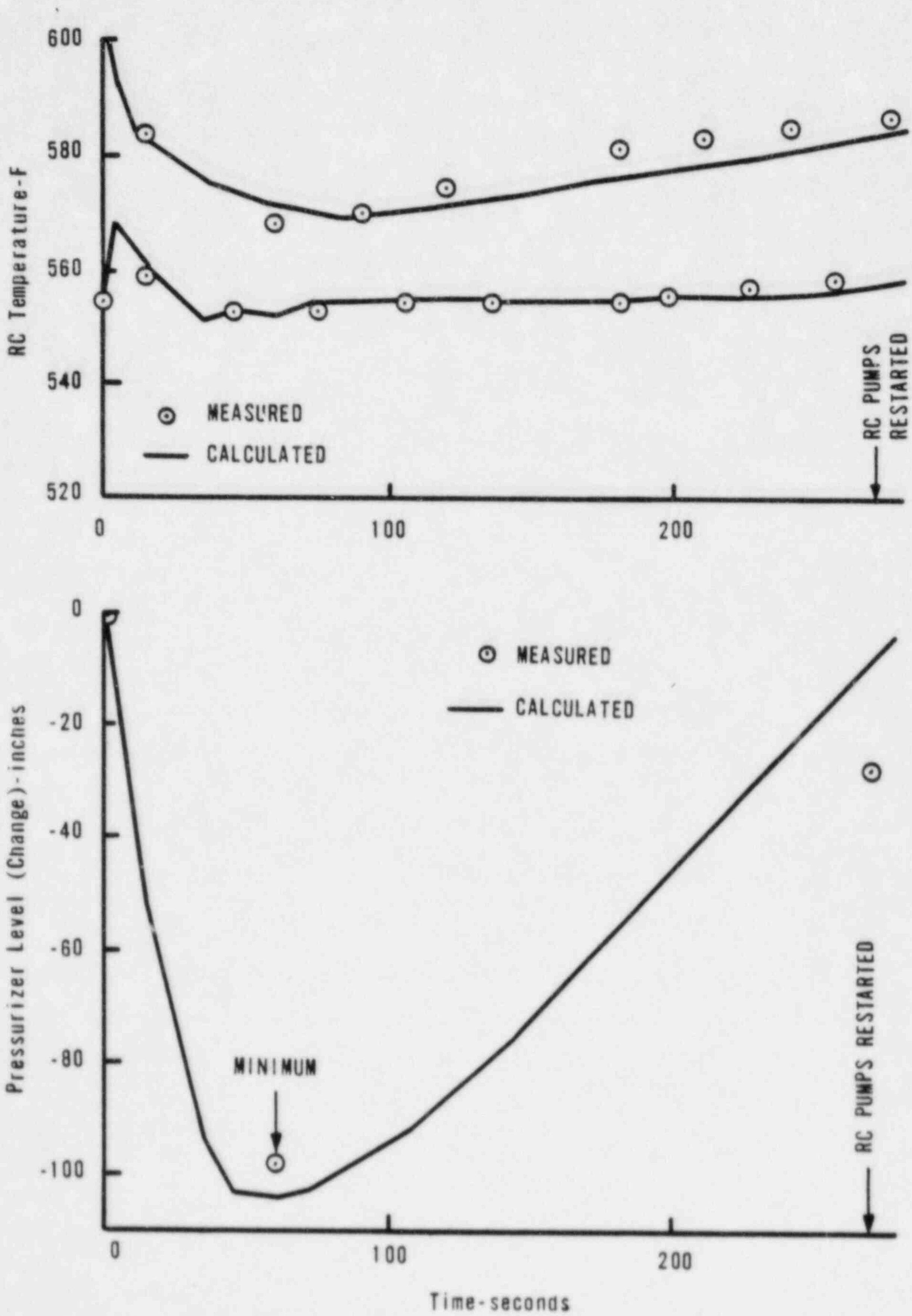


Figure 2-14B AUX BENCHMARK - CASE 3
LOOP EVENT AT ANO-1 FROM 100% POWER
(22 FEB 1975)

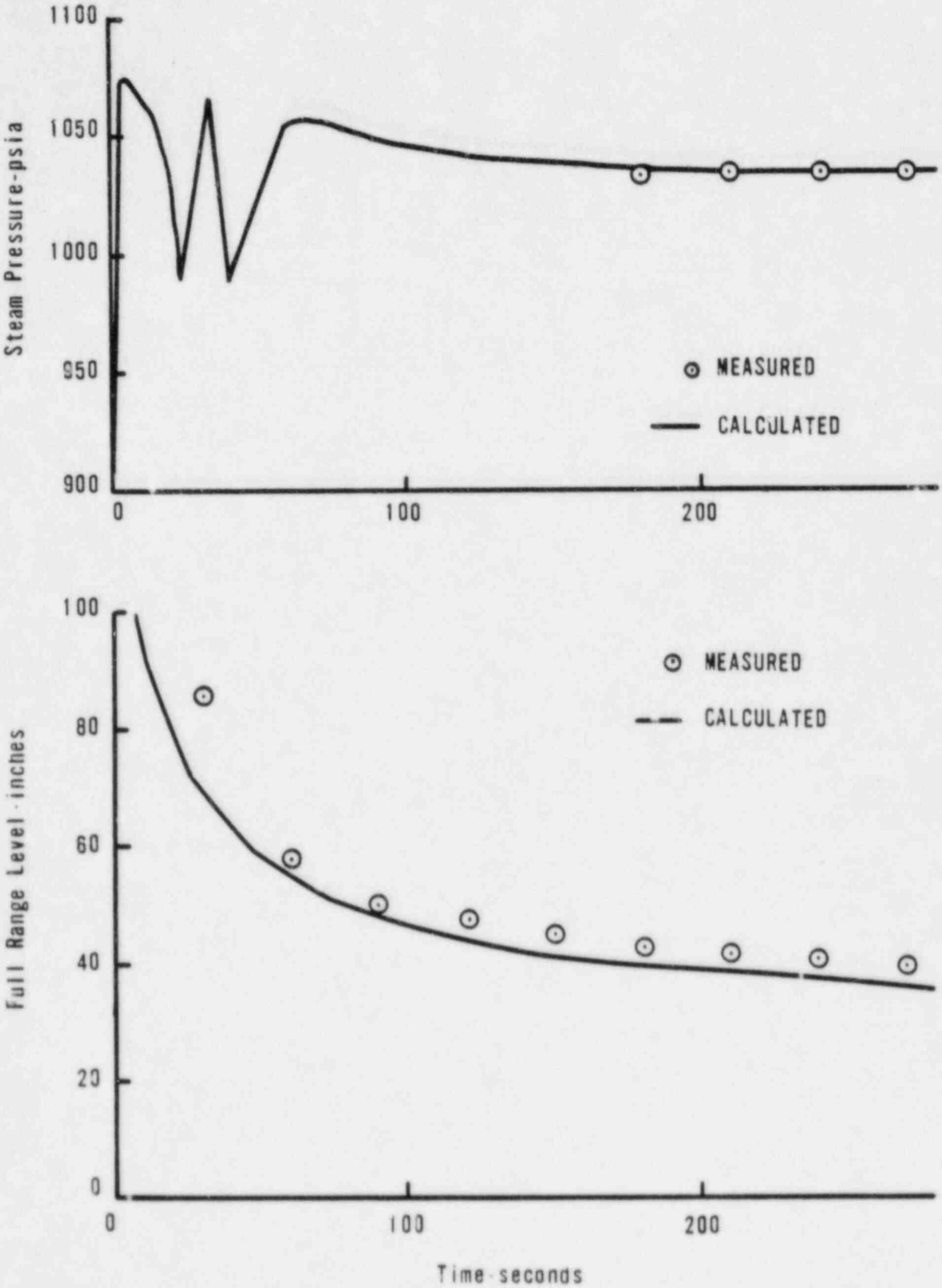


Figure 2-15A AUX BENCHMARK - CASE 4
LOOP EVENT AT ANO-1 FROM 100% POWER
(24 JUN 1980)

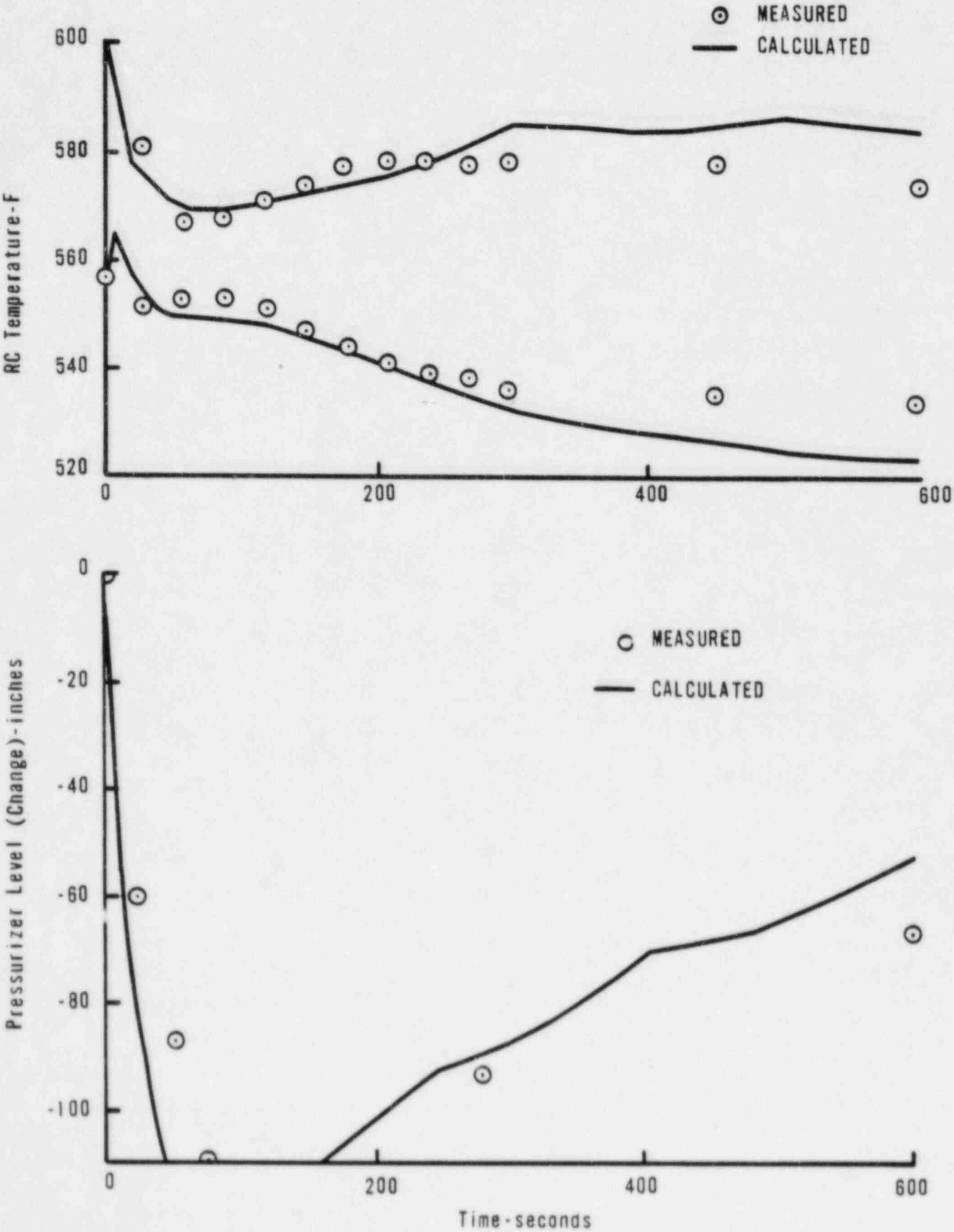
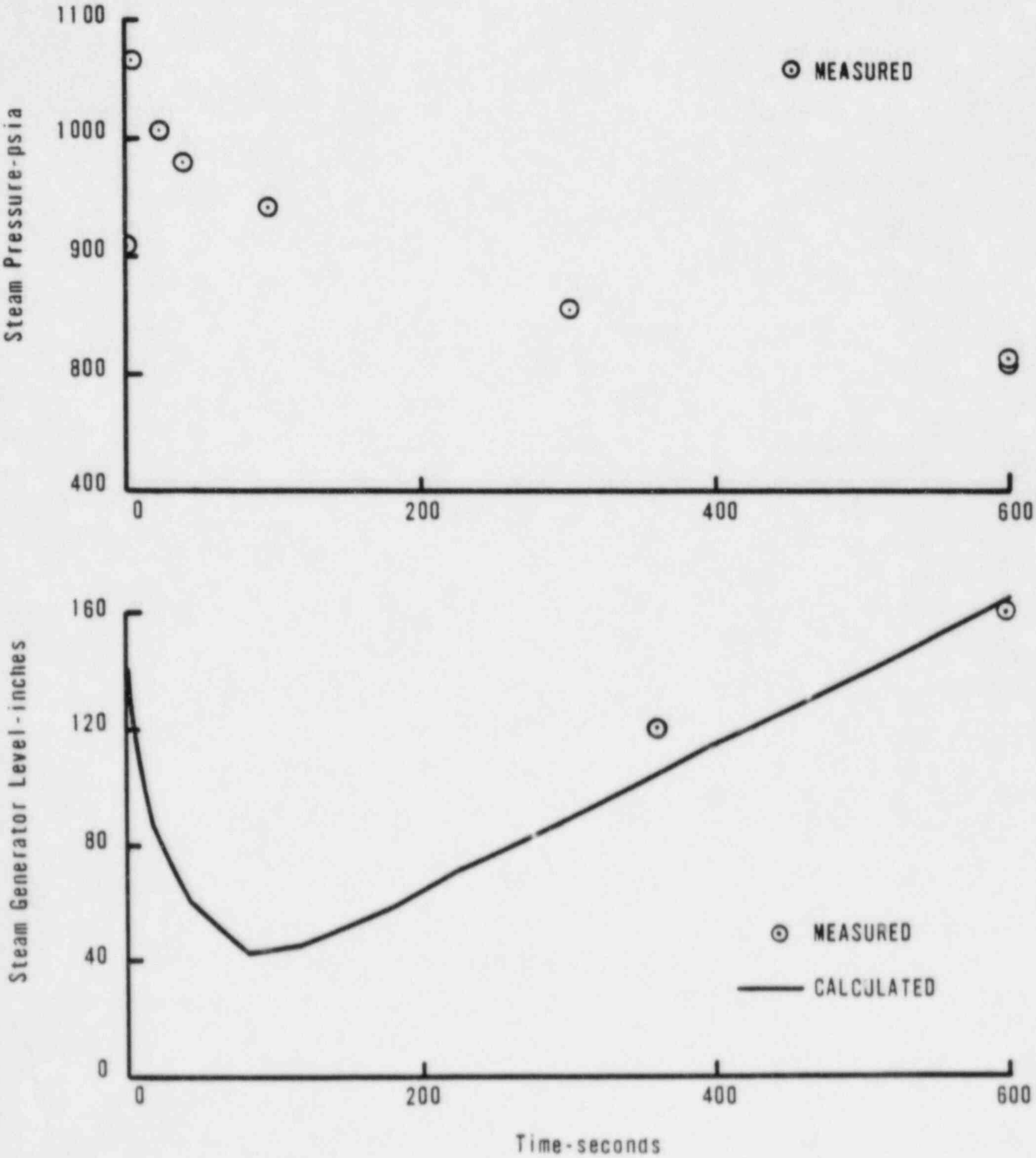


Figure 2-15B AUX BENCHMARK - CASE 4
LOOP EVENT AT ANO-1 FROM 100% POWER
(24 JUN 1980)



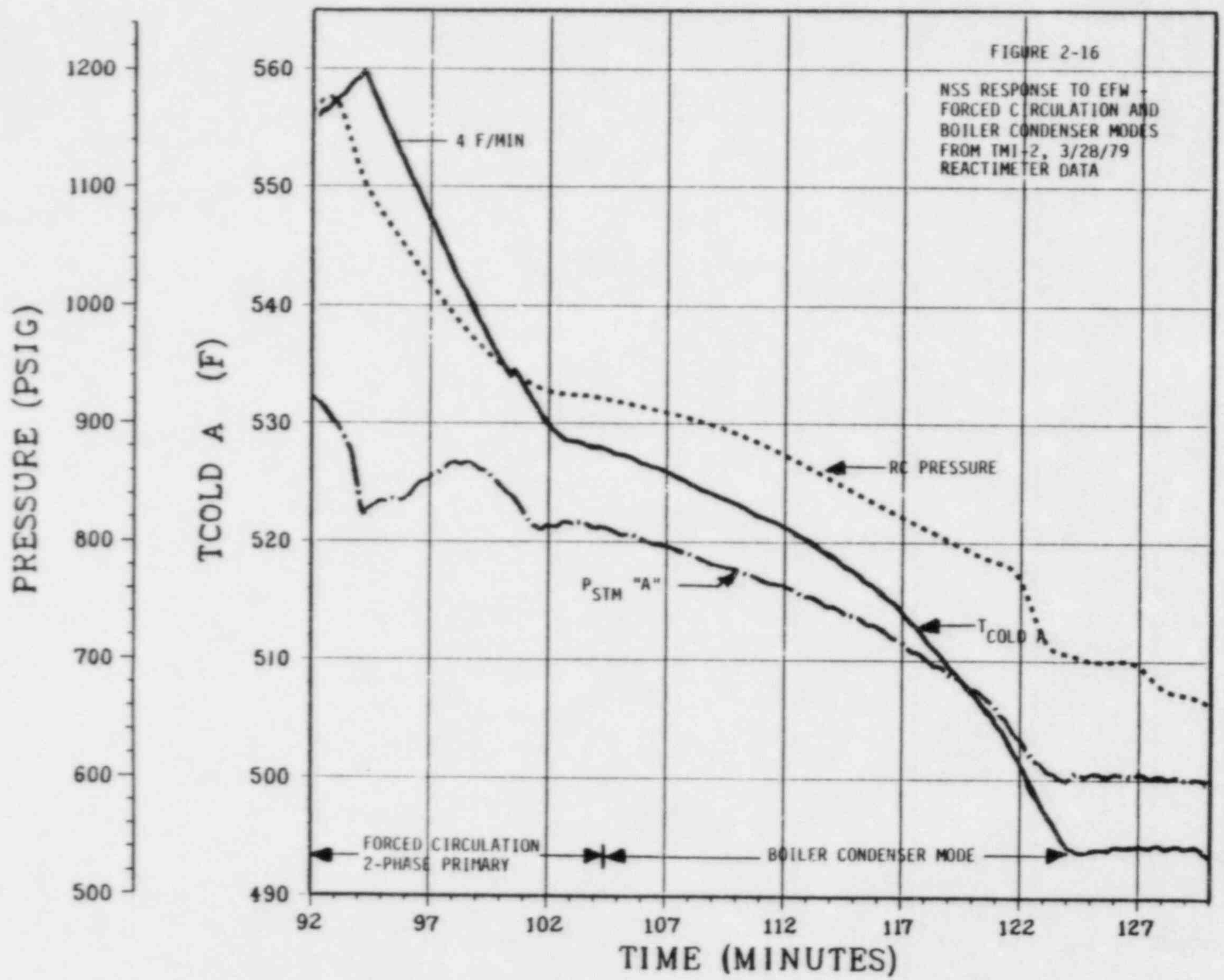
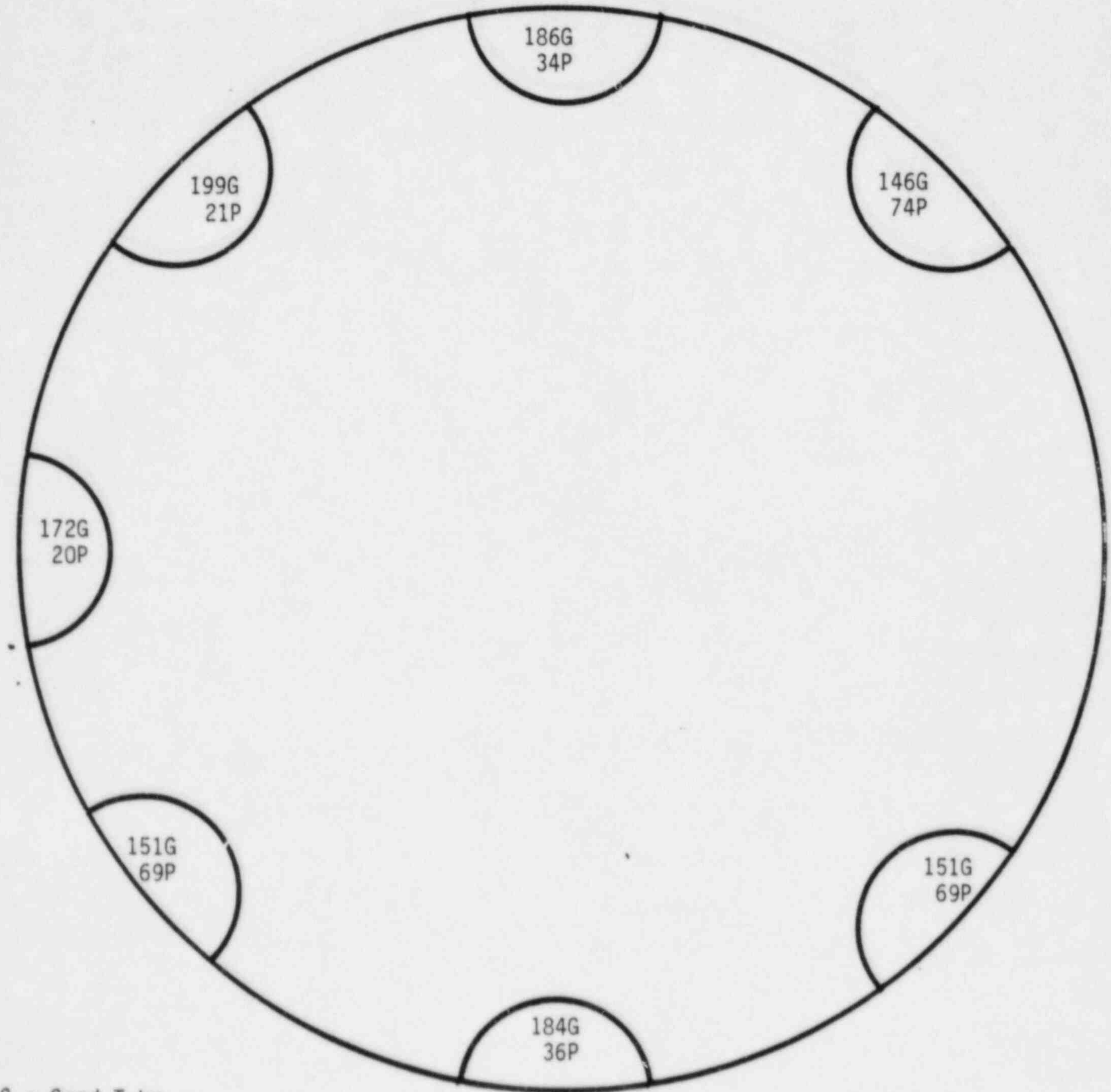


FIGURE 2-16
NSS RESPONSE TO EFW - FORCED CIRCULATION AND BOILER CONDENSER MODES FROM TMI-2, 3/28/79 REACTIMETER DATA
PRESSURE (PSIG)
1200
1100
1000
900
800
700
600
500
TCOLD A (F)
560
550
540
530
520
510
500
490
4 F/MIN
RC PRESSURE
P_{STM} "A"
$T_{COLD\ A}$
FORCED CIRCULATION 2-PHASE PRIMARY
BOILER CONDENSER MODE
92
97
102
107
112
117
122
127
TIME (MINUTES)

FIGURE 2-17

THREE MILE ISLAND NUCLEAR
GENERATING STATION, UNIT 1
STEAM GENERATOR A
TUBES PLUGGED IN EFW INJECTION REGIONS

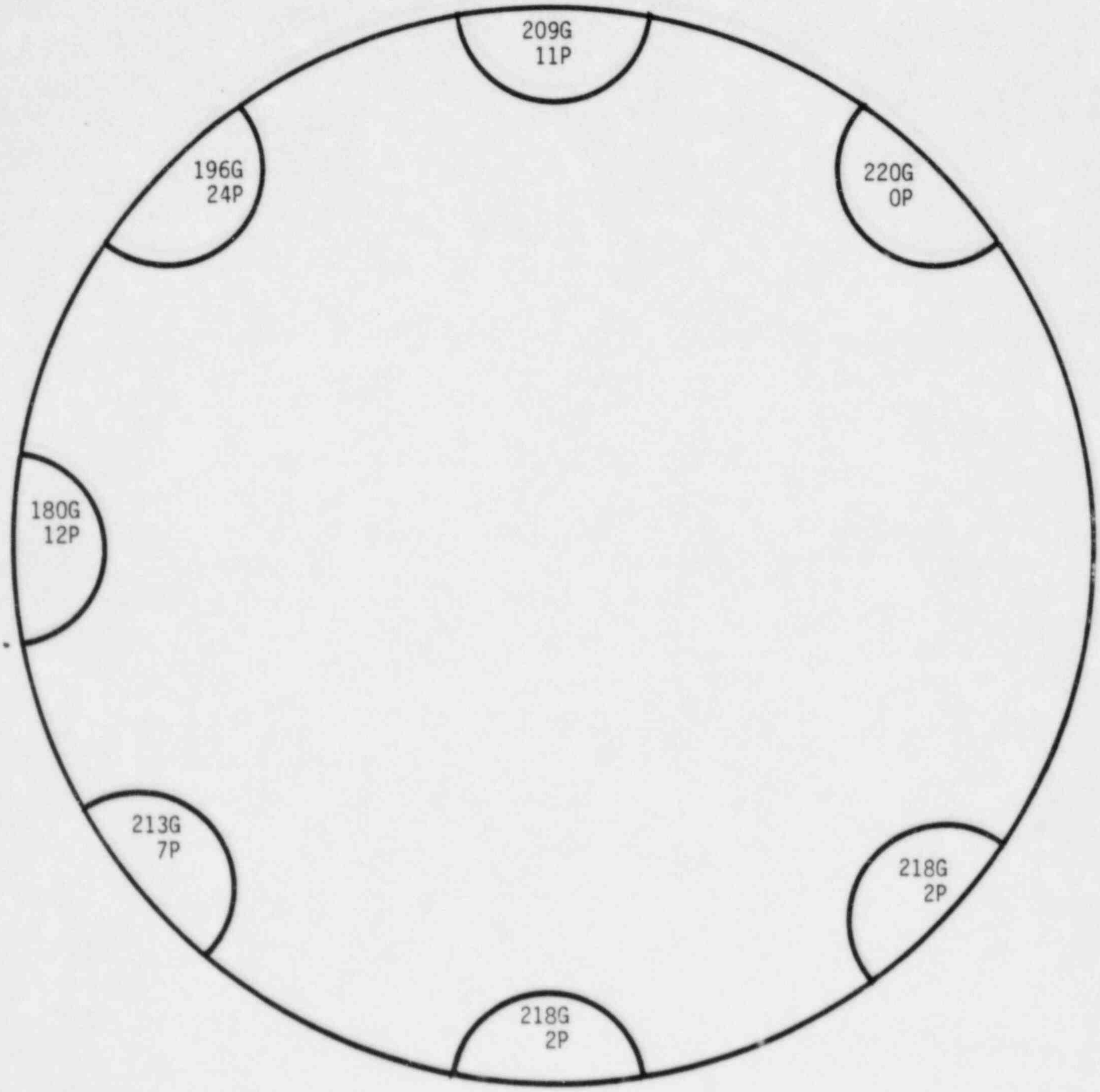


G = Good Tube
P = Plugged Tube

Semicircles
indicate zones
of penetration by
EFW.

FIGURE 2-18

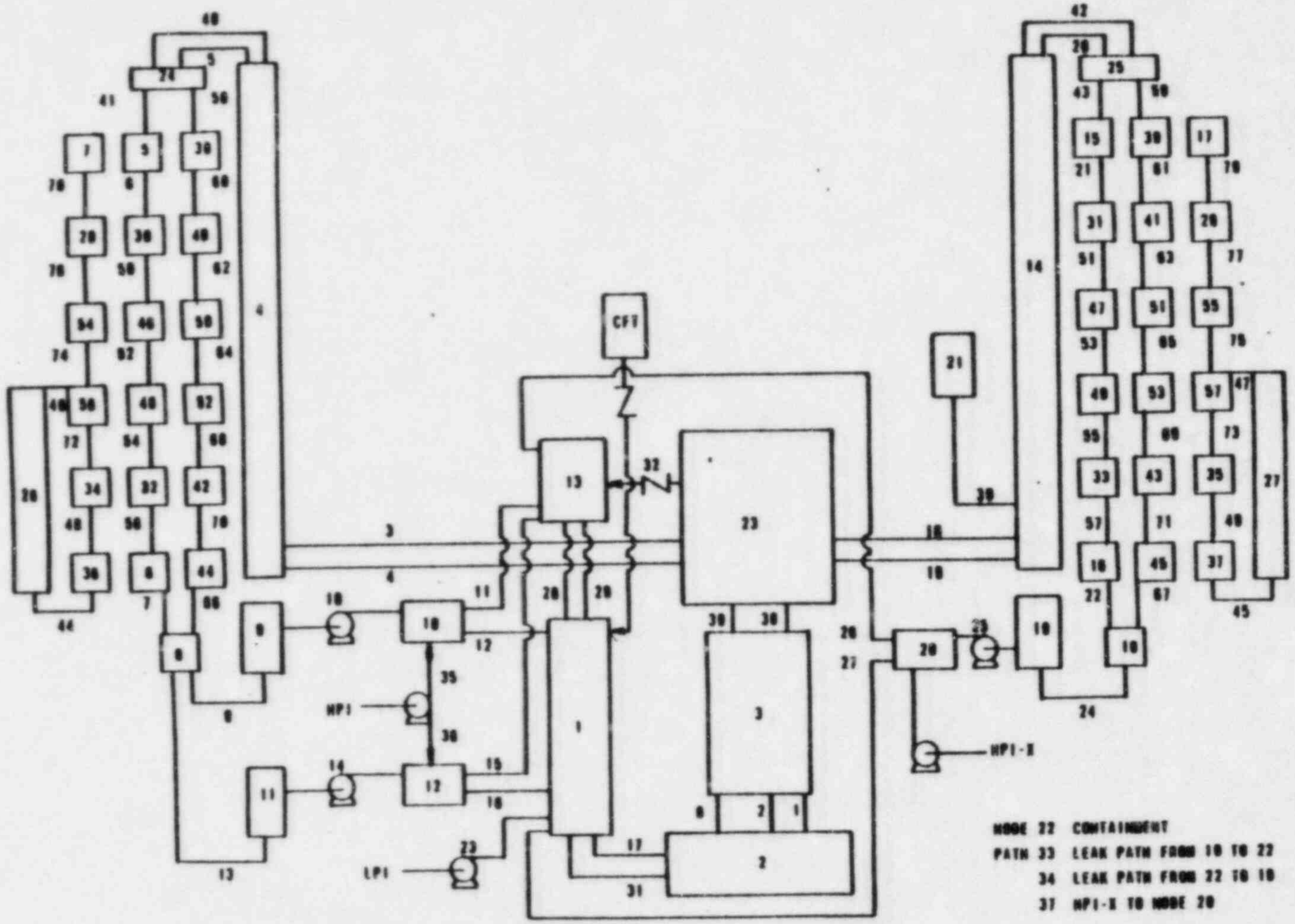
THREE MILE ISLAND NUCLEAR
GENERATING STATION, UNIT 1
STEAM GENERATOR B
TUBES PLUGGED IN EFW INJECTION REGION



G = Good Tube
P = Plugged Tube

Semicircles indicate zones of penetration by EFW.

Figure 2-19 CRAFT2 Nodal Diagram for Plant Transient Simulation



Babcock & Wilcox

Figure 2-20A CRAFT 2 BENCHMARK
LOOP EVENT AT ANO-1 FROM 100% POWER
(24 JUNE 1980)

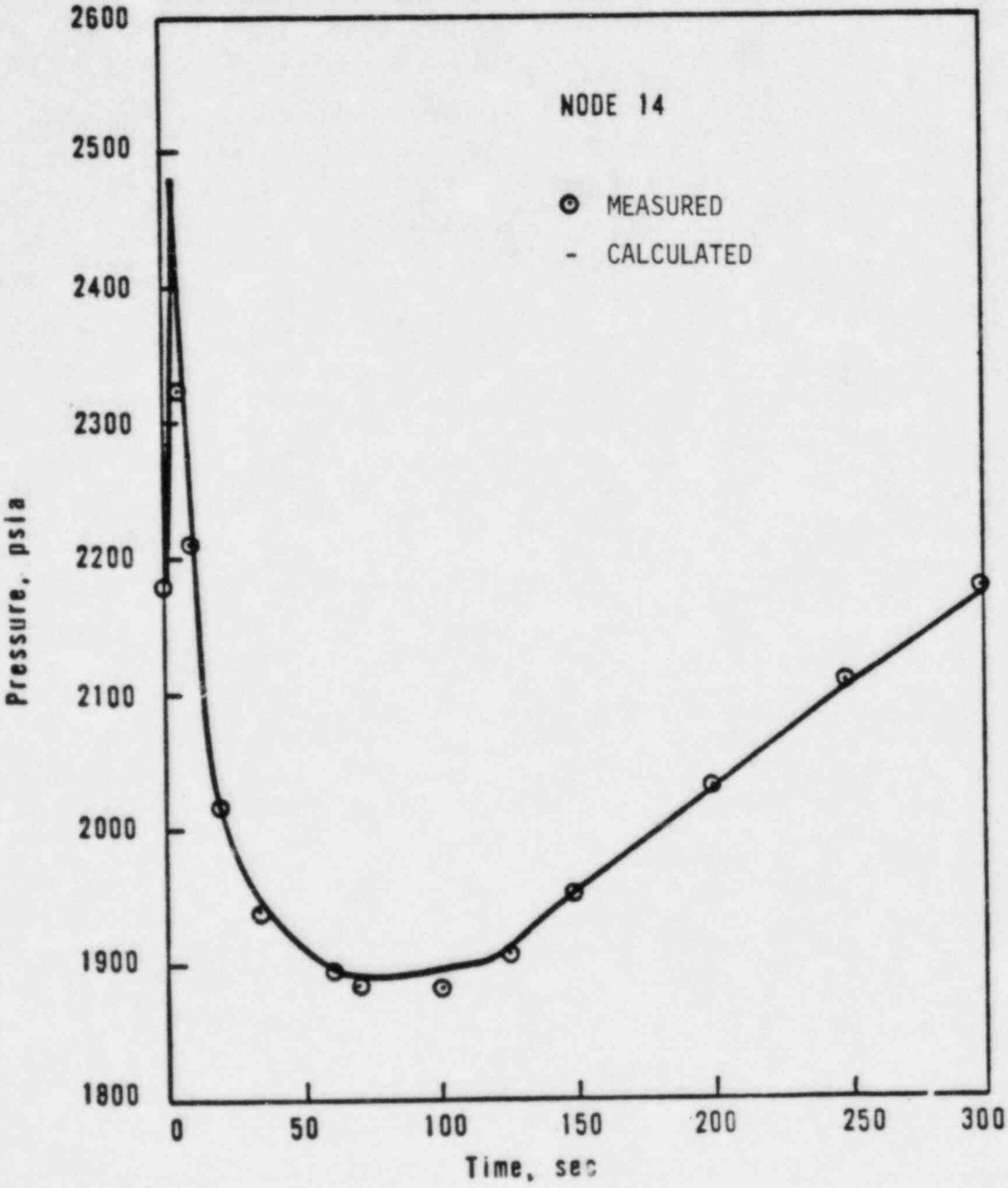
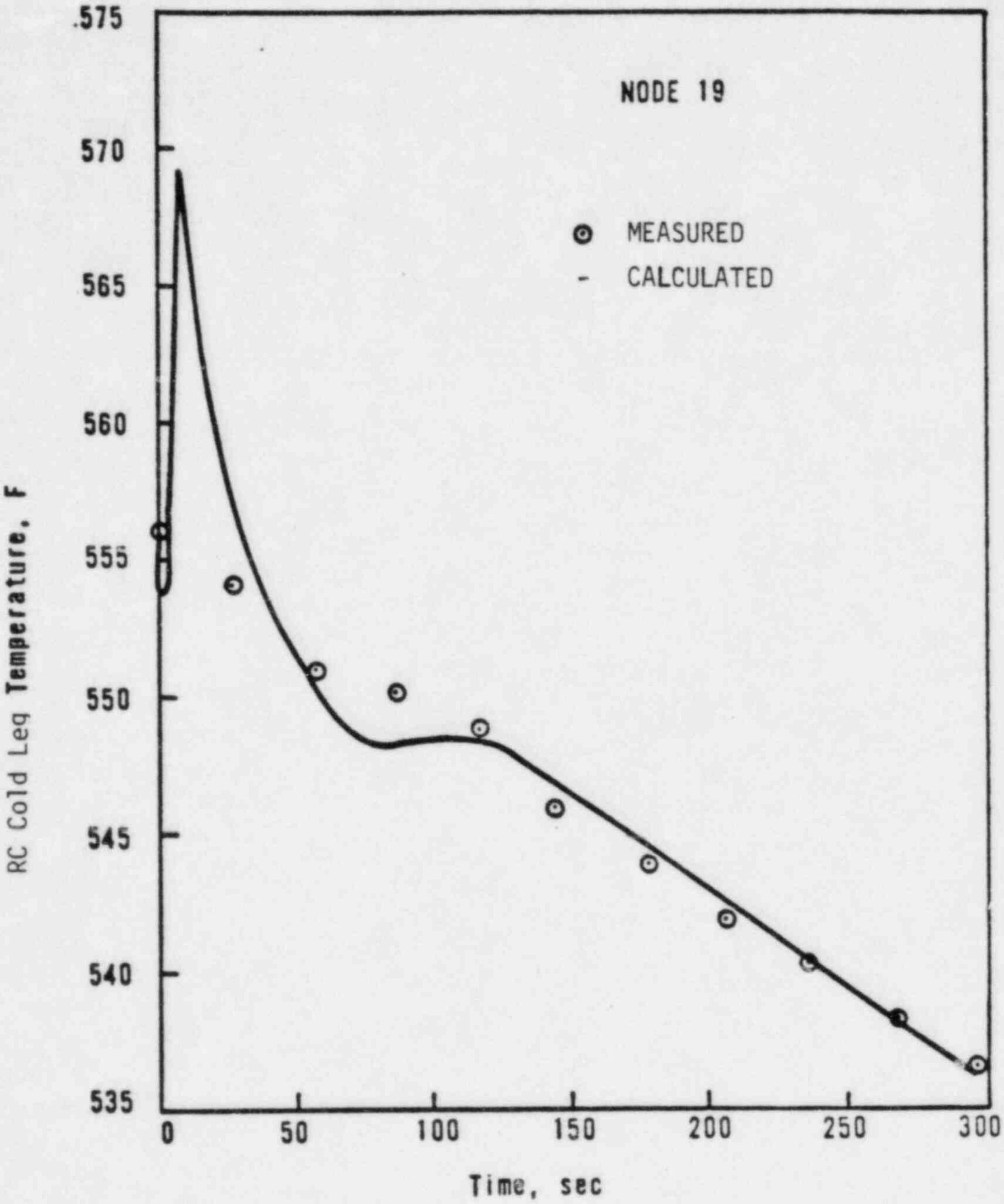


FIGURE 2-20B CRAFT 2 BENCHMARK
LOOP EVENT AT ANO-1 FROM 100% POWER
(24 JUNE 1980)



Babcock & Wilcox

Figure 2-20C CRAFT 2 BENCHMARK
LOOP EVENT AT ANO-1 FROM 100% POWER
(24 JUNE 1980)

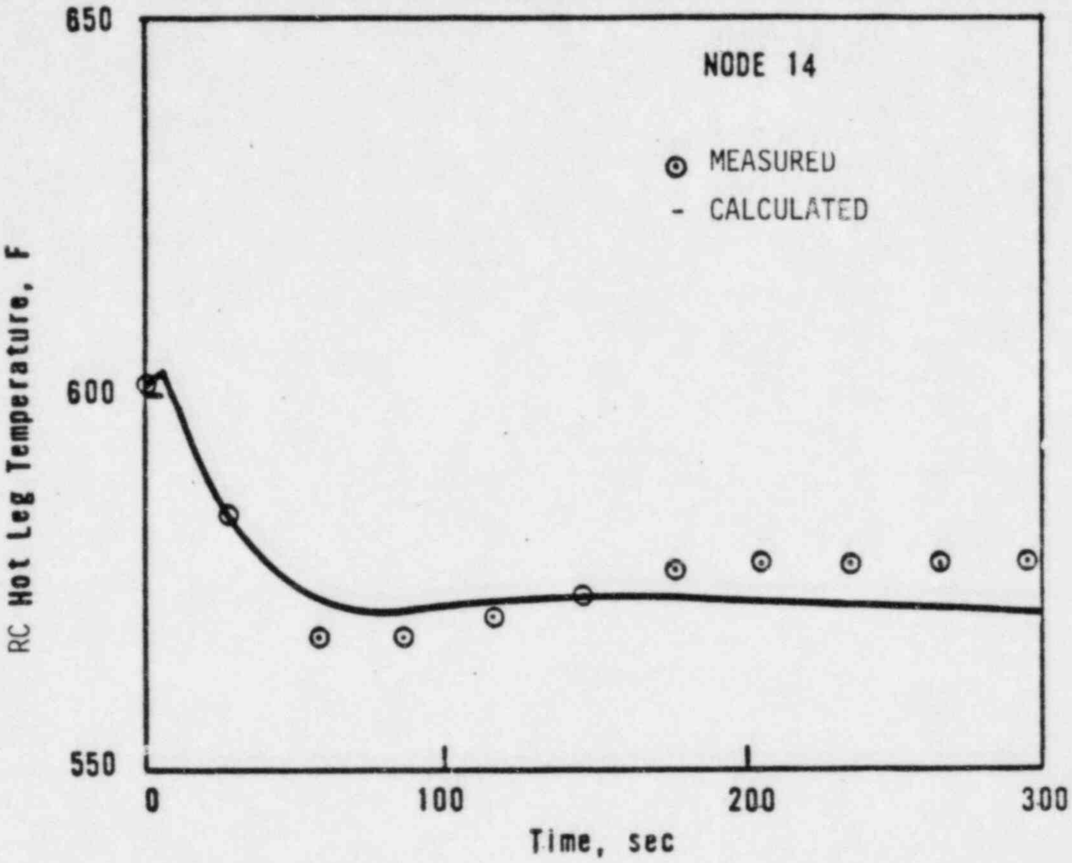


Figure 2-20D CRAFT 2 BENCHMARK
LOOP EVENT AT ANO-1 FROM 100% POWER
(24 JUNE 1980)

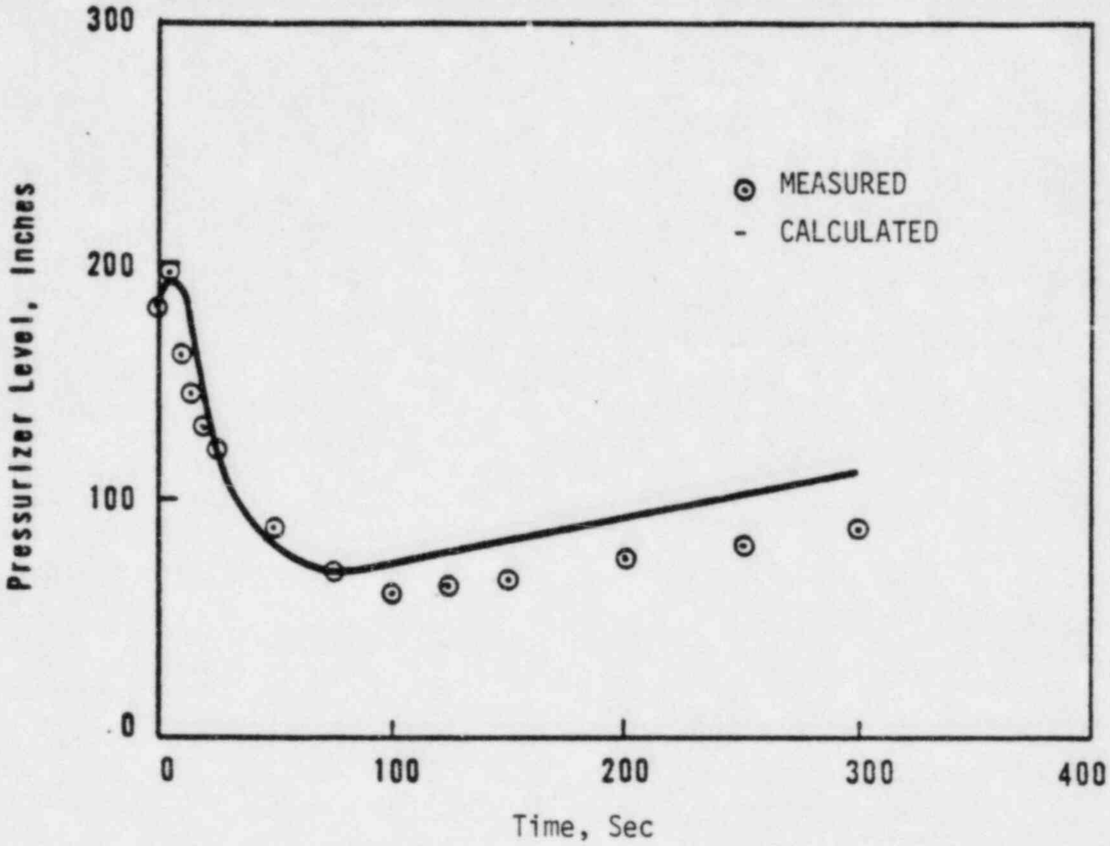
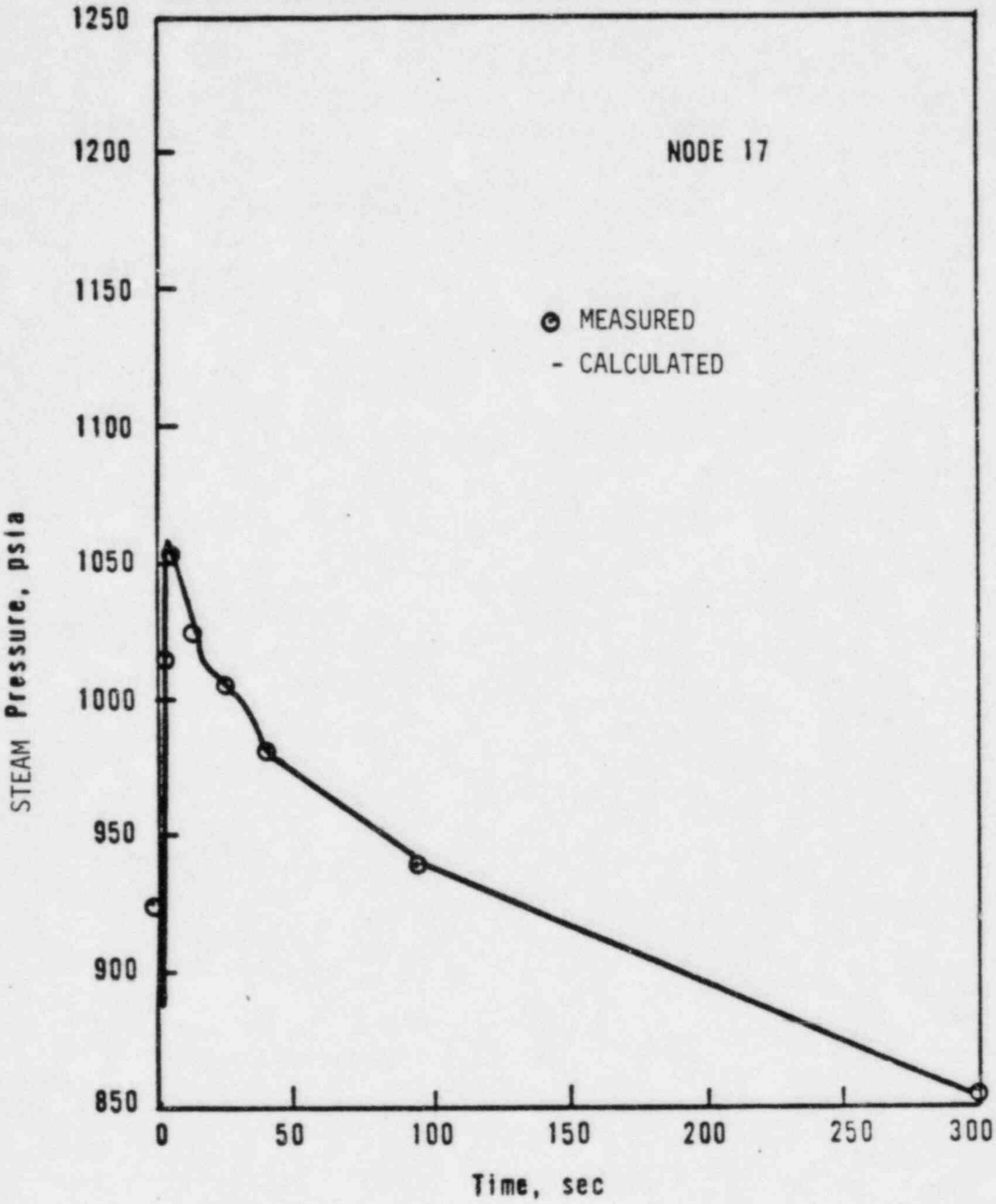


Figure 2-20E CRAFT 2 BENCHMARK
LOOP EVENT AT ANO-1 FROM 100% POWER
(24 JUNE 1980)



3.0 STEAM GENERATOR LEVEL REQUIREMENTS DURING A LOCA

Decay heat removal from the RCS can be accomplished in several ways, e.g., by break flow, steam generator heat removal, or by a combination of the break flow and steam generator heat removal.

Historically, analyses of large break loss-of-coolant accidents (LBLOCAs), including the design basis event (DBE), have shown that heat removal from the RCS occurs by means of the break flow. Therefore steam generator heat transfer and subsequently a steam generator level requirement is not needed to assure adequate core cooling.

Previous analyses, using certain conservative conditions, have indicated that there is a narrrow range of small break loss-of-coolant accidents (SBLOCAs), in which primary to secondary heat transfer is required to maintain core cooling. For B&W operating plants, adequate primary to secondary heat transfer will take place provided emergency feedwater (EFW) is controlled to prescribed SG level requirements. This section will identify the minimum SG level requirements to maintain core cooling for the full spectrum of SBLOCAs. The treatment of EFW in past and current small break analyses, including EFW spray effectiveness, will also be addressed. Finally, SG level requirements for isolatable SBLOCA's will be discussed.

3.1 Typical SBLOCA

A LOCA is considered small when its cross sectional area is 0.5 ft^2 or less. It involves a relatively slow system depressurization. Flow conditions within the reactor coolant system change gradually and smoothly. Temperature and pressure gradients between regions tend to be small. The lack of agitation allows partial phase separation of steam and water and, in some situations, countercurrent flow. Rather than the

distinct blowdown and reflood phases associated with large breaks, small breaks have a smooth transition from a period of relatively high core flow to one of relatively quiescent conditions. During the early phase, heat transfer in the core is flow-controlled and is adequate to keep the cladding cool. Later, during the quiescent period, a two-phase froth level can develop in the core region of the reactor vessel. To ensure adequate core cooling, it is necessary to maintain a two-phase level within the reactor vessel which is at or near the top of the core. In this manner, the core decay heat which is being generated can be removed from the fuel rods by pool boiling or, if the core is slightly uncovered, by convection to superheated steam. The HPI system has been designed to provide the necessary fluid makeup to the RCS to ensure adequate core heat removal.

Figure 3-1 shows the plant response for a typical SBLOCA, including pressure temperature plots and a sequence of events during the flow controlled portion of the transient. For very small breaks, natural circulation (normal or two phase) will not be lost, and the primary system can return to a subcooled state. For "larger" small breaks, the circulation flow phase will end almost immediately after the RC pumps are tripped.

After the "flow circulation" portion of a small break, the reactor coolant "settles out". The water falls by gravity and collects in the lower regions, and the steam separates from the liquid phase and collects in the high points of the system. A "boiling pot" water level will exist that will vary depending on:

- o Break Size
- o Break Location
- o Primary to Secondary Heat Transfer (steam generator cooling)
- o Number of HPI/LPI Pumps Operating
- o Decay Heat Levels

These variables can cause the response characteristics of the RCS to change in different ways after the reactor coolant system "settles" into the "boiling pot". However, four main categories of SBLOCAs are possible, namely:

1. SBLOCA's which maintain natural circulation
1. SBLOCA's which maintain natural circulation
2. SBLOCA's which may pressurize in a saturated condition
3. SBLOCA's which stabilize at approximately secondary side pressure
4. SBLOCA's large enough to depressurize the RCS

A detailed description of the plant response for each of the above categories of SBLOCAs is provided in Reference 1.

3.2 Minimum SG Level Requirements

For some of the SBLOCA categories identified in Section 3.1, the steam generators must be available to remove a portion of decay heat added to the primary system in order to ensure adequate core cooling. Removal of decay heat keeps system pressure at a level where HPI flow is adequate to maintain the core covered by a two phase mixture. The requirement for SG heat removal is determined by the system response or break size.

Typically, for break sizes smaller than 0.005 ft^2, the HPI system can compensate for the break flow and maintain the primary coolant loops essentially full of liquid so that natural circulation is not interrupted.

For break sizes between 0.005 ft^2 and 0.02 ft^2 SG heat removal is required and SG level requirements have been set for this range of break sizes.

Assuming a break size between 0.005 and 0.02 ft^2, the HPI flow is unable to compensate for the leak flow and the RCS will saturate. Steam pockets

will eventually form and grow to a volume sufficient to fill the 180° inverted U-bends at the top of both hot legs. This will result in an interruption of natural circulation. The loss of natural circulation leads to a loss of heat removal via the steam generators and the system will pressurize. As the RCS continues to lose inventory, a condensing surface will be exposed in the steam generators. This will establish the boiler-condenser mode of heat removal. This mode of heat removal will terminate the pressure increase and control RCS pressure at a value sufficient to assure adequate HPI flow for core cooling.

During SBLOCAs where SG heat removal is required, boiler condenser cooling is the principle heat transfer mode. Small break analyses in References 1-4 demonstrate the adequacy of this cooling mode. Using these analyses which assume the loss of one HPI pump, minimum SG level control requirements for B&W operating plants were defined for the three general categories of SBLOCA. Table 3-1 summarizes these requirements and identifies how SG heat transfer impacts core cooling. Further discussion on the analytical basis for the SG level control is provided in Section 3.3

As indicated in Table 3-1, the SBLOCA requirements for SG level control (assuming low pressure ESFAS actuated and RC pumps tripped) are:

1. Establish and maintain 95% on the operate range for the 177 FA lowered loop (LL) plants.
2. Establish and maintain 93 inches indicated on the startup range for Davis Besse if the subcooling margin is lost.

It should be noted that for the 177 FA LL plants, no EFW would be required to maintain core cooling if credit for two HPI trains were taken. How-

ever, to provide simplified EFW control guidelines, these levels have been implemented to meet minimum requirements for maintaining core cooling margin independent of the number of HPI trains available.

3.3 Analytical Basis For SG Level Requirements

SBLOCA analyses have been performed which demonstrate the adequacy of SG level control and EFW spray heat transfer to maintain core cooling during SBLOCAs. These analyses were performed in two separate steps. The first step was to determine the time period during which SG heat removal will assure adequate cooling with HPI. The second step was to verify during this time period that the core remained covered. Beyond this time period, the flow from one HPI pump alone is sufficient to match decay heat and provide core cooling.

3.3.1 Determination of Time Period for SG Heat Removal

A heat transfer analysis of the steam generator, while operating in the boiler-condenser mode, was performed to develop the pressure/time relationship for all B&W operating plants. Prior to any possible uncovering of the core, the full condensing surface of the steam generator will be exposed. Using this surface area, an analysis was performed to determine the RCS temperature, and hence pressure, necessary to condense all the steam being generated as a result of core decay heat removal as a function of time. It should be noted that since none of the generated steam is assumed to be removed via the break, this analysis would overpredict the RCS pressure and thus underpredict the HPI flow that would exist just prior to possible core uncovery. This pressure vs. time plot was compared to the pressure vs. time plot required for one HPI (ie, 70% of one HPI pump for lowered loop plants and 50% of one HPI pump for Davis Besse) to remove decay heat. The intersection of these two curves determines the

time period after which SG heat removal via boiler condenser cooling assures adequate HPI flow to maintain core cooling.

Three different analyses have been made for this time period:

a. <u>177 FA Lowered Loop Plants</u>

Figure 3-2 shows the results for the generic 177 FA LL plants taking credit only for emergency feedwater spray cooling. No credit was taken for heat transfer below the SG water level since it is less limiting. This figure shows that after 3200 seconds SG heat removal will ensure adequate HPI to maintain core cooling. For these plants only 70% of one HPI train was assumed available for core cooling.

b. <u>TMI-1 Plant Specific Analyses</u>

Figure 3-3 presents plant specific results for TMI-1 which takes into account the impact of the plugged tubes on boiler condenser cooling, a power level of 2568 MWt (which is in excess of the licensed power level of 2535 MWt) and plant specific HPI flows. This figure shows that SG heat removal after 1700 seconds will maintain core cooling. This case is bounded by the 177 FA lowered loop time period.

c. <u>Davis-Besse 1 Plant Specific Analyses</u>

Figure 3-4 presents results for Davis Besse. Credit for boiler condenser cooling below the secondary water level (93" on the startup range) only was assumed since this EFW level would be established early in the transient and would be fully exposed prior to core uncovery. Plant specific HPI flow was assumed. For this evaluation, decay heat was based on 120% ANS and credit for only 50% of one HPI was assumed per Appendix K requirements.

From this figure it is seen that the boiler condenser mode of cooling will ensure adequate HPI for core cooling after approxiamtely 7200 seconds. This is the most limiting time period.

The results of these three analyses are summarized in Table 3-2.

3.3.2 CRAFT2 Analysis

CRAFT2 analyses can be used to show that, for the time period determined in Section 3.3.1, the core will remain covered.

a. 177 FA Lowered Loop Plants

For B&W 177 FA LL plants, several small break LOCA analyses have been performed which show that the core can not become uncovered prior to 3200 seconds (bounds TMI @ 1700s) for the break size of interest. In Reference 2, (the 10CFR50.46/Appendix K analyses for this plant type) it can be seen that the 0.04 ft^2 break reaches its minimum system inventory at approximately 3000 seconds. No uncovering of the core is calculated for this break. Since smaller breaks would lose inventory at a slower rate, the 0.04 ft^2 break would bound the results. In addition, the analyses of the 0.01 ft^2 break in Reference 3 show that the boiler-condenser mode of cooling is calculated to occur at approximately 1500 seconds. At this time, there is a substantial quantity of liquid (105,600 lb or 2440 ft^3) remaining above the top of the core. This inventory would have to be lost through the break prior to the core uncovering. Extrapolation of these results indicate that adequate water inventory would be maintained to ensure core cooling to 3200 seconds. Based on this analysis, it is clear that uncovering of the core would not occur prior to 3200 seconds for the break size range for which boiler-condenser heat removal is necessary. Since the boiler-condenser cooling mode assures adequate

pressure control after this time i.e. HPI equals or exceeds the core boil-off, adequate core cooling is assured.

b. Davis-Besse 1 Analysis

For Davis Besse 1, the time to match boil off due to decay heat with HPI is approximately twice as long as for the generic lowered loop plants. This time difference results because 50% of the flow from the available HPI pump, as compared to 30% for the 177 FA LL plants, is assumed to exit the break. Using the 0.01 ft^2 analyses provided in References 1 and 3, extrapolations have been performed to show that no core uncovery would occur before the time that HPI flow equals boil off due to decay heat.

3.3.3 The Impact of NUREG 0737 on These SBLOCA Analyses

The analyses described above and documented in References 1, 2, and 4 were performed utilizing the presently approved CRAFT2 code. A simplistic steam generator model was used to predict primary-to-secondary heat transfer. Additional calculations have also been performed using revised models (see Reference 3) which utilize mechanistic heat transfer models to more realistically predict system response. A description of the approved CRAFT2 SG model and upgraded model proposed in Reference 3 is given below. Applicable analyses which support current SG level requirements are then discussed.

The SG model in the currently approved CRAFT2 code uses a simplistic heat transfer correlation relying solely on the primary-to-secondary liquid temperature difference to derive heat transfer rates. The heat transfer coefficients are based on initial conditions of reactor power, RCS flows and operating delta T's. These coefficients are then held constant throughout the analyses. The SG secondary side level affects heat

transfer only as a switching mechanism for emergency feedwater (EFW) injection. EFW injection impacts heat transfer by increasing the primary-to-secondary temperature difference. As cold EFW enters the SG secondary side, the energy balance produces a lower secondary temperature, which results in larger delta T's and heat transfer rates. With this model the EFW spray effect was not mechanistically treated.

The revised CRAFT2 SG model utilizes more mechanistic correlations which calculate heat transfer coefficients during the transient based on fluid conditions, flows, delta T's, and boiling lengths. These various parameters are then used in a regime-dependent correlation set and an implicit tube calculation to obtain the resulting heat transfer rates. A more detailed discussion of the heat transfer correlations can be found in Reference 5. The model includes such special features as level rate-dependent EFW control, a condensation heat transfer model and EFW spray effectiveness. Section 2 provides the development and basis for the treatment of EFW effectiveness in the upgraded CRAFT2 Model. However, the condensation heat transfer coefficients have been evaluated to underpredict the test data, thus, the overall primary-to-secondary heat transfer for the boiler-condenser cooling mode is conservative.

An evaluation has been performed, and documented in Reference 3, which compares the transient results of the .01 ft^2 Cold Leg Pump Discharge SBLOCA analysis for the generic 177 FA Lowered Loop and the 177 FA Raised Loop Plants. The comparison included the effects of both SG models on transient results. An extrapolation of the current analyses, using the upgraded SG model, has been performed to the point of long term cooling, and shows good agreement with previous results. Although transient event timing has been altered to a slight degree, core cooling results are the same for both models. Furthermore, the revised analyses prediction of sufficient heat removal, via the boiler-condenser mode, and adequate core cooling, through the long term, have verified the previous SBLOCA

transient results. Thus, thc revised models will not alter the conclusions described in 3.3.2.

3.4 Isolatable SBLOCAS

A LOCA can result from the postulated failure of high energy lines attached to the RCS. These SBLOCAs can be classified in three ways with respect to isolatability: (1) can be isolated following ESFAS initiation of containment isolation (i.e., those outside containment), (2) be isolated by operator action from the control room or, (3) can be non-isolatable. Table 3-3 lists some typical SBLOCAs which can be isolated by the operator. The double-ended rupture of the pressurizer spray line between the spray line valves is the largest possible SBLOCA of this category. As indicated in Table 3-3, some isolatable SBLOCAs can fall within the break size range where primary to secondary heat transfer is required to maintain core cooling using Appendix K assumptions (i.e., 120% ANS Decay Heat, failure of one HPI train, etc).

If a SBLOCA which has interrupted natural circulation is isolated, the SBLOCA requirements regarding SG level control should be followed until the reactor coolant is adequately subcooled and circulation is established. With maintenance of these SG levels, boiler-condenser cooling will ensure that one HPI will be sufficient to maintain the core covered. Using the SG analysis method developed in Section 3.3 boiler condenser cooling will allow HPI to ensure core cooling by 900 seconds for TMI-1, 1300 seconds for the generic 177 FA LL plants and 400 seconds for Davis Besse. For this evaluation full flow from one HPI pump was assumed since flow cannot be lost directly out the break. These time periods are illustrated in Figures 3-2 to 3-4.

In order to assess core cooling following a isolatable break, a double-ended rupture of the pressurizer spray line between the two valves was

evaluated. This break size (.049 ft^2) would result in the maximum inventory loss prior to leak isolation. Assuming leak isolation at 15 minutes, one HPI pump and zero primary to secondary heat transfer after 15 minutes (feed and bleed cooling), core uncovery was not predicted until one hour after the initiation of the transient. Since HPI can match decay heat in less than 1300 seconds with boiler condenser cooling for all 177 plants, core cooling is assured for isolatable breaks if the SG level control requirements (given in Table 3-1) are implemented prior to one hour after the initiation of the transient.

3.5 OTSG Rate of Fill

The rate of fill of the OTSG's during the Boiler Condenser mode of heat transfer depends on several factors, including, temperature and pressure, of the RCS and OTSG secondary, the EFW capacity, the time of EFW initiation and the decay heat power level. If it is necessary to raise the OTSG level from 50% to 95% on the operate range level the typical fill time may be estimated as follows: (a) Assume 2% decay heat when the boiler condenser mode is required, (b) Assume the EFW flow rate is 200 gpm per OTSG. An EFW flow rate of 200 gpm per OTSG is adequate to remove 2% decay heat (2772 MW_t). If no steaming occurs, the steam generators will fill at a rate of about 6.2 inches/min resulting in a fill time of about 21 minutes from 50% to 95% on the operate range. If the entire EFW flow (200 gpm per OTSG) is initially evaporated, the fill time will be significantly longer depending on the amount of HPI cooling, the rate of decrease on the amount of HPI cooling, the rate of decrease of the decay heat power level, and the actual EFW flow rate.

Therefore, depending on conditions, the time to fill from 50% to 95% on the operate range can vary over a wide possible range with a realistic time expected to be about 20 minutes.

3.6 Summary

Using 10CFR50.46/Appendix K assumptions, there is a narrow size range of SBLOCA's on B&W operating plants for which steam generator heat removal is necessary to ensure core cooling. Using the SG level (EFW) control requirements, adequate heat transfer will occur such that primary pressure remains sufficiently low for one HPI pump to maintain core cooling. For isolatable breaks, adequate core cooling can also be maintained with one HPI pump provided the SBLOCA requirements for SG level control are followed.

TABLE 3-1

SBLOCA SPECTRUM REQUIREMENTS ON
SG LEVEL CONTROL (WITH RC PUMP TRIPPED)

Break Type	Approximate Break Size ft^2	Minimum SG Level Required 177RL	177LL	Impact On Core Cooling
SBLOCAs which maintain natural circulation	<.005	(2) $\geq$35" on SU	(1) >50% on OR	Break is within capacity of one HPI pump. Flow controlled portion will be maintained and reactor coolant will remain or return subcooled. Boiler condenser cooling is not required. SG level must be high enough to support natural circulation.
SBLOCA which stabilize or repressurize at or above secondary pressure	.005<A<.02	$\geq$93" on SU	$\geq$95% on OR	During boiling pot phase, RCS repressurizes because SG condensation level is covered with primary coolant. This condition can occur when flow controlled period ends or out in time when system refills. High SG level promotes the establishment of boiler condenser cooling. When established, RC pressure will drop to or near secondary pressure allowing one HPI to maintain adequate core cooling.
SLBOCAs large enough to depressurize RCS	>.02	N/A	N/A	No EFW addition is required to maintain core cooling. The break is large enough to depressurize primary system below secondary system. This break size doesn't set a minimum SG level control setpoint for EFW control.

NOTES: (1) 95% on the operate range is required if the subcooling margin is lost.

(2) 93" on the startup range is required if the subcooling margin is lost.

TABLE 3-2

TIME AT WHICH BOILER CONDENSER COOLING ENSURES CORE COOLING

Plant	Time[1] At Which Boiler Condenser Cooling Ensures Adequate Core Cooling With 1 HPI
Generic 177 FA [2,4] LL Plant	3200
TMI-1[2,5]	1700
Davis Besse[3]	7200

Notes:

1. Based on 120% ANS Decay Heat.
2. Only 70% of the flow from one HPI was assumed available to match decay heat.
3. Only 50% of the flow from one HPI was assumed available to match decay heat.
4. EFW flow based on 200 gpm per SG.
5. EFW flow based on 350 gpm per SG.

TABLE 3-3

ISOLATABLE SBLOCAS

Failure	Line Size	Equivalent Break Area(ft^2)
PORV	--	$< .007\ ft^2$
Letdown Line Inside Containment	2 1/2" SCH 160	$\leq .024\ ft^2$
Pressurizer Spray (Double Ended Break)	2 1/2 SCH 160	$\leq .049\ ft^2$

FIGURE 3-1. TYPICAL SYSTEM RESPONSE DURING FLOW CIRCULATION PHASE OF SBLOCA

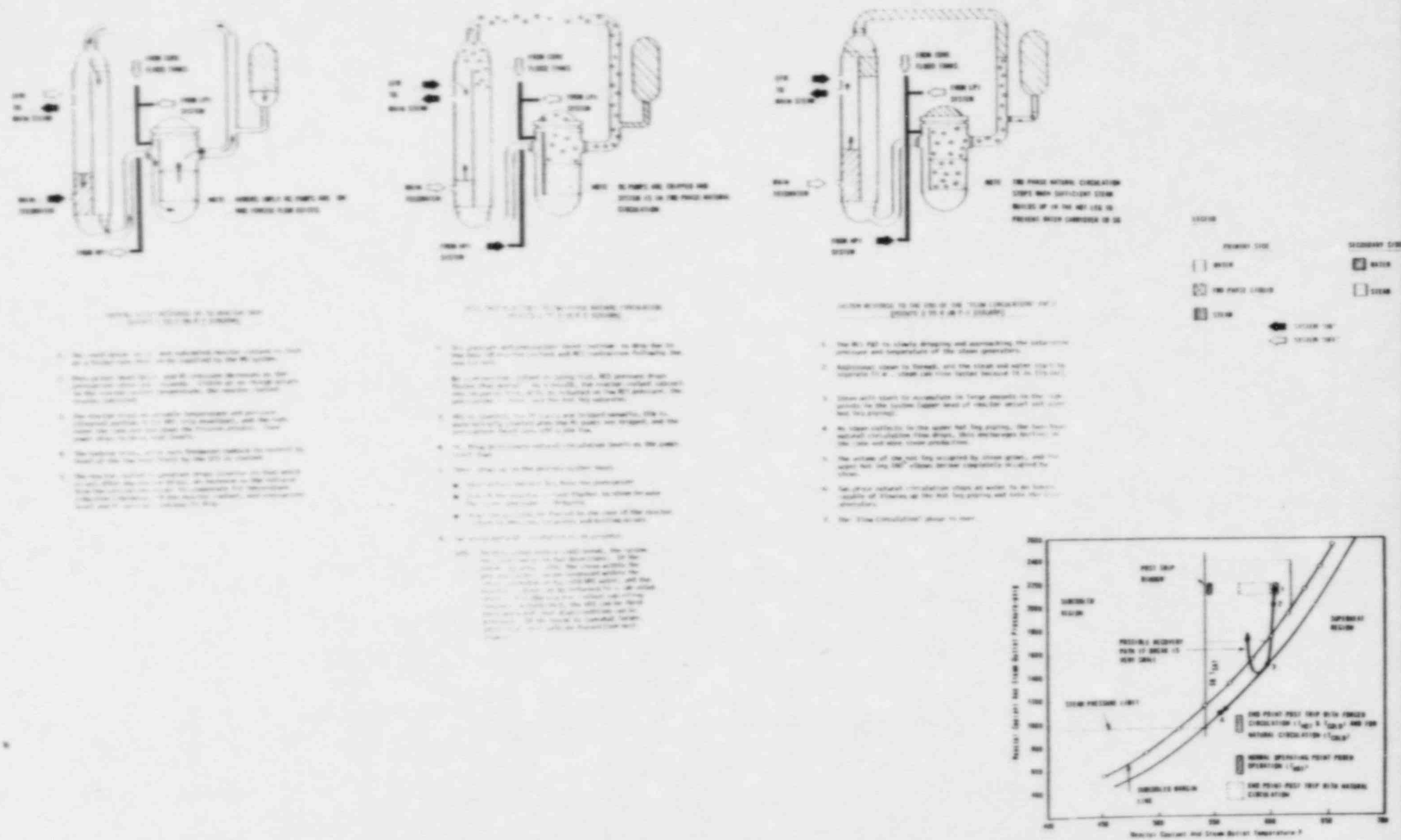


Figure 3-2
RELATIONSHIP OF HPI COOLING AND
BOILER CONDENSER COOLING FOR GENERIC
177 FA LL PLANT

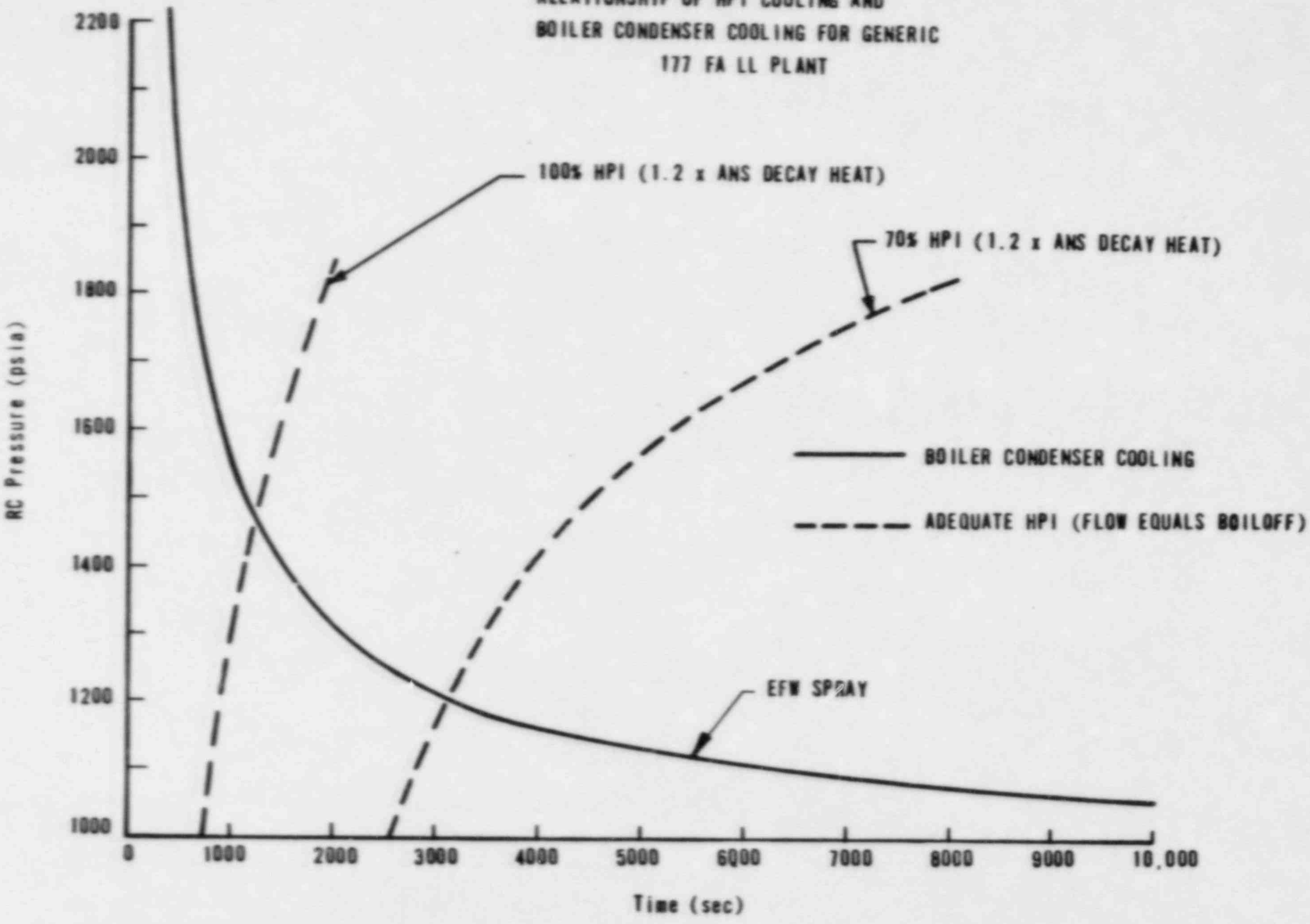


Figure 3-3
RELATIONSHIP OF HPI COOLING AND BOILER
CONDENSER HEAT REMOVAL
FOR TMI-1

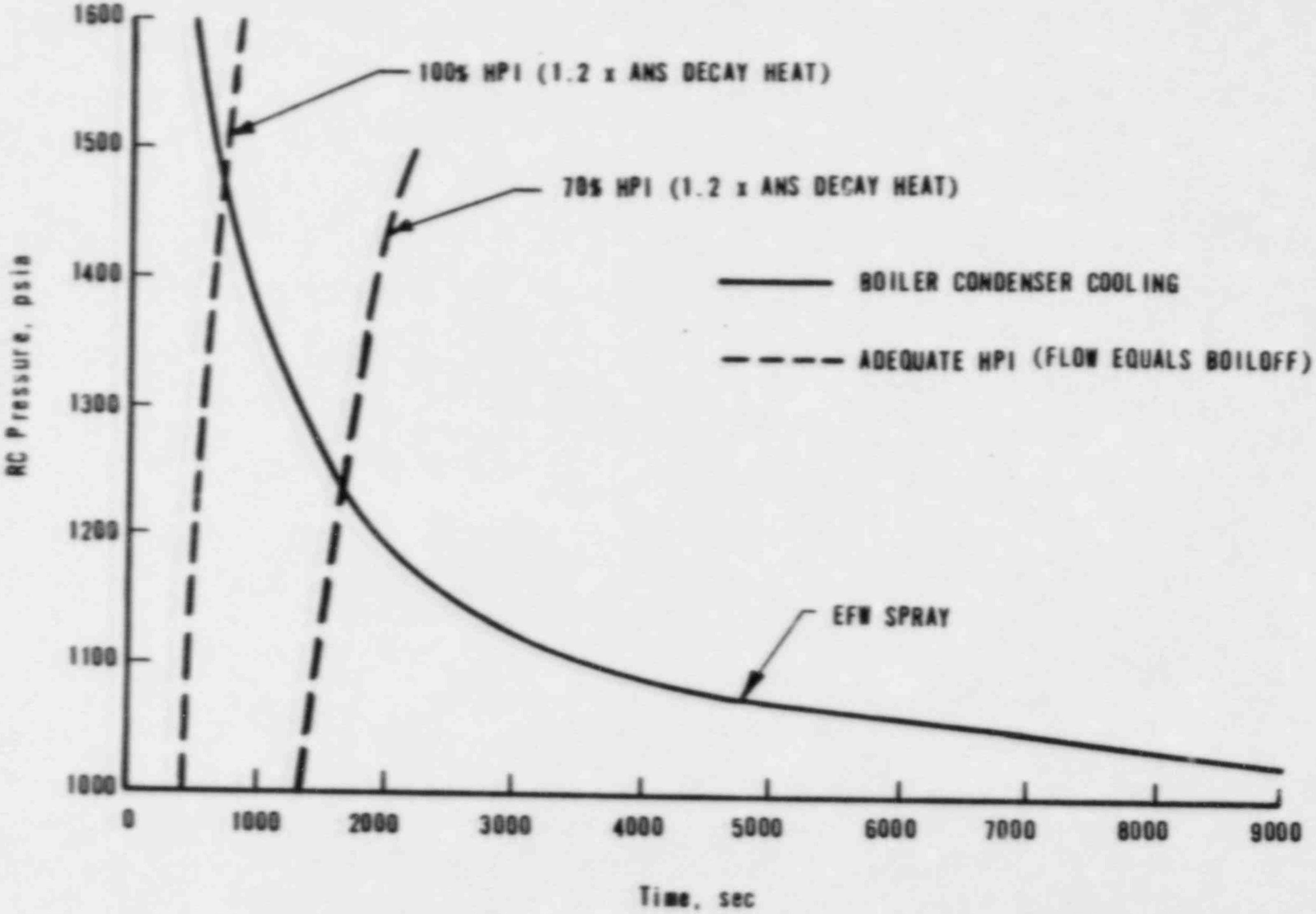
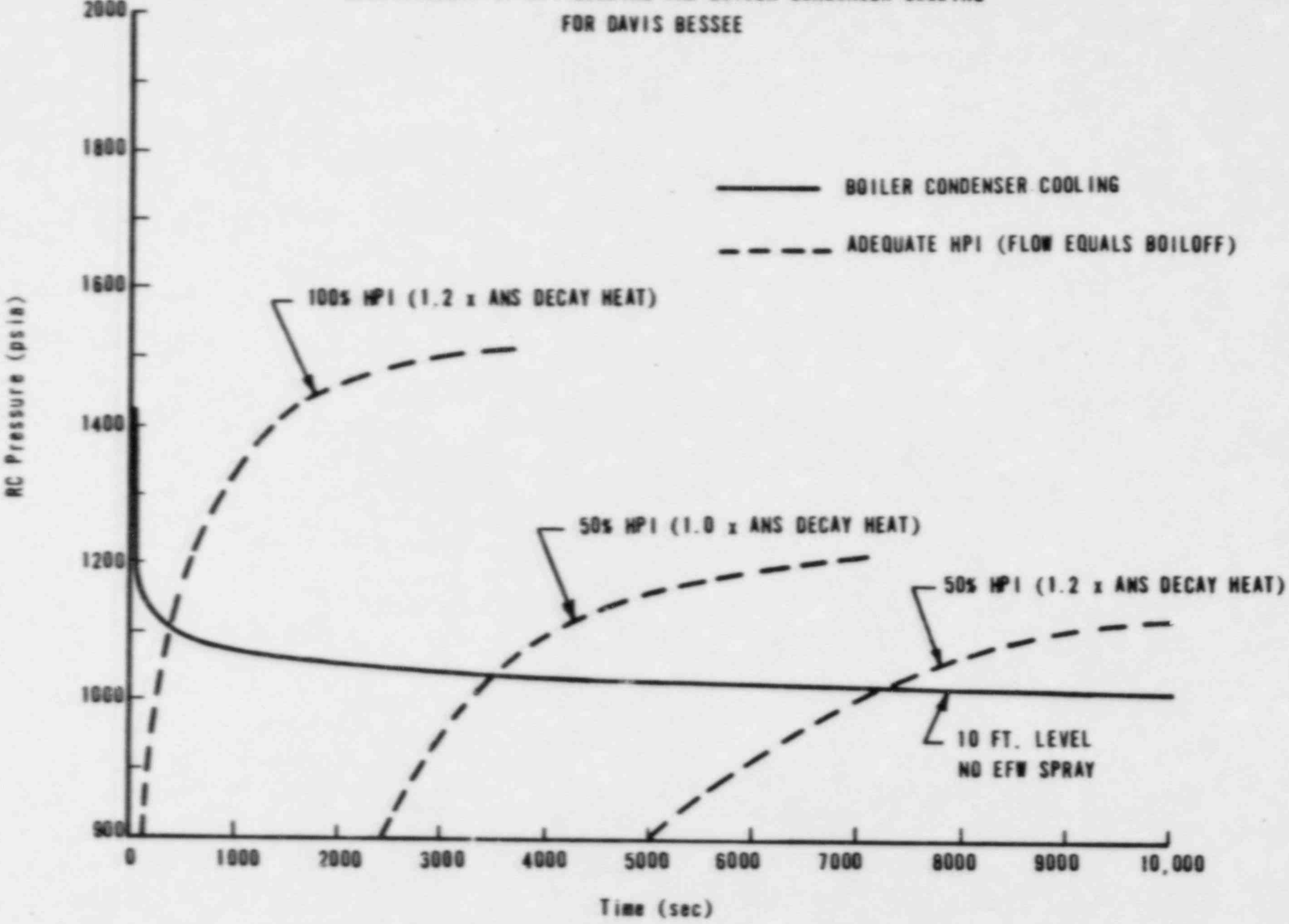


Figure 3-4

RELATIONSHIP OF HPI COOLING AND BOILER CONDENSER COOLING FOR DAVIS BESSEE



4.0 REVIEW OF OPERATING PROCEDURES

Plant procedures and B&W guidelines were reviewed to determine if sufficient guidance is provided to operators to assure safe mitigation of small breaks in the primary system. In particular, the requirements for steam generator level were reviewed to determine if adequate steam generator surface area is available for boiler condenser heat transfer when necessary during small break LOCAs. In addition, plant procedures which would be used during LOCAs, and those which might be used following isolation of small breaks were reviewed to determine if procedures contain conflicting requirements which would preclude adequate steam generator level.

The scope of the review included small break LOCA procedures and, as appropriate to the individual plant, additional procedures which address natural circulation and inadequate core cooling.

In performing this review, the following criteria were used:

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?

Procedure reviews to these criteria were performed for the following plants:

Davis Besse 1	Toledo Edison Company
Oconee 1, 2, 3	Duke Power Company
Crystal River 3	Florida Power Corporation
TMI-1	GPU Nuclear Corporation
Rancho Seco	Sacramento Municipal Utility District
Arkansas Nuclear One, Unit 1	Arkansas Power & Light Company

The reviews indicated that each of the plant's procedures provide the necessary guidance to assure the required steam generator heat removal and high pressure injection under SBLOCA conditions.

In addition, the abnormal transient operating guidelines (ATOG) were reviewed and found to adequately meet the above criteria.

Each utility can provide a description of the plant-specific manner in which their procedures are responsive to the above criteria.

5.0 CONCLUSIONS

Based on the results of these reviews, the following general conclusions have been reached:

- 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.
- Adequate core cooling is provided for the design basis event (large LOCA) and for the other transients considered, which include overcooling transients. The only event for which 95% level on the operate range (93-inches on the startup range for Davis-Besse) is required is a SBLOCA.

6.0 REFERENCES

1. Evaluation of Transient Behavior and Small Reactor Coolant System Breaks In The 177 Fuel Assembly Plant, May 7, 1979.

2. J.H. Taylor (B&W) to S.A. Varga (NRC), Letter, July 18, 1978.

3. N.K. Savani, J.R. Paljug, and R.J. Schomaker, B&W's Small-Break LOCA ECCS Evaluation Model, BAW-10154, Babcock & Wilcox, November 1982.

4. L.R. Cartin, J.M. Hill, and C.E. Parks, Multinode Analysis of Small Breaks for B&W's 177-Fuel Assembly Plants With Raised LOOP Arrangement and Internal Vent Valves, BAW-10075A, Rev. 1, Babcock and Wilcox, March 1976.

5. R.A. Hedrick, J.J. Cudlin, and R.C. Foltz, CRAFT2 - FORTRAN Program for Digital Simulation of a Multinode Reactor Plant During Loss of Coolant, BAW-10092, Rev. 3, Babcock & Wilcox, November 1982.