FOUR MINUTE ISOLATION OF POSTULATED STEAM LINE BREAKS AT THE FORT ST. VRAIN NUCLEAR GENERATING STATION

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ABSTRACT

This study was performed in an effort to validate the steam line rupture curves used in the Fort St. Vrain (FSV) environmental qualification program. These curves assume a four minute isolation time for any major steam line rupture in either the Reactor Building or Turbine Building. All possible locations of a major steam leak were analyzed including the condensate, feedwater, main steam, cold reheat, and hot reheat lines.

All pertinent automatic and manual actions were studied for each case. All manual actions were determined using the FSV Emergency Procedures. Times for each manual action were conservatively estimated to allow for longest reasonable response time. The results of each case are listed in tabular form.

All postulated line breaks can be isolated in four minutes or less. Worst case accident in the Reactor Building was determined to be a cold reheat line break, which is isolated in two minutes. Worst case in the Turbine Building was determined to be a hot reheat line break, which is isolated in four minutes.

Because of the longer termination time found on Reactor Building temperature curves, they are conservative. Because the hot reheat accident has a lower blowdown rate than that used for the Turbine Building curves, these curves, too, are conservative.

Conservatisms found in the steam line rupture curves provides additional margin during testing. All previous test results remain valid and all equipment is qualified.

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SECTION 1

INTRODUCTION

The environmental qualification program for safe shutdown equipment at FSV is based on two sets of curves which represent the time versus temperature response of the environment in the vicinity of the steam line break (Fig. 1 and Fig. 2). These curves were developed assuming the leak would be terminated in four minutes or less.

During the FSV preliminary environmental qualification meeting held with Public Service Company of Colorado (PSC) on February 29, 1984, members of the Nuclear Regulatory Commission's Environmental Qualification Branch (NRC-EQB) questioned the ability of an operator to properly identify a high energy line break and to initiate corrective action within four minutes.

This study was conducted in order to substantiate the assumptions made by Gulf General Atomic (GGA) when generating the steam line rupture curves. A case by case analysis was done for various ruptures occurring on the following high energy lines: condensate, feedwater, main steam, cold reheat, and hot reheat.

All appropriate actions, both automatic and manual, are used to determine the termination time. All operator actions are based on the existing FSV Emergency Procedures (EPs).

SECTION 2

BACKGROUND

The original basis for environmental qualification at FSV was presented in GGA Report GA-A12045. This report included a set of steam line rutpure curves which were to be used to establish test chamber temperature conditions in subsequent steam line rupture simulation tests.

To generate the steam line rupture curves, the computer code CONTEMPT-G was utilized. This code was a modified version of the CONTEMPT code, originally formulated for light water reactor containment conditions.

To compensate for the fact that equipment closer to the leak will experience higher temperatures, the analysis expressed heat transfer surface area as a function of distance. At the rupture, the surface area was zero, and increased with distance from the leak. With increasing surface area, more heat is transferred to the heat sinks and less heat is used to increase surrounding air temperature.

Thermal input to the building was based on the worst case accident. In the Reactor Building, it was determined that a cold reheat line break was the worst case. This was due to the fact that the Steam Pipe Rupture Detection System would automatically isolate a leak on a feedwater, main steam, or hot reheat line. Cold reheat would continue to be fed from the bypass flash tank until the operator acted to close the main steam bypass valves. (See Fig. 3)

Based on discussions with plant personnel, it was determined that it would take two minutes to identify and isolate the leaking line. To be conservative, a four minute time limit was established.

In the Turbine Building, it was decided that the worst case was the hot reheat line. This was based on the size of the line and the enthalpy of the steam. A four minute time limit was used for this analysis as well.

Based on the initial supposition that all safe shutdown equipment was at least twenty feet from the nearest opposite loop steam line, testing began. It soon became apparent that several items were nearer to an opposite loop steam line than twenty feet. GGA then reran the computer program and generated a set of new curves. The results of this analysis were presented in report GA-A14212. Following the conclusion of GGA's test program, all documentation was transmitted to PSC.

With the issuance of IE Bulletin 79-01B, and the environmental qualification rule 10CFR50.49, PSC committed to an audit of the documentation. The audit found several items which did not fully meet environmental qualification requirements. These items were retested. The last test was completed on March 31, 1984.

SECTION 3

CASE STUDIES

The following case studies analyze the consequences of steam leaks in both the Reactor and Turbine Buildings. All leaks are assumed to be a complete offset shear of the associated steam line, with the plant at 100% power, resulting in the greatest thermal input to the building.

Included with each analysis are tables which list the sequence of events following the onset of the break. Corresponding operator actions are based on existing emergency procedures. Although many of the actions are performed simultaneously, all are considered to be sequential.

The time given for each action was estimated following discussions with FSV Training personnel. All times are conservative. (Note: For all tables, TA refers to the time each action takes, TE refers to the total elapsed time from the onset of the break).

CASE 1: Steam Pipe Rupture in the Reactor Building

Any steam leak in the Reactor Building is readily detected by the Steam Pipe Rupture Detection System (SPRDS). Redundant pressure, temperature, and ultrasonic detectors identify the leaking loop, allowing the Plant Protective System (PPS) to shutdown the affected loop. The valves that are closed on a loop shutdown are shown on Fig. 3.

If the leak is on a feedwater, main steam, or hot reheat line, or on a cold reheat line downstream of the circulator block valves, loop shutdown will isolate the steam source. As shown in Table 1, this occurs in approximately 10 sec.

TA	TE	ACTION
5 sec	5 sec	PPS Alarm identifying leaking loop.
5 sec	10 sec	Leaking loop isolated by PPS.

TABLE 1: Sequence of Events Following a Feedwater, Main Steam, Hot Reheat or Cold Reheat-Line Break Downstream of the Circulator Block Valves in the Reactor Building.

CASE 2: Cold Reheat Line Break in the Reactor Building

As can be seen in Fig. 3, a cold reheat line break upstream of the circulator block valves cannot be automatically terminated

by the PPS. After the PPS closes the main steam block valve, main steam pressure rises until the main steam bypass valve is automatically opened. Main steam is then bypassed to the flash tank and on to cold reheat. Thus, a cold reheat line break will continue to be fed until the appropriate operator action takes place. These actions are detailed in Emergency Procedure (EP) B-1 and are outlined in Table 2 below.

TA	TE	ACTION
5 sec	5 sec	PPS Alarm Identifying Leaking Loop.
5 sec	10 sec	Turbine Runback to 50% previous power level. Verified by operator.
12 sec	12 sec	Reactor scrammed by PPS on high reactor building temperature.
5 sec	17 sec	Operator inserts manual scram.
5 sec	22 sec	Operator places ISS in low power position.
5 sec	27 sec	Operator ensures transfer of house power.
5 sec	32 sec	Operator verifies reduction of turbine power.
10 sec	42 sec	Operator verifies stable core cooling.
30 sec	72 sec	Operator reads emergency procedures. Determines course of action.
20 sec	92 sec	Operator closes main steam bypass valves. Leak terminated.

TABLE 2: Sequence of Events Following a Cold Reheat Line Break in the Reactor Building.

It is important to note that the continual thermal input into the Reactor Building from a cold reheat line break results in a reactor scram on high Reactor Building temperature. (The time shown for this scram is based on Reactor Building temperature switch setpoint of 175 degrees $\pm$ 10 degrees F and assuming they are on the opposite side of the Reactor Building from the break. See Figure 4)

Based on this analysis, a cold reheat line break in the Reactor Building will be isolated in less than two minutes.

CASE 3: Main Steam Line Break in the Turbine Building

A main steam line break, either in a loop header or in the common header is immediately detected by redundant pressure switches located on the common header. (See Figure 5) Two out of three logic scrams the reactor on low main steam pressure, closes both main steam block valves, and alarms the control room.

A break downstream of the block valves is automatically suspended by the PPS in about 10 sec. (See Table 3)

TA	TE	ACTION
5 sec	5 sec	Reactor Scrammed on Main Steam Pressure Low.
5 sec	10 sec	PPS closes block valves on both loops. Leak isolated.

TABLE 3: Sequence of Events Following a Main Steam Line Break in the Turbine Building Downstream of the Block Valves.

A break upstream of the block valves is not isolated by the PPS. The appropriate operator actions following a reactor scram on low main steam pressure are listed in EP B-1. These actions are summarized in Table 4. The leak can be terminated in less than 90 sec.

TA	TE	ACTION
5 sec	5 sec	Reactor Scrammed on Main Steam Pressure Low.
5 sec	10 sec	PPS Closes Block Valves on Both Loops.
5 sec	15 sec	Operator Inserts Manual Scram.
5 sec	20 sec	Operator Places ISS in Low Power Position.
5 sec	25 sec	Operator Insures Transfer of House Power.
5 sec	30 sec	Operator Verifies Reduction of Turbine Power.
10 sec	40 sec	Operator Verifies Stable Core Cooling.
30 sec	70 sec	Operator Reads Emergency Procedures. Determines Course of Action.
5 sec	75 sec	Operator Starts Auxiliary Boiler.
5 sec	80 sec	Operator Closes Feedwater Block Valves. Leak Terminated.

TABLE 4: Sequence of Events Following a Low Main Steam Pressure Scram.

CASE 4: Feedwater or Condensate Line Break in the Turbine Building

The consequences of a feedwater or condensate line break in the Turbine Building are much less severe than those anticipated for other line breaks. Calculations show that for the worst case feedwater or condensate break, the maximum steam temperature is 203 degrees F. This temperature is completely enveloped by the Turbine Building 20 foot curve. (See Fig. 6).

Thus, the operator has an extended period to identify and isolate any break in the feedwater or condensate system.

Any break in the feedwater or condensate system results in a low main steam pressure scram. Emergency Procedure B-1 identifies a feedwater or condensate line break as a possible cause of the scram and lists appropriate actions.

If the break is downstream of the feedwater block valve, it is terminated in the same manner as the main steam line break summarized in Table 4.

A break upstream of the block valve is also detected by redundant pressure switches located on the feedwater common header. Two out of three logic closes the normal boiler feed pump (BFP) discharge valve and opens the bypass valve to emergency feedwater.

Since emergency feedwater will also leak out the pipe rupture, lack of back pressure results in high emergency feedwater flow. Redundant flow switches then alarm the control room. The operator then isolates the break by closing the emergency feedwater block valves. This leak is terminated in less than 90 sec. (See Table 5)

TA	TE	ACTION
5 sec	5 sec	PPS Senses Low Feedwater Pressure.
5 sec	5 sec	Reactor Scrammed on Main Steam Pressure Low.
5 sec	10 sec	BFP Discharge Valve Closed, Bypass Opened.
5 sec	15 sec	High Emergency Feedwater Flow Alarm.
5 sec	20 sec	Operator Inserts Manual Scram.
5 sec	25 sec	Operator Places ISS in Low Power Position.
5 sec	30 sec	Operator Insures Transfer of House Power.
5 sec	35 sec	Operator Verifies Reduction of Turbine Power.
10 sec	45 sec	Operator Verifies Stable Core Cooling.
30 sec	75 sec	Operator Reads Emergency Procedures. Determines Course of Action.
5 sec	80 sec	Operator Starts Auxiliary Boiler.
5 sec	85 sec	Operator Closes Emergency Feedwater Block Valve. Leak Terminated.

TABLE 5: Sequence of Events Following a Feedwater Line Break Downstream of the Block Valves.

A break occurring between the feedwater check valves and the BFP normal discharge valve is isolated immediately by the redundant pressure switches on the common feedwater header. (See Table 6)

TA	TE	ACTION
5 sec	5 sec	PPS Senses Low Feedwater Pressure.
5 sec	10 sec	BFP Discharge Valve Closed. Bypass Opened. Leak Terminated.

TABLE 6: Sequence of Events for a Feedwater Line Break Between the Feedwater Check Valves and BFP Discharge Valves.

A break occurring between the condensate pump (CP) and BFP does not result in a harsh environment, as the water only reaches a temperature of 110 degrees F at the break. (See calculations). This break is also detected by a pressure switch that alarms the control room.

CASE 5: Cold Reheat Line Break in the Turbine Building

A break in either a loop cold reheat line or in the common header will result in a depletion of all steam supply to the hot reheat system causing a reactor scram on low hot reheat pressure. It also removes the source of helium circulator motive steam leading to circulator steam drive trips on one loop and complete circulator shutdown on the other loop.

The various operator actions required to terminate this leak are described in EP B-1. Based on the actions listed in Table 7, the leak will be isolated in about 150 sec.

TA	TE	ACTION
10 sec	10 sec	Reactor Scrammed on Hot Reheat Steam Pressure Low.
5 sec	15 sec	Operator Inserts Manual Scram.
15 sec	15 sec	Low Circulator Speed Alarm (Based on Historical Data on Trips from Full Speed).
5 sec	20 sec	Circulator Trips Automatically.
5 sec	20 sec	Operator Places ISS in Low Power Position.
5 sec	25 sec	Operator Insures Transfer of House Power.
5 sec	30 sec	Operator Verifies Reduction of Turbine Power.
10 sec	40 sec	Operator Verifies Stable Core Cooling.
30 sec	70 sec	Operator Reads Emergency Procedures. Determines Course of Action.
5 sec	75 sec	Operator Starts Auxiliary Boiler.
20 sec	95 sec	Operator Trips both Circulators in One Loop. Loop Shutdown Insured.
15 sec	110 sec	Operator Trips Both Circulator Steam Drives In Other Loop. Closes Hot Reheat Steam Stop-Check Valves in Loop.
5 sec	115 sec	Operator Insures Auto Water Turbine Start of Last Two Circulators Tripped.
10 sec	125 sec	Operator Opens Power Operated Main Steam Safety Valves.
20 sec	145 sec	Operator Raises Main Steam Bypass and Start Up Bypass Valves Setpoints to 2800 PSIG. Leak Terminated.

TABLE 7: Sequence of Events Following a Cold Reheat Line Break in the Turbine Building.

CASE 6: Hot Reheat Line Break in the Turbine Building

Hot reheat steam enters the Turbine Building in two separate loops that combine into a common header downstream of the hot reheat stop-check valves (See Fig. 7 and Fig. 8). This common header then divides into two legs which admit hot reheat steam to the intermediate stage of the turbine.

As in the case of main steam, redundant pressure switches are used to monitor line breaks on the common header. Two out of three logic scrams the reactor and alarms the control room.

A line break in one of the loop headers will not result in a trip of the pressure switches because the operating loop, prevented from bleeding off through the leak by the stop-check valve, will continue to feed the turbine. However, due to reduced backpressure, the increased cold reheat flow causes excessive helium circulator speed. The result is the circulators trip on overspeed causing loop shutdown. The loop shutdown isolates the leak by closing the circulator block valves. This leak is terminated in about 10 sec. (See Table 8)

TA	TE	ACTION
5 sec	5 sec	Circulator Trips on Overspeed.
5 sec	10 sec	PPS Initiates Loop Shutdown. Leak Terminated.

TABLE 8: Sequence of Events Following a Hot Reheat Line Break Upstream of the Stop-Check Valves.

A line break between the loop isolation valves and the pressure switches will be immediately detected by the switches. Based on EP B-1 and Table 9 below, the leak will be isolated in less than two minutes.

TA	TE	ACTION
5 sec	5 sec	Reactor Scrammed on Hot Reheat Steam Pressure Low.
5 sec	10 sec	Operator Inserts Manual Scram.
5 sec	15 sec	Operator Places ISS in Low Power Position.
5 sec	20 sec	Operator Insures Transfer of House Power.
5 sec	25 sec	Operator Verifies Reduction of Turbine Power.
10 sec	35 sec	Operator Verifies Stable Core Cooling.
30 sec	65 sec	Operator Reads Emergency Procedures. Determines Course of Action.
5 sec	70 sec	Operator Starts Auxiliary Boiler.
20 sec	90 sec	Operator Trips Both Circulators in One Loop. Assure Loop Shutdown.
15 sec	105 sec	Operator Trips Both Circulator Steam Drives in Other Loop.
5 sec	110 sec	Operator Closes Both Hot Reheat Steam Stop-Check Valves. Leak Isolated.

TABLE 9: Sequence of Events Following A Hot Reheat Line Break Between the Stop-Check Valves and Pressure Switches.

A break downstream of the pressure switches results in a turbine trip on a loss of vacuum. A turbine trip immediately initiates a programmed feedwater flow reduction to 25% of full load flow at a rate of .5% per second.

Calculations show that with a hot reheat line break, the reactor will scram on low hot reheat pressure with feedwater flow at 30%. The sequence of events following this accident are shown in Table 10 below.

TA	TE	ACTION
5 sec	5 sec	Turbine Trip On A Loss Of Vacuum. Feedwater Flow Program Begins.
5 sec	10 sec	Operator Verifies Transfer of House Power.
5 sec	15 sec	Operator Verifies Reduction of Turbine Power.
140 sec	145 sec	Reactor Scrammed on Hot Reheat Steam Pressure Low.
5 sec	150 sec	Operator Inserts Manual Scram.
5 sec	155 sec	Operator Places ISS In Low Power Position.
10 sec	165 sec	Operator Verifies Stable Core Cooling.
30 sec	195 sec	Operator Reads Emergency Procedures. Determines Course of Action.
5 sec	200 sec	Operator Starts Auxiliary Boiler.
20 sec	220 sec	Operator Trips Both Circulators In One Loop. Assures Loop Shutdown.
15 sec	235 sec	Operator Trips Both Circulator Steam Drives In Other Loop.
5 sec	240 sec	Operator Closes Both Hot Reheat Stop Check Valves. Leak Terminated.

TABLE 10: Sequence of Events Following a Hot Reheat Line Break Downstream of the Pressure Switches.

SECTION 4

CONCLUSION

A feedwater, main steam, or hot reheat line break within the Reactor Building, is immediately detected by the Steam Pipe Rupture Detection System and isolated automatically in a matter of seconds. A cold reheat line break continues to be fed by the bypass flash tank until manual action isolates the leak after two minutes.

The cold reheat line break is the most severe in the Reactor Building due to its extended length. This affirms Gulf General Atomic's initial analysis.

Since the cold reheat line break is isolated in two minutes, the four minute curve used for environmental qualification of equipment in the Reactor Building is conservative. The extra time that appears in the four minute curve provides additional margin for all test results. All equipment qualified for use in the Reactor Building remains qualified.

A main steam line break in the Turbine Building is isolated either automatically in 10 sec or manually in 80 sec. A feedwater line break is isolated in either 10 sec or 85 sec. Cold reheat is isolated in about 150 sec.

Although a condensate line break may last longer than four minutes, the consequences are not severe since condensate will not flash to steam.

A hot reheat line break can be terminated in as little as 10 sec automatically or as long as four minutes manually. Thus, a hot reheat line break is the most severe in the Turbine Building. This confirms Gulf General Atomic's assumption.

GGA's steam line rupture curves were generated assuming a constant blowdown rate following the break and prior to leak termination. However, because of the reduction of feedwater flow following a turbine trip, and subsequent reactor scram, the blowdown rate and thermal input into the Turbine Building will decrease as a function of time.

Thus, the steam line rupture curve for a hot reheat line break in the Turbine Building is conservative. The difference between a constant blowdown rate assumed by GGA and the decreasing blowdown rate occurring after a turbine trip provides additional margin to that found on all test results. All equipment qualified for use in the Turbine Building remains qualified.

SECTION 5

REFRENCES

GGA Report GA-A12045 "Qualification of Fort St. Vrain Safe Shutdown Equipment for Steam Environments Resulting from Pipe Ruptures."

GGA Report GA-A14212 "Environmental Temperatures in the Vicinity of the Rupture Point of Steam Lines for Fort St. Vrain Equipment Qualification."

Fort St. Vrain Final Safety Analysis Report

FSV-SD-22-1 "System Description: Secondary Coolant System"

FSV-SD-51 "System Description: Turbine-Generator and Auxiliaries."

FSV-SD-93-1 "System Description: Controls and Instrumentation."

FSV-SD-93-2 "System Description: Overall Plant Control and Plant Protective System."

FSV-SD-93-5 "System Description: "Steam Pipe Rupture Detection System:"

FSV-EP-B-1 "Emergency Procedure: Reactor Scram (Without Two Loop Trouble)".

FSV-EP-B-2 "Emergency Procedure: Two Loop Trouble Scram, With a Trouble Alarm in the Operating Loop."

FSV-EP-C "Emergency Procedure: Loop Shutdown."

Fig. 1

Temperature response of the environment near the rupture for a reactor building cold reheat pipe rupture

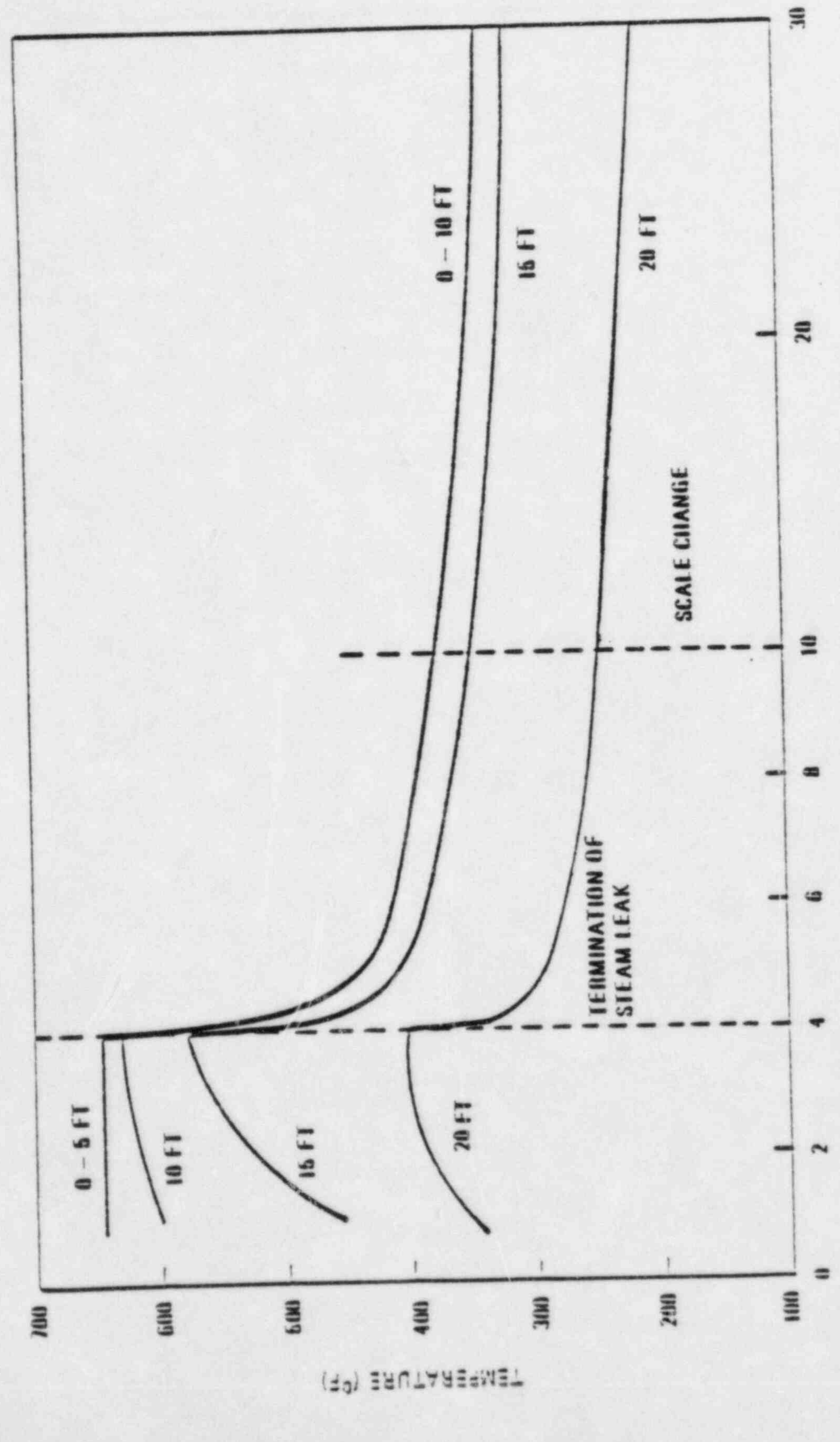


Fig. 2

Temperature response of the environment near the rupture for a turbine building hot reheat pipe rupture

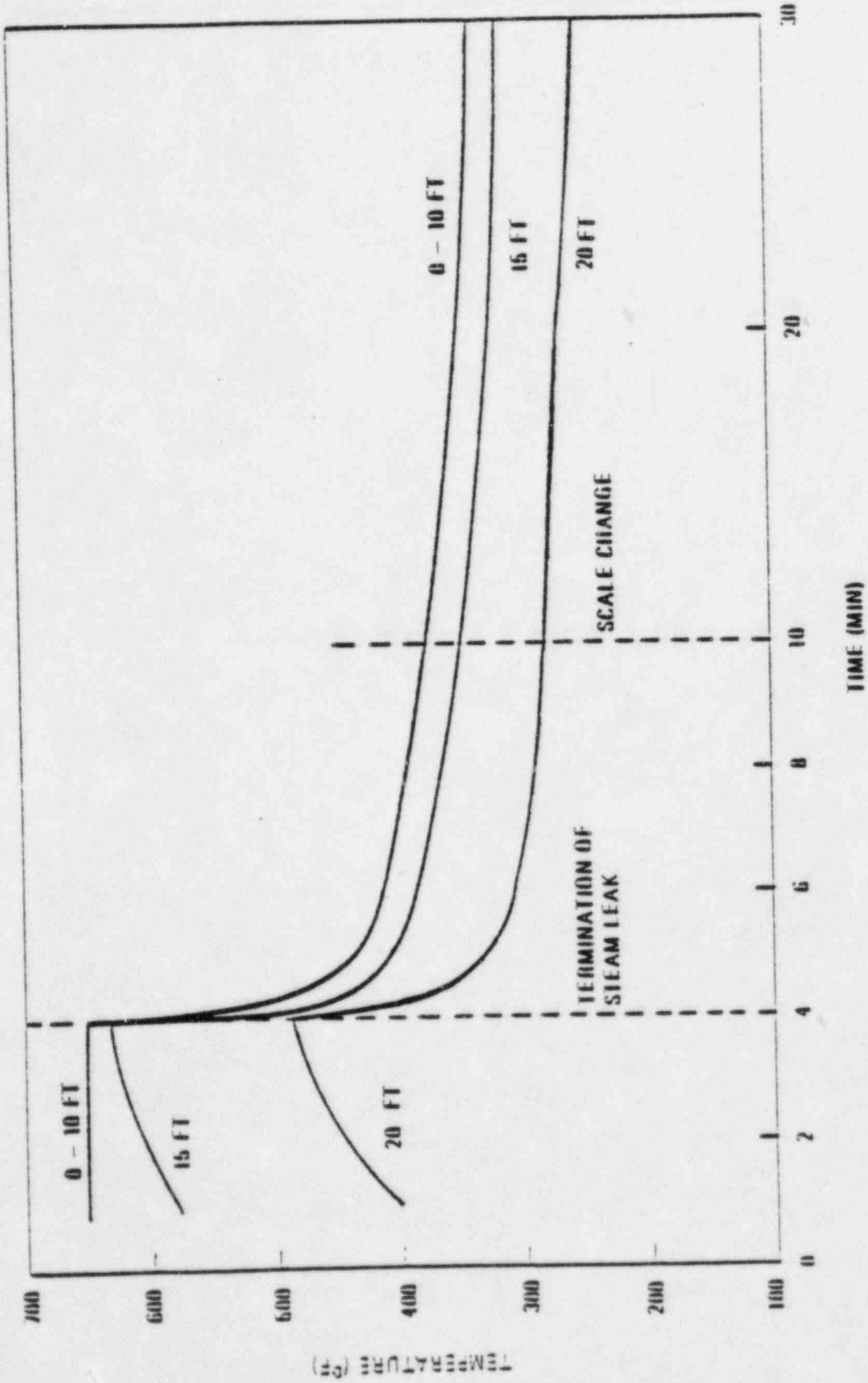


Fig. 3 PPS Valve Isolation Following a Steam Pipe Rupture in the Reactor Building

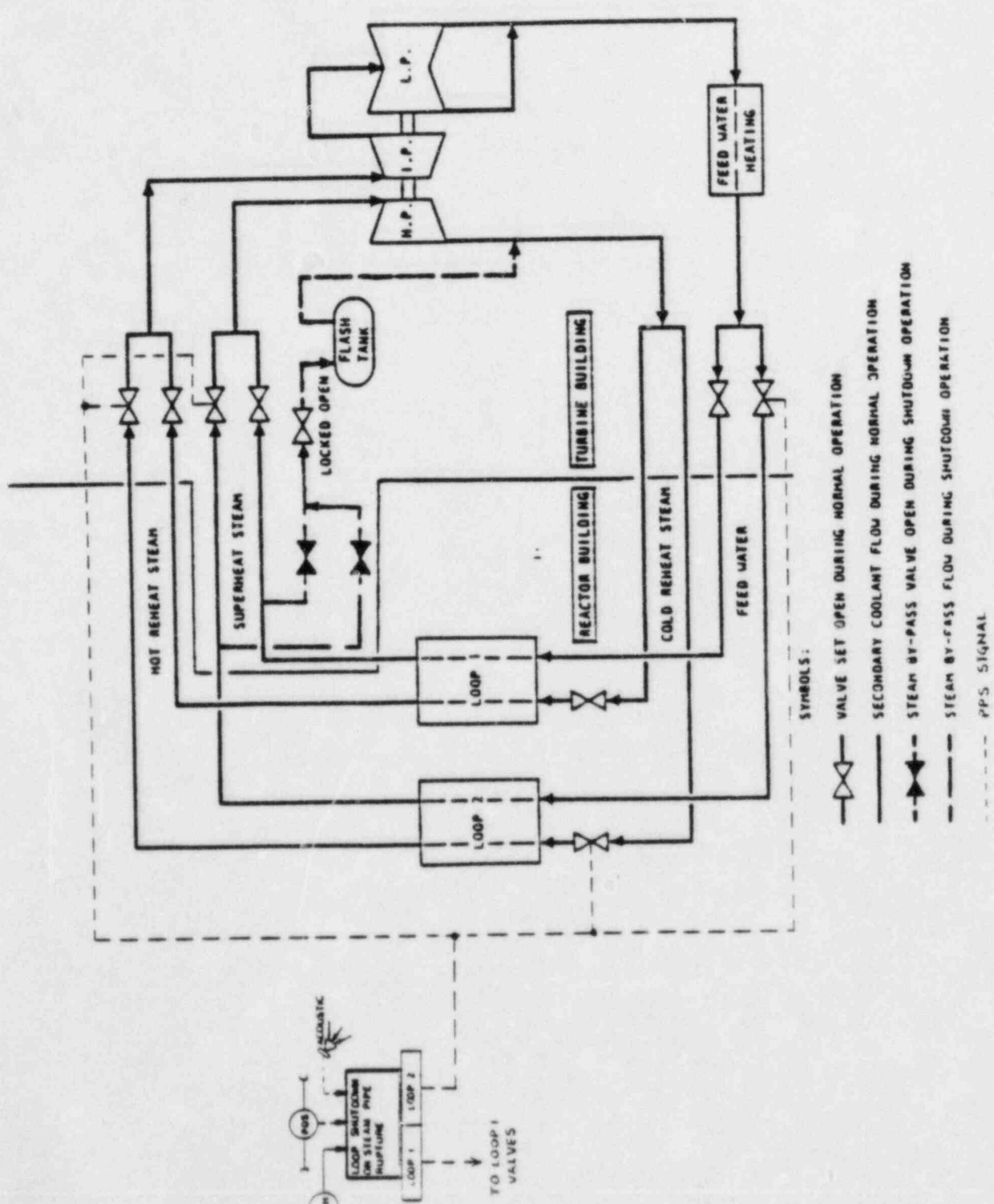


Fig. 4 Trip Time for High Reactor Building Temperature Switches

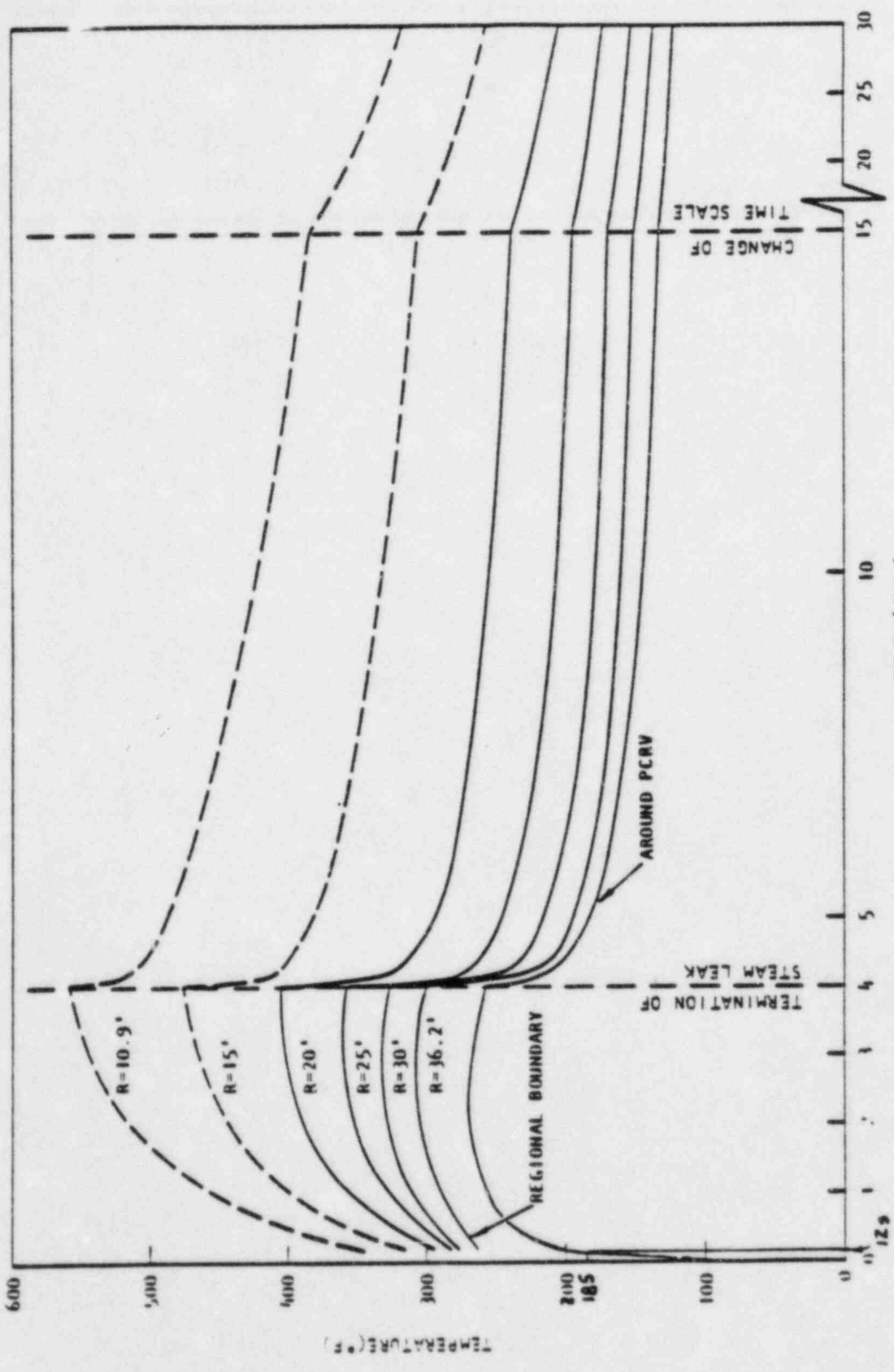


Fig. 5 Alarms and Automatic Actions for Steam Line Breaks on Feedwater, Condensate, and Main Steam Lines in the Turbine Building

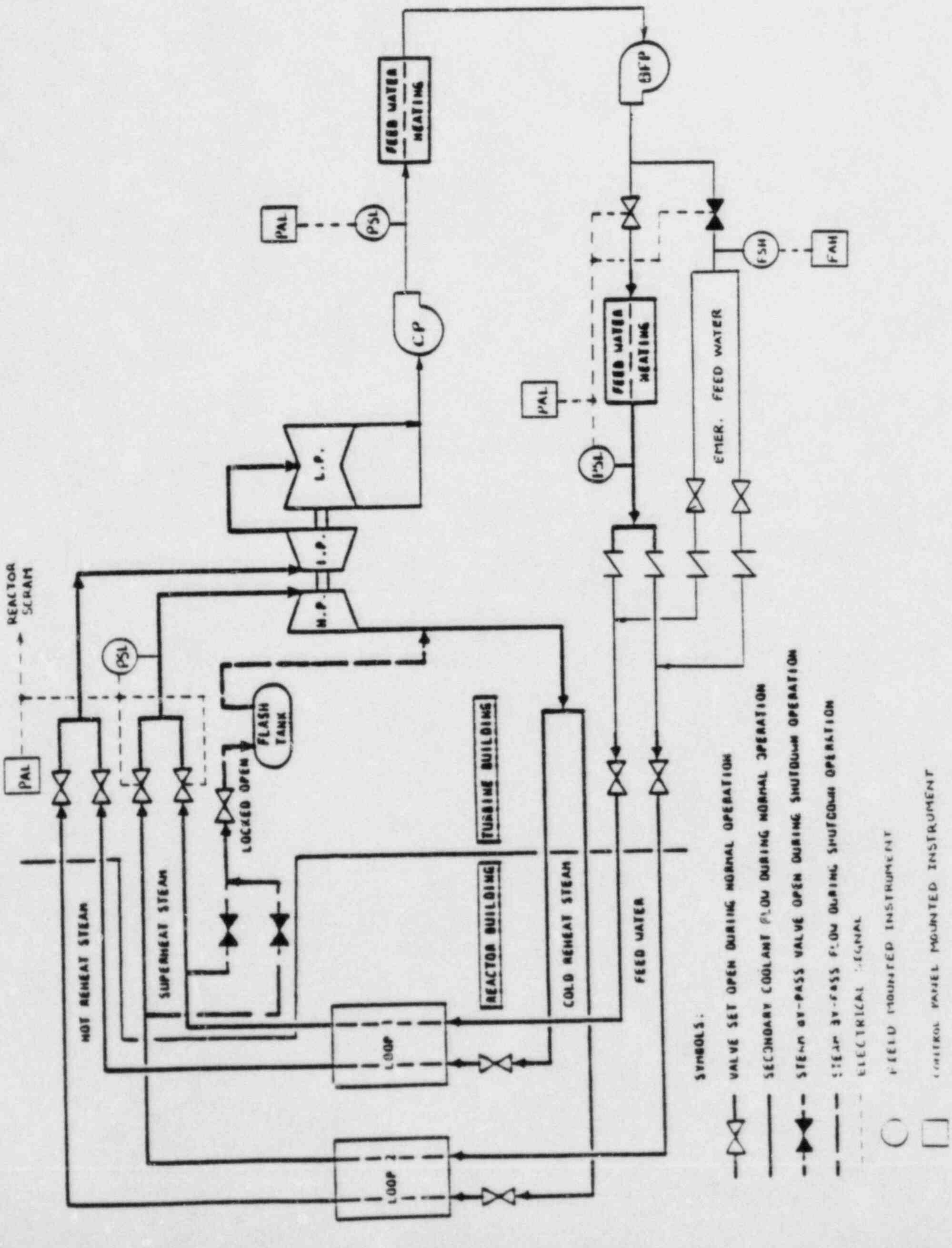


Fig. 6 Comparison of Feedwater Break Temperature with Turbine Building Steam Line Rupture Test Curves

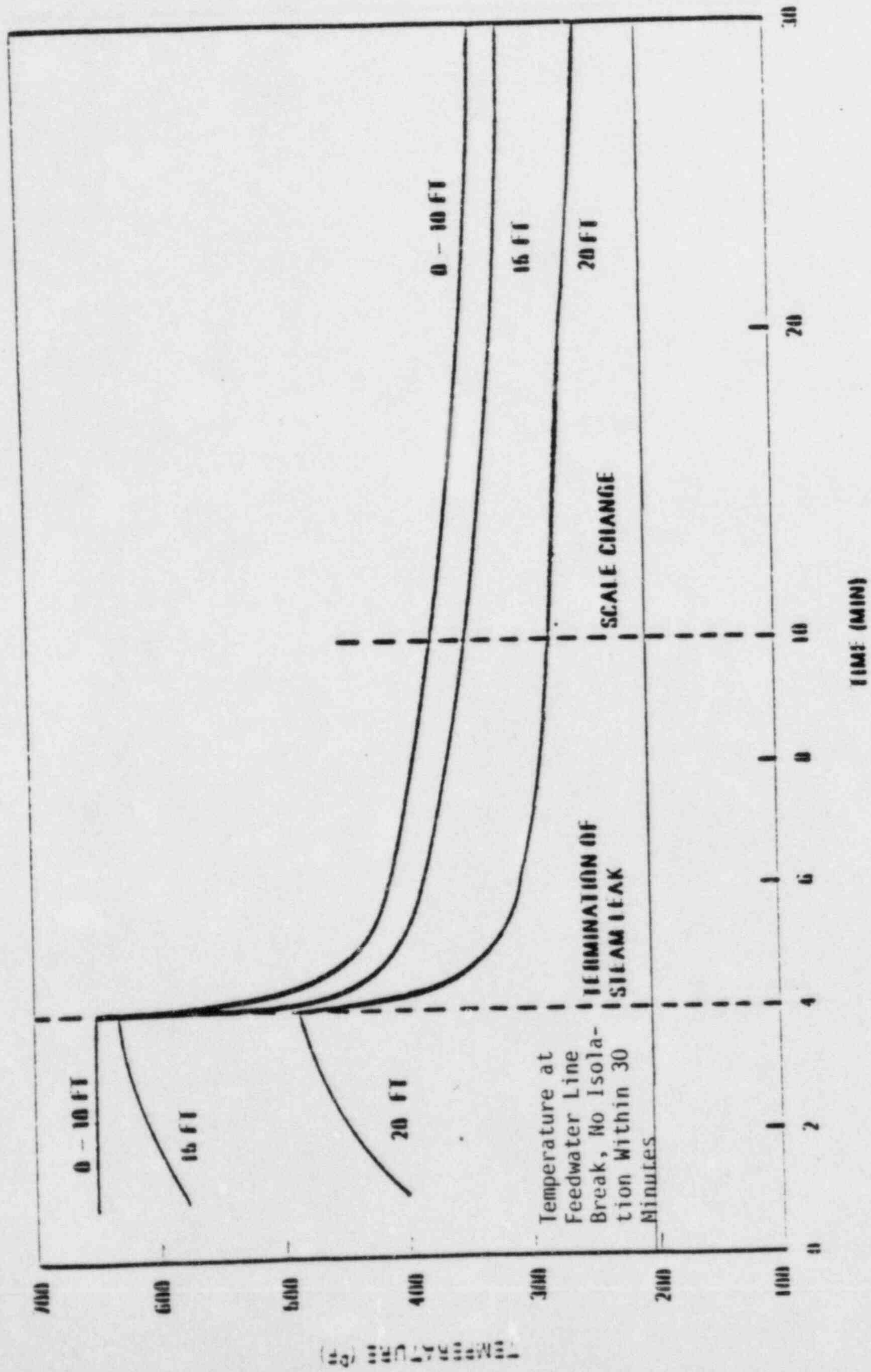


Fig. 7 PPS Alarms and Actions Following a Hot Reheat Line Break in the Turbine Building

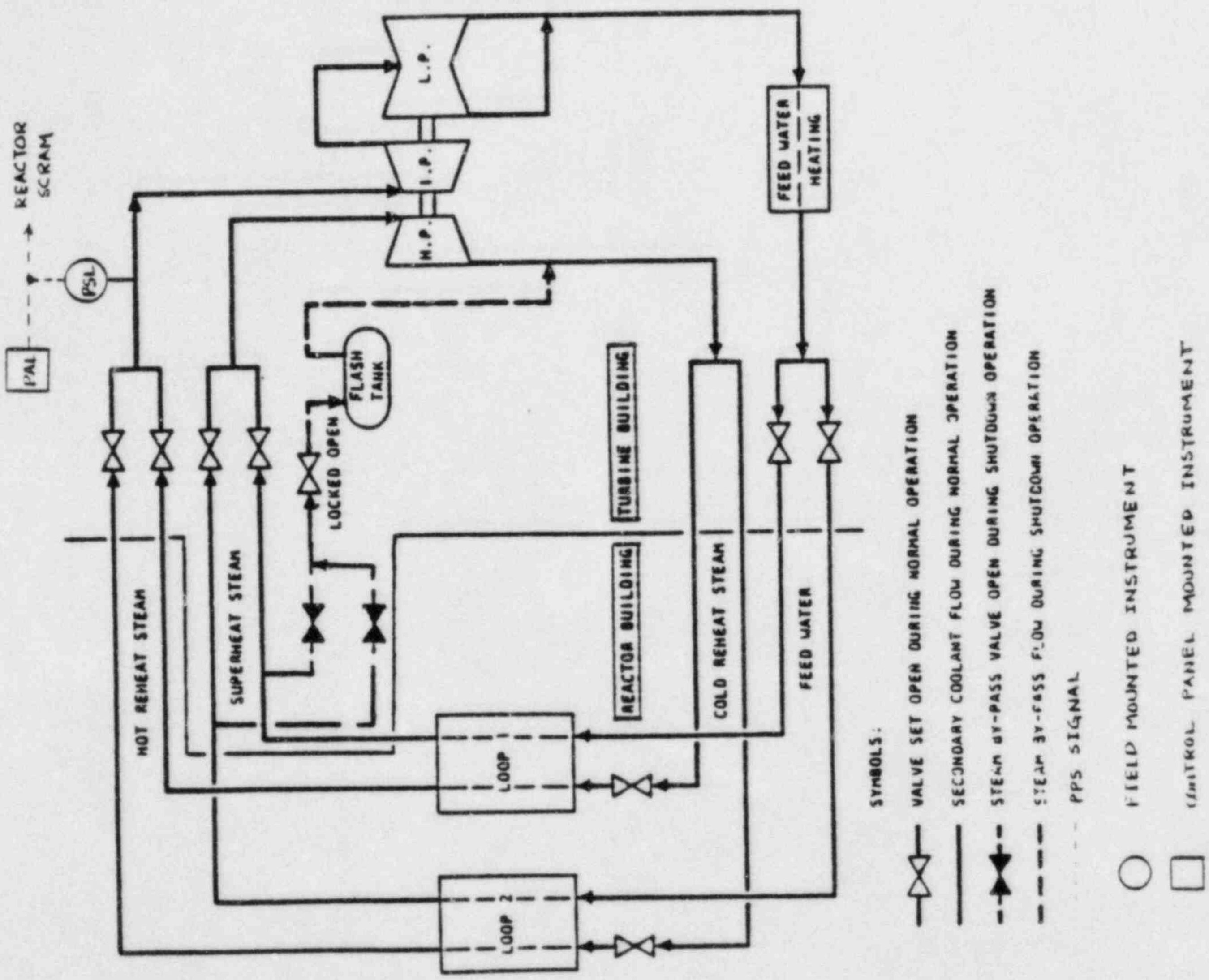
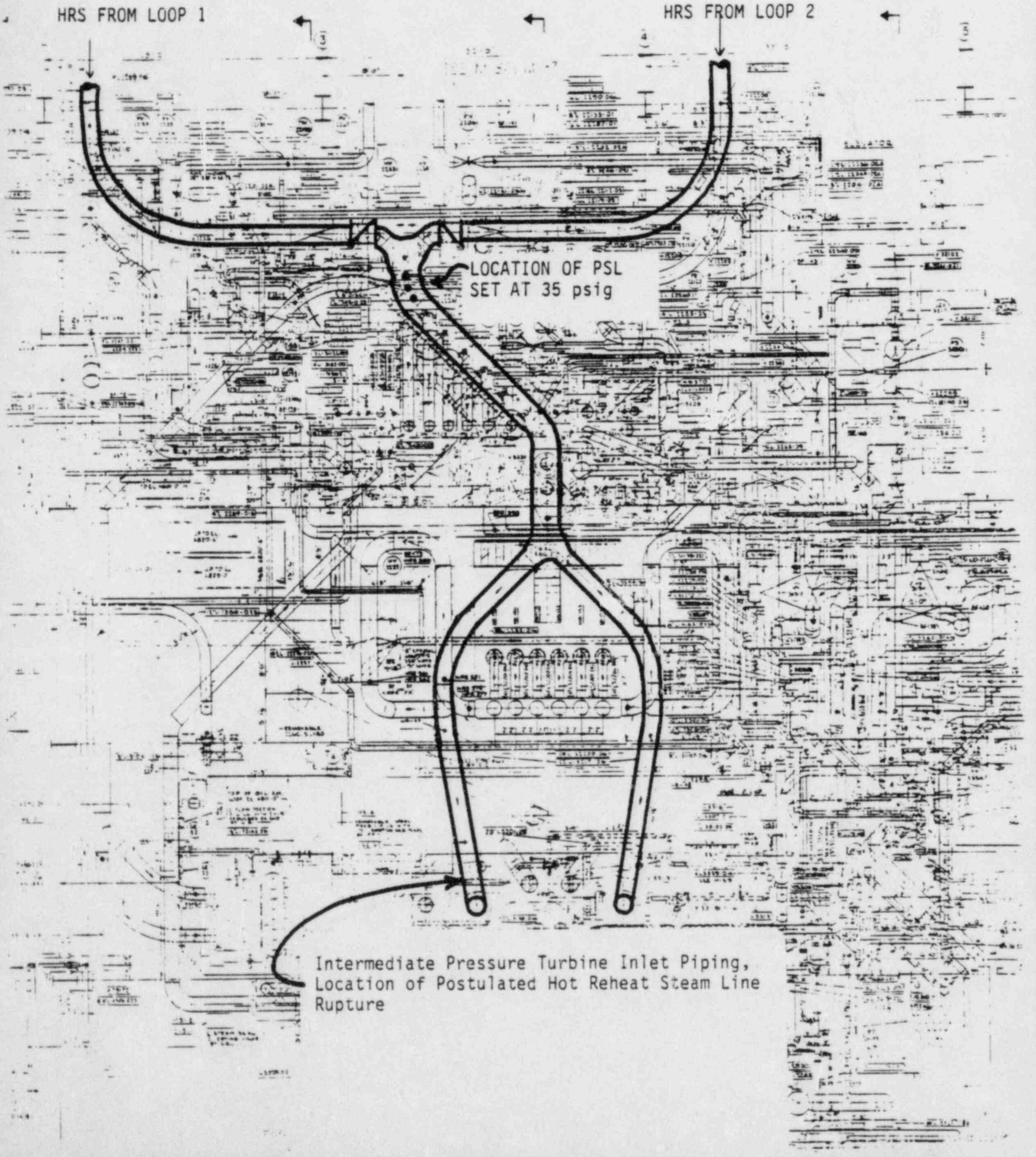


Fig. 8 Postulated Pipe Rupture Downstream of the Low Pressure Switches on the Hot Reheat Line



APPENDIX A

CALCULATIONS

The calculations contained in this appendix were performed to analyze certain steam line ruptures in the turbine building. The calculations show that any main steam line break is detectable within 5 seconds by the low main steam pressure switch. The calculations also show that for a condensate or feedwater line break, the maximum release is a steam water mixture of approximately 200 degrees F. The calculations also show that following a turbine trip with automatic feedwater flow reduction the existing low pressure switches on the hot reheat line will trip within 140 seconds. Crane Technical Paper No. 410, Flow of Fluids through Valves, Fittings and Pipe, was used as a reference throughout.

PIPE RUPTURE IN THE TURBINE
BUILDING ON THE
CONDENSATE OR FEEDWATER LINES

Problem: Find the temperature of the flashing steam escaping from the rupture.

CONDENSATE

Location - discharge of condensate pumps

Conditions - T=110 degrees F, P=310 psia, h=78.78 Btu/lb flashes to atmosphere w/ constant enthalpy fluid: Therefore @ P=12.3 psia, h=78.78 Btu/lb T=110 degrees F w/ no steam flash.

FEEDWATER

Location - discharge of boiler feedpumps

Conditions - T=316 degrees F, P=3300 psia, h=292 Btu/lb flashes to atmosphere @ P=12.3 psia, h=292 Btu/lb, Therefore: T=203 degrees F w/ 12% steam

Location - discharge of feedwater heater #6

Conditions - T=403 degrees F, P=3182 psia, h=382 Btu/lb Flashes to atmosphere @ P=12.3 psia, h=382 Btu/lb, Therefore: T=203 degrees F w/ 22% steam.

MAIN STEAM
RUPTURE AT THE TURBINE

The steam will reach sonic velocity at the exit for maximum choked flow. For compressible sonic flow at the break:

$$W = \frac{vd^2 p}{.0509}$$

*W = flow (lb/hr) = 2.25×10^6 lb/hr
v = velocity (ft/sec)
d = inside pipe diameter = 10.4 in
p = density

Substituting:

$$2.25 \times 10^6 \text{ lb/hr} = \frac{v \times (10.4 \text{ in})^2 \times p}{.0509}$$

*Blowdown rate for steam by GA Report GA12045

Solving: $p = \frac{1059}{v}$ Eqn 1

Sonic velocity is reached at the exit and is equal to:

$$v = \sqrt{\frac{kxg \times 144 \times P_2}{p}}$$

v = velocity (ft/sec) sonic
g = gravity = 32.2 ft/sec^2
P_2= exit pressure

Substituting: p = density (lb/ft^3)
k = specific heat ratio = 1.3 for steam

$$v = \sqrt{\frac{1.3 \times 32.2 \text{ ft/sec}^2 \quad \times 144 \text{ in}^2/\text{ft}^2 \times P_2 \times v}{1059}}$$

Solving: $v = 5.7 \times P_2$ Eqn. 2

Solving Eqn. 1 and Eqn 2 simultaneously yields:

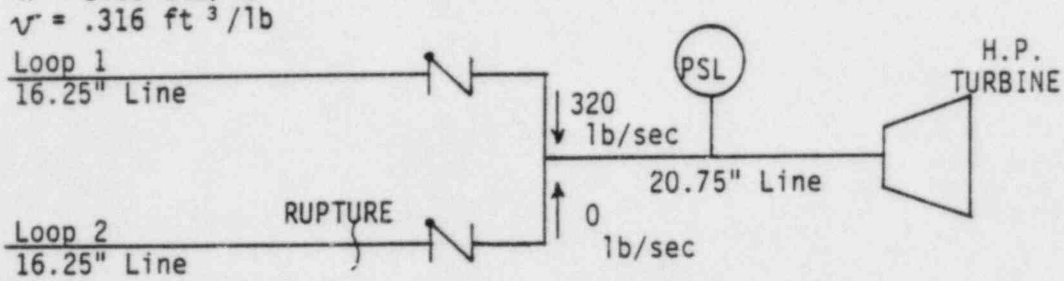
-outlet Temp = 1000 degrees F same as normal flow for worse case	Outlet pressure P_2 = 400 psia Outlet velocity v = 2280 ft/sec Outlet density p = .46 lb/ft^3

Conclusion: The setpoint for the low main steam pressure scram is 1500 psig. The switches are located approximately 40 ft from the break where the exit pressure with choked flow is about 400 psia. Therefore, the pressure switches would trip in less than 5 seconds.

MAIN STEAM
RUPTURE UPSTREAM OF
THE LOOP ISOLATION VALVES

CONDITIONS: P = 2437 psia
T = 1002 degrees F
h = 1460 Btu/lb
ν = .316 ft^3/lb

SITUATION:



A rupture in a single loop line will allow all of the steam in loop 2 to escape to the atmosphere. However, the check valve will not allow the steam from loop 1 to escape. Therefore, flow still exists at the pressure switch which is set @ 1500 psig.

SOLUTION: Loop Lines: D = 16.25 in
d = 10.82 in
A = 91.9 in^2 = .6385 ft^2

Combined Line: D = 20.75 in
d = 14.758 in
A = 171.06 in^2 = 1.188 ft^2

Velocity of steam under normal conditions:

$$V = 320 \text{ lb/sec} \times \frac{.316 \text{ ft}^3/\text{lb}}{.6385 \text{ ft}^2} = 158 \text{ ft/sec in each loop}$$

$$V = 640 \text{ lb/sec} \times \frac{.316 \text{ ft}^3/\text{lb}}{1.188 \text{ ft}^2} = 170 \text{ ft/sec in combined line}$$

After one loop is lost, approximately the same pressure drop, thus the same velocity, will exist in the combined line. The velocity will increase some but this will result in a lower pressure, thus using 170 ft/sec is conservative. Therefore, after one loop is lost, the flow in loop one and the combined loop is equal to 320 lb/sec. Therefore:

Loop 1 flow in lb/sec = Combined Line flow in lb/sec

$$\frac{158 \text{ ft/sec} \times .6385 \text{ ft}^2}{.316 \text{ ft}^3/\text{lb}} = \frac{170 \text{ ft/sec} \times 1.188 \text{ ft}^2}{\nu' \quad \text{ft}^3/\text{lb}}$$

At the pressure switches in the combined line:

$$v' = .633 \text{ ft}^3/\text{lb}$$

and T = 1000 degrees F

Therefore: P = 1293 psia
@ PSL

CONCLUSION: The pressure at the switches reaches about 1295 psia instantly, therefore the switches would trip with a setpoint of 1500 psig.

HOT REHEAT STEAM RUPTURE NEAR THE TURBINE

Find percent of flow to reach the set point of 35 psig.

Resistance of pipe:

$K = f \ L/D$ - 30 ft. of 34" pipe

$K = f \ L/DB^4$ - 40 ft of 22" pipe

$K = \frac{.5(1-B^2)\sqrt{\sin\frac{\theta}{2}}}{B^4}$ - sudden contraction

$K = 1/B^4$ - exit

where

$B = \frac{d_1}{d_2} = \frac{20 \text{ in}}{31 \text{ in}} = .65$

$f = .012$ for turbulent flow
$\theta = 180°$

$K = \frac{.012 \times 30 \text{ ft.} \times 12}{34 \text{ in.}} = .127$ - 34" pipe

$K = \frac{.012 \times 40 \text{ ft.} \times 12}{22 \text{ in.} \times .65^4} = 1.47$ - 22" pipe

$K = \frac{.5\ (1 - .65^2) \times 1}{.65^4} = 1.62$ - contraction

$K = 1/.65^4 = 5.6$ - exit

$K_{TOT} = 8.82$

A+ PSL, $P_1 = 35$ psig + 12.3 = 47.3 psia

Atmospheric $P_2 = 12.3$ psia

$\frac{\Delta P}{P_1} = \frac{35}{47.3} = .74$ for $\frac{\Delta P}{P_1} = .74$ and $K = 8.8$

- Y = .71 w/ subsonic flow

Therefore:

$$W = 1891 \times Y \times d^2 \times \sqrt{\frac{\Delta P}{K \times \overline{V}_1}}$$

W = flow (lb/hr)
Y = .71
d = 31 in.
ΔP = 35 psia
K = 8.82
$\overline{V}_1$ = 14 ft³/lb @ 35 psig
h = 1360 Btu/lb

$$W = 1891 \times .71 \times 31\ \text{in}^2 \times \sqrt{\frac{35\ \text{psia}}{8.82 \times 14\ \text{ft}^3/\text{lb}}}$$

W = 686, 924 lb/hr

$$\text{Percent normal flow} = \frac{686,924\ \text{lb/hr}}{2.25 \times 10^6\ \text{lb/hr}} = .31 = 31\%$$

After a turbine trip, the feedwater flow is automatically reduced to 25% flow by ½% per second. Therefore it would take:

$$+ = \frac{70\%}{.5\%/\text{sec.}} = 140\ \text{seconds}$$

APPENDIX B

This appendix contains the emergency procedures that are used by the reactor operators to respond to a steam line rupture. Depending on the type of steam leak, the operator will receive certain alarms or indications that direct him to the appropriate procedure where the correct immediate and followup actions are found.

TITLE: REACTOR SCRAM (WITHOUT TWO LOOP TROUBLE)

ISSUANCE AUTHORIZED BY	[illegible signature]		
PORC REVIEW	PORC 486 OCT 14 1982	EFFECTIVE DATE	10-21-82

REACTOR SCRAM

SYMPTOM-ACTION MATRIX

ACTIONS	SYMPTOMS 1.1 Start Up Count Rate High (RMS - Fuel Loading) 1-038, 4/4, 5/4, 6/4	1.2 Neutron Flux Rate of CNG High (ISS - Start Up) 1-038, 4/3, 5/3, 6/3	1.3 Neutron Flux High 1-038, 1/1, 1/2, 2/1, 2/2, 3/1, 3/2	1.4 Reheat Steam Temp. High 1-038, 1/3, 2/3, 3/1	1.5 Reactor Pressure High (Programmed) 1-038, 1/4, 2/4, 3/4	1.6 Two Loop Trouble 1-038, 4/1, 5/1, 6/1
IMMEDIATE ACTION						
2.1 Insert Manual Scram.	XX	XX	XX	XX	XX	XX
2.2 Place ISS in Low Power Position.			XX	XX	XX	XX
2.3 Ensure Transfer of House Power.			XX	XX	XX	XX
2.4 Ensure Turbine Trip.			XX	XX	XX	XX
2.5 Ensure or establish stable Core Cooling conditions.	XX XX	XX XX	XX XX	XX XX	XX XX	XX XX
FOLLOWUP ACTION						
3.1 Ensure Reactor Internal Maintenance terminated and openings closed.	XX XX					
3.2 Start Auxiliary Boiler.			XX	XX	XX	XX
3.6 If Reactor pressure high scram or Two Loop Trouble scram due to Moisture In Leakage, Proceed to EP "A".					XX XX	XX XX
3.8 Ensure Feedwater Flow remains at Pre-scram level until Reheat Temperature is ≤ 975°F.				XX		

NOTE: FOLLOWUP ACTION STEPS 3.3, 3.4, 3.5, AND 3.7 ARE NOT APPLICABLE ON THIS PAGE.

(B-1)

REACTOR SCRAM

SYMPTOM-ACTION MATRIX

ACTIONS	SYMPTOMS					
	1.1 Start Up Count Rate High (RMS - Fuel Loading) 1-03B, 4/4, 5/4 6/4	1.2 Neutron Flux Rate of CHG High (ISS - Start Up) 1-03B, 4/3, 5/3, 6/3	1.3 Neutron Flux High 1-03B, 1/1, 1/2, 2/1 2/2, 3/1. 3/2	1.4 Reheat Steam Temp. High 1-03B, 1/3, 2/3, 3/3	1.5 Reactor Pressure High (Programmed) 1-03B, 1/4, 2/4, 3/4	1.6 Two Loop Trouble 1-03B, 4/1, 5/1 6/1
REPORTING/ACTIVATION						
4.1 The event, as written, should be classified as a "Significant Event."	XX	XX	XX	XX	XX	XX

(B-1)
REACTOR SCRAM

SYMPTOM-ACTION MATRIX

ACTIONS	SYMPTOMS: 1.7 Loss of Plant Power 1-03B, 4/2, 5/2, 6/2	1.8 Reactor Building Temperature High 1-03B, 4/6, 5/6, 6/6	1.9 Reactor Pressure Low (Programmed) ISS Power 1-03B, 1/5, 2/5, 3/5	1.10 Main Steam Pressure Low ISS Power 1-03B, 1/6, 2/6, 3/6	1.11 Reheat Steam Pressure Low ISS Power 1-03B, 1/7, 2/7, 3/7	1.12 Manual 1-03B, 4/5, 5/5, 6/5 See TABLE B-1
IMMEDIATE ACTION						
2.1 Insert Manual Scram.	XX	XX	XX	XX	XX	XX
2.2 Place ISS in Low Power position.	XX	XX	XX	XX	XX	XX
2.3 Ensure Transfer of House Power.	XX	XX	XX	XX	XX	XX
2.4 Ensure Turbine Trip.	XX	XX	XX	XX	XX	XX
2.5 Ensure or establish Stable Core Cooling conditions.	XX XX	XX XX	XX XX	XX XX	XX XX	XX XX
FOLLOWUP ACTION						
3.2 Start Auxiliary Boiler.	XX	XX	XX	XX	XX	XX
3.3 Establish 2 Circulators at ~ 7000 RPM.			XX			
3.4 Isolate Primary Coolant Leak. Depressurize through Purification Train if possible.		XX	XX			
3.5 Isolate Secondary Coolant Leak or establish Feedwater Flow.		XX		XX	XX XX	
3.7 Start Emergency Diesel Sets and Energize 480V Essential Busses.	XX					

NOTE: FOLLOWUP ACTION STEPS 3.1, 3.6, AND 3.8 ARE NOT APPLICABLE ON THIS PAGE.

(B-1)

REACTOR SCRAM

SYMPTOM-ACTION MATRIX

ACTIONS	SYMPTOM					
	1.7 Loss of Plant Power I-03B, 4/2, 5/2, 6/2	1.8 Reactor Building Temperature High I-03B, 4/6, 5/6, 6/6	1.9 Reactor Pressure Low (Programmed) ISS Power I-03B, 1/5, 2/5, 3/5	1.10 Main Steam Pressure Low ISS Power I-03B, 1/6, 2/6, 3/6	1.11 Reheat Steam Pressure Low ISS Power I-03B, 1/7, 2/7, 3/7	1.12 Manual I-03B, 4/5, 5/5, 6/5 See TABLE B-1
REPORTING/ACTIVATION						
4.1 The event, as written should be classified as a "Significant Event."				XX	XX	
4.2 The event, as written, is a "Significant Event." If there is a failure of diesels to start and load, it may progress to an ALERT emergency class.	XX					
4.3 The event, as written, is a "Significant Event," providing the leakage was secondary coolant. NOTIFICATION OF UNUSUAL EVENT, or higher, emergency class if leakage was primary coolant.		XX				
4.4 The event would be, as a minimum, NOTIFICATION OF UNUSUAL EVENT depending upon the magnitude of the release of primary coolant.*			XX			
4.5 This item must be considered on a case by case basis depending upon the reason for manual scram. See Table 1.1-1 of EP-Class, Item 7 for details.						XX

* See Item 3, Table 1.2-2; Item 8, Table 1.2-3; and Item 1, Table 1.2-4 of EP-CLASS

Table B-1

PLANT CONDITIONS REQUIRING MANUAL SCRAM

1) Abnormal reactivity changes beyond the capability of the regulating rod to compensate. (Emergency Procedure E)

2) Failure of a circulator helium outlet flapper valve to shut after circulator trip (Emergency Procedure C or D.)

3) PCRV Relief valve opens (Emergency Procedure H-2)

4) Steam leakage that is not isolated by a loop shutdown (Emergency Procedure C.)

5) Loss of all hydraulic pressure in one hydraulic power system (Emergency Procedure M)

6) Turbine shutdown following a loss of outside power without a turbine trip.

7) Loss of outside power with turbine trip. (Emergency Procedure F-3)

8) Loss of condenser vacuum (Emergency Procedure F-2)

9) Fire in, or affecting, a control panel, instrument cabinet, or load center. (Emergency Procedure I)

10) Fire in the general plant area affecting continued safe operation. (Emergency Procedure I)

11) Loss of access to the control room (Scram from I-49) (Emergency Procedure R)

12) Loss of instrument air Header (Emergency Procedure L)

13) Permanent loss of instrument Bus 1 or 2 (Emergency Procedure N)

14) Loss of interruptible Bus 3. (Emergency Procedure N)

15) When any automatic scram is called for.

INTRODUCTION

Reactor scram is the ultimate defense against any circumstance or condition that threatens to damage the reactor core and release radioactive fission products. Reactor scram is initiated automatically by the Plant Protective System (PPS) in a number of situations described in the following discussion of symptoms. Reactor scram is also initiated manually in a number of situations in which the control of the plant is threatened, and a rapid shutdown is called for.

The reactor scram results in a rapid shutdown of the nuclear chain reaction and the attendant power production from fission. However, immediately after reactor scram there remains a significant amount of heat generation due to the radioactive decay of the fission products. There is also a large amount of thermal energy stored in the primary components, particularly the graphite core and reflector. Therefore, the control and reduction of primary system temperature requires a continuation of heat removal through the steam generators. The control system is designed to accomplish these necessary heat removal activities automatically following the scram. The primary function of the immediate operator actions of this Emergency Procedure are to backup the Control System to insure adequate removal of decay heat from the core.

The specific PPS actions that are initiated by reactor scram are as follows:

1. The control rod brakes are de-energized.

2. Power to #1 and #2 Rod Drive Motor Control Centers are interrupted (K49-51 and K48-50 green lights on - Board I-10).

3. First-in scram annunciator and indicating light are actuated, indicating which logic channels have tripped, A, B, or C.

4. Turbine runback is initiated (Scram auxiliary relay red lights on).

5. Turbine is tripped 120 seconds after runback is initiated. (TR54-55 green lights on - Board I-10).

   NOTE: Turbine trip is initiated immediately upon receiving reactor pressure low (programmed) signal or signal indicating trip of all four helium circulators.

6. The feedwater flow program is initiated (scram auxiliary relay red lights on).

NOTE: The feedwater program is inhibited on reheat steam temperature high scram.

7. Circulator trip on programmed low feedwater flow (caused by scram signal) is prevented.

DISCUSSION OF SYMPTOMS

SYMPTOMS

1.1 Startup Count Rate High (RMS Fuel Loading) I-03B 4/4, 5/4, 6/4

This scram is actuated when either source range nuclear channel equals or exceeds a neutron countrate of 1.0E-5 counts/second with the Reactor Mode Switch in the FUEL LOADING position. This scram is provided for use during fuel loading and preoperational testing or other low power operations. The scram action is initiated by 1 of 2 logic trip by the Nuclear Start-up Channels I or II.

1.2 Neutron Flux Rate of Change High (ISS Start-Up) I-03B 4/3, 5/3, 6/3

This scram is actuated as a result of Wide Range Nuclear Channels III, IV or V equaling or exceeding a neutron flux rise of 5 DPM. This scram is utilized during plant start-up and results in additional protection and better scram response than the "Neutron Flux-High" scram in case of accidental control rod withdrawal when operating with the ISS in the start-up position. The setpoint is selected to be above the usual operating rate of flux change.

1.3 Neutron Flux High I-03B 1/1, 1/2, 2/1, 2/2, 3/1, 3/2.

This scram is actuated as a result of the correct two of six nuclear channels (III, IV, V) or (VI, VII, VIII), equaling or exceeding 140% of full power flux. Channels III, IV, and V are combination power and wide range nuclear detectors. Channels VI, VII, and VIII are power range nuclear detectors. High neutron flux levels with the associated excessive heat generation requires a scram to prevent damaging core temperature increases.

1.4 Reheat Steam Temperature High I-03B 1/3, 2/3, 3/3

When the hot reheat steam temperature, as measured by the high selected average of two thermocouples from each loop per scram channel, equals or exceeds the 1075°F setpoint, that channel trips. Tripping of two of the three channels causes scram action.

Hot reheat steam temperature high indicates an increase in core power generation or a decrease in primary coolant flow or reheat steam flow. Hot reheat temperature is used in lieu of reactor primary coolant outlet temperature because of the difficulty in providing redundant gross primary coolant temperature measurements. The design of the steam generator is such that changes in primary coolant temperature first affects the hot reheat steam temperature. This scram serves as a backup to the neutron flux high scram. The trip level is chosen to be just above expected transients in order to minimize the scram response time and corresponding temperature overshoots on rod withdrawal accidents not terminated by the neutron flux high scram.

To further reduce the temperature transient, the feedwater flow program is initially defeated by clamping the output of the rate limiters in the feedwater flow setpoint circuit. When the measured reheat temperature is reduced to 975°F, the feedwater flow program will occur. Maintaining the feedwater flow fixed for a reheat steam temperature high scram from 100% load will require about 23% of the steam to be bypassed by the electromatic relief valves. This condition will last for about nine minutes to reduce the reheat steam temperature to 975°F.

1.5 Reactor Pressure High (Programmed) I-03B 1/4, 2/4, 3/4

The reactor pressure high protective action is a backup to the High Moisture protective action. The reactor pressure high scram action is to scram the reactor and shut down and dump the loop that is preselected by a hand switch. Preselection is necessary because the reactor pressure high action has no basis for identifying the leaking loop. It depends solely on the physical fact that sufficient moisture inleakage will raise the reactor pressure above the high pressure setpoint (7 1/2% above normal) which is programmed with circulator inlet helium temperature. The reactor pressure high action also automatically depressurizes the operating steam generator loop to reduce the pressure differential across the leak in case the wrong loop has been dumped or both loops are leaking.

1.6 Two Loop Trouble I-03B 4/1, 5/1, 6/1

Each of the two primary-secondary coolant loops is provided with a series of PPS actions that shutdown the loop for a variety of reasons. These loop shutdown actions are covered by Emergency

Procedure "C" and explained in Appendix C. With one loop shutdown, the second loop must remain in operation at all times to provide active cooling of the reactor core. Thus, with one loop shutdown, the automatic shutdown of the second loop is inhibited and a reactor scram called "Two Loop Trouble" is substituted when shutdown of the second loop is called for by the PPS. Two Loop Trouble scram is inhibited with the RMS switch in fueling loading.

1.7 Loss of Plant Power I-03B 4/2, 5/2, 6/2

Undervoltage detectors sense the voltage on all three phases of both essential buses 1A and 1C. Detection of voltage loss persisting for 30 seconds by 2 of the 3 detectors on each bus produces 2 of 3 scram channel trips, causing scram.

The accident of concern is the loss of outside power, coincident with turbine generator trip and failure of one diesel generator to start. A scram is required to reduce heat generation to allow for heat removal with less than a normal complement of plant equipment. The immediate actions specified are designated to allow for orderly start-up and/or loading of the diesel generators in the event that they both haven't automatically supplied power to the essential loads. The follow-up actions are details for the manual starting and loading, as well as procedures for continued shutdown cooling.

1.8 Reactor Building Temperature High I-03B 4/6, 5/6, 6/6

High reactor building temperature (greater than or equal to 175°F.) indicates steam or helium leakage into the building. If PPS action to shutdown a loop did not isolate the leak, immediate actions are required to determine the source of the leak and to take actions to isolate it. The temperature and pressure recorders on I-09, actuated by a high pressure or temperature at any of the 14 sensor locations, should help in determining which loop is leaking. The opening of the main steam power operated relief valves (22% capacity) with 25% loop flow would not result in a pressure decay unless a leak of significant size (greater than or equal to 3%) was present in the loop main steam piping. If no pressure decay takes place, and a significant size leak has occurred, the isolation of all reheat piping in the affected loop, or the isolation of all common cold reheat piping should isolate the leak.

1.9 Reactor Pressure Low (Programmed) (ISS - Power) I-03B 1/5, 2/5, 3/5

Pressure elements sense the pressure in the circulator discharge plenum and operate pressure switches in the PPS. Actuation of two of the three switches trips their respective scram channels and causes scram.

Low primary coolant pressure is an indication of gross primary coolant leakage from the system. A scram is initiated because the reactor is in danger of being inadequately cooled, which would increase the hazard associated with activity release from the PCRV. The low pressure trip setpoint is 50 psi below normal and is programmed with helium circulator inlet temperature to reduce the response time. The reduction of response time is desirable to minimize steam generator tube stresses caused by an increasing primary coolant temperature in combination with a constant (controlled) main steam temperature. The control system tends to aggrevate the situation until a scram overrides the control system. A turbine generator trip is initiated simultaneously with the scram to anticipate the ensuing drop in main steam temperature.

The turbine should trip immediately in this scram instead of running back and tripping after 120 seconds as in most scrams. Immediate trip is required bacause the low pressure helium is not capable of transporting enough heat from the core to the steam generators to prevent steam generator flooding and water washing of the turbine if the normal runback-trip sequence is followed. The operator should manually trip the turbine as quickly as possible if automatic PPS actions does not.

This scram protects the reactor in the inconceivable event of the complete failure of both closures of a PCRV penetration. Any conceivable leak of primary coolant results in the reduction of the primary coolant system pressure to atmospheric over a period of from several hours to several days. The source of any leak should be determined and the leak isolated in this length of time. If the source of leakage cannot be located or isolated, the reactor should be manually scrammed and the PCRV depressurized (per SOP 24).

1.10 <u>Main Steam Pressure Low (ISS-Power) I-033 1/6, 2/6, 3/6.</u>

If the main steam pressure drops below 1500 psig, each of three pressure switches located on the common main steam line in the turbine building will cause a scram logic channel trip. Two of the three channel trips will cause a scram.

Low main steam pressure is an indication of either main steam line rupture or of gross failure of the feedwater system. Immediate shutdown of the reactor is appropriate in either situation. In addition, both superheater outlet stop check valves are automatically closed to reroute main steam to the flash tank through the individual loop bypass valves and desuperheaters. This is necessary for the continued operation of the helium circulators on steam. The scram trip point is selected to be below normal operating levels and system transients. The main turbine load is run back by the initial pressure regulator as pressure drops below the 2400 psig normal

operating pressure. The turbine is shutdown by the turbine protective system at ~ 2200 psig.

If the cause of the low main steam pressure was rupture of the main steam line, the automatic closing of the superheater stop check valves would isolate the leak and cooling would continue with both loops on steam drives of the helium circulators. Continuation of core cooling would be accomplished by one of the shutdown cooling modes.

The loss of feedwater considered is the complete loss of the use of all three boiler feed pumps due to failure of either the main condensate line on deaerator side of LCV-3175, the deaerator, the boiler feed pump suction line, the pumps themselves, or both main and emergency feedwater lines. If the cause of the low main steam pressure was failure of the feedwater system, all four helium circulators are automatically shutdown due to loss of feedwater flow and one loop will be shutdown due to both circulators being tripped. In this case, further actions covered by abnormal operating procedures will be required to establish core cooling.

1.11 Reheat Steam Pressure Low (ISS - Power) I-03B 1/7, 2/7, 3/7.

Detection of low pressure by pressure switches located on the common hot reheat steam line in the turbine building causes a scram logic channel trip. Tripping two of three channels causes a scram. The low pressure trip setpoint is 35 psig.

Hot reheat steam pressure low is an indication of a rupture in either a cold reheat line or a hot reheat line in the section common to both loops. Loss of the cold reheat steam line results in loss of the steam supply to all circulators and boiler feed pump turbines. The direct scram in this case precedes a scram resulting from Two Loop Trouble. The trip point is selected to be below normal operating and transient pressures which vary over a wide range.

The Pelton Wheel drives on the helium circulators are automatically started by shutting down all circulator steam drives. The AUTO WATER TURBINE START will come in for the last operating loop on steam drives. The hot reheat stop valve is closed automatically in the second loop to aid in leak isolation.

1.12 Manual I-03B 4/5, 5/5, 6/5

A manual scram is inserted when called for in Table B-1 and following an automatic scram as a backup to the PPS to insure a full scram. A manual scram can be inserted by operating the scram handswitch to the scram position, by depressing two of three push buttons on I-49, or by placing the RMS (Reactor Mode Switch) to the off position. Also there are several mechanical

ways in which brake power may be disrupted there by allowing rods to fall (pull brake fuses, cut power cables, etc.).

DISCUSSION OF IMMEDIATE ACTION

2.1 Insert Manual Scram.

Manual scram is inserted when called for (See Table B-1), and, following automatic scram as a backup to the PPS to insure a full scram. Inward rod motion and decreasing flux are observed to verify that the scram is having the desired effect.

2.2 Place ISS in Low Power Position

The ISS may be placed in the low power position after plant conditions have stabilized and reactor power is < 30%. Stable plant conditions are insured when: 1) Main turbine generator has tripped; 2) Boiler feed pumps and feedwater flow has stabilized; 3) Helium circulator speeds are normal; 4) Main and reheat steam temperatures are decreasing at a maximum rate of 2°F/minute.

When the ISS is placed in the low power position the maximum cooldown rate of 2°F/minute must be controlled manually by adjusting feedwater flow and circulator speed.

Also, during plant cooldown the operator must be aware of the relationship between feedwater flow and circulator speed. If feedwater flow is lost, circulator speed must immediately be reduced to zero to prevent damage to steam generators and/or helium circulators. When restoring core cooling, feedwater flow should always be established before primary coolant flow is established.

2.3 Ensure Transfer of House Power

Following all scrams at power (except the reactor pressure low scram and a scram resulting from all four circulators tripped) an immediate action of the operator is to manually transfer the 4160V buses to the reserve auxiliary transformer, within the 120 seconds before the automatic turbine trip takes place for the following reasons.

Any of the following PPS actions cause the main generator auxiliary tripping relay 86G1 to be energized:

1) Any scram, with a 120 sec. time delay.

2) Reactor pressure low scram (programmed), no time delay.

3) All four circulators tripped, no time delay.

Energizing auxiliary tripping relay 86G1 produces the following actions:

1) Opens OCB 5301 (230 KV generator breaker)

2) Opens OCB 5300 (230 KV generator breaker)

3) Energizes 41G1, which trips the field breaker.

4) Energizes 86GT1, generator and transformer auxiliary tripping relay which in turn:

   a. Opens ACB 152AT1A (4160 volt feed to 1A bus)

   b. Opens ACB 152AT1C (4160 volt feed to 1C bus)

   c. Starts all emergency diesel engines.

5) Energizes the electro-hydraulic master relay, which in turn:

   a. De-energizes two pilot solenoids of the master trip solenoid valve (24VDC) (Trip requires 2 of 2).

   b. Energizes mechanical trip solenoid (125V DC).

   c. Either of these trip signals will trip the turbine SV's and ISV's.

6) Produces light and alarm.

The following actions also take place, as appropriate:

1) Low voltage on 1B bus causes 152RT1B to close.

2) If 152AT1A is open and 152RT1B is closed, bus tie BTAB closes.

3) If 152AT1C is open and 152RT1B is closed, bus tie BTBC closes.

As outlined above, it is seen that a momentary loss of power to the 4160 VAC buses, and hence to the 480 volt buses, is required for the automatic switching to the reserve auxiliary transformer to occur. However, all motor control center breakers are operated and held closed by holding coils energized from the high side of each breaker. Thus, a momentary interruption of power to the motor control center buses will cause the associated closed breakers to open. Equipment must then be restarted through the auto start provisions of the various systems or through operator action.

| 2.4 Ensure Turbine Trip

This is a backup to the PPS action required only when the turbine is in operation. The turbine is tripped in anticipation of the drop in steam pressure resulting from the scram. Without the turbine trip, control of steam pressure and temperature would be difficult and the cooldown rate would be excessive. Further, the time period after scram in which the circulators and feed pumps could continue to be operated on reactor steam would be rapidly decreased by continued turbine operation.

Most of the scrams result in turbine runback at 1% per second to 10% load followed 120 seconds after scram by a turbine trip. This provides a less severe turbine shutdown transient, insures a high rate of heat removal immediately following the scram to absorb energy that may have been produced by events leading to scram, and gives the operator the option to transfer the in-house power supply to the reserve auxiliary transformer.

Two scrams, reactor pressure low and all four circulators tripped (i.e., two loop trouble), result in immediate turbine trip without runback. In both these cases, continued turbine operation only aggrevates the situation, so immediate trip is appropriate.

If the PPS does not runback and/or trip the turbine as required, the turbine should be tripped manually.

| 2.5 Ensure or Establish Stable Core Cooling Conditions

Verify that the transients associated with the scram have not caused circulator trips, and if they have, verify that the water turbine drives have started automatically. The circulators should continue to operate on reactor steam for some time after scram.

If the circulators are not operating, then the operator must take immediate action to restart at least one circulator on water or steam drive, as appropriate. Procedures for supplying circulators with water or steam under a variety of circumstances are provided in the SOP's, in Abnormal Operating Procedures and in the "Safe Shutdown Cooling with Highly Degraded Conditions," document.

Because of the initially high decay heat load and the stored energy in the core, steam conditions should remain near the normal operating setpoints for some time following the scram. The operator should verify that the main and reheat steam temperatures and pressure controllers have handled the scram transient and continue to maintain proper setpoint conditions. If not, the operator should take manual control of the errant controller.

The rapid power decrease of a scram, turbine runback and trip requires pre-programmed feedwater control system action to mitigate the thermal shock to the steam generators. The operator should verify that the feedwater flow program is initiated (inhibited on Hot Reheat steam temperature high scram until Hot Reheat temperature is below 975°F.). If not, the operator should take manual control of the errant controller.

DISCUSSION OF FOLLOWUP ACTIONS

| 3.1 Ensure Reactor Internal Maintenance Terminated and Openings Closed

This scram occurs only when the Reactor Mode Switch is in the "FUEL LOADING" position. The operator should advise any personnel involved in operations from the refueling floor that a scram has occurred and that further operations should be terminated until it is determined to be safe to proceed.

3.2 Start Auxiliary Boiler

As reactor steam pressure drops off following the scram, an auxiliary boiler should be started to supply the 150 psig steam header. The time after scram that the auxiliary boiler will be required varies with a number of factors including the power conditions at which the scram occurred, the decay heat rate (influenced by core power history), the rate of cooldown and equipment operability conditions in the plant (for example, auxiliary boiler steam may be desired as a backup power source for the circulators if the feedwater system has problems).

3.3 Establish Two Circulators at ~ 7000 RPM

Two circulators (on steam driven or on Pelton Wheel) at 7000 RPM will provide sufficient primary coolant flow to prevent core over heating from decay heat. The two circulators may be one in each loop or two in one loop and still provide sufficient primary coolant flow.

<u>If any core region outlet temperature exceeds 2200°F, increase circulator speed to 8000 rpm to obtain maximum available primary coolant flow.</u>

3.4 Isolate Primary Coolant Leak. Depressurize Through Purification Train if Possible

Pumping down the PCRV through the Helium Purification System will minimize the leakage of radioactive helium into the reactor building, release to the atmosphere and reduce subsequent cleanup problems.

3.5 Isolate Secondary Coolant Leak or Establish Feedwater Flow

In conjunction with symptom

(1.8 Reactor Building Temperature High Scram.)

1) If temperature and pressure records on I-09 clearly indicate the leaking loop, trip both circulators in that loop.

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Tripping both circulators results in loop shutdown which shuts the feedwater, main steam and reheat steam stop valves as well as the circulator steam inlet and outlet block valves. This action isolates the steam leak unless it is in the cold reheat line or the hot reheat piping downstream of the reheat stop-check valve.

2) If I-09 recorders do not indicate the leaking loop:

a) When feedwater flow has been reduced to 25%, open the power operated main steam safety valves in both loops.

b) If main steam pressure decreases in either loop, trip both circulators in the loop.

If the leak is in the 5-10% of steam flow range, it may be difficult to locate by the I-09 instruments and the rate is too small to make a significant difference in loop steam pressures. In this case opening the power operated main steam relief valves will dump about 22% of the 25% steam flow so that a smaller leak should make a significant difference in the steam pressure in the leaking loop.

3) If loop steam pressure decrease does not indicate the leaking loop, the leak must be in the reheat steam piping. In this event:

a) Trip all four circulators and shut both circulator bypass block valves.

b) Start water turbine drives in one loop.

c) If leak is still not isolated it must be in the cold reheat piping. In this event, raise setpoint on the main steam bypass valves until all steam flow is through the Main Steam Relief Valves.

These steps assume that the leak is in the reheat piping and cannot be isolated by shutting down a loop. Therefore, steam flow to all circulators is secured and water turbines are started to maintain active core cooling. If the leak turns out to be in the common cold reheat piping between the turbine and the circulators, it is necessary to raise the main steam bypass valve setpoints until they are shut and all steam flow is through the relief valves. This action is necessary because the bypass valves admit steam to the flash tank which cannot be isolated from the cold reheat piping.

3.5 Isolate Secondary Coolant Leak or Establish Feedwater Flow

In conjunction with symptom

(1.10 Main Steam Pressure Low Scram)

1) Main Steam Pressure Low Caused by Line Rupture.

a) Ensure rupture is isolated.

Low main steam pressure would be expected to result from a rupture in the main steam line, the loop steam lines, or main or emergency feedwater lines. The rupture should be isolated by automatic closing of the loop isolation valves on loop shutdown, closing of the loop main steam stop checks or closing of the boiler feed pump discharge valves when low pressure is sensed in the main feedwater line. Operator action would be required to isolate an emergency feedwater header rupture.

b) Maintain shutdown cooling with available helium circulators and steam generators.

With the line rupture isolated, shutdown cooling may be maintained with steam or water driven helium circulators and normal or emergency feedwater supply to the operable steam generator depending on the plant conditions.

2) Main Steam Pressure Low Caused by Loss of Main and Emergency Feed Water Supply.

a) Depressurize at least one steam generator loop and establish condensate supply of steam generators via the emergency condensate line.

The loss of main and emergency feedwater supply could result from failure of the main condensate line, the deareator, the boiler feedpump suction line, the pumps themselves or both the main and emergency feedwater line. The loss of feedwater supply will cause loop shutdown of one loop and trip of the remaining two circulators on low speed. Both loop mainsteam stop-check valves are closed. The depressurization of the steam generators is required prior to establishing condensate flow and is accomplished with the electromatic relief valves. A condensate flow path is established with the emergency condensate line supply and discharge through the main steam generator bypass valves to the flash tanks.

b) Operate at least one helium circulator in an operable loop on condensate to match helium flow with condensate supply to the steam generator.

After condensate supply to one steam operator has been established one or two helium circulators, as necessary, are started on condensate drive and circulator speed increased until the desired helium flow is obtained. The helium flow should be adjusted to maintain the steam generator in the sub-boiling conditions.

3.5 <u>Isolate Secondary Coolant Leak or Establish Feedwater Flow</u>

In conjunction with symptom

(1.11 Reheat steam pressure low scram)

1) If circulators are still operating on steam:

a) Trip both circulator steam drives in one loop and insure appropriate loop shutdown.

b) Trip both circulator steam drives and shut hot reheat stop-check valves in the other loop.

c) Insure automatic water turbine start of last two circulators tripped or manually start circulators on water turbines.

This scram indicates a major steam leak in the reheat steam piping. Such a leak will probably affect steam flow to all circulators, therefore, all circulators are shut down manually if they haven't already tripped automatically. In this situation, the water turbines in one loop should start automatically to maintain active core cooling.

2) Open power operated main steam safety valves

3) Raise main steam and startup bypass valve setpoint to 2800 psig.

These steps provide a path for steam flow so that core cooling is maintained and then shut off steam flow to the flash tank. This is necessary in the event of a steam leak in the reheat piping because the flash tank cannot be isolated from the cold reheat line. Continue cooldown with this flow path until temperatures are low enough to switch to condensate. The secondary coolant temperature and pressure will be low enough that a flow path may be re-established through the bypass flash tank.

4) Ensure motor driven boiler feed pump is operating and then shut down steam driven boiler feed pumps.

As discussed above, all reactor steam is being dumped to atmosphere through the safety valve. Therefore, steam flow to the deaerator and turbine driven feed pumps is lost, at least until the auxiliary boiler can be brought into service.

5) Ensure hotwell makeup valves are open.

As discussed above, no steam is being returned to the condenser, so hotwell level must be maintained by makeup from the condensate storage tank.

6) Reduce feedwater flow to maintain main steam pressure below the setting of the main steam safety valves.

This step assumes that the scram occurs with feedwater flow greater than 25%. In this event, the feedwater flow must be reduced to the capacity of the open power operated relief valve (about 22%). Thereafter, the feedwater flow should be controlled to maintain and/or reduce the steam temperature and pressure depending on whether the plant is to be cooled down. The steam pressure should not be permitted to increase above the setpoint of the spring loaded safety valves.

3.6 <u>If Reactor Pressure High Scram or Two Loop Trouble Scram Due to Moisture Inleakage, Proceed to Emergency Procedure "A".</u>

Emergency Procedure "A" contains the appropriate operator action in the event of moisture inleakage.

3.7 <u>Start Emergency Diesel Sets and Energize 480 Volt Essential Busses</u>

In conjunction with symptom

(1.7 Loss of plant power scram.)

1) Ensure all standby diesel generators have started. If not, start them from the Control Room. If none will start from the Control Room, start them from their control panel.

In this situation the primary concern is maintaining enough electrical power to continue active core cooling. Only one diesel generator is required to produce this minimum amount of power. Thus if the turbine generator and all outside power are lost, at least one diesel-generator must be started. The second diesel generator provides backup to the first and produces additional power

to facilitate subsequent operations. Both diesel generators should start automatically when the turbine generator trips and should pick up essential loads when power is lost to the 4KV busses. In the event of malfunction in the automatic controls, the diesel generators can be started and/or loaded manually from the Control Room. If manual control from the Control Room is lost, the diesel generators can be started from their local control panels. Whatever the operator action required, it is essential that at least one diesel generator operates when all other power sources are lost.

The following are suggested troubleshooting steps to be followed when the diesel-generator runs, but will not pick up the electrical load as it should. The auto-start prohibit relays prevent the breaker between the diesel generator and the 480 volt bus from closing until the 480 volt bus is de-energized. Because the 480 volt bus normally carries more than the essential loads, load shedding must occur and the bus must be isolated before the supply breaker can be closed without overloading the diesel-generator. When these conditions are satisfied, the diesel-generator feed breakers must close and the 480 volt bus tie between the energized bus (1A or 1C) and bus 1B must close to supply the essential loads. When power is available to the 480 volt bus, the automatic load sequencer should operate to start essential equipment in the proper order. If any of these normally automatic actions fails to occur, the operator should take the appropriate manual action.

2) Ensure that the diesel generator auto-start prohibit relays (HS-9221/5 and HS-9225/5) are tripped.

3) Ensure that 480 Volt bus load shedding has occurred.

4) Ensure isolation of de-energized 480 volt busses.

5) Ensure diesel generator feed breakers are closed.

6) Ensure the 480 volt 1B bus tie is closed to the bus of the first functioning diesel generator.

7) Ensure that the automatic load sequencer is operating to start essential equipment. If not, manually start loads according to the program list.

8) If power cannot be re-established to the security system start the ACM diesel-generator and transfer security system loads to the ACM system.

The security system backup power is supplied from the ACM system and requires operator action to transfer from normal power to backup power.

9) If power cannot be re-established to sufficient equipment to maintain active core cooling proceed with Emergency Procedure G, "Extended Loss of Active Core Cooling."

If the cause of the loss of plant power was of such nature as to prevent the use of normal methods of removal of decay heat the procedures of Emergency Procedure G provide alternate methods of heat removal and preservation of PCRV integrity.

10) When condenser vacuum improves and condensate pressure is adequate, shut power operated hot reheat relief valves.

Condenser vacuum is affected by the reduction of circulating water flow. When the diesel generator is operating, one circulating water pump is restarted and vacuum should improve permitting steam to be dumped to the condenser instead of vented to atmosphere through the power operated hot reheat relief valves. Returning the condenser to service as quickly as possible is important to conserving the feedwater supply.

11) When power is available to the small condensate pumps, re-establish makeup to the deaerator.

This is necessary to maintain feedpump suction pressure.

12) Initiate shutdown cooling.

The operator selects the appropriate mode of cooling to suit the specific circumstances he faces.

| 3.8 <u>Ensure Feedwater Flow Remains at Pre-Scram Level Until Reheat Temperature at 975°F.</u>

This action is backup to PPS. The operator should take manual control of feedwater flow if the automatic system does not respond as programmed.

REPORTING/ACTIVATION

4.1 <u>The event as written should be classified as a "Significant Event".</u>

Unless the listed symptoms deteriorate further, this event should be reported within one hour to the NRC in accordance with the Significant Event Reporting Procedure.

4.2 The event as written, is a "Significant Event". If there is a failure of diesels to start and load, it may progress to an ALERT emergency class.

If diesels start and load, report this incident within one hour to the NRC in accordance with the Significant Event Reporting Procedure. If the diesels do not start and load, classify the event as an ALERT, and implement RERP implementing procedure CR-ALERT.

4.3 The event, as written, is a "Significant Event", providing that the leakage was secondary coolant. NOTIFICATION OF UNUSUAL EVENT, or higher, emergency class if leakage was primary coolant.

If the leakage was secondary coolant, report this incident within one hour to the NRC in accordance with the Significant Event Reporting Procedure. If the leakage was primary coolant, classify the event as a NOTIFICATION OF UNUSUAL EVENT, and implement CR-UE RERP implementing procedure.

4.4 The event would be, as a minimum, NOTIFICATION OF UNUSUAL EVENT, depending upon the magnitude of the release of primary coolant.

Review Item 3, Table 1.2-2; Item 8, Table 1.2-3; and Item 1, Table 1.2-4 in EP-CLASS to determine if the event is more serious than a NOTIFICATION OF UNUSUAL EVENT. Implement CR-UE for UNUSUAL EVENT class, or CR-ALERT for an ALERT or higher class.

4.5 This item must be considered on a case by case basis depending upon the reason for Manual Scram. See Item 7 of Table 1.1-1 of EP-Class for details.

Table 1.1-1 of EP-CLASS in Item 7 lists cases whereby a manual scram does not constitute a "Significant Event". If the event does not meet these exceptions, report the incident within one hour to the NRC in accordance with the Significant Event Reporting Procedure.

TITLE: TWO LOOP TROUBLE SCRAM, WITH A TROUBLE ALARM IN OPERATING LOOP

ISSUANCE AUTHORIZED BY			
PORC REVIEW	PORC 464 MAY 13 1982	EFFECTIVE DATE	5-14-82

(B-2)

TWO LOOP TROUBLE SCRAM WITH A TROUBLE ALARM IN OPERATING LOOP

SYMPTOM-ACTION MATRIX

ACTIONS	SYMPTOMS: 1.1 Helium Circulator Tripped A & B 1-05A, 4-5 C & D 1-05D, 4-5	1.2 Steam Generator Penetration Pressure High Loop 1 1-05A, 2-6 or Loop 2 1-05D, 2-6	1.3 Reheater Header Activity High Loop 1 1-05A, 2-5 or Loop 2 1-05D, 2-5	1.4 Main Steam Temperature Low Loop 1 1-05A, 4-6 or Loop 2 1-05D, 4-6	1.5 Pipe Rupture Loop 1 1-05A, 3-6 or Loop 2 1-05D, 3-6
IMMEDIATE ACTION					
2.1 Insure Immediate Actions per Reactor Scram (EP "B-1") Complete.	XX XX	XX XX	XX XX	XX XX	XX XX
2.2 Restore Active Core Cooling.	XX XX				
FOLLOWUP ACTION					
3.1 Reduce Feedwater Pressure to about 850 psig and Continue Cool Down on Operating Loop.		XX			
3.2 If possible, Continue Cool Down on Affected Loop until Region Outlet Temperatures are 900°F or less. Otherwise, proceed to Step 3.5.			XX		XX
3.3 If leak is in Reheater, Isolate Reheaters and Start Circulators on Pelton Wheel. Continue Cool Down using EES Section of Steam Generator.			XX		XX
3.4 If leak is in Main Steam Piping, Establish Flow through Power Operated Relief Valve(s).					XX
3.5 If Core Cooling Cannot be Maintained or Restored, Proceed to EP "C", EXTENDED LOSS OF ACTIVE CORE COOLING.	XX	XX	XX	XX	XX

TWO LOOP TROUBLE SCRAM WITH A TROUBLE ALARM IN OPERATING LOOP

SYMPTOM-ACTION MATRIX

ACTIONS	SYMPTOMS				
	1.1 Helium Circulator Tripped A & B 1-05A, 4-5 C & D 1-05D, 4-5	1.2 Steam Generator Penetration Pressure High Loop 1 1-05A, 2-6 or Loop 2 1-05D, 2-6	1.3 Reheater Header Activity High Loop 1 1-05A, 2-5 or Loop 2 1-05D, 2-5	1.4 Main Steam Temperature Low Loop 1 1-05A, 4-6 or Loop 2 1-05D, 4-6	1.5 Pipe Rupture Loop 1 1-05A, 3-6 or Loop 2 1-05D, 3-6
REPORTING/ACTIVATION					
4.1 The event would be reportable as a "Significant Event", as written.		XX		XX	XX
4.2 The event would be reportable as a "Significant Event", if core cooling could be maintained or quickly restored. If the event degrades to a LOFC for greater than two hours (from 100% power) the event becomes an ALERT or higher emergency class.	XX				
4.3 The event would result in an unplanned release, and would be, as a minimum, a NOTIFICATION OF UNUSUAL EVENT.			XX		

INTRODUCTION

This emergency procedure is an extension of Emergency Procedure B-1 "Reactor Scram". The action item steps presume one coolant loop has tripped and been isolated. The reason for the first loop trip could be for any number of causes and it is futher assumed that the loop is not available for active core cooling. With trouble in the remaining loop and Two Loop Trouble Scram, Emergency Procedure B-2 is a guide for the operator to insure that active core cooling is either restored or continued on the remaining loop.

DISCUSSION OF SYMPTOMS

Also see discussions of symptoms in Emergency Procedure "C".

1.1 Helium Circulator Tripped
A & B I05A 4-5
OR
C & D I05D 4-5

This condition could result from a number of causes due to individual circulator trips and loop trip or fixed feedwater flow low. In the latter case, trip of both circulators in the remaining loop is not inhibited.

1.2 Steam Generator Penetration Pressure High.
Loop 1 I05A 2-6
OR
Loop 2 I05D 2-6

1.3 Reheat Header Activity High
Loop 1 I05A 2-5
OR
Loop 2 I05D 2-5

1.4 Main Steam Temperature Low
Loop 1 I05A 4-6
OR
Loop 2 I05D 4-6

1.5 Pipe Rupture
Loop 1 I05A 3-6
OR
Loop 2 I05D 3-6

These conditions in the remaining loop will result in Two Loop Trouble Scram but the loop continues in operation to provide active core cooling.

DISCUSSION OF IMMEDIATE ACTION

2.1 Insure immediate actions for reactor scram (EP B-1) complete.

These actions provide backup for the PPS automatic scram and insure active shutdown core cooling.

2.2 Restore active core cooling.

With all four circulators tripped, active core cooling must be restarted. It will be necessary to clear at least one circulator trip and possibly correct the condition that caused the trip.

DISCUSSION OF FOLLOWUP ACTION

3.1 Reduce feedwater pressure to about 850 psig and continue cooldown on operating loop.

With a Steam Generator Penetration Pressure High alarm in the operating loop, this action will reduce the driving force behind the leak. The reduction in feedwater pressure is accomplished by reducing the setpoint on PC-22129 or PC-22130, or by opening relief valves PV-22167 or PV-22168.

3.2 If possible, continue cooldown on affected loop until region outlet temperatures are 900 degrees Fahrenheit or less. Otherwise, proceed to Step 3.5.

3.3 If leak is in reheater, isolate reheaters and start circulators on pelton wheel. Continue cooldown using EES section of steam generator.

With Hot Reheat Header Activity High alarm in the operating loop and radioactive gases being released from the plant, it is desirable to both maintain active core cooling and to isolate the leak. Not isolating the affected reheater until core region outlet temperatures are 900 degrees Fahrenheit or less will assure that the reheater remains well within its design temperatures.

3.4 If leak is in Main Steam piping, establish flow through power operated relief valve(s).

This will reduce the amount of steam released directly into plant buildings and help protect plant equipment from damage.

3.5 If core cooling cannot be maintained or restored, proceed to EP-G, Extended Loss of Active Core Cooling.

EP-G references procedures "Cooling Using Abnormal Procedures" and "Safe Shutdown and Cooling with Highly Degraded Conditions" which provide many alternative means of obtaining active core cooling.

REPORTING/ACTIVATION

4.1 The event would be reportable as a "Significant Event", as written.

Any manual or automatic scram from >2% power is reportable as a "Significant Event", and should be reported to the NRC within one hour in accordance with the Significant Event Reporting procedure.

4.2 The event would be reportable as a "Significant Event" if core cooling could be maintained or quickly restored. If the event degrades to a LOFC for greater than two hours (from 100% power) the event becomes an ALERT, or higher, emergency class.

If circulation is maintained/quickly restored, report the event within one hour to the NRC in accordance with the Significant Event Reporting procedure. Otherwise, implement RERP implementing procedure CR-ALERT.

4.3 <u>The event would result in an unplanned release, and would be, as a minimum, a NOTIFICATION OF UNUSUAL EVENT</u>

Depending upon the magnitude of the release, the event will be as a mimimum, an UNUSUAL EVENT. Refer to Tables 1.2-2, Item No. 4; 1.2-3, Item No. 8; or 1.2-4, Item No. 1 of EP-CLASS for further details. Implement RERP implementing procedure CR-UE for an UNUSUAL EVENT, CR-ALERT for an ALERT or higher emergency classification.

TITLE: LOOP SHUTDOWN

ISSUANCE AUTHORIZED BY			
PORC REVIEW	PORC 464 MAY 13 1982	EFFECTIVE DATE	5-14-82

(C)
LOOP SHUTDOWN

SYMPTOM-ACTION MATRIX

ACTIONS	SYMPTOMS					
	1.1 Helium Circulator Tripped A & B 1-05A, 4-5 C & D 1-05D, 4-5	1.2 Steam Gen. Penet. Pressure High Loop 1 1-05A, 2-6 Loop 2 1-05D, 2-6	1.3 Reheater Header Activity High Loop 1 1-05A, 2-5 Loop 2 1-05D, 2-5	1.4 Main Steam Temperature Low Loop 1 1-05A, 4-6 Loop 2 1-05D, 4-6	1.5 Pipe Rupture Loop 1 1-05A, 3-6 Loop 2 1-05D, 3-6	1.6 Feedwater Flow Low Loop 1 1-05B, 1-1 Loop 2 1-05C, 1-3
IMMEDIATE ACTION						
2.1 Insure Turbine runback & Reactor Power decreases to approximately 1/2 the previous power level.	W E XX XX	XX XX	XX XX	XX XX	XX XX	XX XX
2.2 Insure Circulators in Shutdown Loop stop and their helium outlet valves close.	XX	XX	XX	XX	XX	XX
2.3 If the Circulator helium outlet valves in the Shutdown Loop do not close, Scram Reactor per EP "B-1".	XX	XX	XX	XX	XX	XX
2.4 Insure Secondary Flow in Shutdown Loop is Isolated.	XX	XX	XX	XX	XX	XX
2.5 Insure at least one dump valve is open.		XX				
FOLLOWUP ACTION						
3.1 Insure that Steam Generator Penetration Interspace Isolation Valve is closed in the Shutdown Loop.		XX XX				
3.2 If dump tank pressure exceeds 250 psig or dump tank activity is indicated or when dump is complete - manually close dump valves.		XX				
3.3 Reduce Plant Power to recover Shutdown Loop per OPOP VIII. *	XX XX	XX XX	XX XX	XX XX	XX XX	XX XX
3.4 Insure stable operation of Primary Coolant System.				XX		XX
3.5 If Steam Leak has not been isolated, manually Scram Reactor per EP "B-1".					XX	

* For symptom 1.3, proceed to Step 4.1 prior to resuming routine plant operations.

(C)

LOOP SHUTDOWN

SYMPTOM ACTION MATRIX

ACTIONS	SYMPTOMS					
REPORTING/ACTIVATION	1.1 Helium Circulator Tripped A & B I-05A, 4-5 C & D I-05A, 4-5	1.2 Steam Gen. Penet. Pressure High Loop 1 I-05A, 2-6 Loop 2 I-05D, 2-6	1.3 Reheater Header Activity High Loop 1 I-05A, 2-5 Loop 2 I-05D, 2-5	1.4 Main Steam Temperature Low Loop 1 I-05A, 4-6 Loop 2 I-05D, 4-6	1.5 Pipe Rupture Loop 1 I-05A, 3-6 Loop 2 I-05D, 3-6	1.6 Feedwater Flow Low Loop 1 I-05B, 1-1 Loop 2 I-05C, 1-3
4.1 The event, as written, is a "Significant Event."	XX	XX			XX	
4.2 The event, as written, is a "Significant Event," but may result in problems for primary coolant/fuel temperatures, and should be evaluated as required.				XX		XX
4.3 The event would result in an unplanned radiological release, and would be, as a minimum, a NOTIFICATION OF UNUSUAL EVENT emergency class (See Item 2 on Tables 1.2-2 and 1.2-3 of EP-CLASS for further discussion of this symptom).			XX			

INTRODUCTION

There are a number of potential problems and conditions involving the steam generators, the circulators, the steam piping and/or the control system that require shutdown of major sections of the plant. Because of the two loop design, many of these problems and conditions can be handled without taking the plant completely out of service by shutting down only one loop of the Primary and Secondary Coolant Systems.

Loop shutdown may be initiated manually by the operator or automatically by the PPS. Regardless of how they are initiated, all loop shutdowns during power operation result in the following actions being automatically performed by the Plant Protective System:

1. The hot reheat stop check valve is closed. (This causes trips of both helium circulator steam drives . ISS in power.)

2. The radiation monitor sample line valve is closed.

3. Control signals for the hot reheat temperature controller and the feedwater valve differential controller are generated, as described in detail below.

4. The loop feedwater control valve is closed.

5. The loop feedwater stop valve is closed.

6. Both helium circulator steam drives are tripped.

7. The helium circulator bypass block valve is closed.

The trip of both circulator steam drives, in turn, produces signals to:

1. Close both circulator steam drive speed control valves.

2. Close both circulator steam drive outlet block valves.

3. Close both water turbine outlet steam trap isolation valves.

4. Close the loop main steam stop valve.

5. Close the loop reheater attemperator flow control valve.

6. Close the loop reheater attemperator feedwater block valve.

7. Reduce turbine load to 50% of the initial value, which produces a signal through the first stage pressure to the flux controller to reduce reactor power.

8. Adjust main steam pressure controls as described below.

9. Adjust feedwater controls as described below.

The remaining loop continues to operate at approximately its initial power level because the load setting on the turbine governor is automatically reduced to 50% of its initial power level at 10% per second. This balances the turbine steam flow with the operating loop's steam flow at its initial level. At the same time, the gain is doubled on the output of the main steam pressure controller, so that the feedwater control valve will respond twice as fast to pressure changes, making its response compatible with the reduced steam delivery capcacity.

If, at the time of loop shutdown or during subsequent operation, the remote setpoint for the operating feedwater controller becomes 50% or less of rated loop flow, the setpoint will be locked and further reduction of feedwater flow can only be accomplished by placing the feedwater controller in local setpoint or manual. This action is taken to prevent the total steam flow from dropping below 25% of rated, the minimum steam flow required to keep the main turbine in operation.

The feedwater differential pressure signal across the shutdown loop feedwater control valve is eliminated so that the boiler feed pump steam turbine governors respond only to the differential pressure across the operating feedwater control valve.

As the main turbine governor valves close toward 50% of the initial power, the feed forward signal from the first stage pressure to the flux controller will cause the automatic control rod, and probably the six pre-selected runback rods, to be inserted to reduce the reactor power to meet main turbine steam flow requirements. After the transient has passed, the operator must manually adjust the shim rods, so as to keep the automatic control rod in a position of control and to allow withdrawal of the runback rods.

The hot reheat steam temperature input to the reheat steam temperature controller from the shutdown loop is automatically cancelled. The setpoint of the hot reheat steam temperature controller then becomes the average of the six modules in the operating loop.

DISCUSSION OF SYMPTOMS

SYMPTOMS

1.1 Helium Circulator Tripped
A & B I-05A, 4-5
C & D I-05D, 4-5

Because active core cooling must be maintained, only one loop is permitted to be shut down. This is accomplished by "first in with lockout" circuitry which identifies the loop that had the first trip signal and prevents shutdown of the remaining loop.

1.2 Steam Generator Penetration Pressure High
Loop 1 I-05A, 2-6
Loop 2 I-05D, 2-6

This trip is caused by a pipe rupture within the steam generator PCRV penetration. In either loop, actuation of at least two of three pressure switches set at 757 psig will cause loop shutdown. To minimize the amount of steam/water inleakage to the penetration, a steam/water dump is initiated. In addition, the penetration inerspace block valve (HV-11151 or HV-11152) to the penetration is closed to prevent moisture backflow to the purified helium system. The penetration is protected against over-pressurization by relief valves. The trip point is set above normal operating pressures but below penetration relief valve setting.

1.3 Reheater Header Activity High
Loop 1 I-05A, 2-5
Loop 2 I-05D, 2-5

Each loop has three detectors monitoring radiation from the hot reheat header. If two of the three detectors sense radiation levels in excess of 5 mR/hr, a loop shutdown is initiated.

The high radiation level is caused by a reheater tube rupture resulting in leakage of primary coolant into the reheat steam system. Isolation of the helium leak requires shutting the reheater outlet stop check valve and a loop shutdown to minimize the amount of activity introduced into the steam system. Following loop shutdown and equalization of pressure in the reheater with the primary coolant, moisture may diffuse into the primary coolant system. Therefore, an inhibit is introduced in the moisture monitoring system upon detection of "High Reheat Header Activity" to prevent an unnecessary "Steam/water dump".

If a small leak in the reheater developed over a period of time, activity would be alarmed on I-05, either panel "B" (Loop I) or panel "C" (Loop II) at ≈ 1 mr/hr above background by the shine monitors or by the reheat sample monitor. Reheat sample monitor activity is recorded on the I-05 (RR-2263/2264) panel. Activity should also show up at the SJAE air discharge and will be indicated (RI-31193) and alarmed on I-05 and recorded on I-14.

1.4 Main Steam Temperature Low
Loop 1 I-05A, 4-6
Loop 2 I-05D, 4-6

Thermocouples in the main steam header on each loop sense the temperature in that loop and cause loop shutdown on the low temperature loop, if the temperature is less than 800°F, and the difference between loop temperatures exceeds 50°F on at least two of the three thermocouples.

Low main steam header temperature in a loop is indicative either of a feedwater valve or controller failure which results in excessive loop feedwater flow or a deficiency of primary coolant flow. As temperatures decrease below 800°F, this situation could result in moisture carryover with resulting mechanical damage to the turbine.

1.5 Pipe Rupture
Loop 1 I-05A, 3-6
Loop 2 I-05D, 3-6

This situation requires rapid action to isolate the leak, so as to minimize the pressure and temperature buildup within the PCRV support ring area. A combination of ultrasonic noise detectors with pressure or temperature instruments is used to initiate PPS action to shut down the leaking loop.

The ultrasonic instruments detect high frequency noise generated by the turbulence associated with escaping high pressure steam from leaks. Each channel contains two identical ultrasonic probes and amplifiers, one for each loop, energized by a common power supply.

There are three ultrasonic detector channels under the PCRV support ring. The sensing probes are mounted on opposite sides of a flat reflector surface located in a vertical plane that divides the loops. A steam leak in the piping of one loop should cause the ultrasonic detectors on that loop's side of the reflector surface to trip. The trip setpoint is an ultrasonic noise level of twice background. The ultrasonic trip alone does not cause loop shutdown until it is accompanied by trips of either the pressure or temperature detectors. Loop shutdown occurs when two of three ultrasonic detectors agree on which loop is leaking and two of the three pressure detectors or two of the three temperature detectors trip, confirming the presence of a major leak. The pressure detectors trip when the pressure inside the PCRV support ring increases by 2½ inches of water above the reactor building reference pressure. The temperature detectors trip at 130°F.

When a loop shutdown due to steam leakage under the PCRV occurs, the operator should verify that the leak has been successfully isolated. Verification that steam leakage has been arrested is aided by the pressure and temperature recorders located on the I-09 panel. These recorders are started when the pressure at any sensor location exceeds 2 inches of water or the temperature at any sensor location exceeds 120°F.

For steam pipe ruptures outside of the PCRV, the detectors, setpoints, and trip logic are identical to those used under the PCRV. Because of the size of the PCRV, two separate noise detector channels are used, one on the north side and one on the south. As in the case of the steam pipe rupture under the PCRV, it is important that the operator verify that the PPS action has isolated the leak. The pressure and temperature recorders on I-09 aid in this verification.

1.6 Feedwater Flow Low
Loop 1 I-05B, 1-1
Loop 2 I-05C, 1-3

Feedwater flow is monitored for each loop by the PPS. With the Interlock Sequence Switch (ISS) in the POWER position and an indication of less than 20% feedwater flow, the loop will be tripped and isolated to protect the affected steam generator. Also, the automatic feedwater flow control is programmed to maintain feedwater flow above 25%. Should a controller or valve malfunction cause a flow less than 20% of rated, the loop is tripped and isolated. This protects the affected steam generator from either an over-temperature transient or from thermal shock due to over-cooling.

DISCUSSION OF IMMEDIATE ACTIONS

2.1 Insure turbine runback and reactor power decreases to approximately one-half the previous power level.

The operator acts as a backup to PPS action that reduces steam demand to the level that was being produced by one loop.

2.2 Insure circulators in shutdown loop stop and their helium outlet valves close.

These actions backup the PPS and insure primary coolant flow in the shutdown loop is isolated. Specifically, both circulator speed control valves, steam outlet and helium outlet valves, and common bypass block valves should be closed.

2.3 If the helium outlet valves in the shutdown loop do not close, scram reactor per Emergency Procedure B-1.

A stuck open helium outlet valve would be detected by higher speed than the normal self turbining speed of about 300 to 400 rpm as a result of reverse flow. Some of the primary coolant helium flow would be diverted from the operating loop as backflow through the shutdown loop. Therefore, the proper action is to scram the reactor to reduce heat production to decay heat status.

2.4 Insure secondary flow in shutdown loop isolated.

This action is a backup to the PPS and prevents thermal shock to the steam generators. The feedwater inlet block, control, and reheat attemperation block valves should be closed. The main steam stop check, reheat stem stop check and sample valves should be closed.

2.5 Insure at least one dump valve open.

This is applicable to the Steam Generator Penetration Pressure High condition, only where the steam/water dump is part of the PPS automatic action. The loop dump limits the steam inleakage into the penetration cavity thus limiting the pressure buildup.

DISCUSSION OF FOLLOWUP ACTION

3.1 Insure that steam generator penetration interspace valve is closed in the shutdown loop.

The operator acts as backup to PPS action thus preventing steam from flowing back into the purifed helium system.

3.2 If dump tank pressure exceeds 250 psig or dump tank activity is indicated or when dump is complete - manually close dump valve.

Dump tank pressure and/or activity increase indicates the presence of helium from the primary coolant system. The dump valves are shut to minimize the spread of activity. The dump valves should also be shut when the steam-water dump is complete.

3.3 Reduce plant power to recover shutdown loop.

It is necessary to reduce plant power to recover the shutdown loop. An orderly plant shutdown is preferred to a reactor scram whenever possible. Loop recovery conditions and procedures are specified in OPOP VIII. For the case of Reheat Header Activity High, it is necessary to carry out the notifications in Step 4.3 prior to resuming routine operations.

3.4 Insure stable operation of primary coolant system.

With loop trip due to main steam temperature or feedwater flow problems, careful attention should be given to stable operation of the primary and secondary coolant systems.

3.5 If steam leak has not been isolated, manually scram the reactor per Emergency Procedure B-1.

The PPS action should have isolated the leaking loop. However, if the steam leak was not isolated, the reactor should be scrammed to minimize the amount of high temperature steam leakage and consequent damage to plant equipment.

REPORTING/ACTIVATION

4.1 The event, as written, is a "Significant Event."

This event is reportable as a "Significant Event" as a result of PPS actions resulting in a loop shutdown, and should be reported to the NRC within one hour, in accordance with the Significant Event Reporting procedure.

4.2 The event, as written, is a "Significant Event", but may result in problems for primary coolant/fuel temperatures, and should be evaluated as required.

This event is reportable as a "Significant Event" as a result of PPS actions resulting in a loop shutdown, and should be reported to the NRC within one hour in accordance with the Significant Event Reporting procedure. Item Number 5 on Table 1.2-1 of EP-CLASS discusses possible effects of this symptom, which could eventually require elevation to a NOTIFICATION OF UNUSUAL EVENT emergency class.

4.3 The event would result in an unplanned radiological release, and would be, as a minimum, a NOTIFICATION OF UNUSUAL EVENT emergency class.

Evaluate the magnitude of the radiological release utilizing work sheets contained in RERP implementing procedure CR-UE, as required, to assist in assessment of release magnitude. Further discussion of alternate emergency classifications is contained in Item 2 on both Table 1.2-2 and Table 1.2-3 of EP-CLASS. If this event is an UNUSUAL EVENT, implement RERP implementing procedure CR-UE. If the event is an ALERT or higher, implement CR-ALERT.

ENCLOSURE B

March 28, 1985
Search Technology, Inc.
25b Technology Park/Atlanta
Norcross, Georgia 30092

Public Service Company of Colorado
16805 Weld County Rd. 19 1/2
Platteville, Colorado 80651

Attention: Mr. M. E. Niehoff

SUBJECT: Walkthrough Validation of Response Times for Steam/Feed Pipe Ruptures

Dear Mr. Niehoff:

In response to NRC questions regarding the four-minute basis for environmental qualification of vital equipment, NED attempted to validate the four minute isolation time. The result of this NED analysis is a report (NDG-84-0631) that describes a total of ten design basis pipe breaks, along with the plant and operator actions that would result from those breaks. For each plant or operator action, the NED report lists an approximate elapsed time that is based on a combination of engineering analysis and interviews with site operator training instructors.

In order to determine whether the time estimates in the NED report are reasonable from an operational perspective, walkthroughs of the pipe break scenarios were conducted in the Fort St. Vrain Control Room Mock-up. These walkthroughs were conducted by the author using a licensed operator with 3 years of on-the-boards experience at Fort St. Vrain. Also in attendance during the walkthroughs were a plant equipment operator and the authors of the NED report.

In general, the walkthroughs and interviews support the PSC contention that a design-basis steam or feedline pipe rupture can be isolated within four mintes. Certain assumptions in the NED report are overly conservative in that they result in much longer response times than are credible. For example, each operator and plant action is assumed to take place serially. In reality, the East- and West-end operators work in parallel especially during emergencies.

In one case (hot reheat line break downstream of the Pressure switches), the conservatism of the NED report results in an elapsed time of four minutes until the leak is isolated. This assumes that the Reactor is not scrammed for 140 seconds after the Turbine trips due to loss of condenser vacuum. From the walkthrough, however, it is obvious that the operators would manually scram the Reactor in well under one minute after such a severe vacuum loss. This one change would considerably reduce the postulated time required to isolate the rupture.

One area in which the NED report appears to be non-conservative involves the time estimated for operators to read emergency procedures and decide on the proper follow-up actions. In all cases, the NED report estimates 30 seconds for the operator to pull an emergency procedure and determine the proper course of action. In these cases, however, emergency procedure B1 is applicable. It is considered to be a relatively involved procedure and some of the discussion concerning follow-up actions is ambiguous. The operator who participated in the walkthrough estimated that approximately two minutes is a more realistic estimate for this step.

After conducting these walkthroughs, it is apparent that, at least for design basis pipe breaks, the four-minute isolation criterion is likely to be met. However, there is one question that relates to design basis pipe breaks that persists and probably should be addressed by PSC. That question, at least in my own mind, is how long after a break is "isolated" might heat still be input to the Reactor or Turbine Building.

As to this question, consider a cold reheat line break in the Turbine Building. The NED report considers this rupture to be isolated once the operator closes the main steam bypass valves into the bypass flash tank. This is, in fact, the last thing the operator can do and closure of these valves does stop the input to the flash tank, which is feeding the line break. It seems intuitive to me, perhaps with no technical basis, that, even after the input to the flash tank is closed, BTUs would still be available to exit through the cold reheat line. Hence, even though the line break is "isolated" in under three minutes (according to the NED estimate), the Turbine building temperature may continue to increase, or not start decreasing, for some time beyond this point. I'm not technically astute enough to know if this is a problem, but I wanted to raise the question.

In summary, it looks as if the NED conservatism and non-conservatism cancel each other out, rendering their time estimates reasonable for the design basis pipe breaks they analyzed.

Very truly yours,

M. E. Maddox /KJ

M. E. Maddox,
Senior Scientist

MEM/sa

cc: D. Glenn
S. Marquez

ENCLOSURE C

RECEIVED AUG 1 6 1984

GA Technologies Inc.
P.O. BOX 81608
SAN DIEGO, CALIFORNIA 92138
(619) 455-3000

August 9, 1984
GP-2325

Mr. H. L. Brey, Manager
Nuclear Energy Division
Public Service Company of Colorado
2420 West 26th Avenue, Suite 100D
Denver, CO 80211

Subject: FSV Steam Break Accident - 10/20 Min. Leak Termination Analysis

Dear Mr. Brey:

At the request of Mr. J. Reesy, GA performed an analysis of the reactor and turbine building environmental temperatures for steam line break accident conditions. The analysis was based on leak termination after 10 minutes and after 20 minutes. The results, as summarized in Table I (below), show higher environmental temperatures than for the 4-minute case on which previously reported results were based.

TABLE I
PEAK ATMOSPHERIC AIR TEMPERATUARES
20 FEET FROM LEAK SOURCE

Building/Reheat Steam Line	Steam Leak Termination Minutes	Peak Atmosphere Temperature, ^{o}F
Reactor/Cold	4	429
Reactor/Cold	10	429
Reactor/Cold	20	442
Reactor Hot	4	506
Turbine/Hot	10	571
Turbine/Hot	20	612

Reesy to follow

The work is documented by SD&PD:CJR:083:84 which is transmitted for your information.

Should you have any questions regarding this transmittal, please contact R. Rosenberg at (619) 455-2174.

Very truly yours,

DJKowal

D. J. Kowal, Director
Fort St. Vrain Services

Enclosure
cc: J. Reesy

[MB-33]

Enclosure to GP-2325

GA CORRESPONDENCE FORM
1476

FROM:	C. Rodgers CJR	IN REPLY REFER TO:	SD&PD:CJR:083:84
TO:	R. Rosenberg	DATE:	June 28, 1984

SUBJECT: Fort St. Vrain reactor building cold reheat and turbine building hot reheat steam line breaks with 10 minute leak termination

Summary

Environmental qualification of safety-related equipment is required at Fort St. Vrain to ensure plant safety. Postulated accidents that may significantly alter the environment inside the reactor and turbine buildings and possibly affect equipment performance are studied. Operating conditions in the buildings are then predicted for these events. Qualification tests are performed on safety related items to ensure their satisfactory performance during these postulated accident conditions. The atmospheric conditions inside the buildings must be compatible with the operating limits of all safe shutdown cooling equipment to ensure plant safety. One postulated accident that could cause changes inside the buildings that may affect some of these components is a steam pipe break.

In the reactor building, a cold reheat steam line break upstream of the loop isolation valves was considered to be the worst case. In this event, automatic controls put the plant into a shutdown mode. Then diagnosis and manually initiated corrective action are necessary to isolate the rupture. In the turbine building, the hot reheat steam line break was considered the most severe accident because of the pipe size and steam enthalpy. No remedial action occurred prior to the manually initiated leak termination for this event.

Four, ten, and twenty minute delays from the initial rupture until manual termination of the steam flow were assumed for both pipe breaks in this analysis. The turbine and reactor buildings both have louvers that vent to the outside so pressure build-up is not considered a problem. However, temperature gradients developing from the steam leak source may become undesirable.

Peak atmospheric air temperatures predicted in the Fort St. Vrain reactor and turbine buildings by the CONTEMPT-G code for several steam pipe leak termination times at a distance of 20 ft from the source are summarized in Table 1. The cold reheat steam pipe leak in the reactor building and the hot reheat steam pipe leak in the turbine building were terminated at 4, 10, and 20 minutes. Plots of atmospheric air temperature in the reactor building are shown in Figure 1 for the three leak termination times. Temperatures in the turbine building are shown in Figure 2 for each leak termination. These temperatures are all at a distance of 20 ft from the steam source.

In both the reactor and turbine buildings the predicted atmospheric air temperatures appear higher and remain higher for longer periods of time with successively longer steam leaks. This is because there is more energy added to the environment by the released steam than can be absorbed for a steady state so air and component temperatures rise. The higher temperatures resulting from the longer leaks decay at the same rate as the four minute leak. They just begin decaying at a higher temperature so remain higher for a longer period of time. Thus longer steam leak terminations lead to equipment exposure at higher temperatures for longer periods of time.

Evaluation

The predicted response of the atmospheric air temperature inside the reactor building to a postulated accidental cold reheat steam line rupture was requested for a ten minute and twenty minute steam leak. Temperatures are to be predicted at a distance of 20 ft from the leak source. The worst location for this break would be upstream of the loop isolation valves. This rupture would cause excessive movement of the turbine rotor which will trip the turbine generator and low hot-reheat steam pressure will scram the reactor. Normal secondary coolant system control actions will cause the main steam loop isolation check valves to close and put the plant into a shutdown mode; however, the pipe rupture will still be fed with steam from the flash tank. Isolation of the pipe rupture can be accomplished remotely from the control room by increasing the setpoints on two pressure control valves, which are located upstream of

the main steam desuperheaters, to a valve above the settings for the loop pressure relief valves. The superheater steam would then be discharged to the atmosphere instead of to the flash tank and the rupture would be isolated. Diagnosis and corrective action will require some time to terminate this leak.

Also, the predicted response of the atmospheric air temperature inside the turbine building to an accidental hot reheat steam pipe rupture for a ten minute and twenty minute leak was requested. These temperatures were to be determined at a distance of 20 ft from the leak source. The hot reheat steam pipe rupture will dump more heat energy into the building for a limited time interval than any comparable superheat or cold reheat steam leak because of its pipe size and steam enthalpy. It was assumed that remedial action such as reactor scram did not occur before the manually initiated leak termination.

Calculations made in 1972 for similar postulated pipe leaks using a four minute termination were recalled to form the basis for this new study. These results were published in Reference 1. Figures 3.11 and 3.12 of this reference are reproduced here as Figures 3 and 4 for the reactor building and turbine building atmospheric temperatures after the rupture. Table 2, line 1 shows the peak temperatures for the reactor building 20 ft and 30 ft from the leak and the turbine building 20 ft from the source from this reference. A trial calculation was made to reproduce these results. This would verify the model and data before proceeding with the new conditions.

The CONTEMPT-G code was used to predict the atmospheric air temperature inside the turbine and reactor buildings. It was retrieved from the CSD archive library reference number THSD0560. This version of the program can be used to analyze steam pipe breaks within an open containment building. The code is described in Reference 2.

Two computer runs were retrieved from the April 1972 calculation files. They calculated temperatures in the reactor building 30 ft from the leak and in the turbine building 20 ft from the leak. Peak atmospheric temperatures are recorded in Table 2, line 2 for these two cases. These examples were rerun using the same input data and the retrieved CONTEMPT-G code. Results are plotted for atmospheric air temperature in the reactor building in Figure 5 and

turbine building in Figure 6. Peak temperatures are listed in Table 2, line 3. The input data for these cases were the same as before, but the retrieved CONTEMPT-G code is a more recent version.

Calculated turbine building peak temperature was 469°F, 23°F higher at four minutes and the same distance than the recovered run and reactor building peak temperature was 372°F, 40°F higher than the recovered run. These temperature differences are due to using a more recent version of CONTEMPT-G since the data input was the same. It is not know specifically what the differences are between the two code versions. The code version used for the 1972 calculations no longer exists. However, the plots of temperature versus time for these two cases have the same characteristic shape as shown in Figures 3 and 4. The discrepancy in peak temperature was small and deemed acceptable, so this version of CONTEMPT-G was used for the remaining analysis.

Data for the steam blowdown rate which includes steam addition rate and energy added is very critical for this analysis. It directly affects the atmospheric air temperature peak and profile with respect to time. And yet available sources consisting of some recovered runs and Reference 1 differed about steam addition rates.

The steam blowdown curves shown in Reference 1 for the reactor building cold reheat steam line rupture (Figure 3.3) and the turbine building hot reheat steam line rupture (Figure 3.4) are reproduced here as Figures 7 and 8, respectively. Scaling values from these curves into tabular data for use in the CONTEMPT-G code yields the data shown in Table 3. This data was supposedly used for the four minute leak termination study in 1972. However, it is considerably different than the values recovered from the runs in April 1972 which are shown in Table 4.

In particular, the reactor building blowdown rate at 239.99 seconds is 193 lb/sec from the recovered run and 113 lb/sec from Reference 1. This difference yields atmospheric air temperature profiles that are quite different. Results for the reactor building cold reheat steam pipe break for a four minute leak termination at a distance of 20 ft from the source are shown using these

two rates. Steam blowdown rates from Reference 1 were used for Figure 9 yielding a peak temperature of 416°F. Data from recovered runs of April 1972 were used for Figure 10 yielding a peak temperature of 429°F. Comparing these two temperature profiles with the results reported in Figure 3, lead one to surmise that the blowdown data actually used in the previous analysis was the data from the recovered runs in April 1972 and not what is shown in Figure 7. Substantiating this is some data found in Reference 3, Figure 1, which is reproduced here as Figure 11. This shows steam flow through the rupture from the initial break to seven minutes later. Scaling data from this curve agrees with the values for the four minute leak from the recovered runs of April 1972 listed in Table 4. This data was chosen to be the best available blowdown data for the cold reheat steam pipe break. Flow was assumed to be a constant 70 lb/sec after 400 seconds for the ten minute and twenty minute peak cases. Table 5 shows this blowdown data tabulated.

The blowdown curves tabulated in Tables 3 and 4 for the turbine building hot reheat steam line rupture are different initially, but agree after about two seconds. Its initial difference does not affect the air temperatures because it occurs for such a short time period. Results are shown for the turbine building atmospheric air temperature using Reference 1 data in Figure 12 and recovered data in Figure 13 for a four minute leak and a distance of 20 ft from the source. Peak temperatures and curve shapes are the same. Another source to justify this data was found. Blowdown data from Reference 4, Figure 1 is reproduced here as Figure 14 which supports the recovered data of April 1972. Therefore, the data listed in Table 4 was chosen to be the best available blowdown data for the hot reheat steam pipe break with four minute termination. Flow was assumed to be a constant 625 lb/sec after four seconds for each case studied. Table 6 shows the blowdown data for hot reheat pipe rupture with ten minute and twenty minute leak termination.

Heat transfer surface areas of major components (heat sinks) in each building for various distances from the leak source were determined in the 1972 study. The derivation of these areas is described in Reference 1. This data was tabulated for the reactor building in Table 3.1 of Reference 1 and the turbine building in Table 3.2 of Reference 1 and are reproduced here as Tables 7

and 8, respectively. These values were used directly as input for the CONTEMPT-G code in this study.

Reactor building atmospheric temperatures were then predicted for ten minute and twenty minute leak terminations. Blowdown data was taken from Table 5 and the heat transfer surface areas of each heat sink were taken from Table 7 for a 20 ft distance from the postulated leak. These results were plotted along with the four minute termination results in Figure 1. Peak temperatures are listed in Table 2, item 4.

Turbine building atmospheric temperatures were also predicted for ten minute and twenty minute leaks. Table 6 blowdown data was used and Table 8 heat transfer surface areas of each heat sink for a distance of 20 ft from the leak were used. Results were plotted along with the four minute termination predictions in Figure 2. Peak temperatures are listed in Table 2, item 4.

Discussion

In the reactor building the predicted atmospheric air temperature for an accidental cold reheat steam pipe break 20 ft from the leak source is summarized in Figure 1. The postulated four minute leak appears to cause the air temperature to reach a maximum of 429°F at four minutes, then decay very rapidly for several seconds. Following this initially rapid decay period the decay rate slows. Initially the pipe rupture releases a jet of steam into the atmosphere which expands into a steam cloud. This event adds a lot of energy into the immediate area which causes the air temperature to rise rapidly. As the steam cloud expands, components in its path begin absorbing some energy. The energy of the steam released into the air reduces over time. This slows the rise in temperature. When the leak terminates suddenly the heat source is removed so temperatures drop quickly. They then decay towards an equilibrium as the released steam energy is dissipated. This curve characteristic shape agrees with the one predicted by the previous study shown in Figure 3. The peak temperature predicted now is 24°F higher than what was reported before. This is due to using a more recent version of the CONTEMPT-G code. It was assumed that this difference was acceptable.

The ten minute leak termination does not appear to cause the peak atmospheric air temperature to increase beyond the 429°F calculated for the four minute leak. It does, however, cause the temperatures to remain higher for a longer period of time than the shorter leak. After four minutes the air temperature drops to a minimum at about seven minutes, then rises a few degrees until the leak terminates at ten minutes. After ten minutes temperatures drop sharply then decay as before. Between four and seven minutes after the rupture, the rate of steam energy added to the environment is decreasing. Components in the reactor building are absorbing energy at a slower rate during this time because there is less heat to transfer from the steam. Air temperatures begin to drop. The heat transfer coefficient, thermal conductivity, volumetric heat capacity, and surface areas of each component modeled are considered when analyzing temperatures. Some components begin to loose heat during this time such as the steel decking, ducting, electrical conduits, and cable trays. These items tend to follow the air temperature more closely than the others. Between about seven and ten minutes the rate of steam energy release is constant. This heat addition rate is more than the components and environment can absorb for a steady state so temperatures begin rising again. They rise until the steam leak terminates. At this time air temperature has peaked at about 409°F. Terminating the leak source causes the air temperature to drop rapidly and decay towards a new steady state.

The twenty minute leak termination air temperature curve follows the same predicted curve as the ten minute leak except that after ten minutes it continues to rise to a peak air temperature of 442°F at twenty minutes. It then drops rapidly and decays towards a new equilibrium. Once again the constant steam energy added between ten and twenty minutes is transferred to the air and components causing their temperatures to rise.

It was assumed that there was an adequate cold reheat steam supply to maintain the constant steam leak energy addition rate of 70 lb/sec, 1359 Btu/lb from 400 seconds until either the ten or twenty minute termination. An estimate of the total volume of water/steam required to sustain these leaks was based on the flow rates in Table 5. The volume of water released from the pipe for a twenty minute leak is approximately 144,000 lb. There appears to be more than sufficient water available to supply this leak.

In the turbine building the predicted atmospheric air temperature for an accidental hot reheat steam pipe rupture 20 ft from the leak source is summarized in Figure 2. The postulated four minute leak appears to cause the air temperature to reach a maximum of 506°F at four minutes then decay very rapidly for several seconds. The decay rate then slows as the system moves towards a steady state. Initially the hot reheat steam pipe releases a jet of energy into the atmosphere producing a steam cloud. This cloud expands into the turbine building atmosphere. The steam energy released decreases quickly for four seconds then remains constant for the duration of the leak. This constant heat source causes temperatures to increase inside the building until the leak is terminated. The sudden termination of steam energy addition causes the air temperature to drop quickly then decay to an equilibrium. This curve characteristic shape agrees with the one predicted by the previous study shown in Figure 4. The peak temperature predicted now is 26°F higher than what was reported before. This is due to using a more recent version of CONTEMPT-G. It was assumed that this difference was acceptable and longer leak termination times analyzed.

The ten and twenty minute steam leak terminations appear to cause air temperatures to continue rising until the leaks are stopped. This is because there is more energy added to the environment by the released steam than can be absorbed for a steady state, so temperatures rise. The predicted peak atmospheric air temperature for a ten minute leak was 571°F and for a twenty minute leak was 612°F. These higher temperatures decay at the same rate as the four minute leak. Temperatures also remain higher for the increased steam leak terminations for a longer period of time.

It was assumed that there was adequate hot reheat steam to maintain a steam leak for ten and twenty minutes. A constant energy source of 625 lb/sec, 1360 Btu/lb from four seconds until leak termination at ten or twenty minutes is a considerable amount of water. An estimate of the total volume of water/steam required to sustain these leaks was based on the flow rates in Table 6. The volume of water released from the pipe for a twenty minute leak is approximately 754,000 lb. There appears to be more than enough water available to supply this leak.

Conclusion

The CONTEMPT-G predicted peak air temperature of 429°F in the reactor building appears unchanged for a four minute and ten minute cold reheat steam pipe rupture 20 ft from the source. Thus extending the leak termination time from four to ten minutes only increases the length of time components will be exposed to higher temperatures. A twenty minute leak would produce a peak temperature of 442°F, only slightly higher than the shorter leaks.

The predicted peak air temperature in the turbine building appears to be 506°F for a four minute hot reheat steam pipe rupture 20 ft from the source. The ten and twenty minute leaks appear to produce air temperatures of 571°F and 612°F, respectively. These predicted temperatures are significantly higher than the peak for a four minute leak. They also maintain temperatures higher for a longer period of time.

It appears from a rough estimation that there is more than enough steam supply for the long leak terminations of ten and twenty minutes. The blowdown rate from the cold reheat pipe rupture in the reactor building is much smaller than the hot reheat pipe rupture in the turbine building. Thus the volume of water required to supply the cold reheat pipe leak is less than the hot reheat pipe leak.

References

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2. Houghton, W. J. and C. L. Turner, "Operation of the CONTEMPT-G Containment Analysis Program on the UNIVAC 1108 Computer," GA Technologies Report GA-A12692 Supplement A (GA-LTR-6), February 8, 1974.

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4. Potter, R. C. to H. F. Menzel, "PSC Predicted Leakage to Turbine Building from a Hot Reheat Line Rupture," GA Technologies Internal Memo Number RCP:PSC:SE311:72, March 6, 1972.

5. Joseph, W. M., "Interactive Plotting Routine, SUPER*PLOT," GA Technologies Internal Memo Number SAB:025:WMJ:81, March 2, 1981.

CR:MAB

SBI

cc: S. B. Inamati
A. S. Shenoy
F. A. Silady

TABLE 1
PEAK ATMOSPHERIC AIR TEMPERATURES
20 FEET FROM STEAM LEAK SOURCE

BUILDING/REHEAT STEAM LINE	STEAM LEAK TERMINATION (MINUTES)	PEAK ATMOSPHERIC TEMPERATURE (°F)
reactor/cold	4	429
reactor/cold	10	429
reactor/cold	20	442
turbine/hot	4	506
turbine/hot	10	571
turbine/hot	20	612

TABLE 2

PEAK ATMOSPHERIC TEMPERATURES (°F)[1]

	20 Feet From Source						30 Feet From Sour
	Reactor Building			Turbine Building			Reactor Buildin
	4 min	10 min	20 min	4 min	10 min	20 min	4 min
1) Reference 1 results (Figure 3.11 and Figure 3.12)	405	---	---	480	---	---	333 @ 1.1 326
2) Recovered runs April 1972	---	---	---	446	---	---	332 @ 2.8 327
3) Steam blowdown and heat transfer surface area data from recovered runs	---	---	---	469[2]	---	---	372 @ 2.5 365
4) Steam blowdown from recovered runs, heat transfer surface area data from Reference 1	429	429 @ 3.8 409	442	506	571	612	same as case 3
5) Steam blowdown and heat transfer surface area data from Reference 1	416 @ 2.2 392			507			

Notes:

(1) peak atmospheric temperature at cut-off time unless otherwise noted.

(2) heat transfer areas not as reported in Reference 1

TABLE 3
BLOWDOWN DATA FOR 4 MINUTE LEAK FROM REFERENCE 1

TIME		STEAM ADDITION RATE		ENERGY (ENTHALPY) ADDITION RATE	
hour	second	lb/hr	lb/sec	Btu/hr	Btu/lb
Reactor building, cold reheat pipe					
0.	0.	9.216 E+6	2560	1.252 E+10	1359
1.389 E-4	0.5	4.896 E+6	1360	6.654 E+9	
2.083 E-4	0.75	3.492 E+6	970	4.746 E+9	
2.778 E-4	1.0	2.412 E+6	670	3.278 E+9	
6.944 E-4	2.5	1.465 E+6	407	1.991 E+9	
1.389 E-3	5.0	1.260 E+6	350	1.712 E+9	
2.778 E-3	10.0	1.152 E+6	320	1.566 E+9	
2.778 E-2	100.	1.080 E+6	300	1.468 E+9	
6.666 E-2	239.99	4.068 E+5	113	5.528 E+8	
6.667 E-2	240.0	0.	0	0.	
0.51	1836.0	0.	0	0.	
Turbine building, hot reheat pipe					
0.	0.	1.746 E+7	4850	2.654 E+10	1520
1.389 E-4	0.5	1.004 E+7	2790	1.506 E+10	1500
2.778 E-4	1.0	6.516 E+6	1810	9.644 E+9	1480
4.167 E-4	1.5	4.356 E+6	1210	6.360 E+9	1460
5.556 E-4	2.0	3.391 E+6	942	4.883 E+9	1440
8.333 E-4	3.0	2.534 E+6	704	3.548 E+9	1400
1.111 E-3	4.0	2.257 E+6	627	3.070 E+9	1360
6.666 E-2	239.99	2.257 E+6	627	3.070 E+9	1360
6.667 E-2	240.0	0.	0	0.	0.
0.51	1836.0	0.	0	0.	0.

TABLE 4

BLOWDOWN DATA FOR 4 MINUTE LEAK FROM RECOVERED RUNS OF APRIL 1972

TIME		STEAM ADDITION RATE		ENERGY (ENTHALPY) ADDITION RATE	
hour	second	lb/hr	lb/sec	Btu/hr	Btu/lb
Reactor building, cold reheat pipe					
0.	0.	9.504 E+6	2640	1.291 E+10	1359
1.3888 E-4	0.5	4.752 E+6	1320	6.457 E+9	
2.0833 E-4	0.75	3.24 E+6	900	4.402 E+9	
2.7777 E-4	1.0	2.304 E+6	640	3.13 E+9	
6.9444 E-4	2.5	1.44 E+6	400	1.957 E+9	
1.3888 E-3	5.0	1.26 E+6	350	1.712 E+9	
2.7777 E-3	10.0	1.152 E+6	320	1.565 E+9	
2.7777 E-2	100.	1.08 E+6	300	1.467 E+9	
6.6666 E-2	239.99	6.936 E+5	193	9.424 E+8	
6.67 E-2	240.0	0.	0	0.	
0.51	1836.0	0.	0	0.	
Turbine building, hot reheat pipe					
0.	0.	1.854 E+7	5150	2.8181 E+10	1520
1.388 E-4	0.5	1.08 E+7	3000	1.62 E+10	1500
2.777 E-4	1.0	6.66 E+6	1850	9.857 E+9	1480
4.166 E-4	1.5	4.5 E+6	1250	6.57 E+9	1460
5.555 E-4	2.0	3.42 E+6	950	4.925 E+9	1440
8.333 E-4	3.0	2.52 E+6	700	3.528 E+9	1400
1.111 E-3	4.0	2.25 E+6	625	3.06 E+9	1360
6.666 E-2	239.99	2.25 E+6	625	3.06 E+9	1360
6.67 E-2	240.0	0.	0	0.	0
0.51	1836.0	0.	0	0.	0

TABLE 5

BLOWDOWN DATA FOR COLD REHEAT PIPE RUPTURE IN REACTOR BUILDING WITH 10 MINUTE AND 20 MINUTE TERMINATION

TIME		STEAM ADDITION RATE		ENERGY ADDITION	
hour	second	lb/hr	lb/sec	Btu/hr	Btu/lb
1) 10 minute steam termination					
0.	0.	9.504 E+6	2640	1.291 E+10	1359
1.3888 E-4	0.5	4.752 E+6	1320	6.457 E+9	
2.0833 E-4	0.75	3.24 E+6	900	4.402 E+9	
2.7777 E-4	1.0	2.304 E+6	640	3.13 E+9	
6.9444 E-4	2.5	1.44 E+6	400	1.957 E+9	
1.3888 E-3	5.0	1.26 E+6	350	1.712 E+9	
2.7777 E-3	10.0	1.152 E+6	320	1.565 E+9	
2.7777 E-2	100.	1.08 E+6	300	1.467 E+9	
1.111 E-1	400.	2.52 E+5	70	3.425 E+8	
1.666 E-1	599.99	2.52 E+5	70	3.425 E+8	
1.667 E-1	600.	0.	0	0.	
0.51	1836.	0.	0	0.	
2) 20 minute steam termination					
0.	0.	9.504 E+6	2640	1.291 E+10	1359
1.3888 E-4	0.5	4.752 E+6	1320	6.457 E+9	
2.0833 E-4	0.75	3.24 E+6	900	4.402 E+9	
2.7777 E-4	1.0	2.304 E+6	640	3.13 E+9	
6.9444 E-4	2.5	1.44 E+6	400	1.957 E+9	
1.3888 E-3	5.0	1.26 E+6	350	1.712 E+9	
2.7777 E-3	10.0	1.152 E+6	320	1.565 E+9	
2.7777 E-2	100.	1.08 E+6	300	1.467 E+9	
1.111 E-1	400.	2.52 E+5	70	3.425 E+8	
0.33332	1199.99	2.52 E+5	70	3.425 E+8	
0.33333	1200.	0.	0	0.	
0.51	1836.	0.	0	0.	

TABLE 6

BLOWDOWN DATA FOR HOT REHEAT PIPE RUPTURE IN TURBINE BUILDING
WITH 10 MINUTE AND 20 MINUTE TERMINATION

TIME		STEAM ADDITION RATE		ENERGY ADDITION	
hour	second	lb/hr	lb/sec	Btu/hr	Btu/lb
1) 10 minute steam termination					
0.	0.	1.854 E+7	5150	2.8181 E+10	1520
1.388 E-4	0.5	1.08 E+7	3000	1.62 E+10	1500
2.777 E-4	1.0	6.66 E+6	1850	9.857 E+9	1480
4.166 E-4	1.5	4.5 E+6	1250	6.57 E+9	1460
5.555 E-4	2.0	3.42 E+6	950	4.925 E+9	1440
8.333 E-4	3.0	2.52 E+6	700	3.528 E+9	1400
1.111 E-3	4.0	2.25 E+6	625	3.06 E+9	1360
1.666 E-1	599.99	2.25 E+6	625	3.06 E+9	1360
1.667 E-1	600.0	0.	0	0.	0
0.51	1836.0	0.	0	0.	0
2) 20 minute steam termination					
0.	0.	1.854 E+7	5150	2.8181 E+10	1520
1.388 E-4	0.5	1.08 E+7	3000	1.62 E+10	1500
2.777 E-4	1.0	6.66 E+6	1850	9.857 E+9	1480
4.166 E-4	1.5	4.5 E+6	1250	6.57 E+9	1460
5.555 E-4	2.0	3.42 E+6	950	4.925 E+9	1440
8.333 E-4	3.0	2.52 E+6	700	3.528 E+9	1400
1.111 E-3	4.0	2.25 E+6	625	3.06 E+9	1360
0.33332	1199.99	2.25 E+6	625	3.06 E+9	1360
0.33333	1200.	0.	0	0.	0
0.51	1836.	0.	0	0.	0

TABLE 7

REACTOR BUILDING CONTEMPT-G CODE HEAT SINKS

Heat Transfer Surface Areas (ft²) are shown as function of distance from postulated steam leak

No.	Heat Sink	Material	Around PCRV(1)	36.2 ft(2)	30 ft	25 ft	20 ft	15 ft	10.9 f
1	Concrete walls, floor, PCRV	Concrete	42,230	17,380	11,960	8,300	5,310	2,990	1,58
2	PCRV support ring	Concrete	9,870	9,870	6,790	4,710	3,020	1,700	90
3	Partition walls & floors	Concrete	9,710	6,600	4,540	3,150	2,020	1,130	60
4	Thin steel wall (4)	Steel	16,090	3,760	2,590	1,800	1,150	650	34
5	Composite steel wall	Steel	19,810	-(3)	---	---	---	---	--
6	Steel decking	Steel	43,140	19,440	13,370	9,290	5,940	3,340	1,77
7	Structural steel, equipment	Steel	28,800	12,320	8,470	5,890	3,770	2,120	1,12
8	Ducting, conduits, trays	Steel	63,700	34,180	23,510	16,330	10,450	5,880	3,11
9	Piping (4)	Steel	49,200	34,440	23,690	16,450	10,530	5,920	3,13
	Total Heat Transfer Surface Area (ft²)		282,550	137,990	94,240	65,920	42,190	23,730	12,55
	Total Volume (ft³)		534,730	198,220	113,100	65,450	33,510	14,140	5,44
	Heat Transfer Coefficient (Btu/hr-ft²-°F)		5.0	9.3	13.2	18.6	28.1	48.0	86.

(1) Section includes the total volume of reactor building below operating floor but without process area.

(2) This is the maximum radius (regional boundary) considered in the new analysis. Section includes the volume below PCRV which is inside and outside support ring, see Figures 3.6 and 3.7 of Reference 1.

(3) The composite steel wall is an outside heat transfer surface which does not extend below EL. 4790'-0".

(4) Does not include insulated piping.

TABLE 8

TURBINE BUILDING CONTEMPT-G CODE HEAT SINKS

Heat Transfer Surface Areas (ft^2) are shown as a function of distance from postulated steam leak

No.	Heat Sink	Material	70.2 ft[1]	60 ft	50 ft	40 ft	30 ft	20 f
1	Concrete floor	Concrete	30,400	24,640	19,320	14,350	9,780	5,69
2	Concrete Structures	Concrete	5,810	4,710	3,600	2,740	1,870	1,09
3	Concrete partition walls & floors	Concrete	44,600	36,140	28,350	21,050	14,340	8,35
4	Piping [2]	Steel	80,540	65,270	51,190	38,010	25,900	15,09
5	Composite steel wall	Steel	7,930	6,430	5,040	3,740	2,550	1,49
6	Steel decking	Steel	12,300	9,970	7,820	5,810	3,960	2,30
7	Structural steel, equipment	Steel	62,710	50,820	39,850	29,600	20,170	11,75
8	Conduits & cable trays	Steel	52,700	42,710	33,490	24,870	16,950	9,87
Total Heat Transfer Surface Area (ft^2)			296,990	240,690	188,750	140,170	95,520	55,63
Total Volume (ft^3)			750,000	547,200	380,000	243,200	136,800	60,80
Heat Transfer Coefficient (Btu/hr-ft^2-°F)			5.0	6.1	7.6	10.1	14.4	23.7

(1) This is the maximum radius (regional boundary) considered in the analysis, see Figures 3.8 and 3.9 of Reference 1.

(2) Does not include insulated piping.

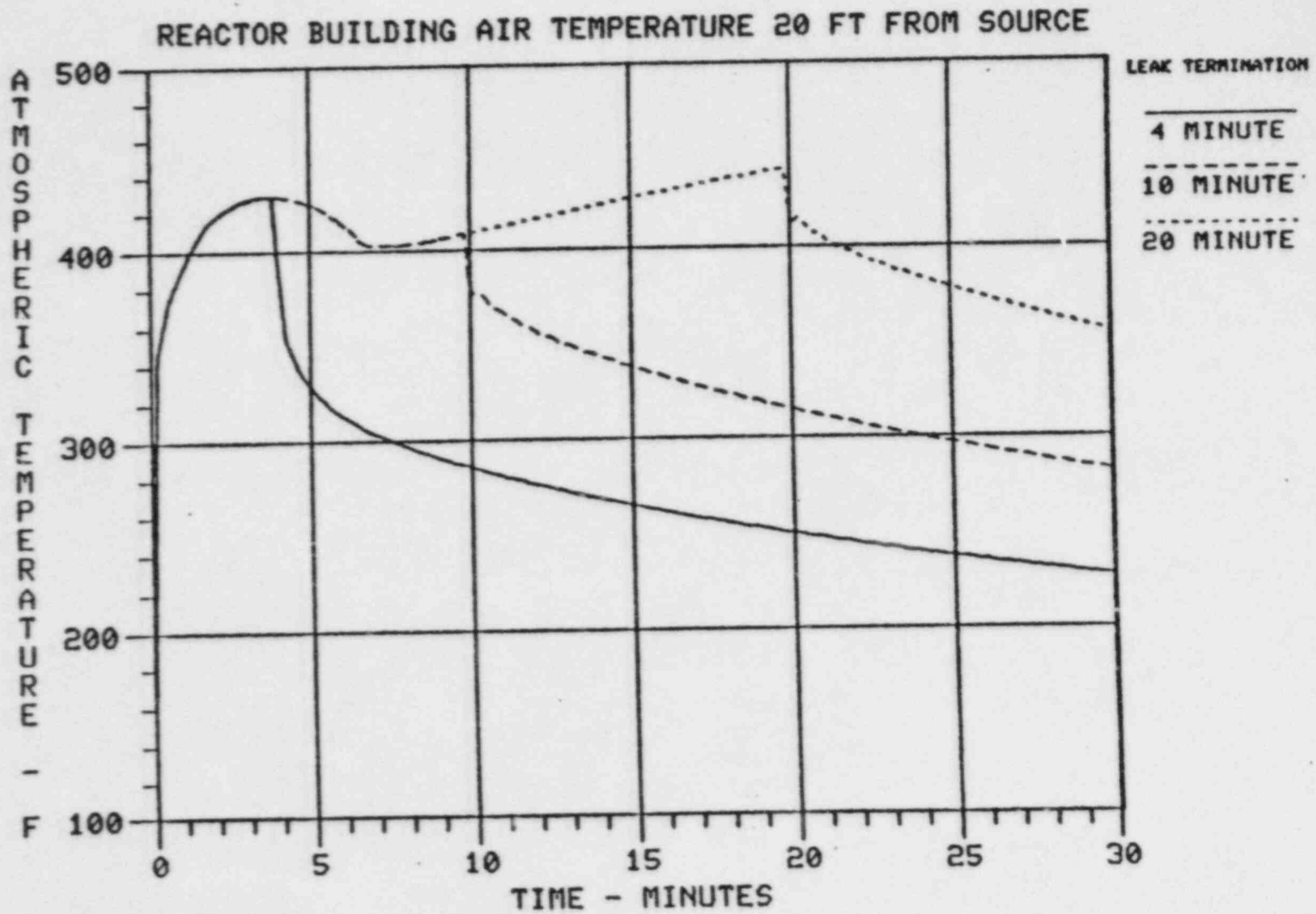

Fig. 1. Reactor building atmospheric temperature at a distance of 20 ft from leak source

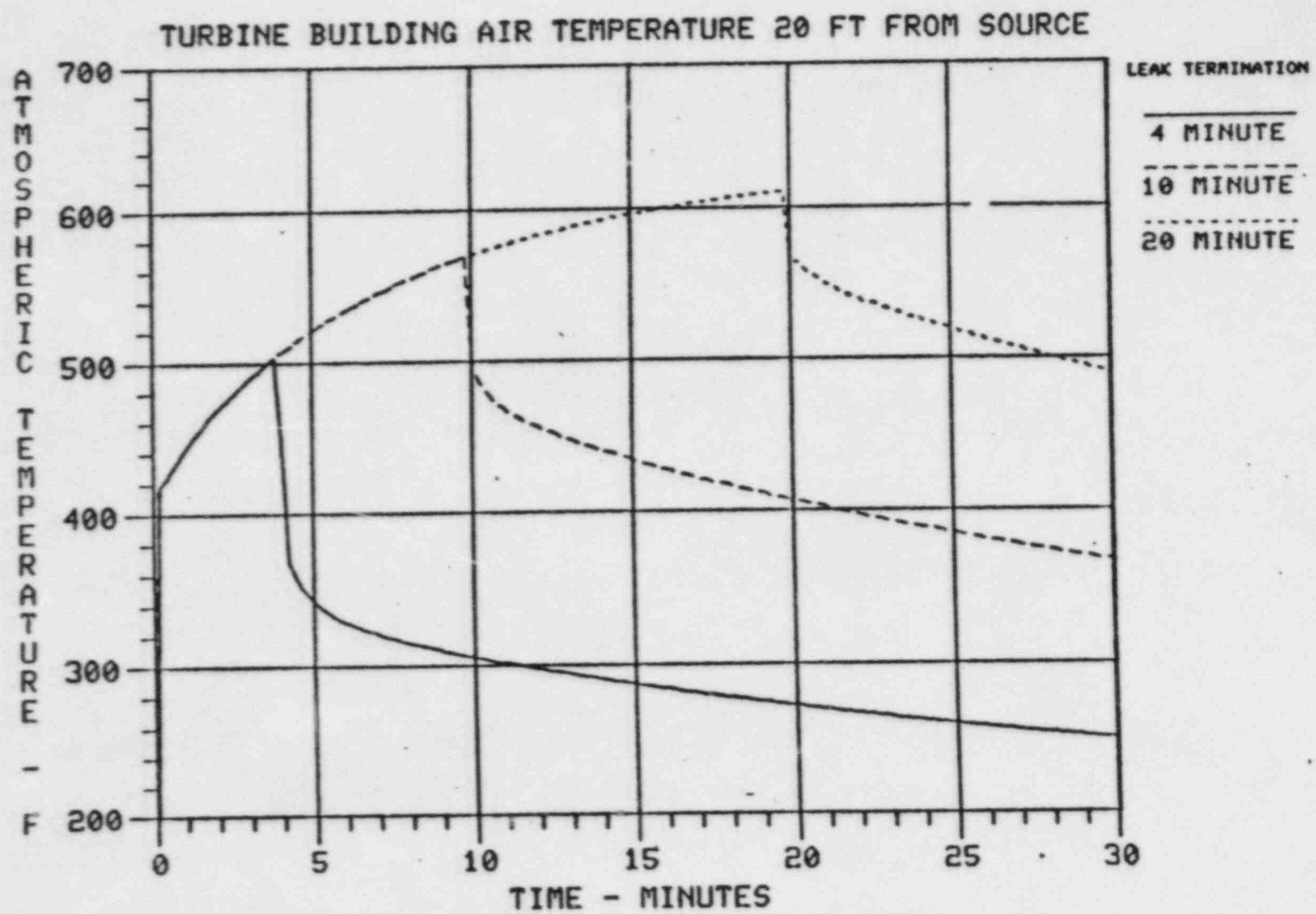

Fig. 2. Turbine building atmospheric temperature at a distance of 20 ft from leak source

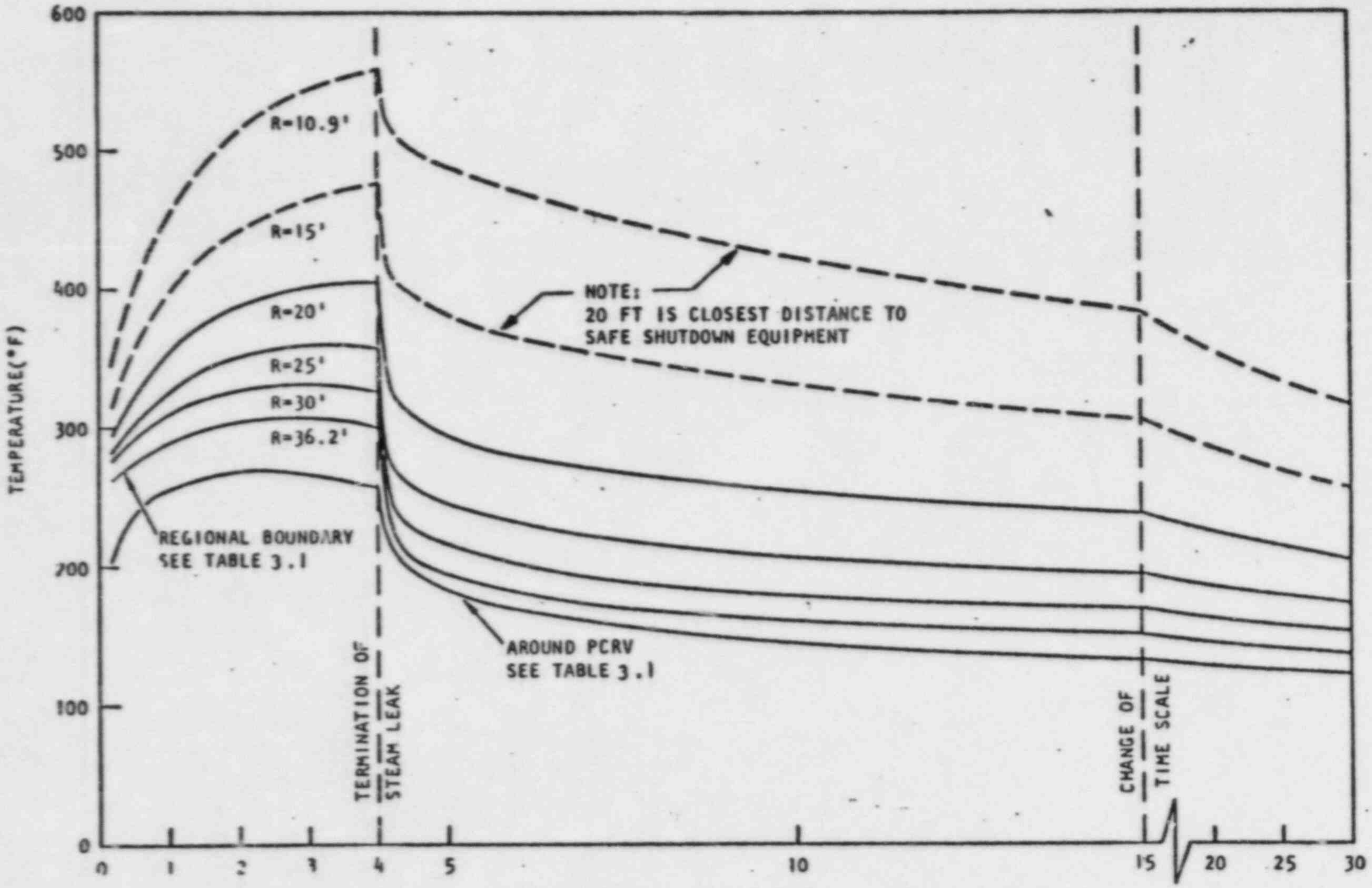

Fig. 3. Temperature of reactor building atmosphere during postulated cold reheat steam pipe rupture accident as a function of distance "R" from possible source of steam leak (from Ref. 1).

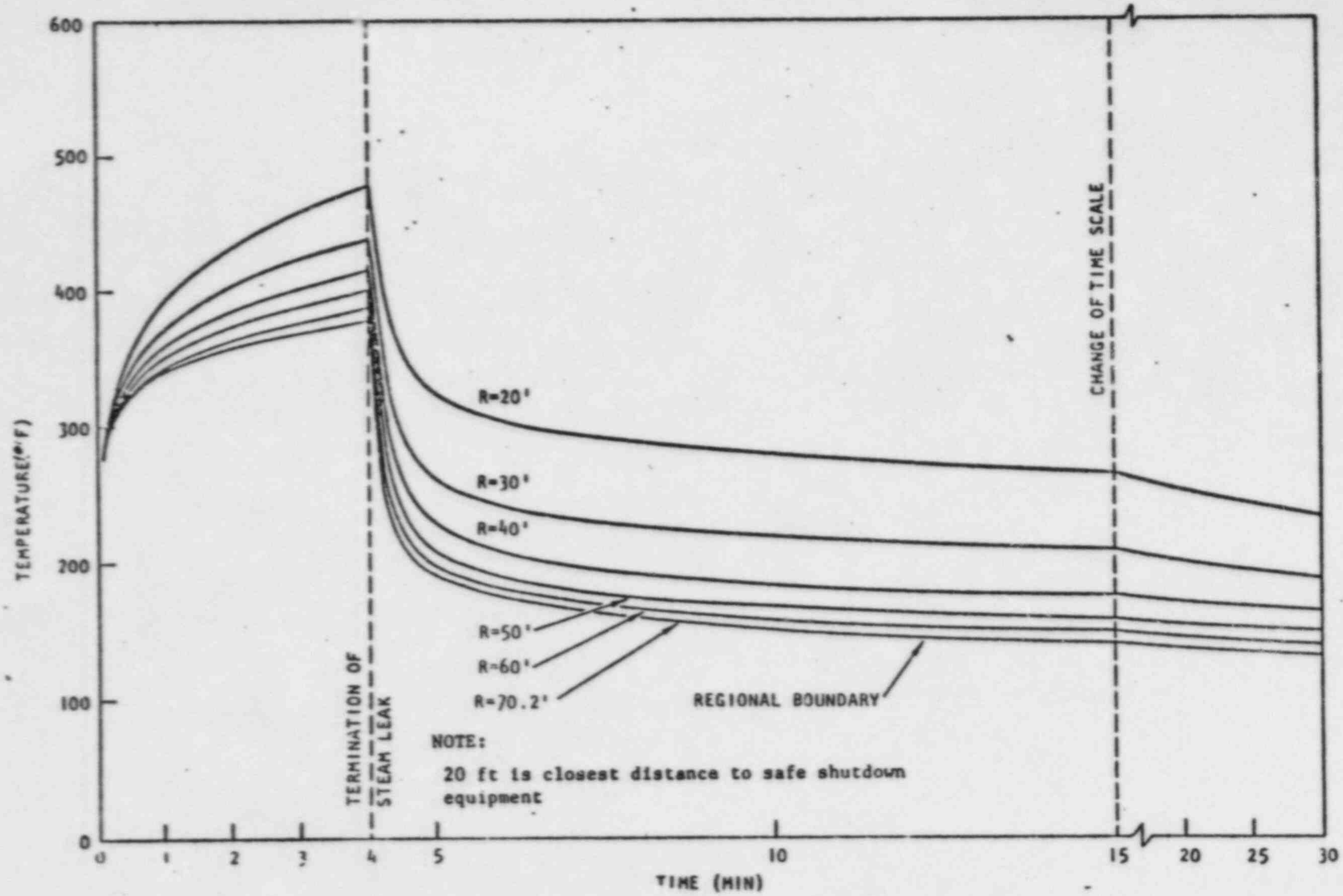

Fig. 4. Temperature of turbine building atmosphere during postulated hot reheat steam pipe rupture accident as a function of distance "R" from possible source of steam leak; no reactor scram until leak termination (from Ref. 1).

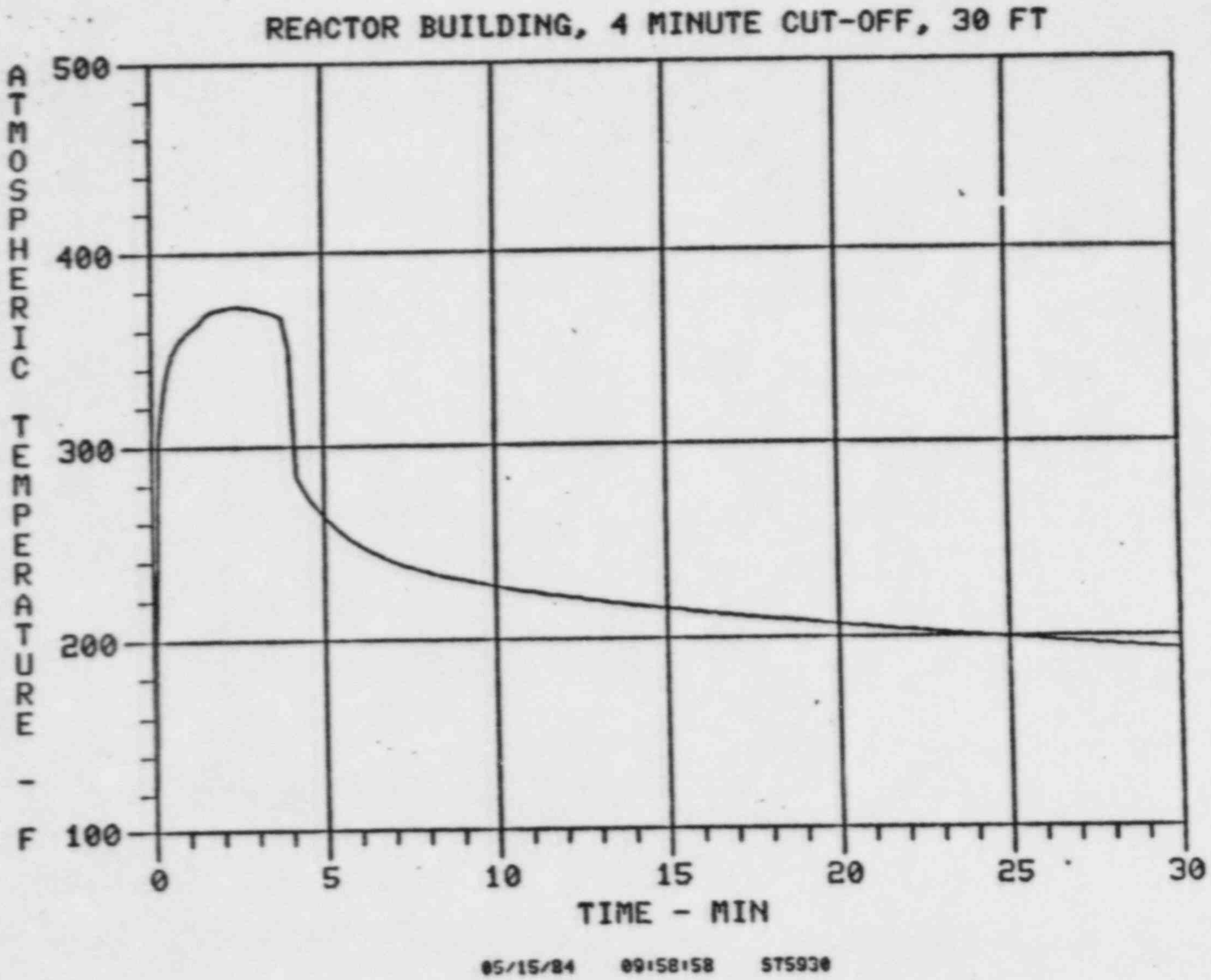

Fig. 5. Reactor building atmospheric temperature at a distance of 20 ft from leak source using data from recovered run.

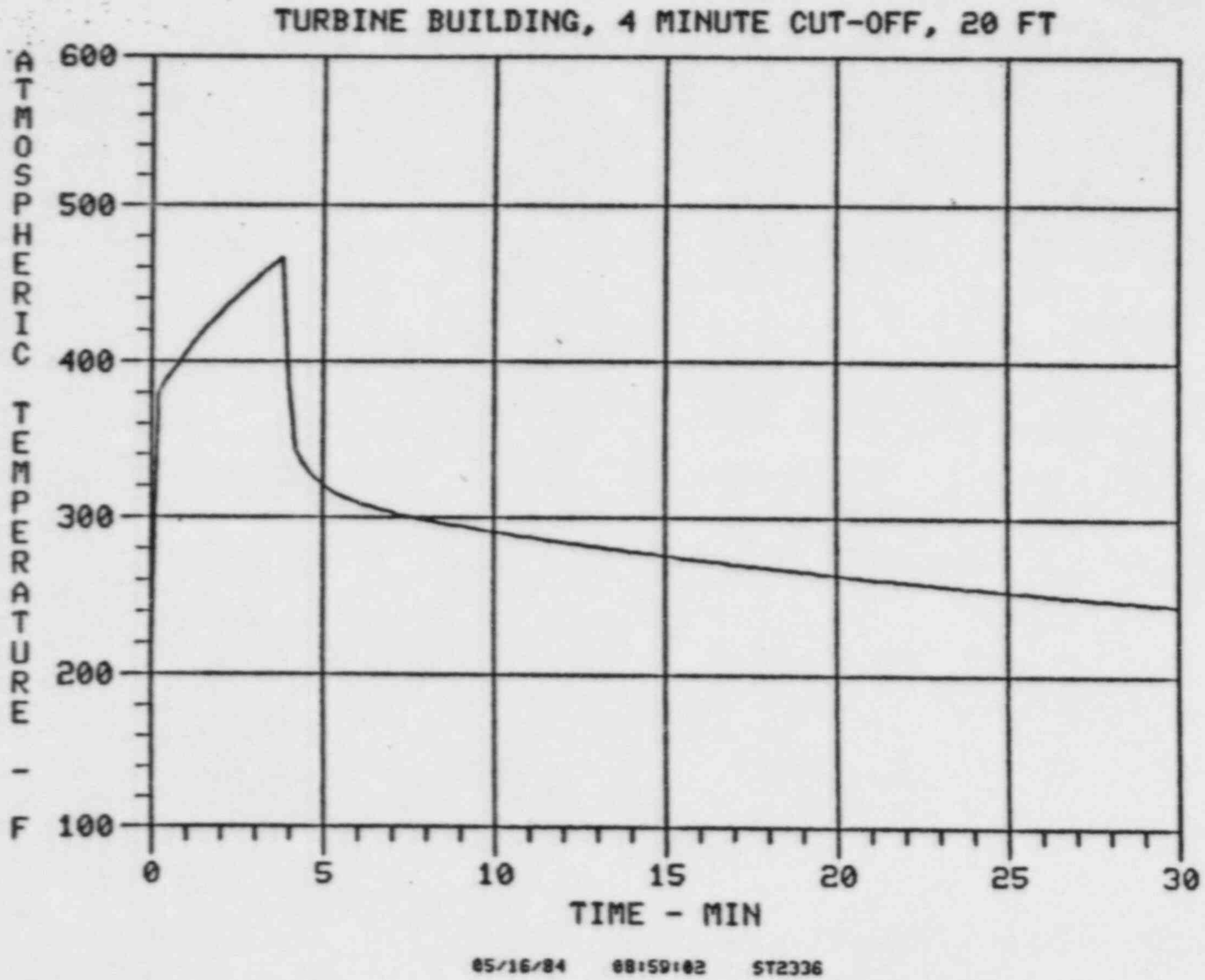

Fig. 6. Turbine building atmospheric temperature at a distance of 30 ft from leak source using data from recovered run.

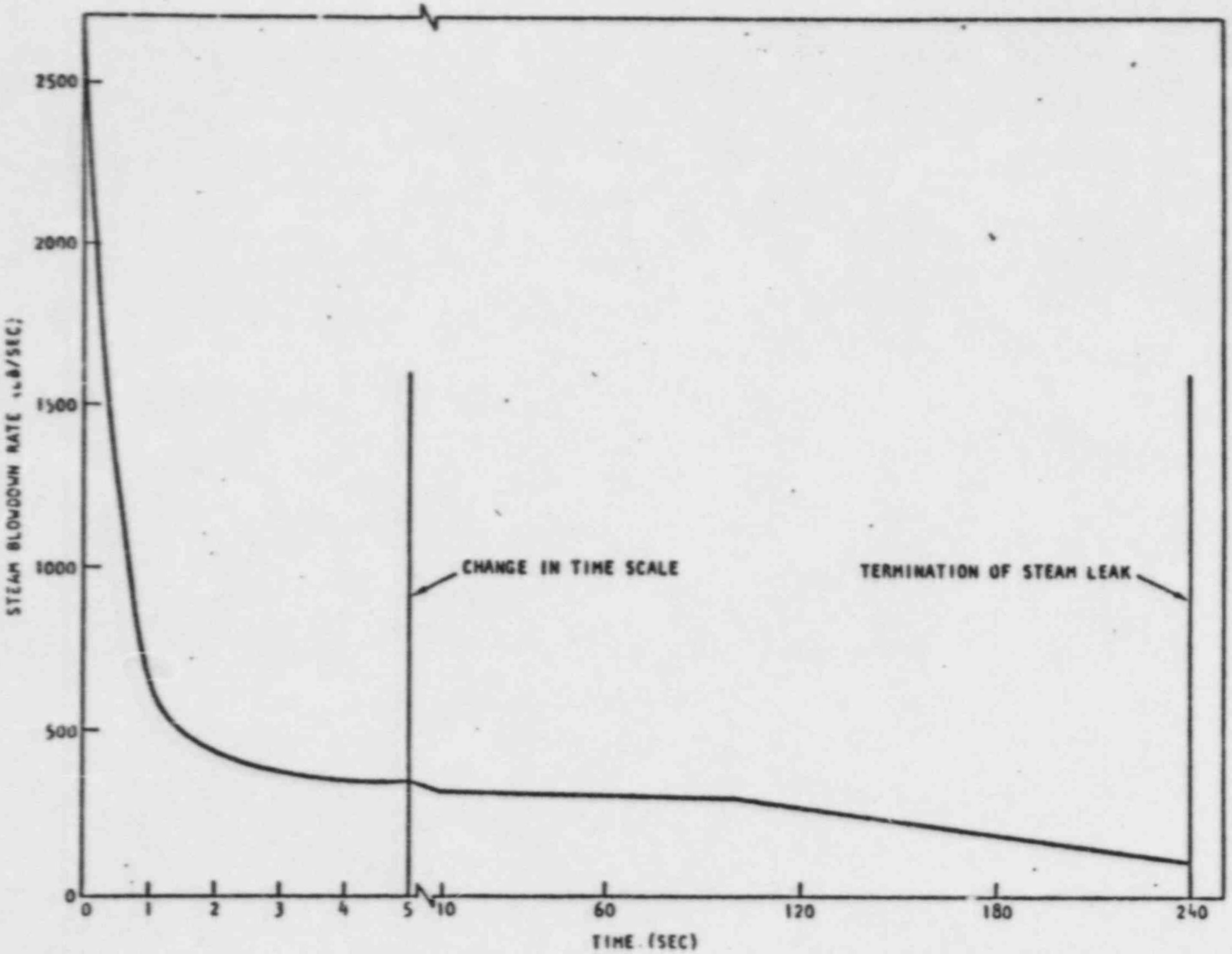

Fig. 7. Steam blowdown rate during postulated cold reheat pipe rupture in reactor building, enthalpy constant at 1359 Btu/lb (from Ref. 1).

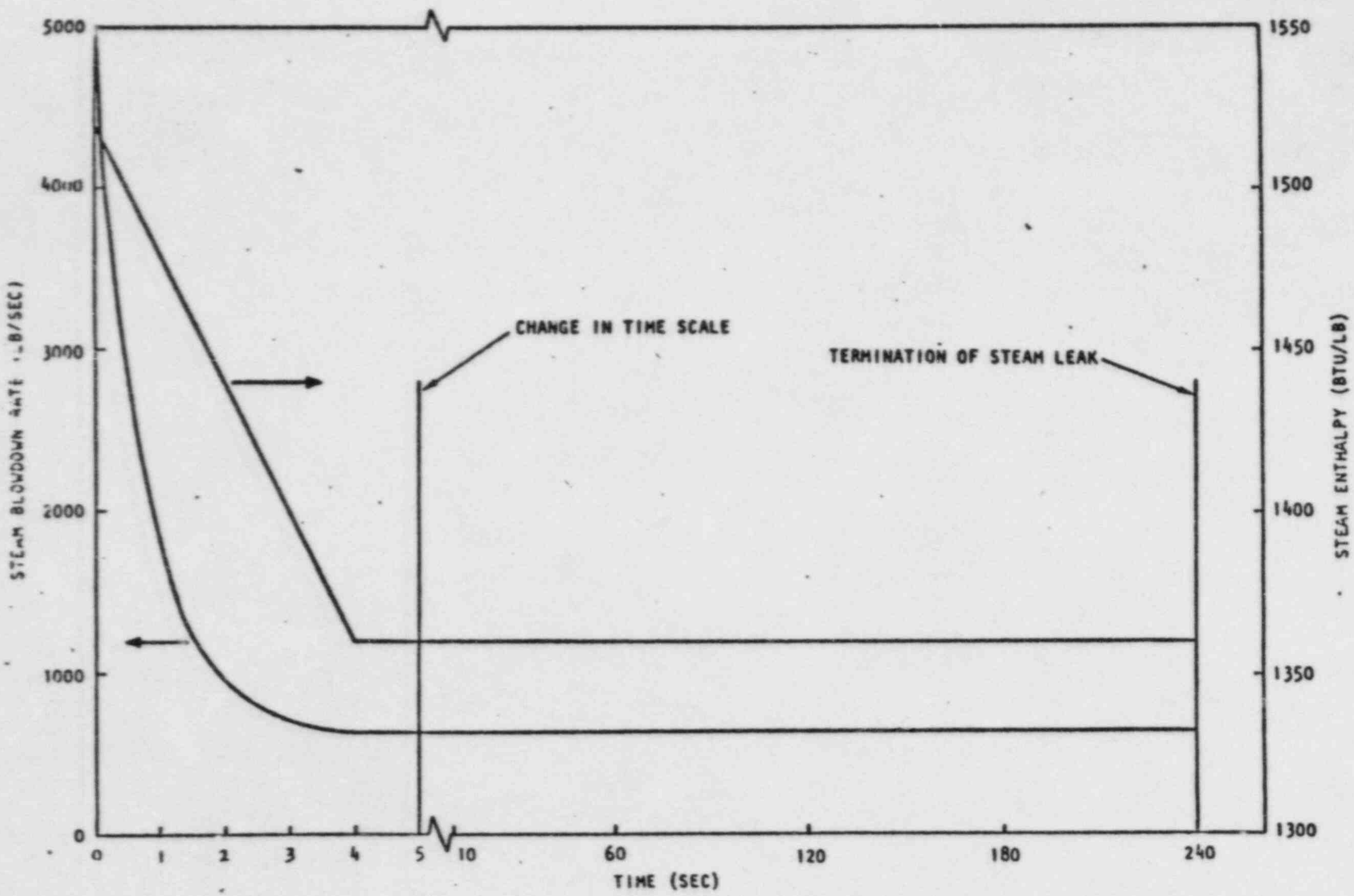

Fig. 8. Steam blowdown rate and enthalpy input during postulated hot reheat pipe rupture in turbine building without scram action (from Ref. 1).

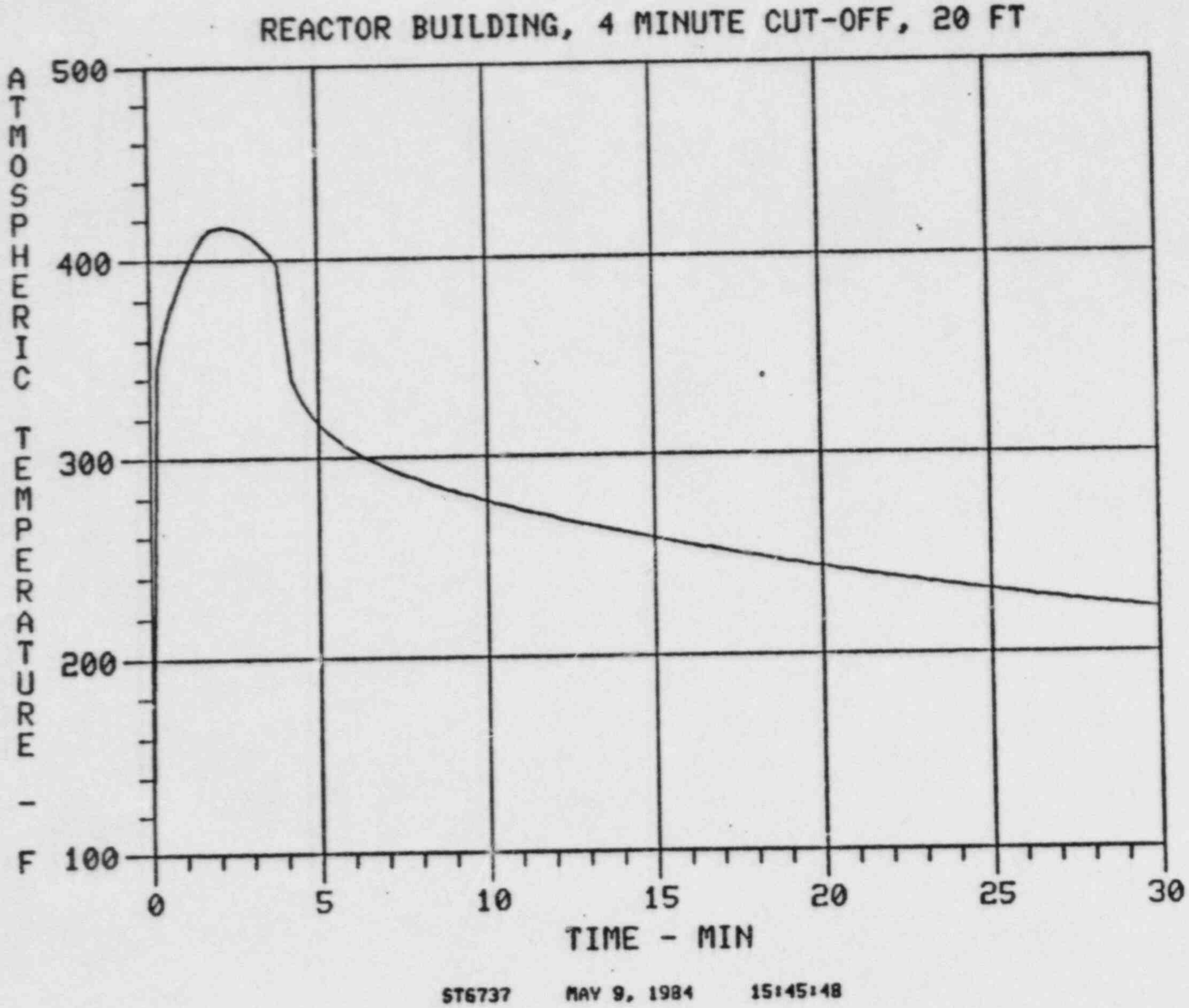

Fig. 9. Reactor building atmospheric temperature at a distance of 20 ft from steam leak using blowdown data from Ref. 1.

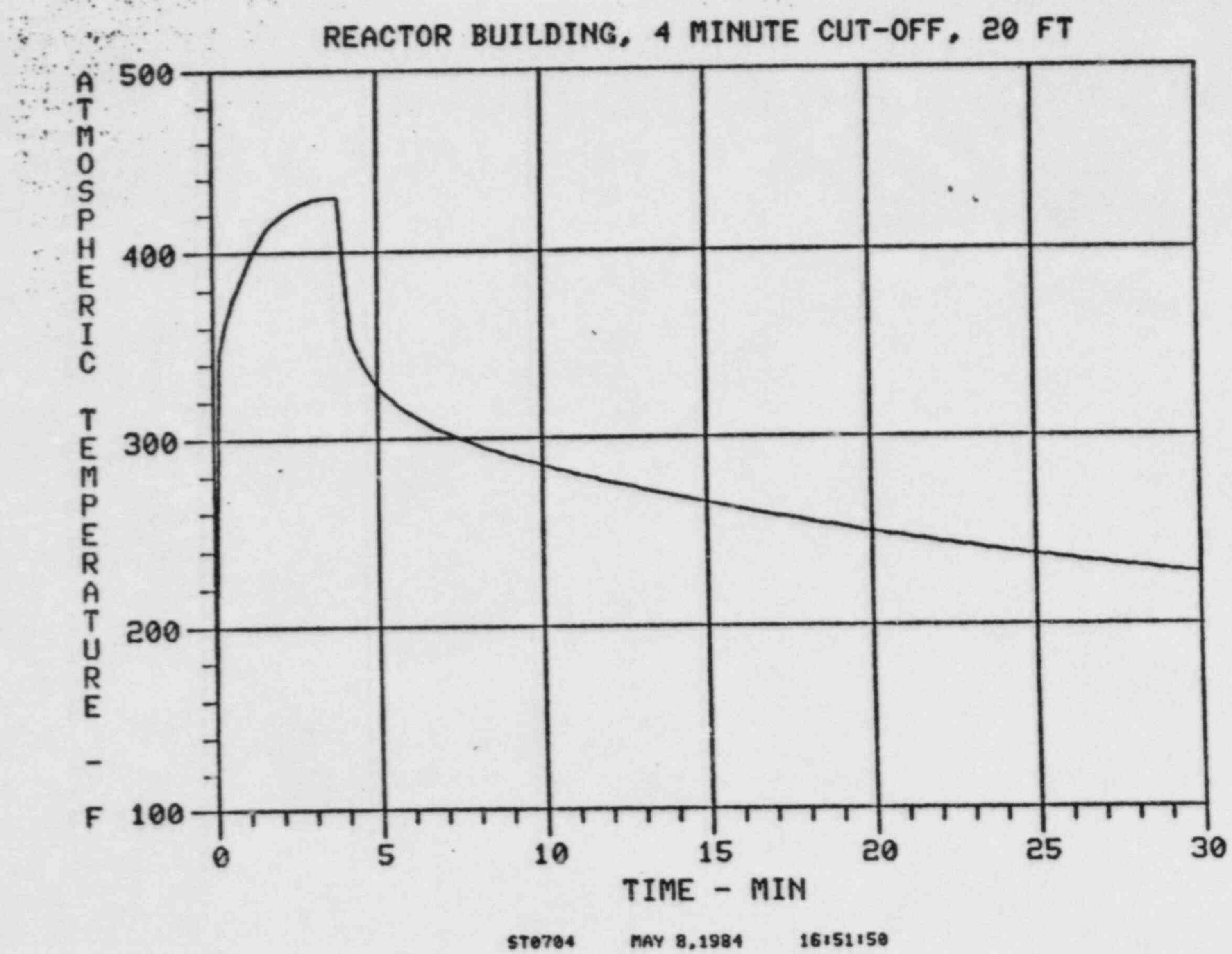

Fig. 10. Reactor building atmospheric temperature at a distance of 20 ft from steam leak using blowdown data from recovered runs of April 1972.

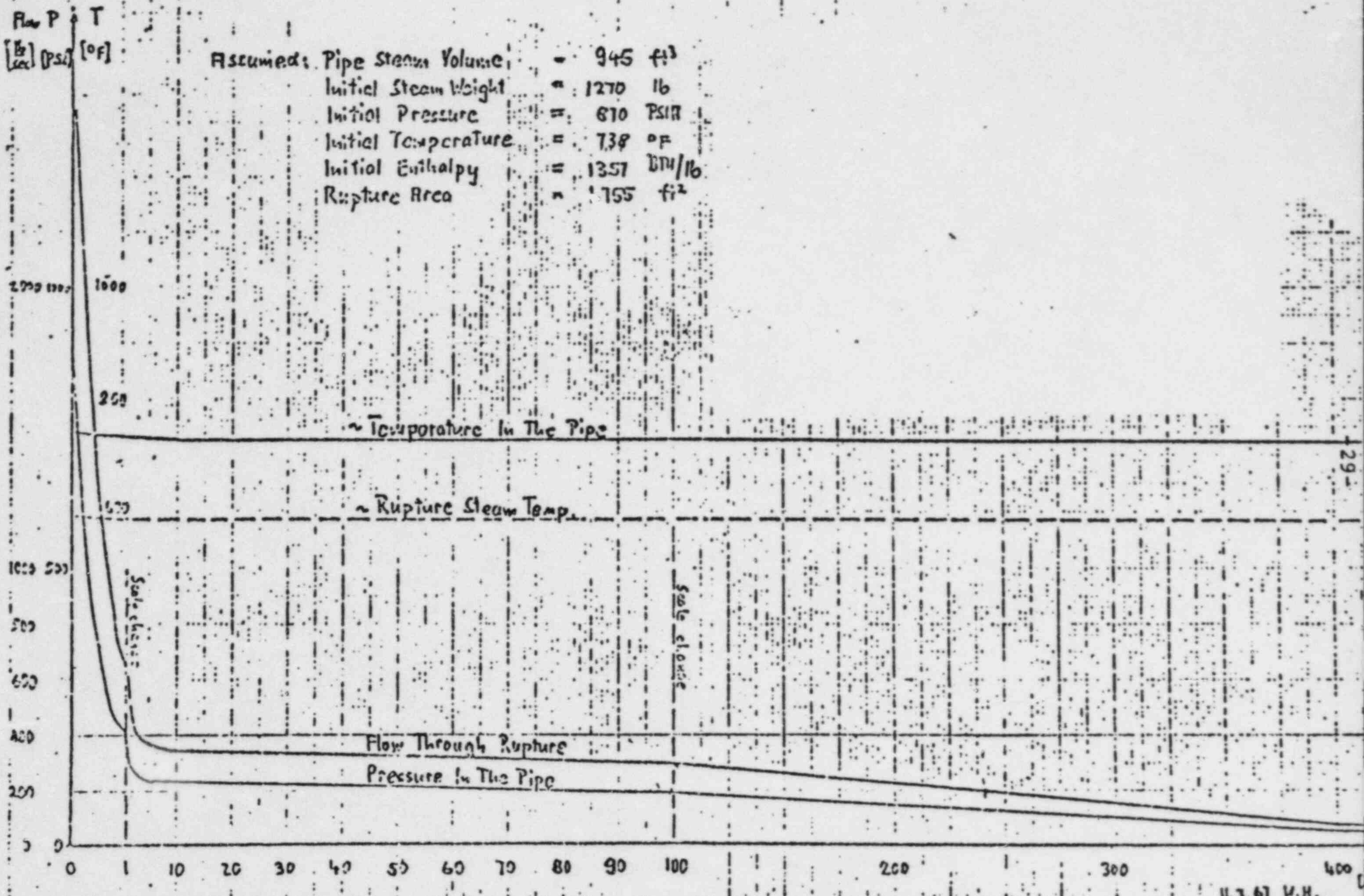

Fig. 11. Complete off-set rupture in cold reheat steam pipe in reactor building, 100% load, flow in the pipe between rupture and HP-turbine exhaust line (flash tank).

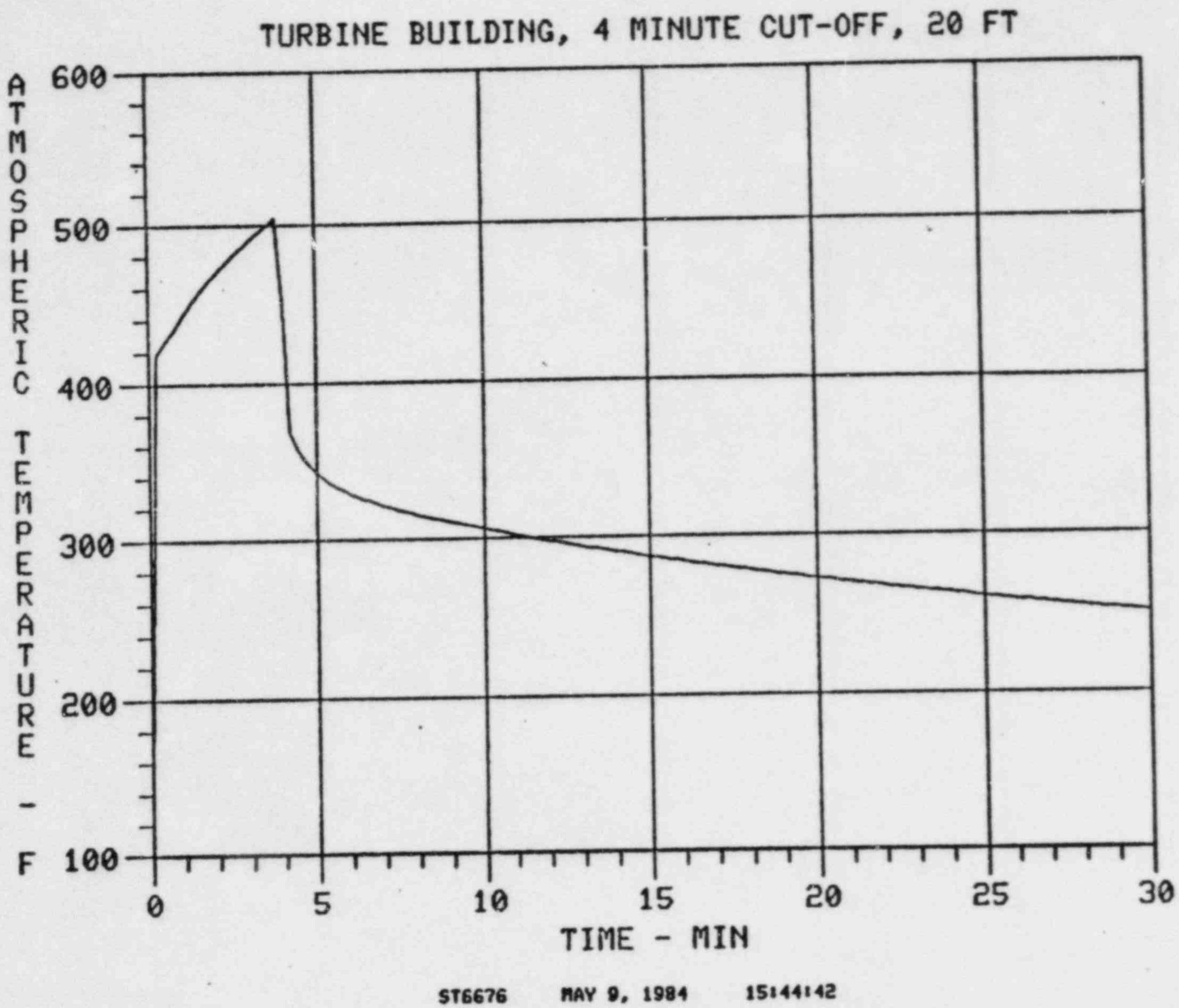

Fig. 12. Turbine building atmospheric temperature at a distance of 20 ft from steam leak using blowdown data from Ref. 1..

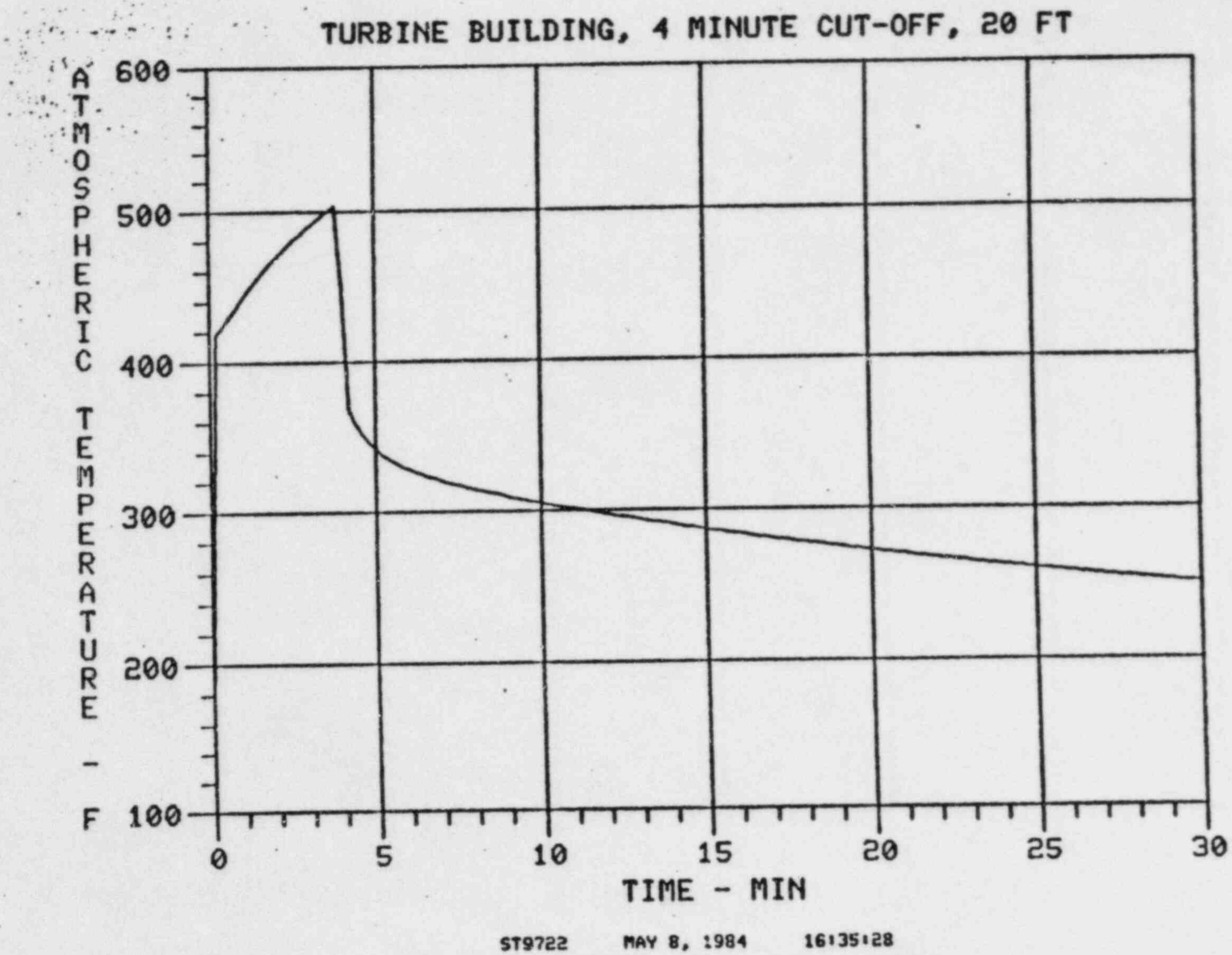

Fig. 13. Turbine building atmospheric temperature at a distance of 20 ft from steam leak using blowdown data from recovered runs of April 1972.

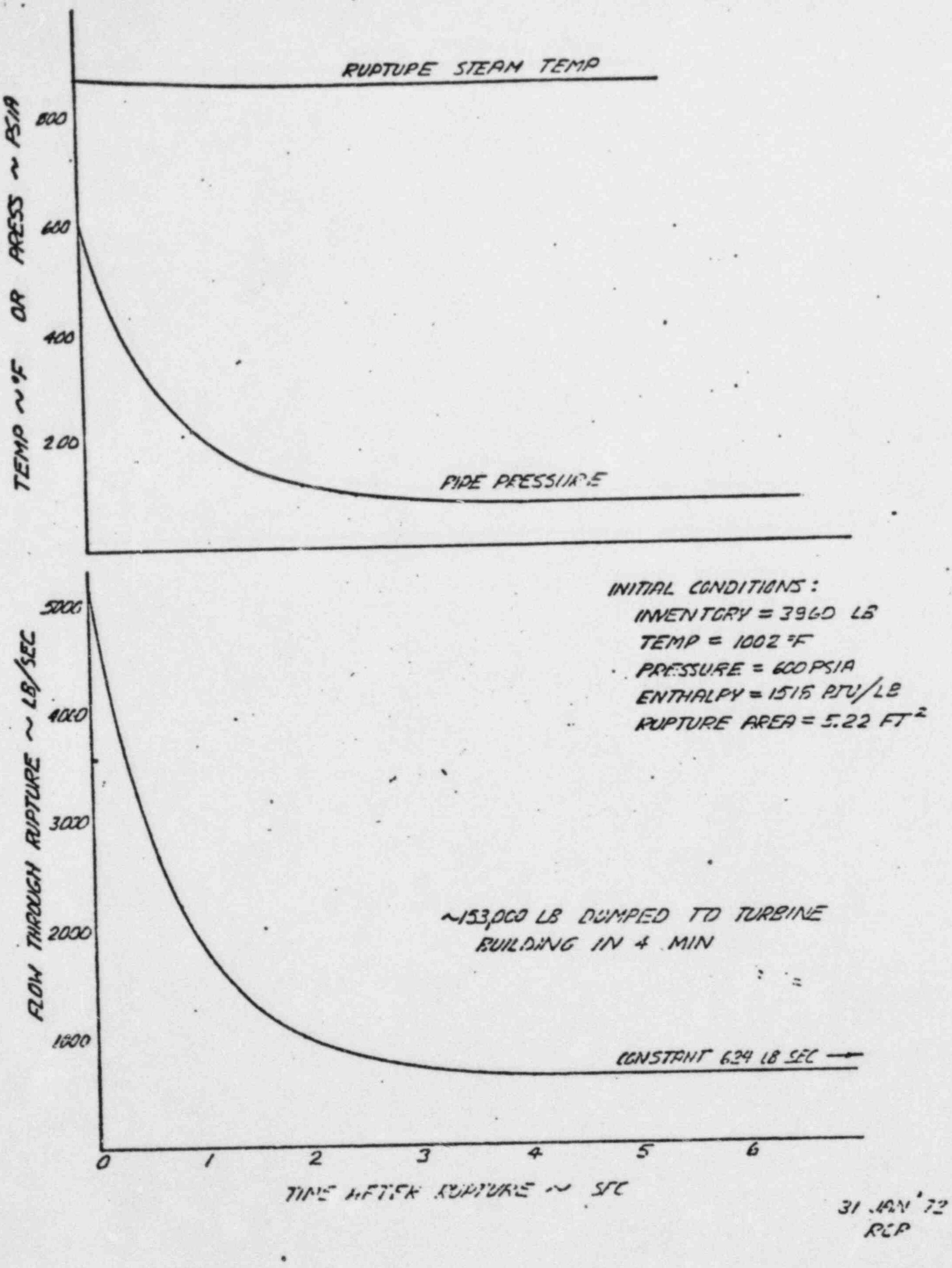

Fig. 14. PSC hot reheat rupture 34" OD main header line, no scram.

APPENDIX A

COMPUTER ANALYSIS STORAGE

Storage and retrieval of the computer studies performed for this analysis are described here. Computer programs and runstreams for each case analyzed are saved in Computer Services Division (CSD) Archive Library. Printed output for each case is stored by Records Management Department.

The computer code used for this analysis was the CONTEMPT-G code retrieved from CSD Archive Library reference number THSD0560. Runstreams, data and output plot files for each transient case are saved in the CSD Archive Library as a data tape. The entry is FSV-STEAMLK/5-84 reference number THSD3856. Retrieving a data tape from the Archive Library follows the procedure given in the CSD Users Guide GA-A10888.

The computer runs for each transient case are listed in Table A1. This table lists the case number, batch run STJOB number, date and time run was created, building, distance from leak, leak cut-off time, source of leak rate, source of heat transfer data, output plot file name and runstream element name. The run elements and data elements are all stored in the runstream file.

A change was made to the CONTEMPT-G code to allow plotting data for atmospheric air temperature to be written on an output file for later use with the SUPER*PLOT program described in Reference 5. These changes were incorporated in a new version of CONTMT called CONTMT/P. The actual data writing onto the file is done in a new subroutine called PLOTT. A map element CONTMT/PMAP incorporates these changes into absolute element named CONTMT/PABS. Table A2 shows the changes made to CONTMT and the new map element. Table A3 lists subroutine PLOTT.

Runstreams, input data and plot files described in Table A1 as well as the changes to the CONTEMPT-G code for plotting are all saved in Archive Library entry FSV-STEAMLK/5-84 reference number THSD3856. This data tape was written by the runstream shown in Table A4. Temporary tape number 0086 was submitted

to the Archive Library and copied on a permanently stored archive tape. Contents of this data tape entry are listed in Table A5. This table shows the archive data tape file number, catalogued file name written on the tape and a brief description of the file contents.

The Archive Librarian will retrieve this data tape from the archives when given a completed Retrieval Request (Form GA-1467) as described in the CSD Users Guide.

Computer printouts generated during this analysis are stored in box GA-03130 with the Records Management Department for reference. Table A6 itemizes the contents of this box.

TABLE A1
CONTEMPT-G RUNS

CASE	STJOB	DATE	START TIME	BUILDING(1)	DISTANCE FROM LEAK (FT)	CUT-OFF (MIN)	LEAK RATE DATA(2)	HEAT TRANSFER(3)	PLOT FILE NAME	ELEMENT NAME(4)
1	2336	May 16	8:59	T	20	4	2	2	FSV-PLOT*TB4M20F-1	TRBN-4MIN/20FT-1
2	9722	May 8	16:35	T	20	4	2	1	FSV-PLOT*TB4M20F-3	TRBN-4MIN/20FT-3
3	0704	May 8	16:51	R	20	4	2	1	FSV-PLOT*RB4M20F-3	RCTR-4MIN/20FT-3
4	6676	May 9	15:44	T	20	4	1	1	FSV-PLOT*TB4M20F-2	TRBN-4MIN/20FT-2
5	6737	MAY 9	15:45	R	20	4	1	1	FSV-PLOT*RB4M20F-2	RCTR-4MIN/20FT-2
6	5833	May 15	9:57	R	20	10	2	1	FSV-PLOT*RB10M20F-1	RCTR-10MIN/20FT-1
7	5930	May 15	9:58	R	30	4	2	2	FSV-PLOT*RB4M30F-1	RCTR-4MIN/30FT-1
8	8364	May 15	10:39	T	20	10	2	1	FSV-PLOT*TB10M20F-1	TRBN-10MIN/20FT-1
9	1840	May 23	11:37	T	20	20	2	1	FSV-PLOT*TB20M20F-1	TRBN-20MIN/20FT-1
0	1880	May 23	11:38	R	20	20	2	1	FSV-PLOT*RB20M20F-1	RCTR-20MIN/20FT-1

Notes:

1) Building
   R - Reactor
   T - Turbine

2) Steam leak rate taken from
   1 - GA-A12045
   2 - recovered run dated April 1972

3) Heat transfer surface areas, coefficients taken from
   1 - GA-A12045
   2 - recovered run dated April 1972

4) Element name is stored on catalog file FSV-STMLK*5-84

TABLE A2

CONTMT UPDATES TO CREATE CONTMT/P
AND MAP ELEMENT CONTMT/PMAP

```
@HDG      CONTMT/P
@FOR,S    CONTMT/O,CONTMT/P,CONTMT/P
-940         CONTMT/O
C       STORE PLOTTING VALUES
        CALL PLOTT (TSEC,TMPO,0)
-1026        CONTMT/O
C       WRITE PLOTTING VALUES ON UNIT 7
        CALL PLOTT (TSEC,TMPO,1)
@HDG      CONTMT/PMAP
@PREP
@MAP,IS    CONTMT/PMAP,CONTMT/PABS
        IN CONTMT/P
```

TABLE A3

SUBROUTINE PLOTT

```
@HDG      PLOTT
@FOR,SI   PLOTT,PLOTT
      SUBROUTINE PLOTT (TSEC,TMPO,IRITE)
C
C     SAVE PLOTTING VALUES FOR TIME AND ATMOSPHERIC TEMPERATURE ON FILE 7
C
C     TSEC = TIME, SEC
C     TMPO = ATMOSPHERIC TEMPERATURE, F
C     IRITE = 0, STORE TSEC AND TMPO IN ARRAYS
C             1, WRITE ARRAYS ON FILE 7
      PARAMETER NPLT=160
      DIMENSION TIMPLT(NPLT), TEMPLT(NPLT), DATE(2), TIMEX(2)
      DATA IP/0/, INIT/0/, DTPLOT/0.2/, TPLOT/0./
C     IP = NUMBER OF POINTS
C     DTPLOT = FREQUENCY FOR STORING PLOT DATA, MIN
C     TPLOT = NEXT TIME TO STORE PLOT DATA, MIN
C
      IF(INIT.EQ.1) GO TO 100
C     INITIALIZE FILE WITH DATE, TIME, JOBNO
      CALL DATGT1(DATE)
      CALL GTIME(TIMEX)
      CALL PCT(JOBNO,1,0)
      INIT=1
      WRITE(7,7701) DATE,TIMEX,JOBNO
  100 CONTINUE
C
      IWRITE=IRITE+1
      GO TO (200,300), IWRITE
C
C     SAVE PLOTTING VALUES AT DTPLOT TIME INTERVALS
  200 CONTINUE
      TMIN=TSEC/60.
      IF(TMIN.LT.TPLOT) GO TO 500
      IP=IP+1
      IF(IP.GT.NPLT) CALL MERR
      TIMPLT(IP)=TMIN
      TEMPLT(IP)=TMPO
      TPLOT=TPLOT+DTPLOT
      GO TO 500
C
C     WRITE PLOTTING ARRAYS ON FILE 7
  300 CONTINUE
      WRITE(7,7704) IP
      WRITE(7,7702) (TIMPLT(I),I=1,IP)
      WRITE(7,7703) IP
      WRITE(7,7702) (TEMPLT(I),I=1,IP)
      WRITE(6,7705) DATE,TIMEX,JOBNO
  500 RETURN
C
 7701 FORMAT(13A6)
 7702 FORMAT(6E12.7)
 7703 FORMAT(I6,' ATMOSPHERIC TEMPERATURE, DEG-F')
 7704 FORMAT(I6,' TIME, MINUTES')
 7705 FORMAT('1PLOTTING ARRAYS WRITTEN ON UNIT 7 FOR ',13A6)
      END
```

TABLE A4

RUNSTREAM THAT WROTE TAPE FOR ARCHIVE ENTRY THSD3856

```
@ASG,TX   0086,U9S,0086W
@HDG   FILE 1 IS PSC*FILE1  COPY,GM TO TAPE 0086
@PRT,TJLC   PSC*FILE1.
@COPY,GM   PSC*FILE1.,0086.
@HDG   FILE 2 IS FSV-STMLK*5-84  COPY,GM TO TAPE 0086
@PRT,TJLC   FSV-STMLK*5-84.
@COPY,GM   FSV-STMLK*5-84.,0086.
@HDG   FILE 3 IS FSV-PLOT*TB4M20F-1  COPY,GM TO TAPE 0086
@PRT,TJLC   FSV-PLOT*TB4M20F-1.
@COPY,GM   FSV-PLOT*TB4M20F-1.,0086.
@HDG   FILE 4 IS FSV-PLOT*TB4M20F-3  COPY,GM TO TAPE 0086
@PRT,TJLC   FSV-PLOT*TB4M20F-3.
@COPY,GM   FSV-PLOT*TB4M20F-3.,0086.
@HDG   FILE 5 IS FSV-PLOT*RB4M20F-3  COPY,GM TO TAPE 0086
@PRT,TJLC   FSV-PLOT*RB4M20F-3.
@COPY,GM   FSV-PLOT*RB4M20F-3.,0086.
@HDG   FILE 6 IS FSV-PLOT*TB4M20F-2  COPY,GM TO TAPE 0086
@PRT,TJLC   FSV-PLOT*TB4M20F-2.
@COPY,GM   FSV-PLOT*TB4M20F-2.,0086.
@HDG   FILE 7 IS FSV-PLOT*RB4M20F-2  COPY,GM TO TAPE 0086
@PRT,TJLC   FSV-PLOT*RB4M20F-2.
@COPY,GM   FSV-PLOT*RB4M20F-2.,0086.
@HDG   FILE 8 IS FSV-PLOT*RB10M20F-1  COPY,GM TO TAPE 0086
@PRT,TJLC   FSV-PLOT*RB10M20F-1.
@COPY,GM   FSV-PLOT*RB10M20F-1.,0086.
@HDG   FILE 9 IS FSV-PLOT*RB4M30F-1  COPY,GM TO TAPE 0086
@PRT,TJLC   FSV-PLOT*RB4M30F-1.
@COPY,GM   FSV-PLOT*RB4M30F-1.,0086.
@HDG   FILE 10 IS FSV-PLOT*TB10M20F-1  COPY,GM TO TAPE 0086
@PRT,TJLC   FSV-PLOT*TB10M20F-1.
@COPY,GM   FSV-PLOT*TB10M20F-1.,0086.
@HDG   FILE 11 IS FSV-PLOT*TB20M20F-1  COPY,GM TO TAPE 0086
@PRT,TJLC   FSV-PLOT*TB20M20F-1.
@COPY,GM   FSV-PLOT*TB20M20F-1.,0086.
@HDG   FILE 12 IS FSV-PLOT*RB20M20F-1  COPY,GM TO TAPE 0086
@PRT,TJLC   FSV-PLOT*RB20M20F-1.
@COPY,GM   FSV-PLOT*RB20M20F-1.,0086.
@FREE  0086
```

TABLE A5
ARCHIVE THSD3856 CONTENTS

TAPE FILE	CATALOG FILE	CONTENT DESCRIPTION
1	PSC*FILE1	CONTEMPT-G updated program, MAP and absolute element
2	FSV-STMLK*5-84	Runstreams and data
3	FSV-PLOT*TB4M20F-1	Plot file for case 1
4	FSV-PLOT*TB4M20F-3	Plot file for case 2
5	FSV-PLOT*RB4M20F-3	Plot file for case 3
6	FSV-PLOT*TB4M20F-2	Plot file for case 4
7	FSV-PLOT*RB4M20F-2	Plot file for case 5
8	FSV-PLOT*RB10M20F-1	Plot file for case 6
9	FSV-PLOT*RB4M30F-1	Plot file for case 7
10	FSV-PLOT*TB10M20F-1	Plot file for case 8
11	FSV-PLOT*TB20M20F-1	Plot file for case 9
12	FSV-PLOT*RB20M20F-1	Plot file for case 10

GA 1638 (10/81)

TABLE A6
BOX GA-03130
RECORDS STORAGE MANIFEST

Location (RMD use only)	Organization Name	Org. No.	Project No.
	System Design & Plant Dynamics	647	1900

Records Storer	Project Name	Contract Number	Box No.
C.J. Rodgers	Fort St. Vrain	N-5052	GA-03130

Date of Records	Date Stored (Yr/Mo/Day)	Disposition Date (Yr/Mo/Day)
May 1984	84/06/29	99/01/01

Records Descriptions

1. Turbine building, 4 minute cut-off, 20ft, blowdown from recovered run, surface areas from GA-A12045
2. Reactor building, 4 minute cut-off, 20ft, blowdown from recovered run, surface areas from GA-A12045
3. Turbine building, 4 minute cut-off, 20ft, blowdown from GA-A12045, surface areas from GA-A12045
4. Reactor building, 4 minute cut-off, 20ft, blowdown from GA-A12045, surface areas from GA-A12045
5. Reactor building, 10 minute cut-off, 20ft, blowdown from recovered run, surface areas from GA-A12045
6. Reactor building, 4 minute cut-off 30ft, blowdown from recovered run, surface areas from recovered run
7. Turbine building, 10 minute cut-off, 20ft, blowdown from recovered run, surface areas from GA-A12045
8. Turbine building, 4 minute cut-off, 20 ft, blowdown from recovered run, surface areas from recovered run
9. Turbine building, 20 minute cut-off, 20ft, blowdown from recovered run, surface areas from GA-A12045
10. Reactor building, 20 minute cutoff, 20ft, blowdown from recovered run, surface areas from GA-A12045

Note: Box GAC-01447 contains related data from previous FSV steam leak studies.

CALCULATION REVIEW REPORT

TITLE: Fort St. Vrain reactor building cold reheat and Turbine building hot reheat steam line breaks with 10 minute leak termination

APPROVAL LEVEL ____

QAL LEVEL ____

DISCIPLINE	SYSTEM	DOC. TYPE	PROJECT	DOCUMENT NO.	ISSUE NO./LTR.

INDEPENDENT REVIEWER:

NAME C. G. Hoot

ORGANIZATION Safety Design (648)

REVIEWER SELECTION APPROVAL: BR MGR [signature] DATE 071684

REVIEW METHOD:	YES	NO	ERROR DETECTED
ARITHMETIC CHECK	✓		no
LOGIC CHECK	✓		no
ALTERNATE METHOD USED		✓	
SPOT CHECK PERFORMED	✓		no
COMPUTER PROGRAM USED		✓	

REMARKS: (ATTACH LIST OF DOCUMENTS USED IN REVIEW)

CALCULATIONS FOUND TO BE VALID AND CONCLUSIONS TO BE CORRECT:

INDEPENDENT REVIEWER Charles Hoot DATE 7-13-84

SIGNATURE