APPENDIX 1

NATURAL CIRCULATION

IN

B&W OPERATING PLANTS

Rev 1

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INTRODUCTION

This appendix documents both the natural circulation tests and other natural circulation events that have occurred on B&W designed reactors. Natural circulation testing has been conducted at Oconee I and Davis-Besse. Unscheduled events at other plants have provided additional natural circulation data as well. In all, we have identified nine instances in which data on natural circulation cooling have been recorded. This appendix contains the data available for each event, presented as uniformly as the parameters monitored and period of recording permit. A short description of each event includes the prevailing plant conditions, a summary of the sequence of events, and plots of the important parameters versus time.

This information demonstrates that in all cases the plant was adequately cooled in the natural circulation mode and primary system temperatures decreased to approach the prevailing steam generator secondary side saturation temperature. A curve of natural circulation flow versus core decay heat generation is also provided to show this important relationship. In all natural circulation tests, reactor coolant flow in this mode met the acceptance criteria.

SYNOPSES OF EVENTS

1. <u>Dry Run Natural Circulation Test</u>

After the fuel assemblies were loaded but prior to initial criticality at Oconee Unit 1, B&W and Duke Power Company conducted a simulated natural circulation test on May 1, 1973. With the core in place but with no power history to produce decay heat, two tests were conducted and were successful in developing sufficient natural circulation flow rate.

The first test induced natural circulation flow in both reactor coolant system loops by rapidly lowering the steam generator steam pressure to lower the temperature of the reactor coolant external to the reactor vessel. A decrease in steam pressure from 885 psig to 730 psig in approximately two minutes was accomplished with the turbine bypass valves. This caused a 15 to 20F decrease in cold leg temperatures, which maintained a natural circulation flow rate for 10 minutes or more. A large change in the indicated water levels of both steam generators occurred in the same two-minute depressurization time. The final level was 310 inches above the lower tubesheet, whereas the initial level was 375 inches. The reason for rapidly creating a 20F decrease in the cold leg temperature was to use the transport time between the cold leg temperature probe and the hot leg temperature probe to calculate the natural circulation flow rate.

For the second test initial operating conditions were reestablished several hours later, and after securing the last two RC pumps, the OTSG water levels were rapidly increased to cool down the cold leg temperatures discharging from both steam generators. Main feedwater flow was used to increase the initial water level of 310 inches to approximately 390 inches in seven minutes. The

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rapid fill of both steam generators decreased steam pressure from 850 psig to 650 psig in approximately 16 minutes. As before, the cold leg temperatures were decreased about 25F to produce sufficient natural circulation flow lasting 10 or more minutes.

Test data from this simulated natural circulation flow rate test are presented in Figures 1 through 5. Figure 1 is a schematic of the B&W 177-fuel assembly lowered-loop plant. The location of the instrumentation used to measure the behavior of the reactor coolant system during natural circulation is shown in Figure 1. Figure 2 displays loop B hot leg and cold leg RC temperatures during the first test described above. RC pressure and pressurizer level are shown in the lower graph. RC pressure varied ± 60 psi about the normal operating value of 2150 psig.

Figure 3 shows the loop A hot leg and cold leg temperatures. The lower figure displays a comparison of the highest hot leg temperature and the RC system saturation temperature. Approximately 110F subcooling existed in the RC system during this simulated test.

Figure 4 displays the rapid change in steam generator pressure and steam generator water levels as the turbine bypass valves were used to cool the cold leg temperatures 15 to 20F.

Figure 5 shows a comparison of a cold leg temperature with the saturation temperature within the steam generator. These data show that the time delay for cold leg temperature to be altered by changing steam pressure is approximately two minutes.

Test data from the second test, natural circulation by raising water level, are presented in Figures 6 through 9. Figure 6 shows hot leg and cold leg temperatures in loop A. The lower graph displays RC pressure and pressurizer level with changes nearly equal to the first test.

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Figure 7 displays loop B hot and cold leg temperatures The lower graph shows, as before, the amount of subcooling is greater than 110F for the second test inducing a natural circulation flow rate. Figure 8 shows the change in both steam generator pressures and water level indications.

Figure 9 displays a comparison of loop A cold leg temperature and the saturation temperature corresponding to loop A steam pressure. Note again the short time delay of 2½ minutes for the cold leg temperature to follow steam pressure.

2. First Natural Circulation Test with Decay Heat

On November 4, 1973, Duke Power Company conducted a loss-of-offsite-power test at Oconee Unit 1. The only data available are shown in Figure 10. Loop A and B hot and cold leg temperatures are very well matched, and the core temperature difference is about 30F. The maximum value of hot leg temperature was only 560F.

3. Second Natural Circulation Test with Decay Heat

On May 2, 1974, Duke Power Company performed another natural circulation test on Oconee Unit 1. B&W and Duke dete[illegible] at if the natural circulation test could be completed within one hour after reactor shutdown, decay heat greater than 1% rated power would be available due to the previous power history. The natural circulation test was successful, and after 37 minutes of unpowered flow, the RC pumps were restarted.

Figure 11 displays the loop A and B cold leg and hot leg temperatures. The core temperat[illegible] difference on loop B is slightly larger than that on loop A - 25F compa[illegible] to 22F. The highest hot leg temperature was only 558F.

Figure 12 shows a comparison of the highest hot leg temperature with the saturation temperature of the RC system. Minimum subcooling was approximately 100F.

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Figure 13 shows an increase in both loop A and loop B steam pressure during the period of natural circulation and is the reason why both loop A and B cold leg temperatures were increasing rather than decreasing during this test.

4. First Unplanned Natural Circulation Incident

On January 4, 1974, Oconee Unit 2, at 75% power level, was separated from the grid by erroneous actions in the 230-kV switchyard. The reactor tripped on either high RC pressure (2355 psig) or pump monitors and both feedwater pumps were shut down. The emergency feedwater system was delayed in supplying water to the steam generators for approximately 7 minutes. Instrument air pressure reduced to a value below 50 psi until a diesel-powered backup compressor was manually started about 7 minutes after the trip. At that time, the emergency feedwater system began to fill the steam generators to 95% on the Operate Range level indication (~375 inches above the tubesheet). Control room operators took manual control of the startup valves in an effort to reduce the cooldown rate about 10 minutes after the trip.

Figure 14 shows the rapid decrease in the cold leg temperature without the concurrent reduction in RC pressure and pressurizer level. This was due to the operator's use of High Pressure Injection, as required, to maintain RC pressure and pressurizer level.

Figure 15 displays RC system temperatures and the saturation temperature corresponding to the RC pressure during this occurrence of natural circulation.

Figure 16 shows recorded steam generator parameters that were available for this unplanned natural circulation occurrence. When the emergency feedwater was supplied to the loop A steam generator, it automatically filled the generator to 95% on the operate level. This setpoint has been changed to 50% on all B&W lowered-loop plants subsequent to this incident.

Figure 17 displays the calculated core decay heat versus time that natural circulation flow rate existed.

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5. Unplanned Loss-of-Offsite Power Event at ANO-1

On February 22, 1975, Arkansas Nuclear One, Unit 1, operating at 100% power, experienced a complete loss of electrical power and the reactor tripped, the RC pumps shut down, and the main feedwater pumps shut down. Diesel generator No. 1 started automatically. Thirty-six seconds after the reactor trip, the makeup pump was switched to the borated water storage tank and water was supplied to the RC pump seals. Also, makeup was provided into the RC system to maintain pressurizer level.

Emergency feedwater was established with minimum flow through emergency feedwater bypass control valves. By five minutes after reactor trip, one RC pump in each loop was restarted.

Figure 18 displays the loop B hot leg and cold leg temperatures and other RC system parameters during the 5-minute coastdown event.

Figure 19 shows nearly identical core temperature difference for loop A as recorded for loop B. The maximum hot leg temperature was 586F just prior to starting the RC pumps and was more than 60F below the saturation temperature. At this point natural circulation was not fully developed.

Figure 20 displays steam generator conditions resulting from minimum emergency feedwater flow rate injection. Note that the flow rate was adequate to maintain high steam pressure and maintain a nearly constant cold leg temperature.

Figure 21 shows a comparison of the lowest cold leg temperature and the corresponding steam generator saturation temperature.

Figure 22 displays the calculated core decay heat for ANO-1 over a time interval that includes the 5-minute coastdown event.

6. Natural Circulation at Crystal River 3 on Loss of Offsite Power Test

On April 23, 1977, Florida Power Corporation conducted a successful loss of offsite power test and demonstrated that satisfactory control of the plant was achieved under simulated station blackout conditions. The reactor was tripped from 15% power and all four RC pumps were shut down. One emergency diesel generator was left operating to power a makeup pump, cooling water pumps, an instrument air compressor, and some pressurizer heaters. In one minute the emergency feedwater pump was operating and supplying feedwater to both steam generators. Operators tried to use both steam-driven and motor-driven emergency feedwater pumps to evenly and simultaneously supply water to both steam generators. One steam generator received too little feedwater and boiled dry until the filling operation of the other steam generator was completed. This resulted in unmatched loop A and B hot and cold leg temperatures.

Figure 23 shows RC system parameters during natural circulation at Crystal River 3. Loop A indicates a core temperature difference of nearly 50F.

Figure 24 shows several other RC system temperatures during the same time. The largest core temperature difference recorded for loop B was 35F, whereas the margin of subcooling exceeded 60F.

Figure 25 displays the unmatched filling operation and resulting unequal system pressures for loop A and B steam generators.

Figure 26 shows the close relationship of cold leg temperature of each loop with the saturation temperature for each steam generator. The time delay is approximately equal to two minutes as noted earlier for Oconee Unit 1 tests.

Figure 27 displays the calculated core decay heat at the time natural circulation flow rate existed at Crystal River 3.

7. Natural Circulation during Loss of Offsite Power Test at TMI-2

Metropolitan Edison Company conducted a successful loss of offsite power test on April 22, 1978, from an initial power level of 15%. Both diesel generators were off initially but started automatically within 17 seconds after the reactor trip. The emergency feedwater system started a supply of water to the steam generators at approximately one minute, but the final level was about 40% in the Operate Range rather than 50%.

Figure 28 displays the loop B core temperature difference of 35F or more and the simultaneous change in RC pressure and pressurizer level during the natural circulation flow condition.

Figure 29 shows other RC system temperatures and indicates nearly equal loop A and loop B performance. The highest hot leg temperature is more than 100F below the saturation temperature for the RC system.

Figure 30, which shows steam generator conditions following the reactor trip, reveals that the normal fill rate for the emergency feedwater system has drastically reduced steam pressure to 700 psig. A review of previous case power history showed that the core had only 0.5 EFP days at the time of this test.

Figure 31 compares the lowest cold leg temperature with the steam generator saturation curve.

8. Unplanned Natural Circulation Occurrence at Davis-Besse 1

On November 29, 1977, Toledo Edison Company experienced an inadvertent loss of offsite power incident which unpowered the four RC pumps and resulted in natural circulation flow rate conditions in the RC system. The initial reactor power was 40%. The incident was initiated by unremoved electrical test jumpers which caused the reactor power to increase until it tripped on overpower at 50%. The reactor trip initiated a turbine trip which caused the operator to open the main circuit breakers. The station experienced a 7-second blackout. After 14½ minutes, the RC pumps were restarted.

Figure 32 displays the rapid change in loop 2 cold leg temperature. The cooldown of the RC system was severe enough to cause the pressurizer level to drop below zero indication.

Figure 33 shows a similar rapid cooldown of the loop 1 cold leg temperature. Also shown is the amount of subcooling of the loop 2 hot leg below the RC system saturation temperature. The minimum value is greater than a 40F margin.

Figure 34 displays the variation in steam generator water level indication and pressures during the time the emergency feedwater system was supplying water to the steam generators. The loop 2 generator was filled first to the setpoint of 10 feet on the startup level. The loop 1 generator was filled a short time later, and the operator took control of valves to reduce the cooldown rate on the RC system. The second generator was only filled to the 8-foot level. Notice the recovery of steam pressure after terminating the fill of the loop 1 steam generator.

Figure 35 shows the usual relationship of cold leg temperature to the steam generator saturation temperature during natural circulation conditions.

Figure 36 displays the calculated core decay heat profile based on the power history of Davis-Besse 1 prior to the reactor trip.

9. Natural Circulation at Davis-Besse 1 during Planned Loss of Offsite Power Test

Toledo Edison Company performed a loss of offsite power test on January 15, 1979, to evaluate the ability of the Davis-Besse 1 plant to withstand this transient. Since the previous occurrence, control room operators had demonstrated that 3 or 4 feet of level in both steam generators would adequately control the cooldown of the RC system. Also, a special natural circulation test was conducted a few

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weeks earlier and it verified that the core flow rate was adequate with water levels greater than 3½ feet but less than 10 feet. (This is the final case presented in this report.)

Figure 37 shows the loop 2 hot leg and cold leg temperatures and the RC pressure and pressurizer level measurements during the natural circulation flow conditions. This was a successful test and the cooldown of the RC system was controlled very well with the lower setpoint for the emergency feedwater system.

Figure 38 displays the loop 1 core differential temperature and a 60F margin of subcooling between hot leg temperature and saturation temperature.

Figure 39 shows steam generator conditions during natural circulation and operation of the emergency feedwater system. After 200 seconds when the operator manually raised the level to approximately 4 feet, a significant reduction in steam pressure resulted. The rate of fill appears to quench steam generation but holding a constant level does not. For once-through steam generators, the fill rate needs to be matched with the decay heat and the desired cooldown rate.

Figure 40 shows a comparison of the lowest cold leg temperature and the steam generator saturation temperature. The action taken by the operator at 200 seconds reduced the usual temperature difference of 20F to nearly zero.

Figure 41 presents the calculated core decay heat profile for Davis-Besse 1 following the reactor trip on January 15, 1979.

10. Natural Circulation Test at Power at Davis-Besse 1

On December 3, 1978, Toledo Edison Company conducted a special test on natural circulation with the core at a constant power level of 4%. A few days earlier testing was completed that provided the neutron flux-cold leg temperature correction factors so that core power would be accurately known during the natural circulation flow rate test.

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An initial level of 40% on the Operate Level was established in each steam generator using the emergency feedwater system. The last two RC pumps were secured and natural circulation flow rate developed. Sufficient time was allowed to obtain good steady-state data for determining the core flow rate.

Next, the water level was set lower and maintained while another set of steady-state data was obtained. In all, six levels were established ranging from 3½ feet to 13 feet and the calculated natural circulation flow rates show surprisingly little variation.

Figure 42 displays the loop B steady-state temperature difference between the hot leg and cold leg temperatures. The lower graph shows the change in RC pressure and pressurizer level during this special natural circulation test.

Figure 43 shows a loop A core differential temperature at 40F and a margin of subcooling between hot leg temperature and saturation temperature of 80F.

Figure 44 presents the quasi-steady-state conditions of both steam generators during the first steady-state test condition for determining natural circulation flow rate at a water level of 13 feet. This fluctuation in emergency feedwater flow rate existed throughout the test.

Figure 45 displays the measured cold leg temperatures for loops A and B and the calculated saturation temperatures based on the measured steam pressures for loop A and B steam generators.

11. <u>Summary of Natural Circulation Flow Measured/Calculated at B&W 177-FA Plants</u>

Reactor coolant flow during natural circulation occurrences, to date, at operating B&W plants is summarized in Figure 46 and the associated table. The flow rates were calculated based on observed core ΔTs and calculated decay heat power levels using a simple heat balance method. In the six cases from the Natural Circulation at Power Test at Davis-Besse 1, reactor power, measured by nuclear instrumentation, was used with observed core ΔTs to calculate natural circulation flow rates.

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During the unplanned natural circulation event at ANO-1 the reactor coolant pumps were restarted approximately 5 minutes into the event. The RC pumps had not fully stopped resulting in the "anomalous" point on Figure 46.

Figure 46 shows that natural circulation flow vs. heat energy in B&W plants behaves in an acceptable and consistent manner. Also shown for comparison is the required flow for core cooling; the measured flow is in all cases well above the required flow.

CONCLUSIONS

In each of the nine cases documented here we have seen adequate natural circulation flow established. The reactor coolant system has been maintained in a subcooled state (minimum of $40^{\circ}F$ subcooling) and heat transfer has been established from the core via the steam generators to the secondary system. Thus, it is concluded that natural circulation is a reliable, effective means of core cooling.

List of Figures

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Figures (Cont'd)

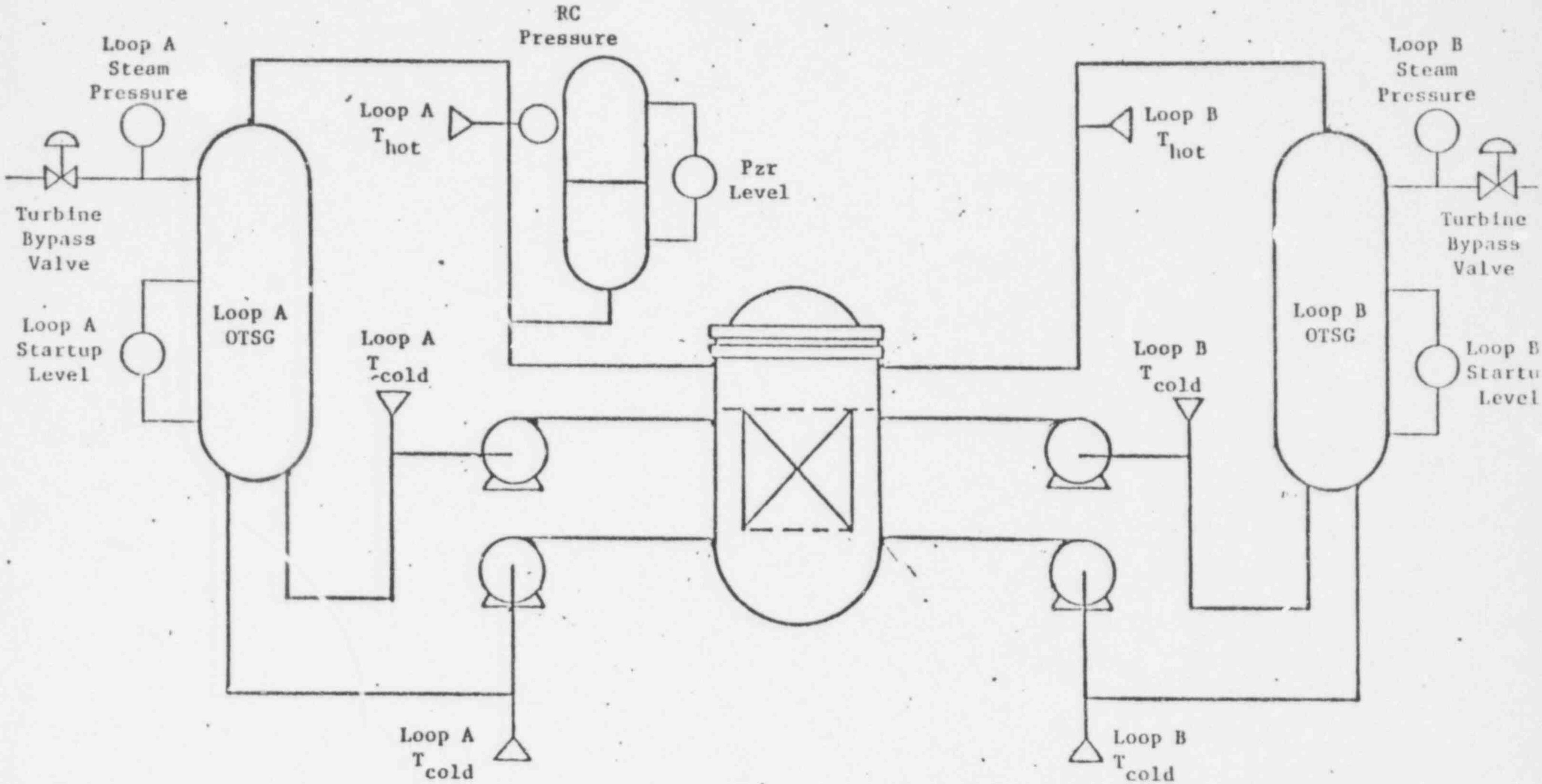

Figure 1. Instrumentation Used for Natural Circulation Flow Rate Measurements at B&W 177-Fuel Assembly Plants

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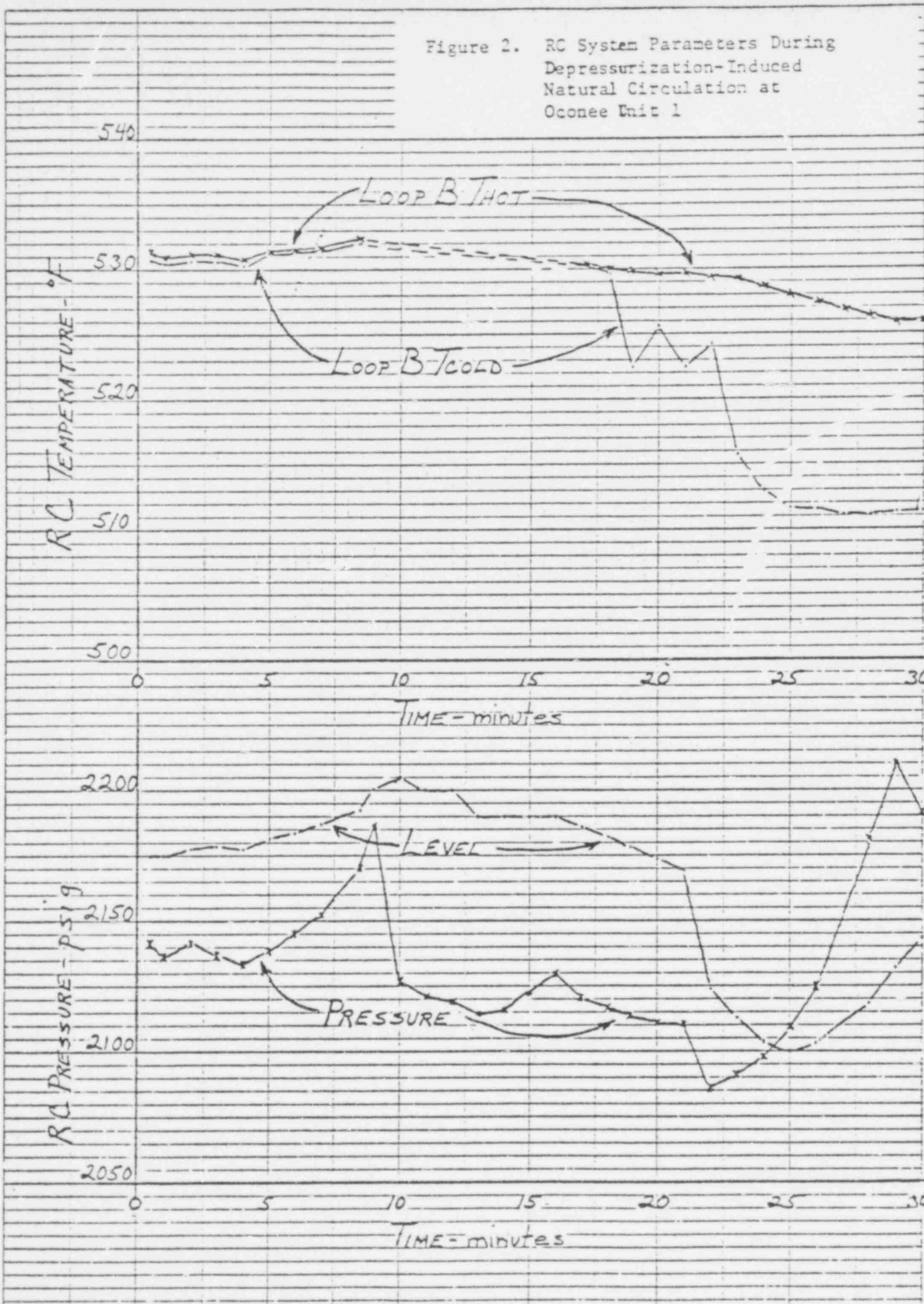

Figure 2. RC System Parameters During Depressurization-Induced Natural Circulation at Oconee Unit 1

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Figure 3. RC System Temperatures and Saturation Temperature During Depressurization-Induced Natural Circulation at Oconee Unit 1

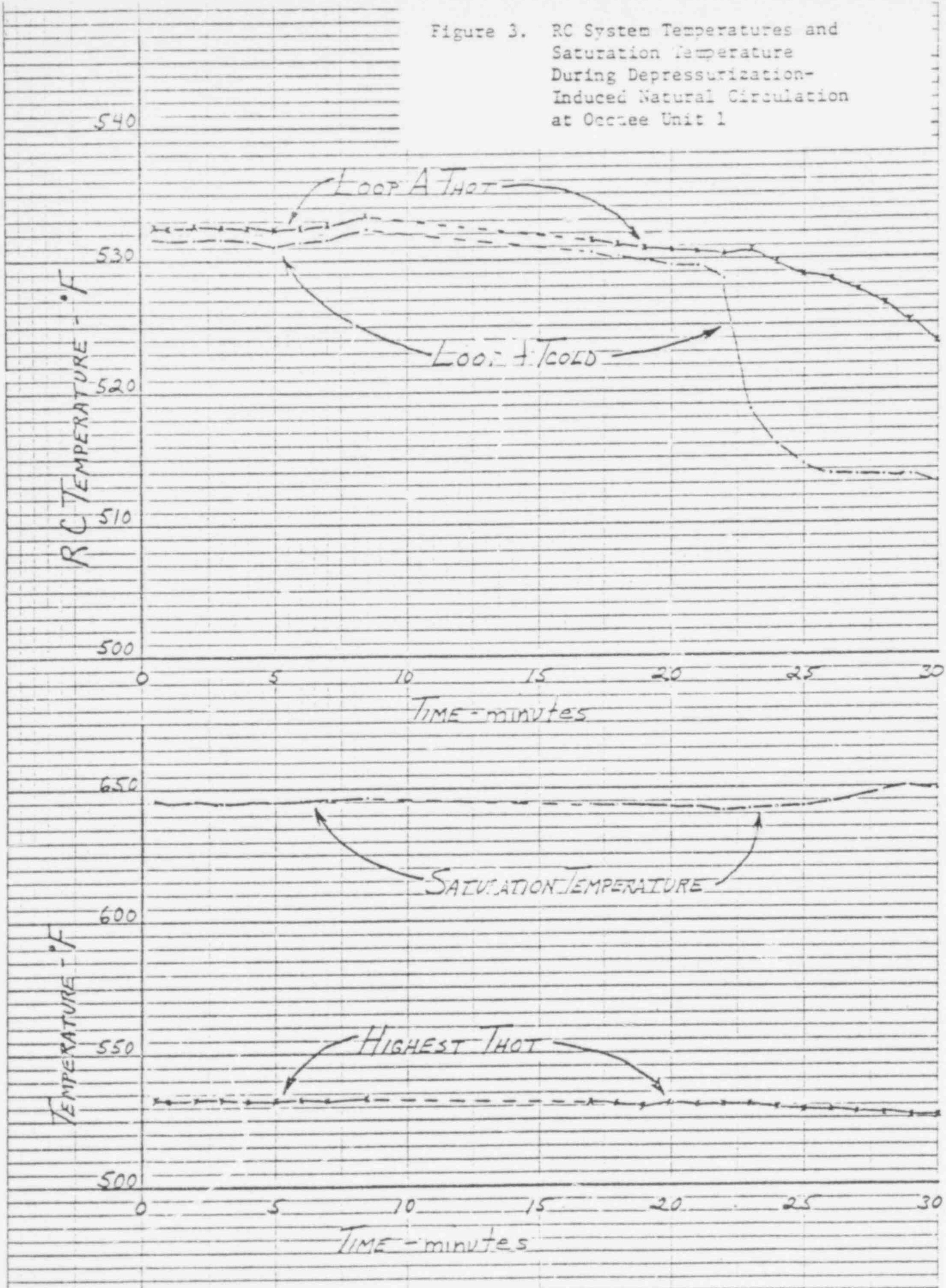

Figure 4. Steam Generator Conditions During Depressurization-Induced Natural Circulation at Oconee Unit 1

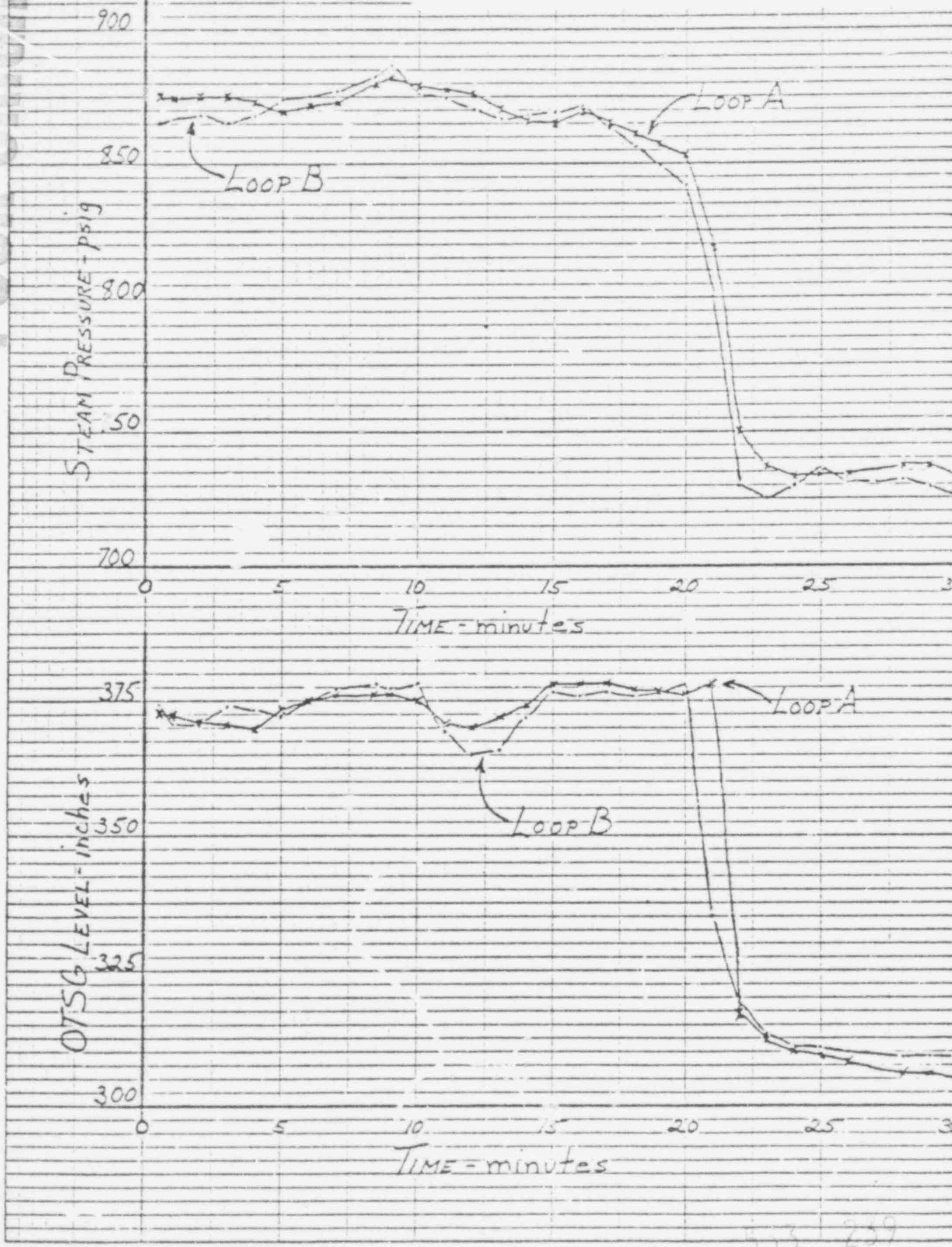

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Figure 5. Steam Generator Outlet and Saturation Temperatures During Depressurization-Induced Natural Circulation at Oconee Unit 1

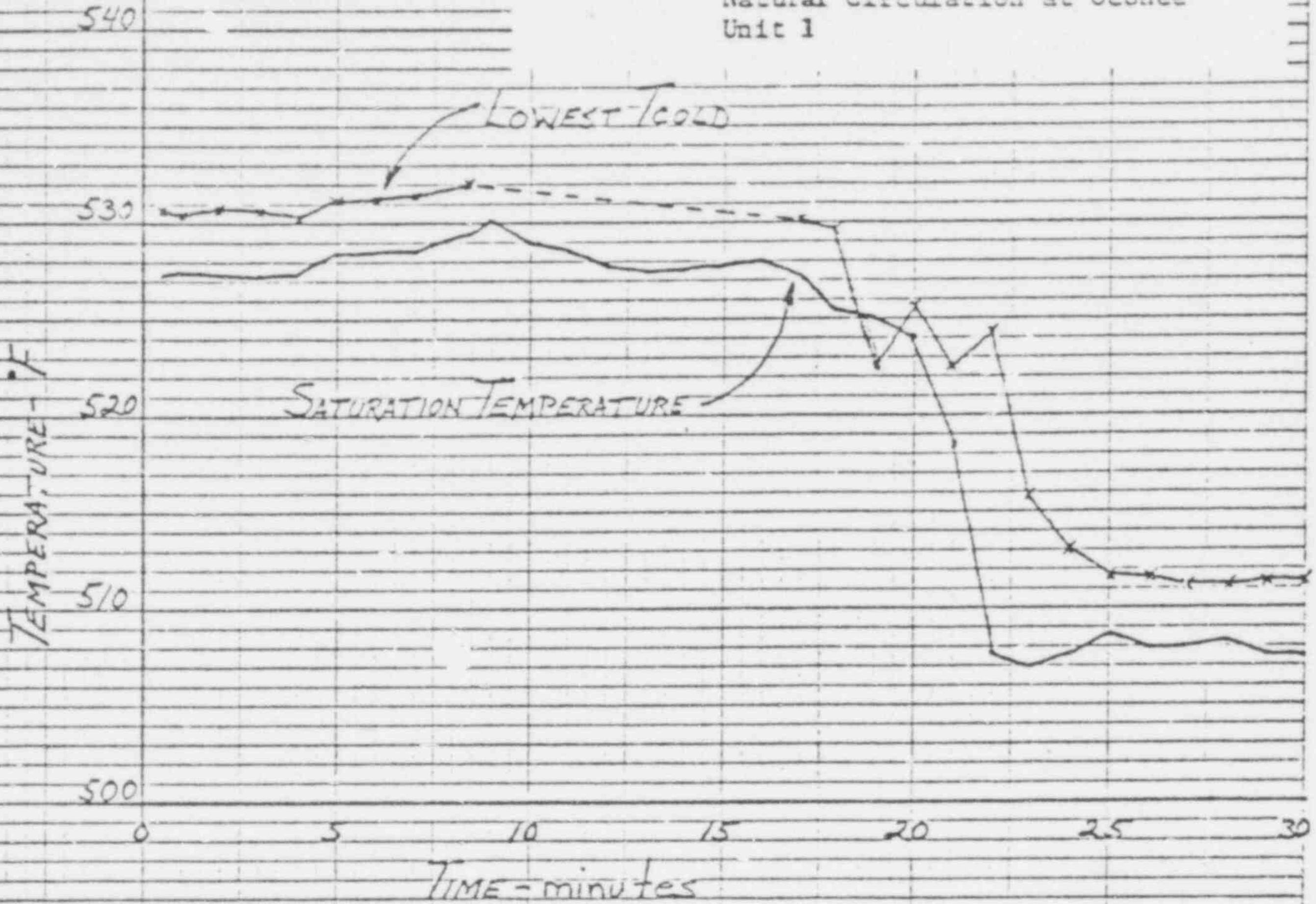

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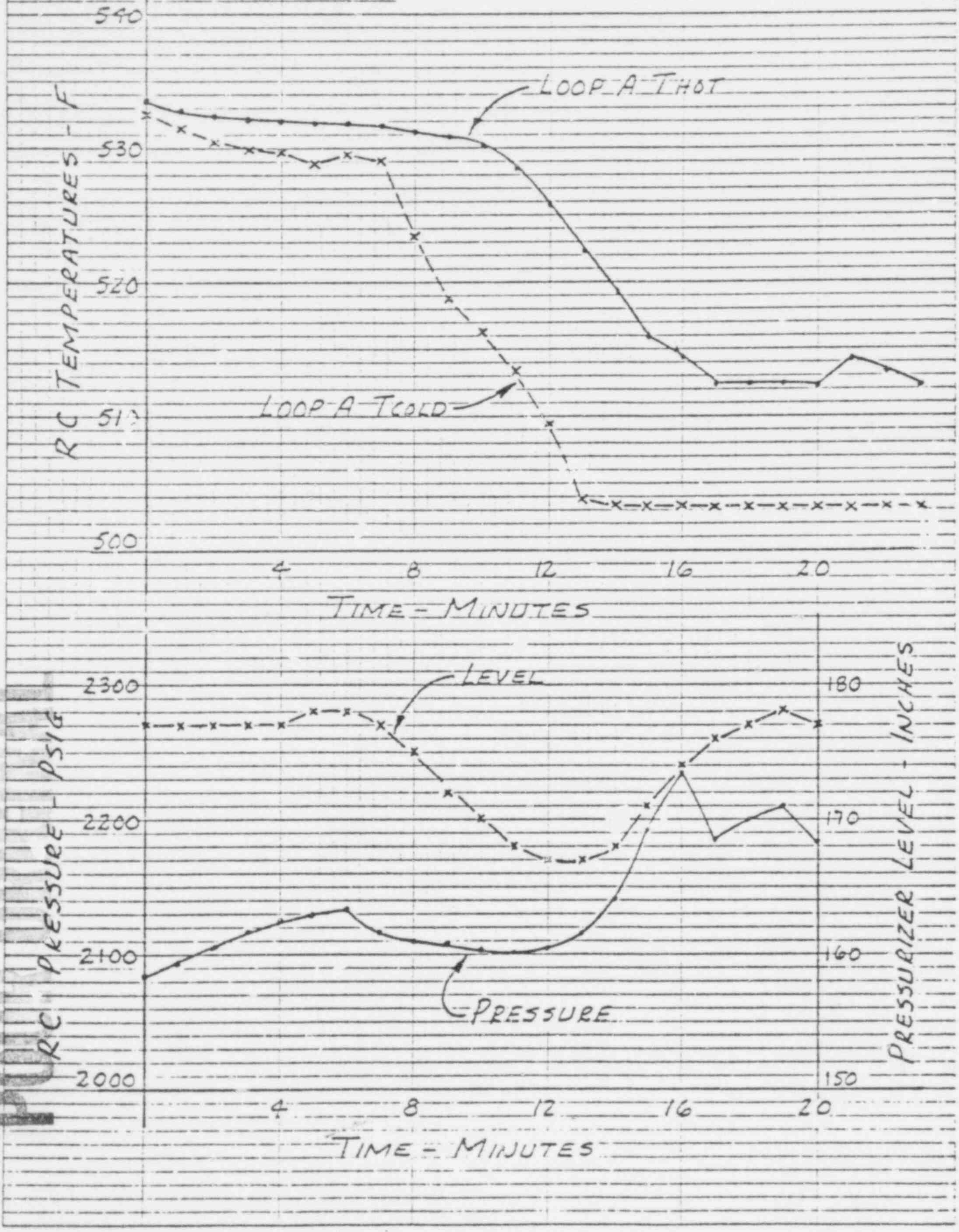
Figure 6. RC System Parameters During Level Increase-Induced Natural Circulation at Oconee Unit 1



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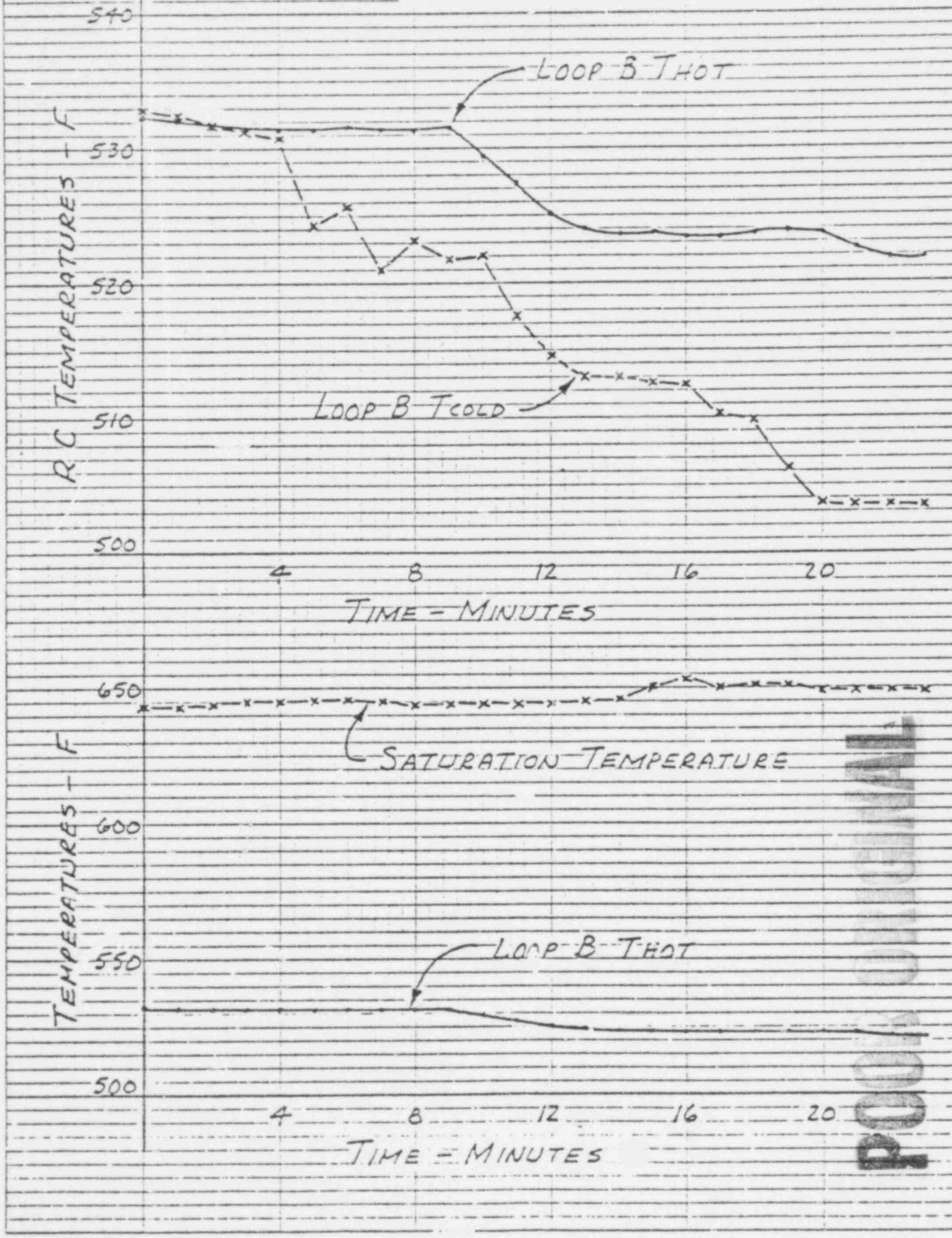

Figure 7. RC System Temperatures and Saturation Temperature During Level Increase-Induced Natural Circulation at Oconee Unit 1

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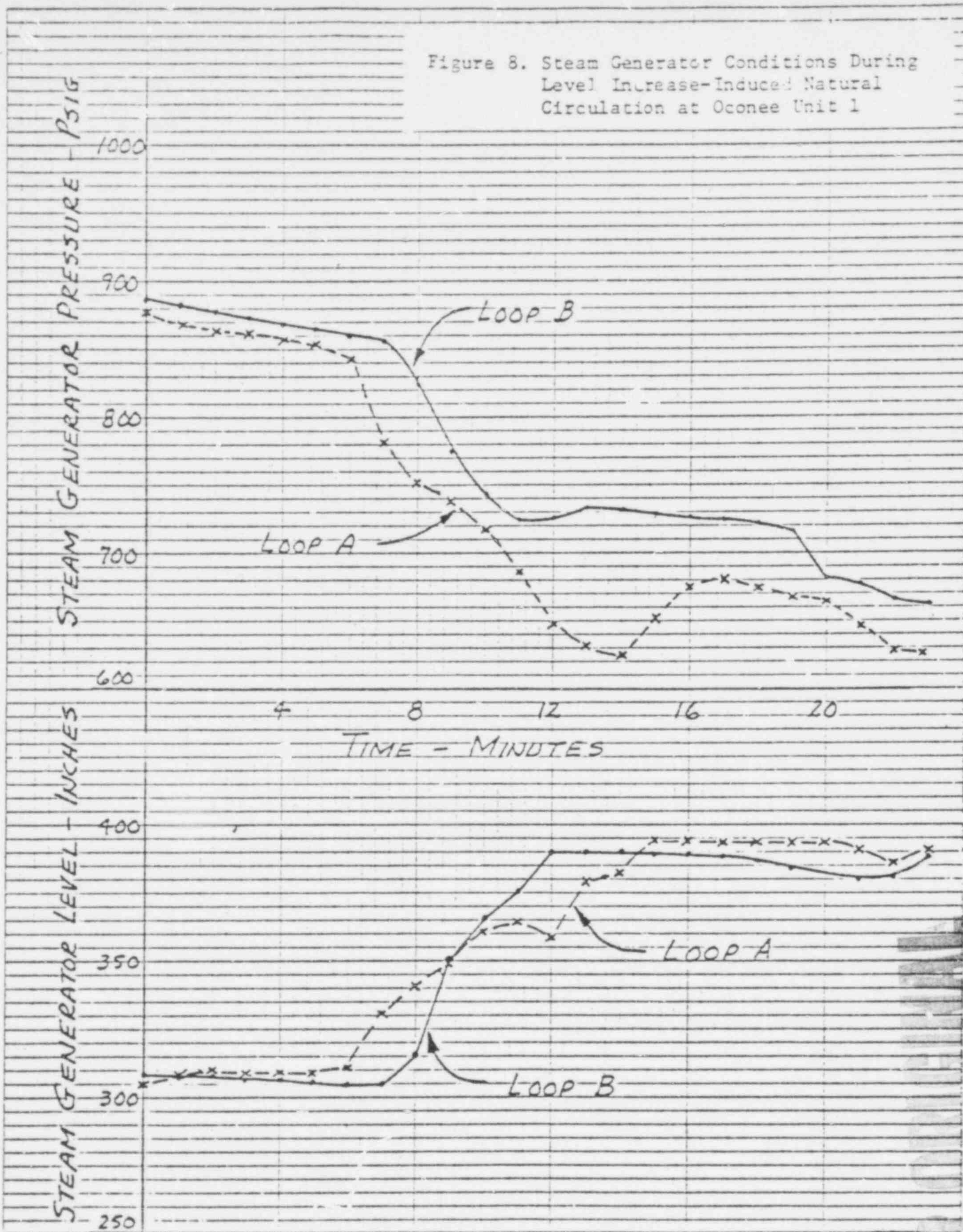

Figure 8. Steam Generator Conditions During Level Increase-Induced Natural Circulation at Oconee Unit 1

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Figure 9. Steam Generator Conditions During Level Increase-Induced Natural Circulation at Oconee Unit 1

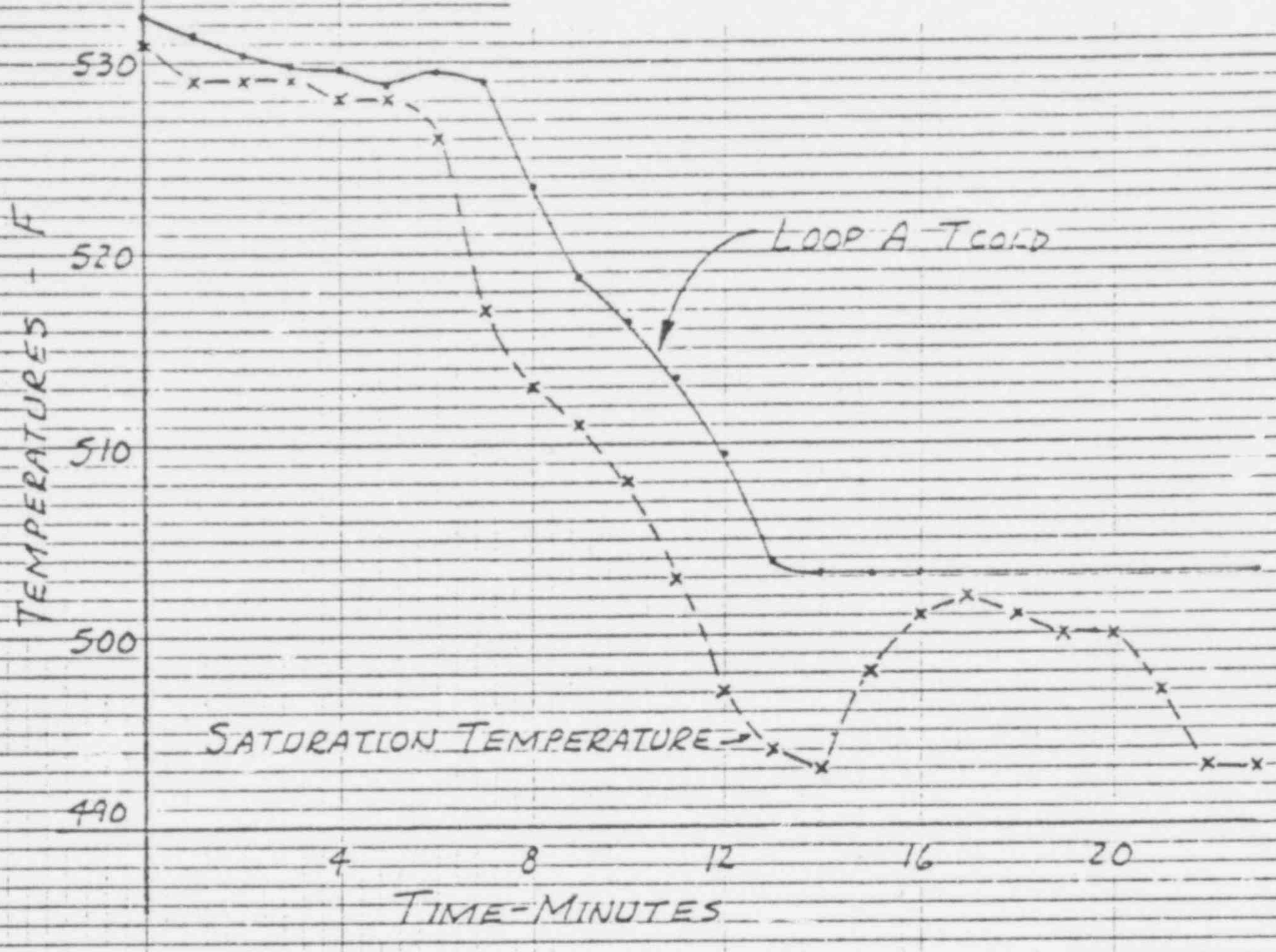

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Figure 10. Loop A and B RC Temperatures During Natural Circulation at Oconee Unit 1

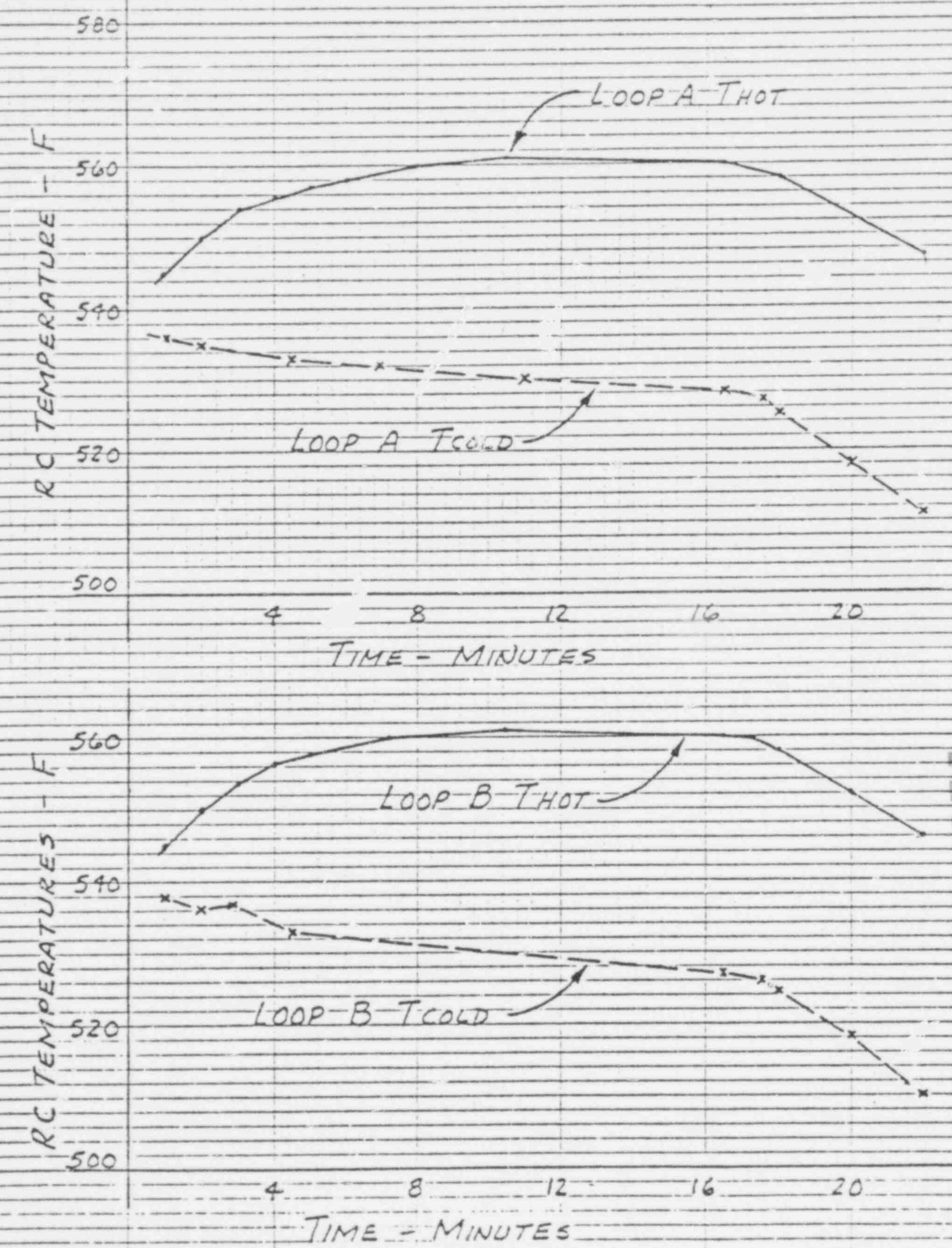

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Figure 11. Loop A and B RC Temperatures During Planned Test of Natural Circulation at Oconee Unit 1

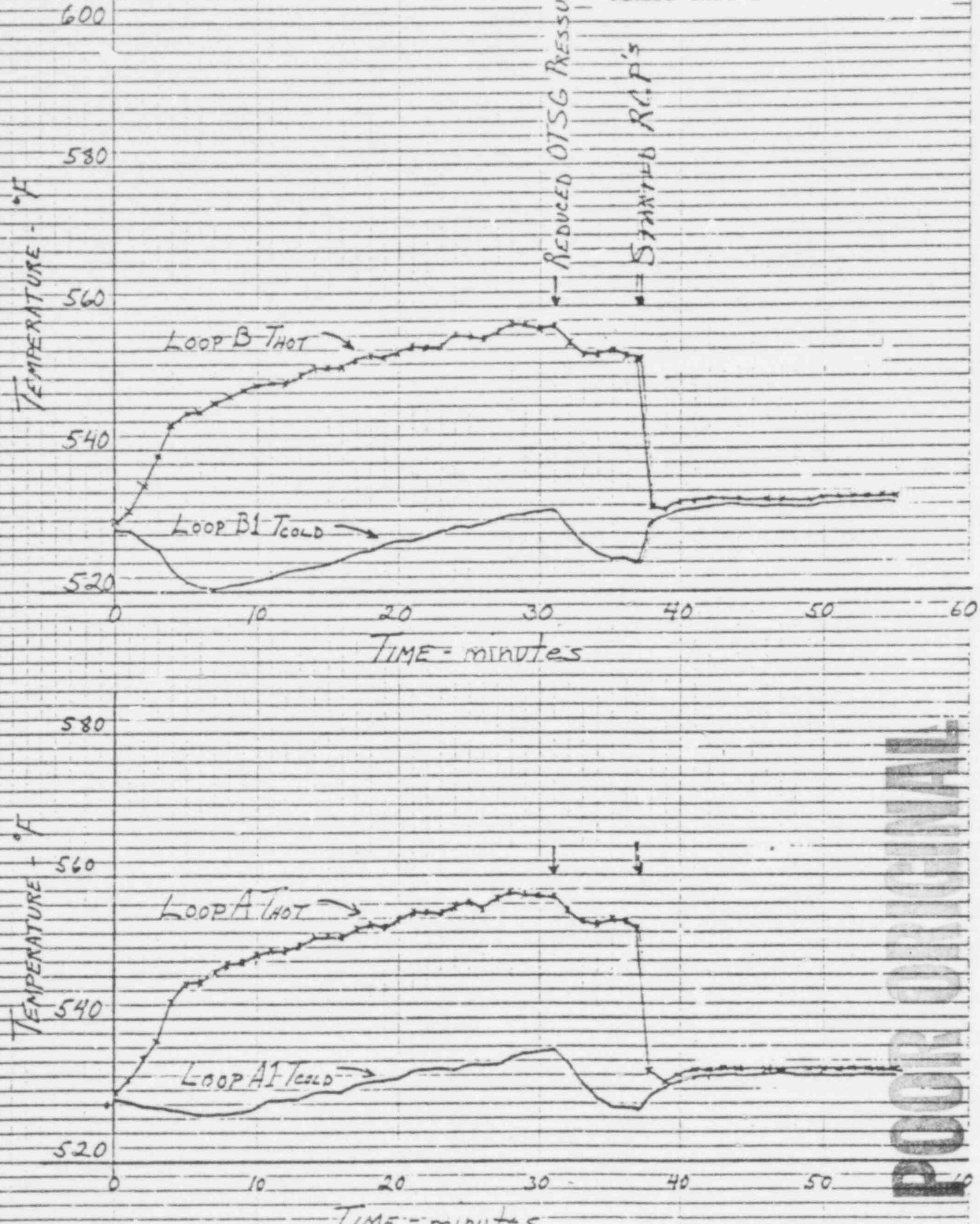


Figure 12. RC System Parameters During Planned Test of Natural Circulation at Oconee Unit 1

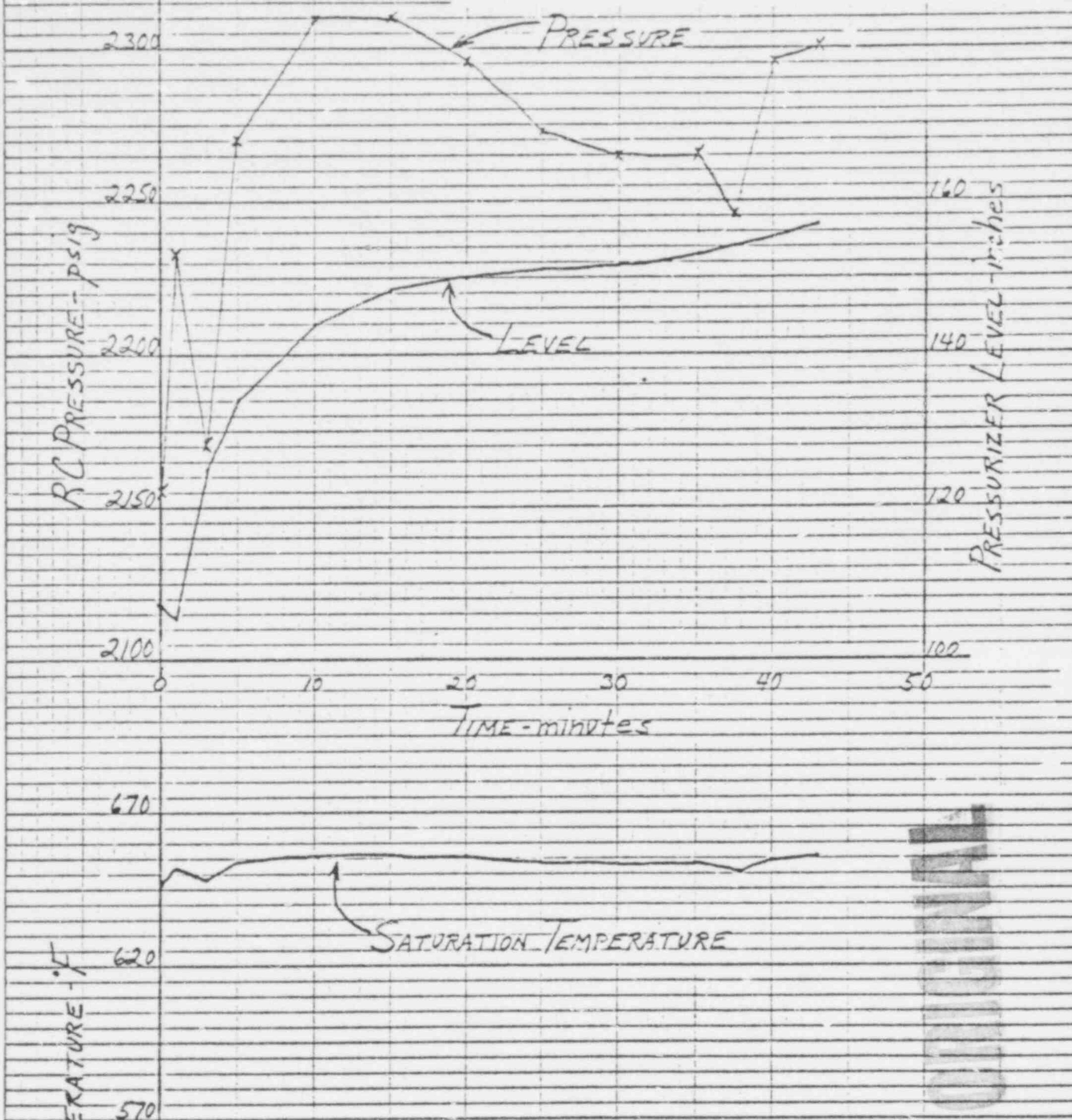
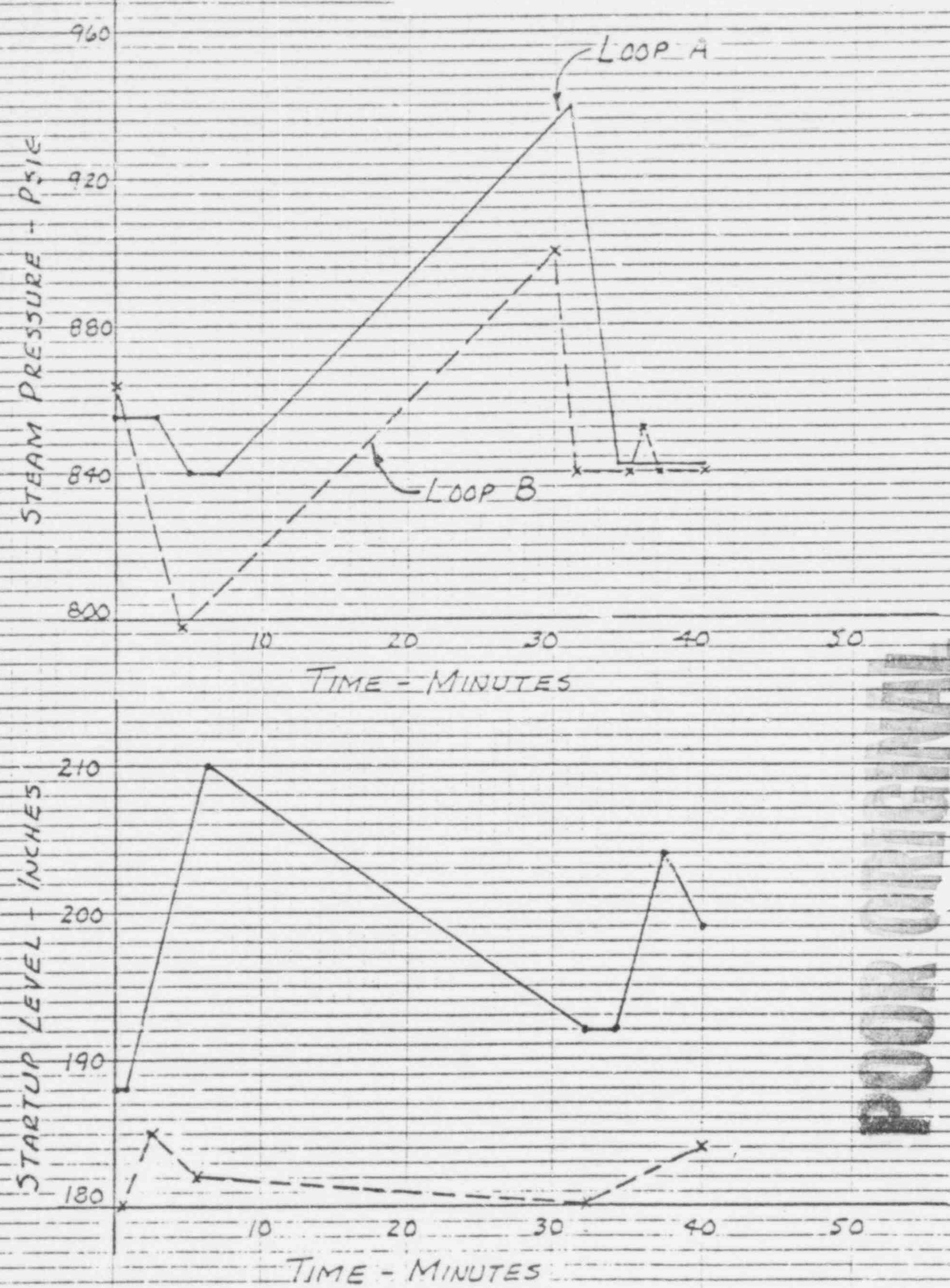


Figure 13. Steam Generator Conditions During Planned Test of Natural Circulation at Oconee Unit 1



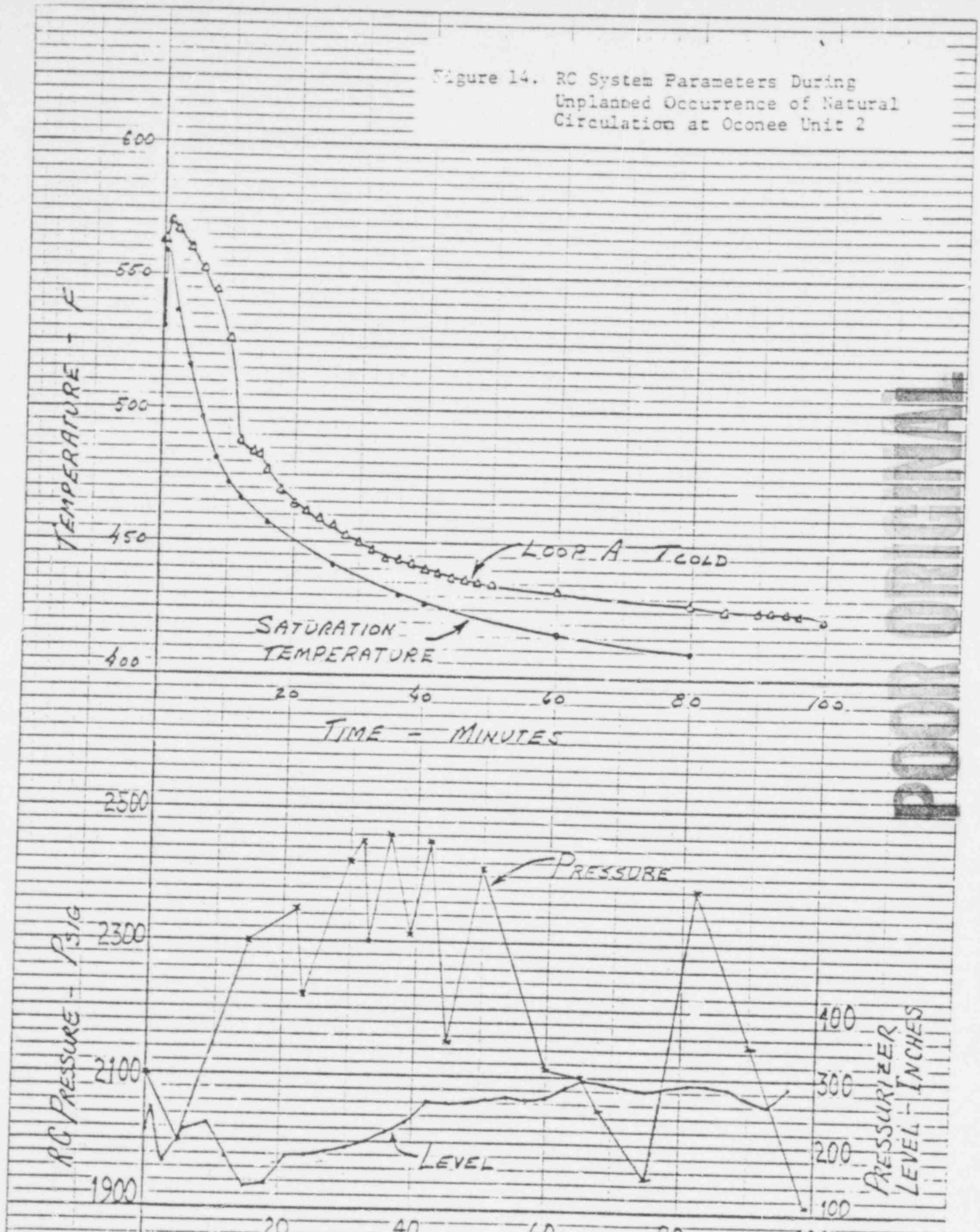


Figure 14. RC System Parameters During Unplanned Occurrence of Natural Circulation at Oconee Unit 2

Figure 15. RC System Parameters and Saturation Temperature During Unplanned Occurrence of Natural Circulation at Oconee Unit 2

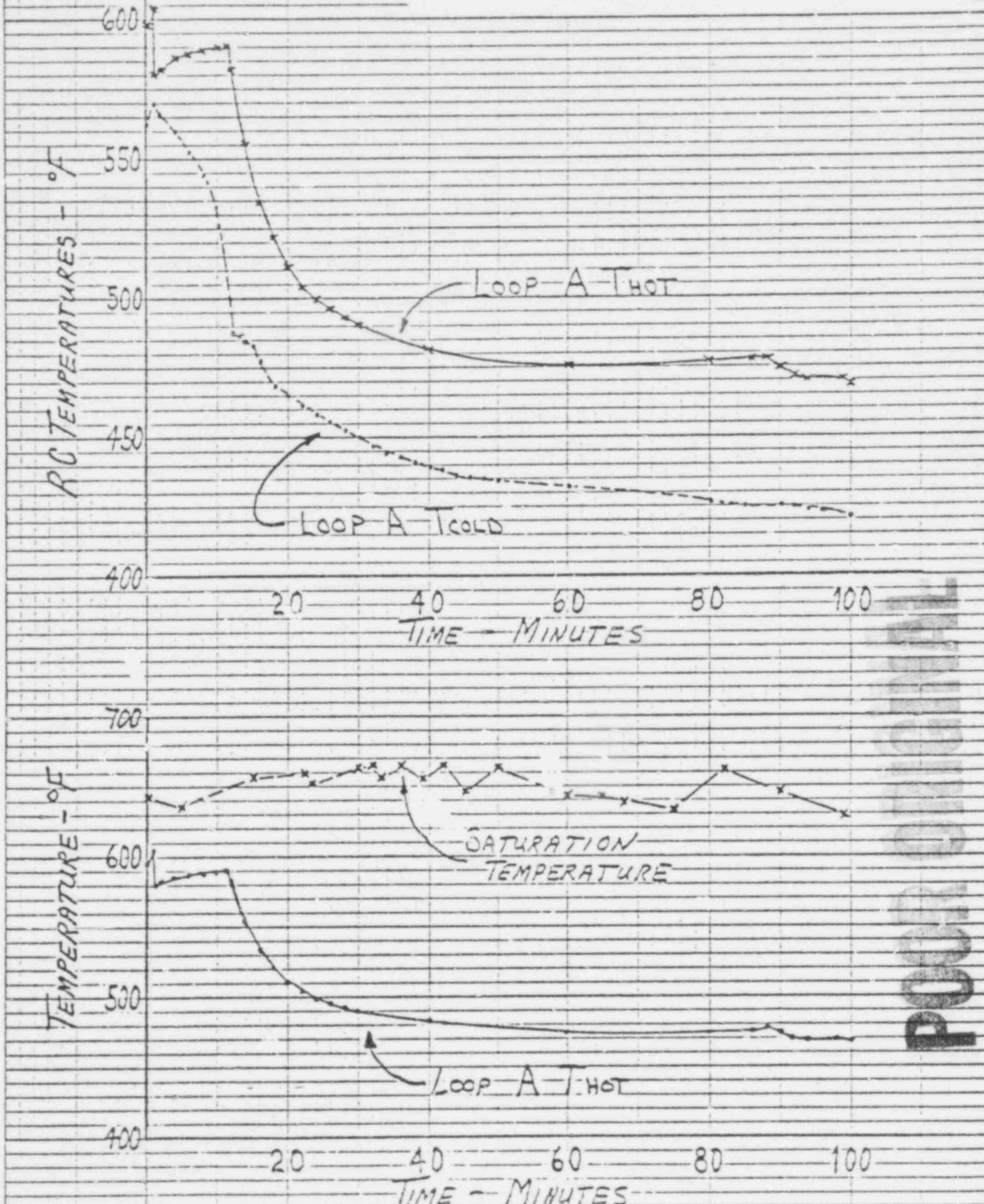


Figure 16. Steam Generator Conditions During Unplanned Occurrence of Natural Circulation at Oconee Unit 2

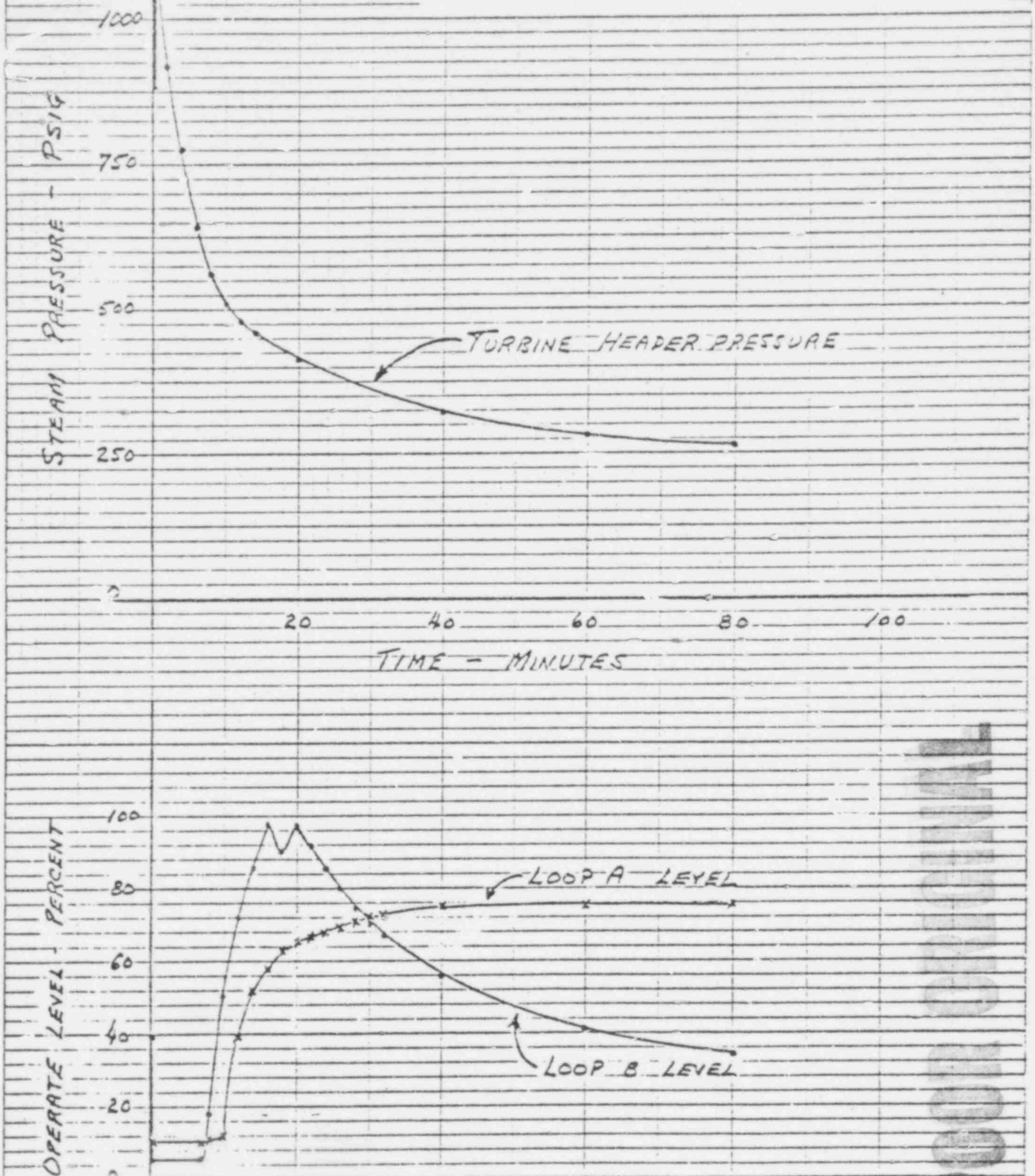
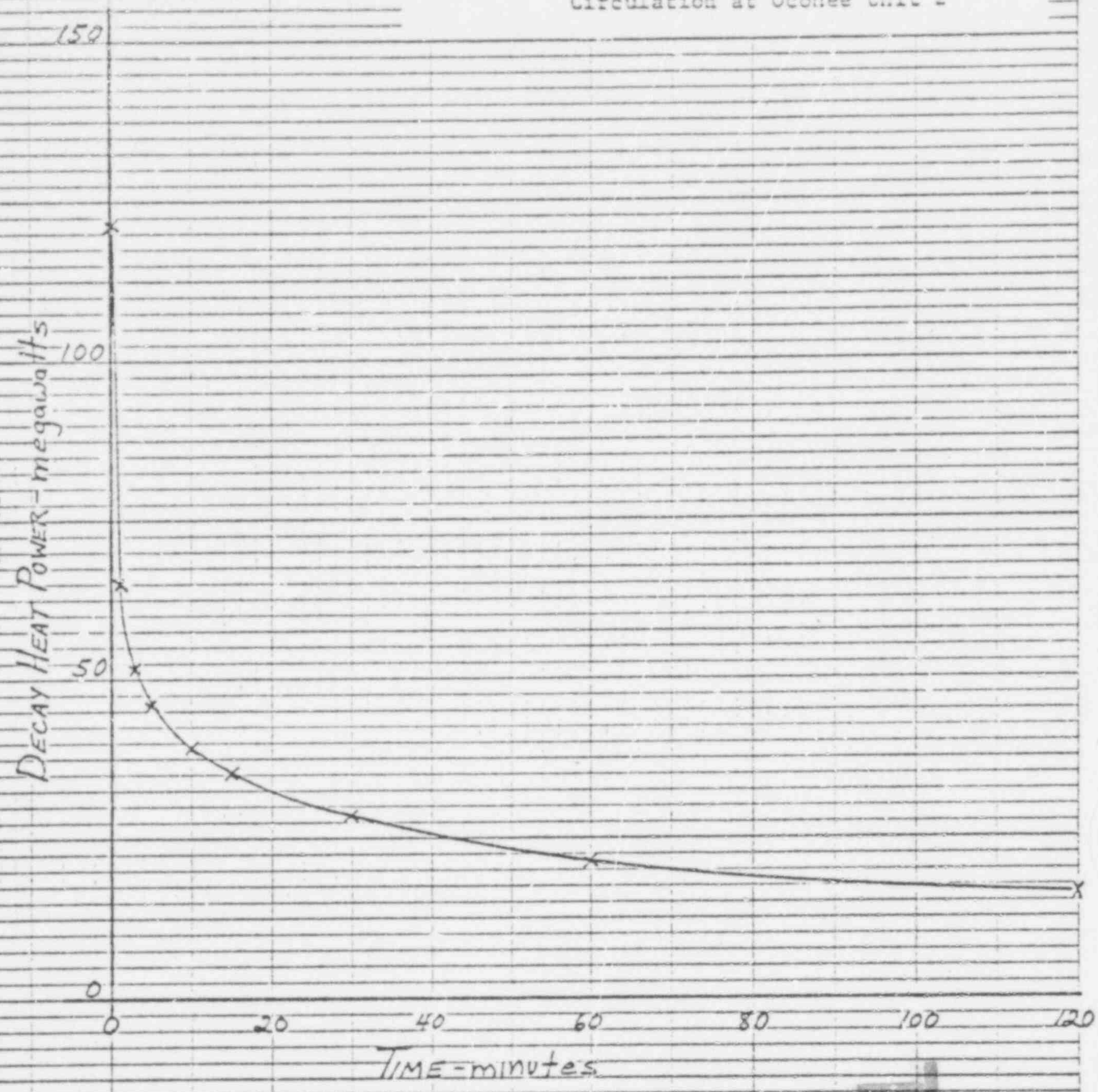


Figure 17. Calculated Core Decay Heat During Unplanned Occurrence of Natural Circulation at Oconee Unit 2



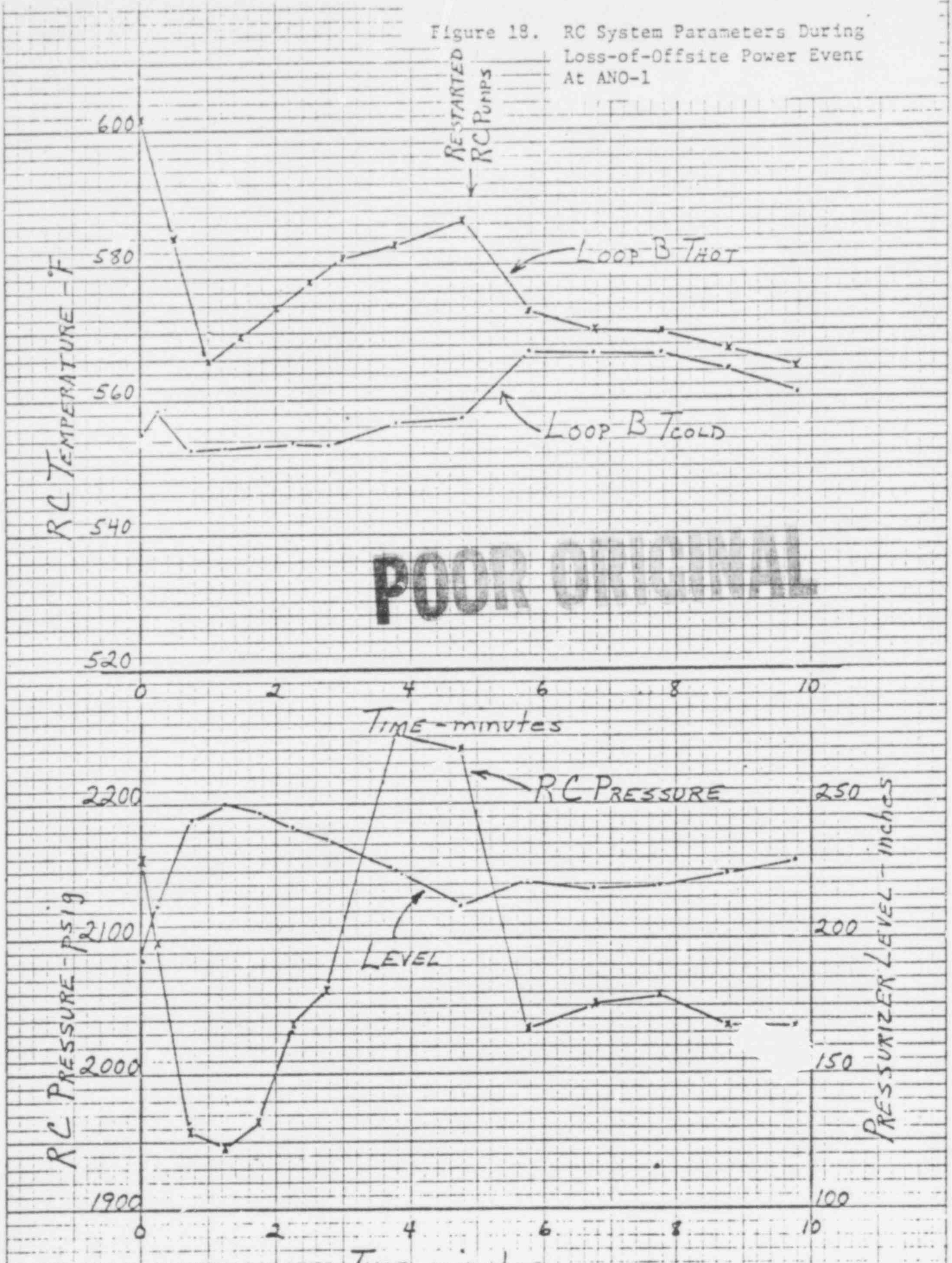


Figure 18. RC System Parameters During Loss-of-Offsite Power Event At ANO-1

Figure 19. RC System Temperatures and Saturation Temperature During Loss of Offsite Power Event at ANO-1

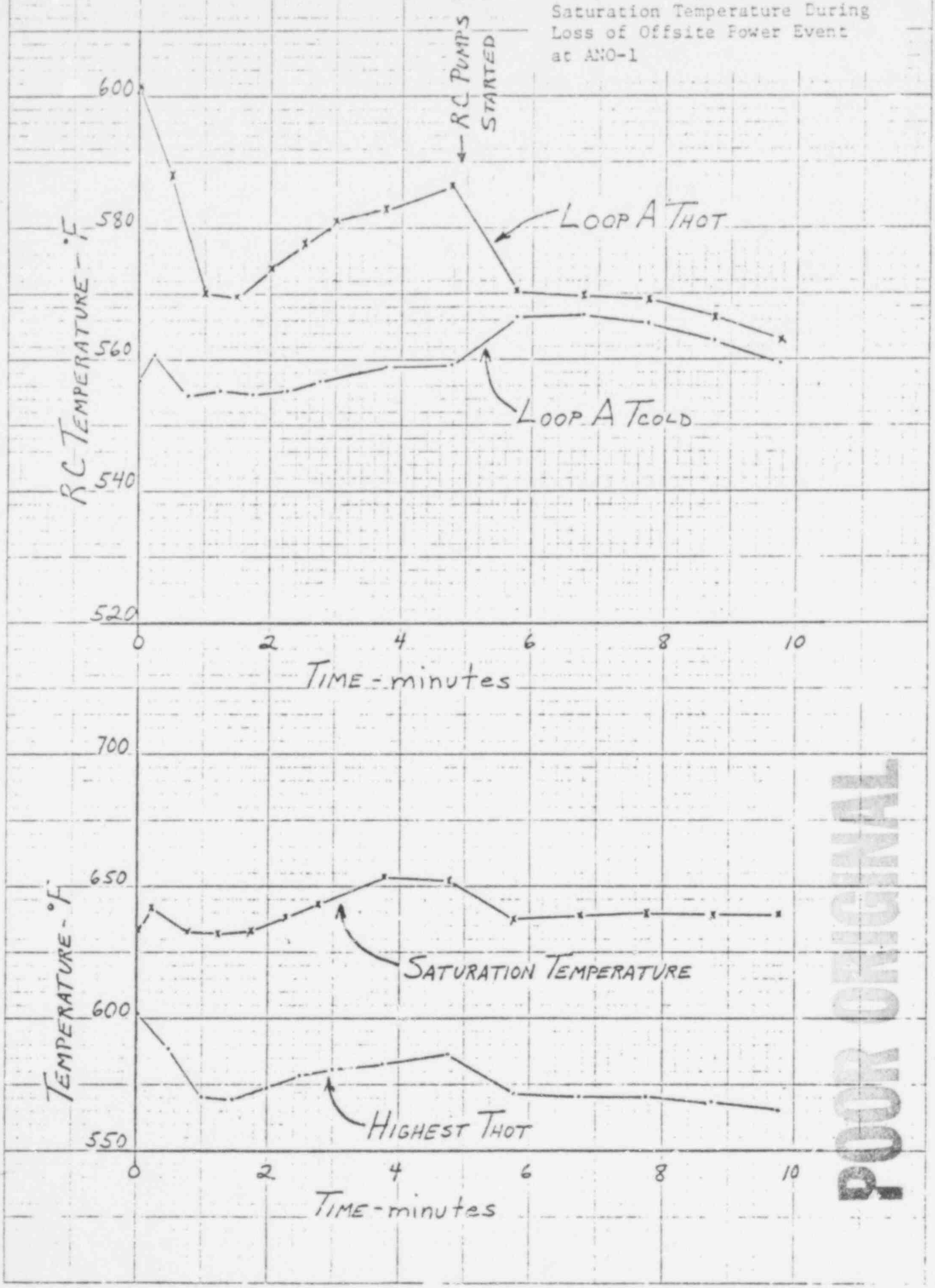


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Figure 20. Steam Generator Conditions During Loss of Offsite Power Event at ANO-1

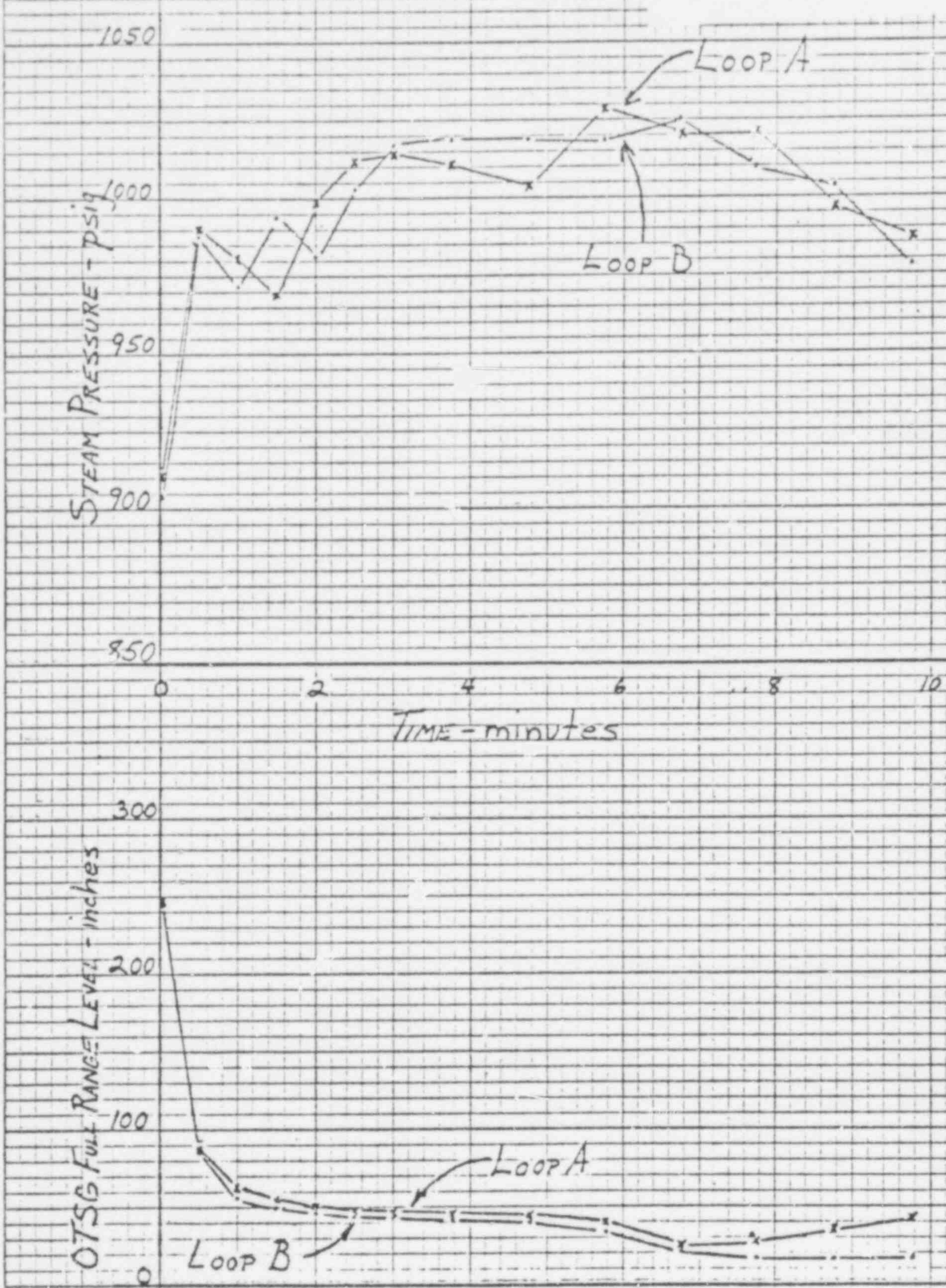


Figure 21. Steam Generator Outlet and Saturation Temperatures During Loss of Offsite Power Event at ANO-1

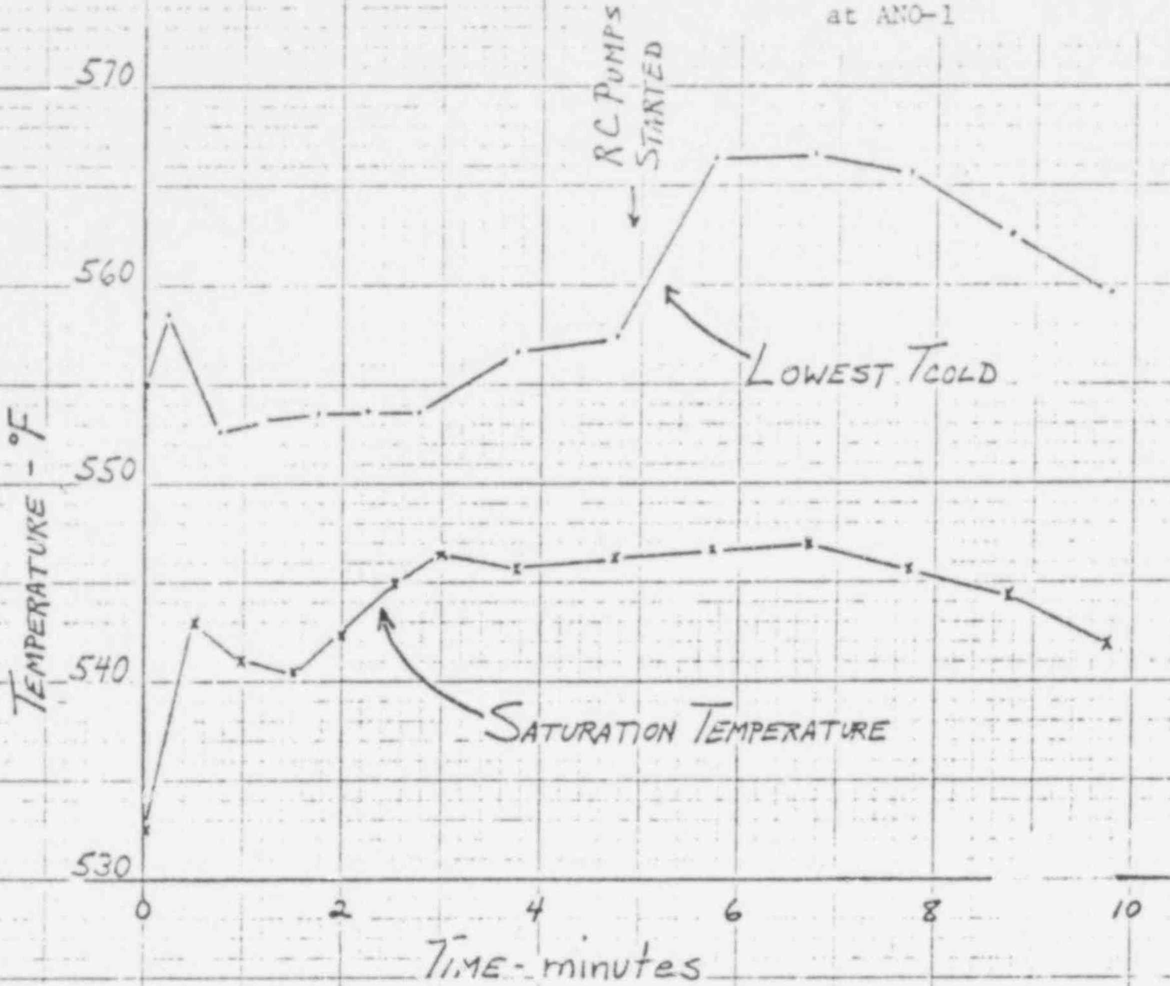


Figure 22. Calculated Core Decay Heat During Loss of Offsite Power Event at ANO-1

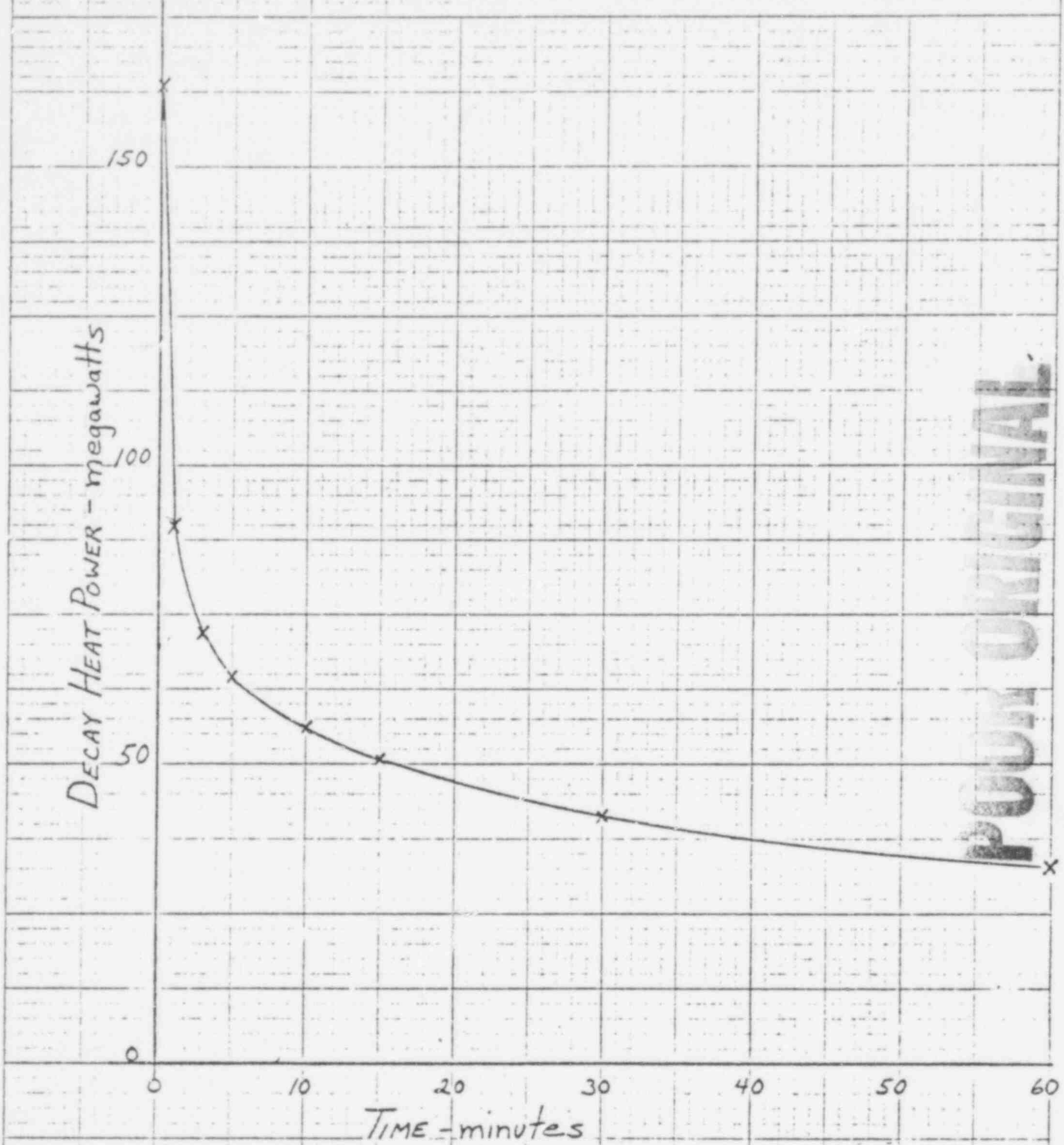


Figure 23 RC System Parameters During Loss of Offsite Power Test With Natural Circulation at Crystal River 3

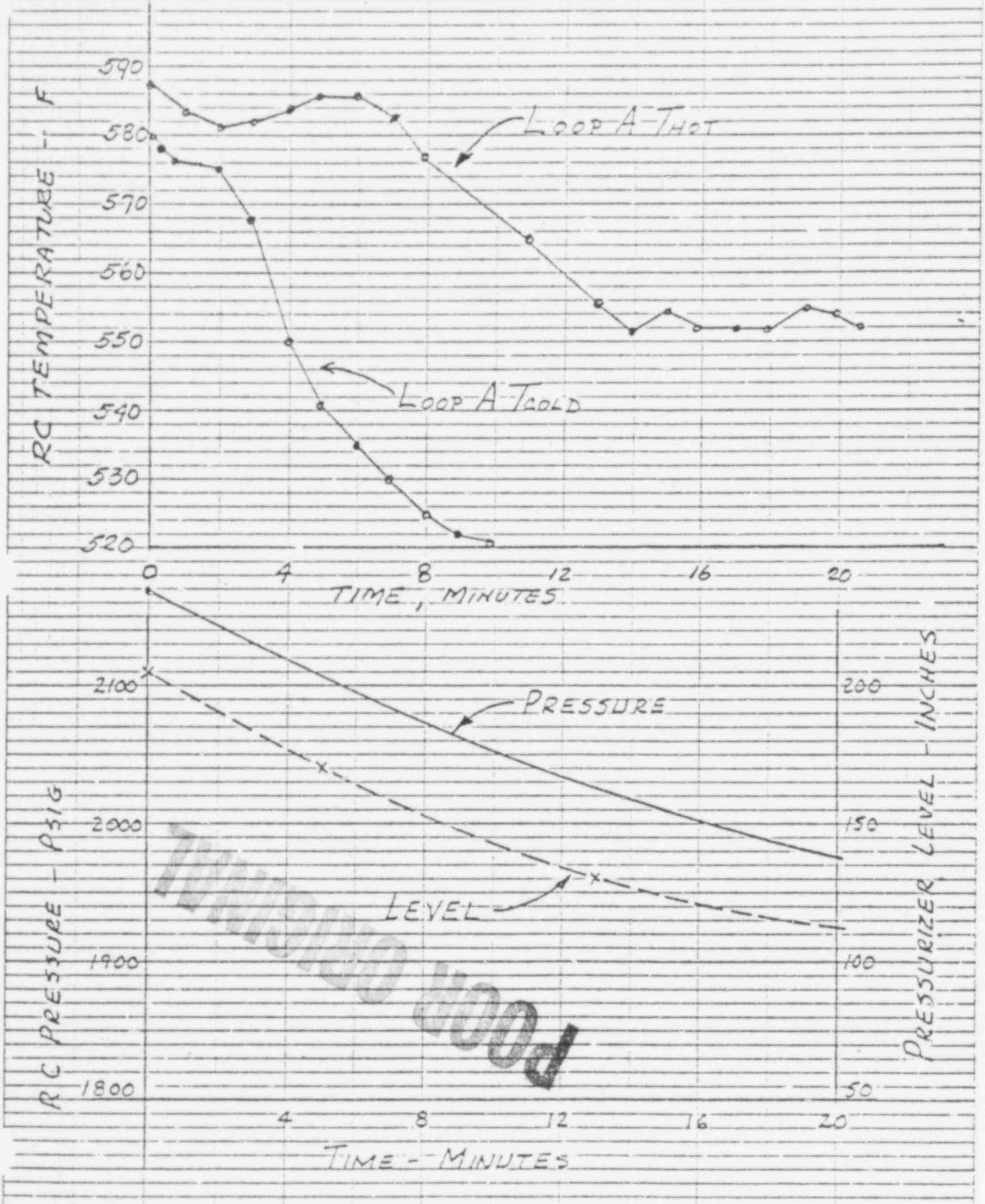


Figure 24 RC System Temperatures and Saturation Temperature During Loss of Offiste Power Test With Natural Circulation at Crystal River 3

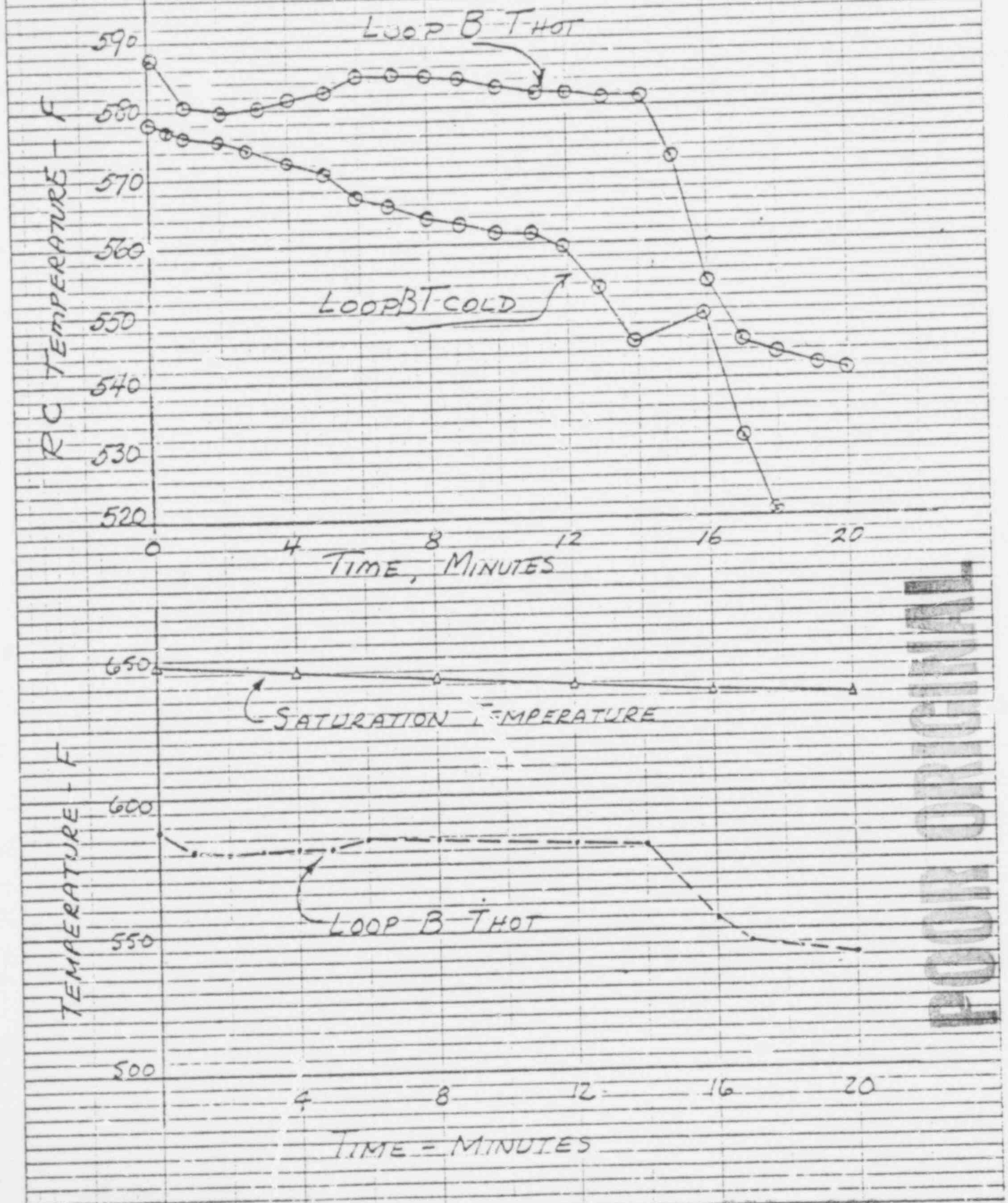


Figure 25 Steam Generator Conditions During Loss of Offsite Power Test With Natural Circulation at Crystal River 3

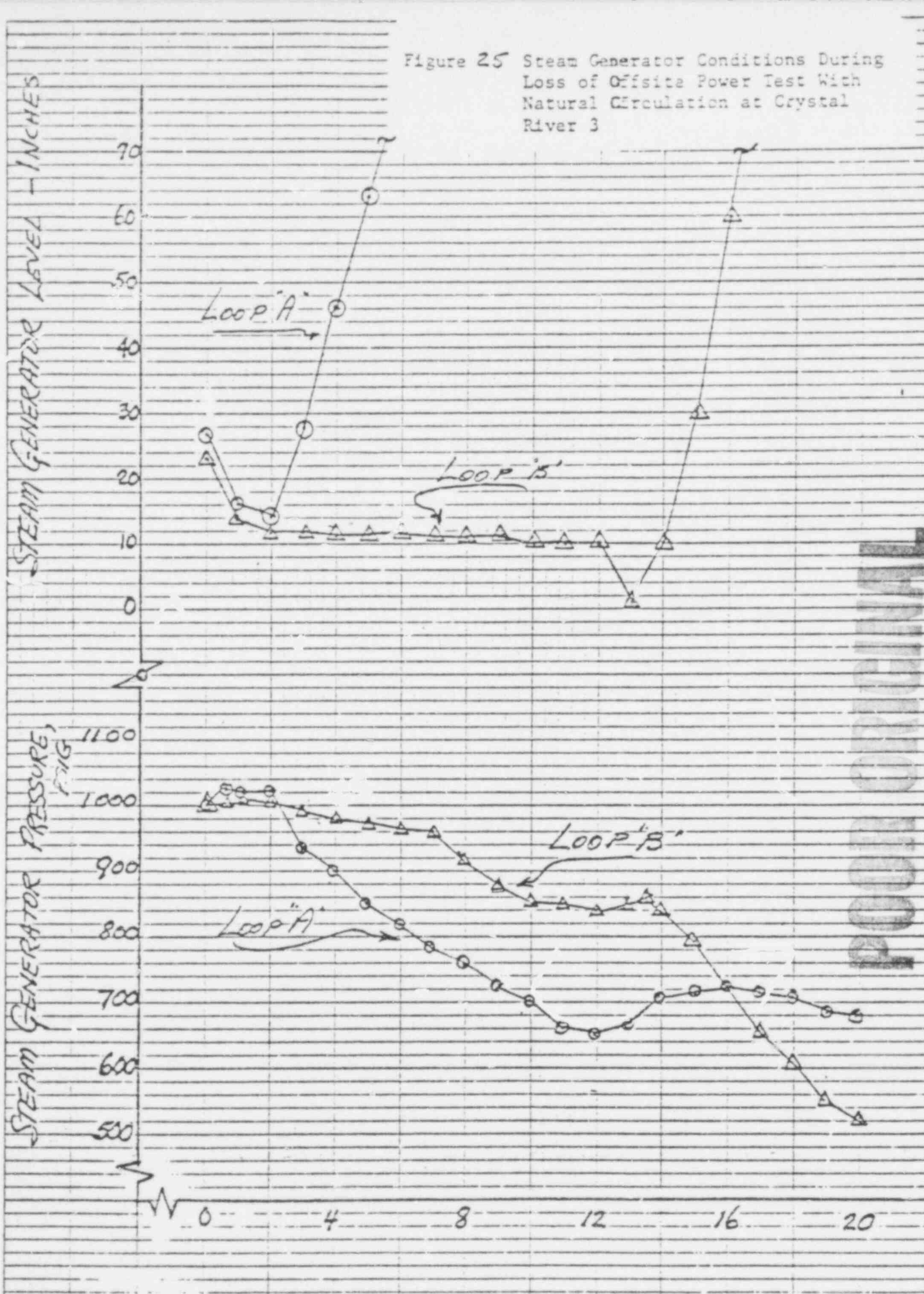
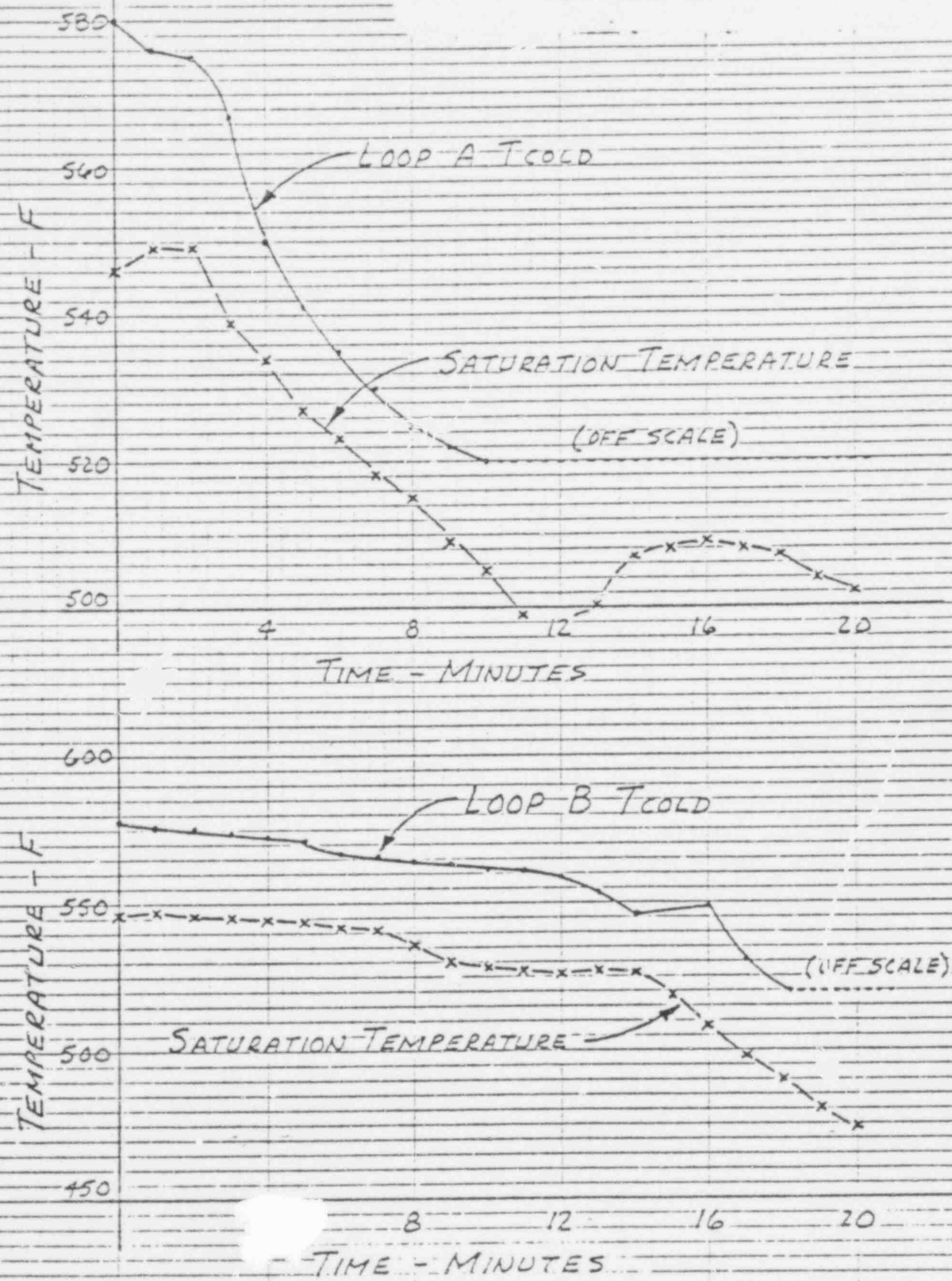


Figure 26 Steam Generator Outlet and Saturation Temperatures During Loss of Offsite Power Test With Natural Circulation at Crystal River 3



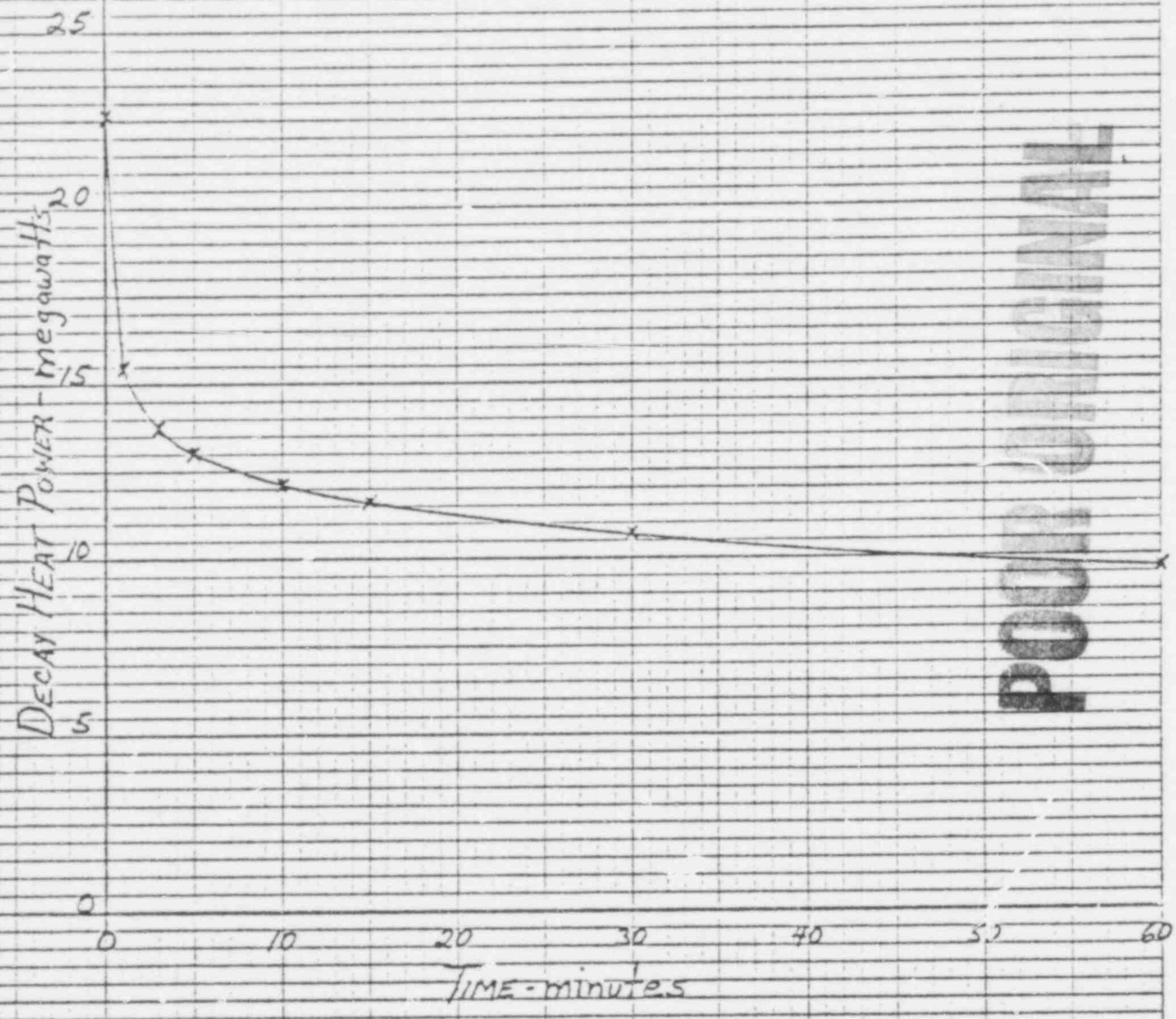


Figure 27 Calculated Core Decay Heat During Loss of Offsite Power Test With Natural Circulation at Crystal River 3

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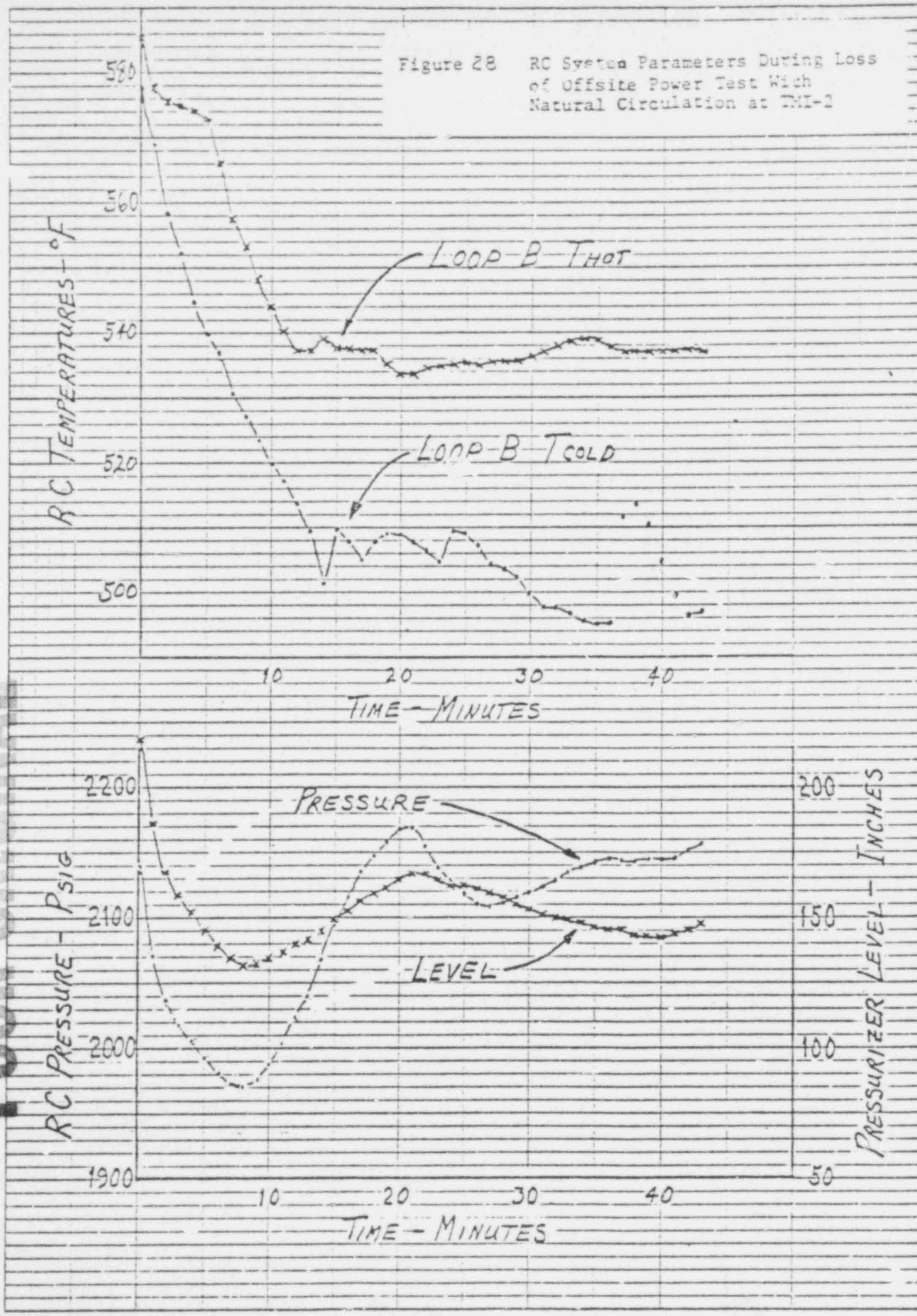

Figure 28 RC System Parameters During Loss of Offsite Power Test With Natural Circulation at TMI-2

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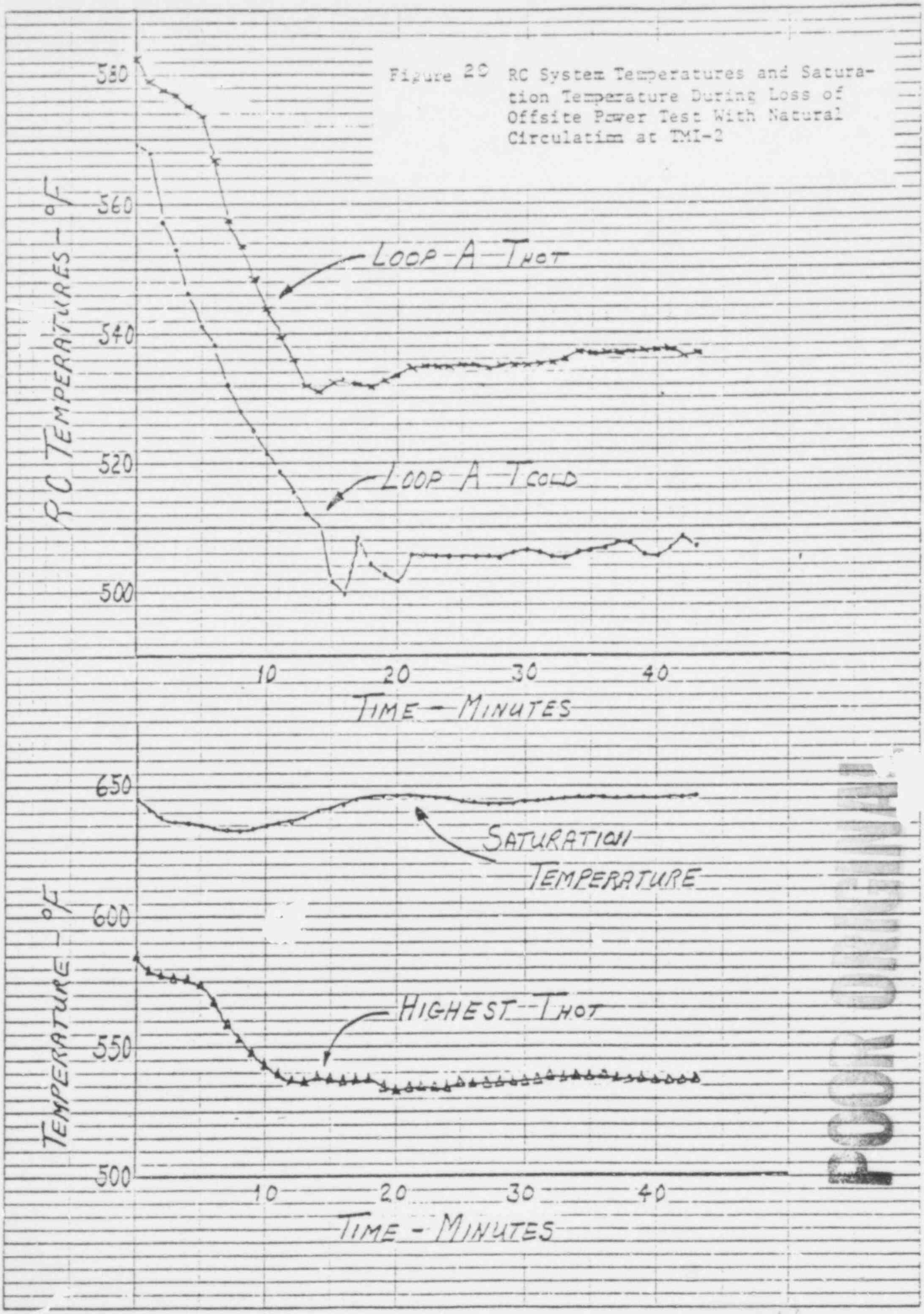

Figure 20 RC System Temperatures and Saturation Temperature During Loss of Offsite Power Test With Natural Circulatim at TMI-2

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Figure 30 Steam Generator Conditions During Loss of Offsite Power Test With Natural Circulation at TMI-2

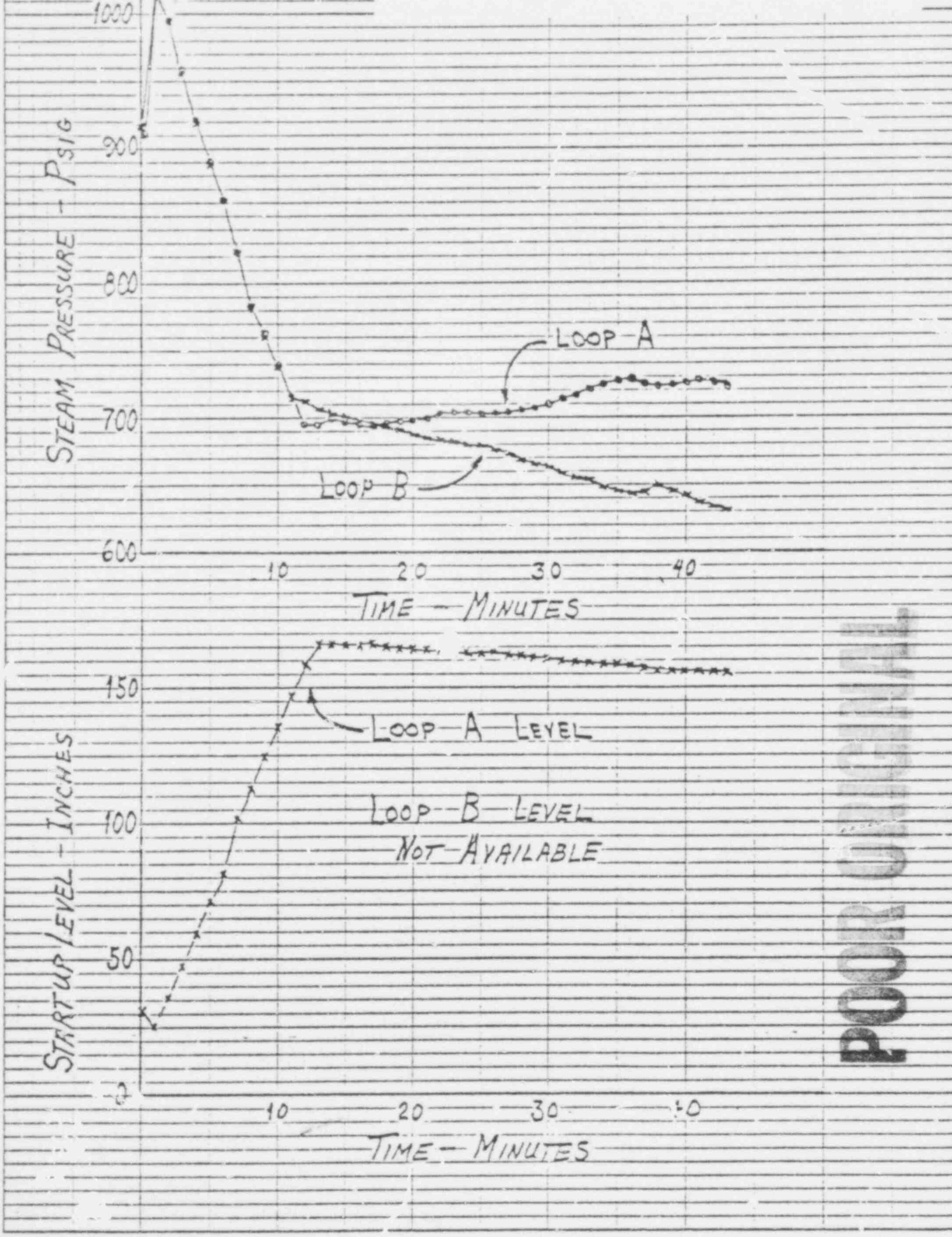
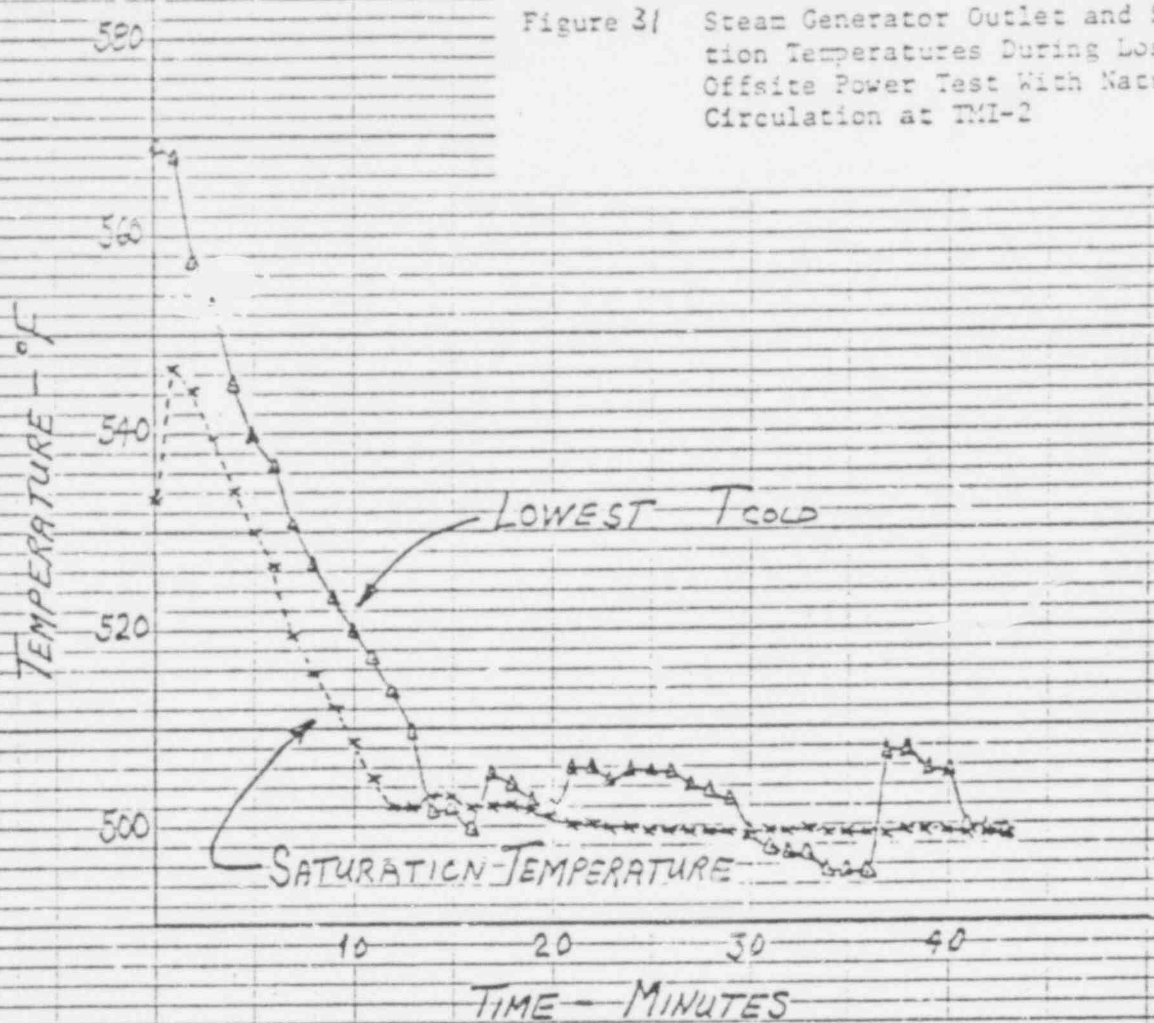


Figure 31 Steam Generator Outlet and Saturation Temperatures During Loss of Offsite Power Test With Natural Circulation at TMI-2



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K&E 10 X 10 TO THE INCH • 7 X 10 INCHES KEUFFEL & ESSER CO.

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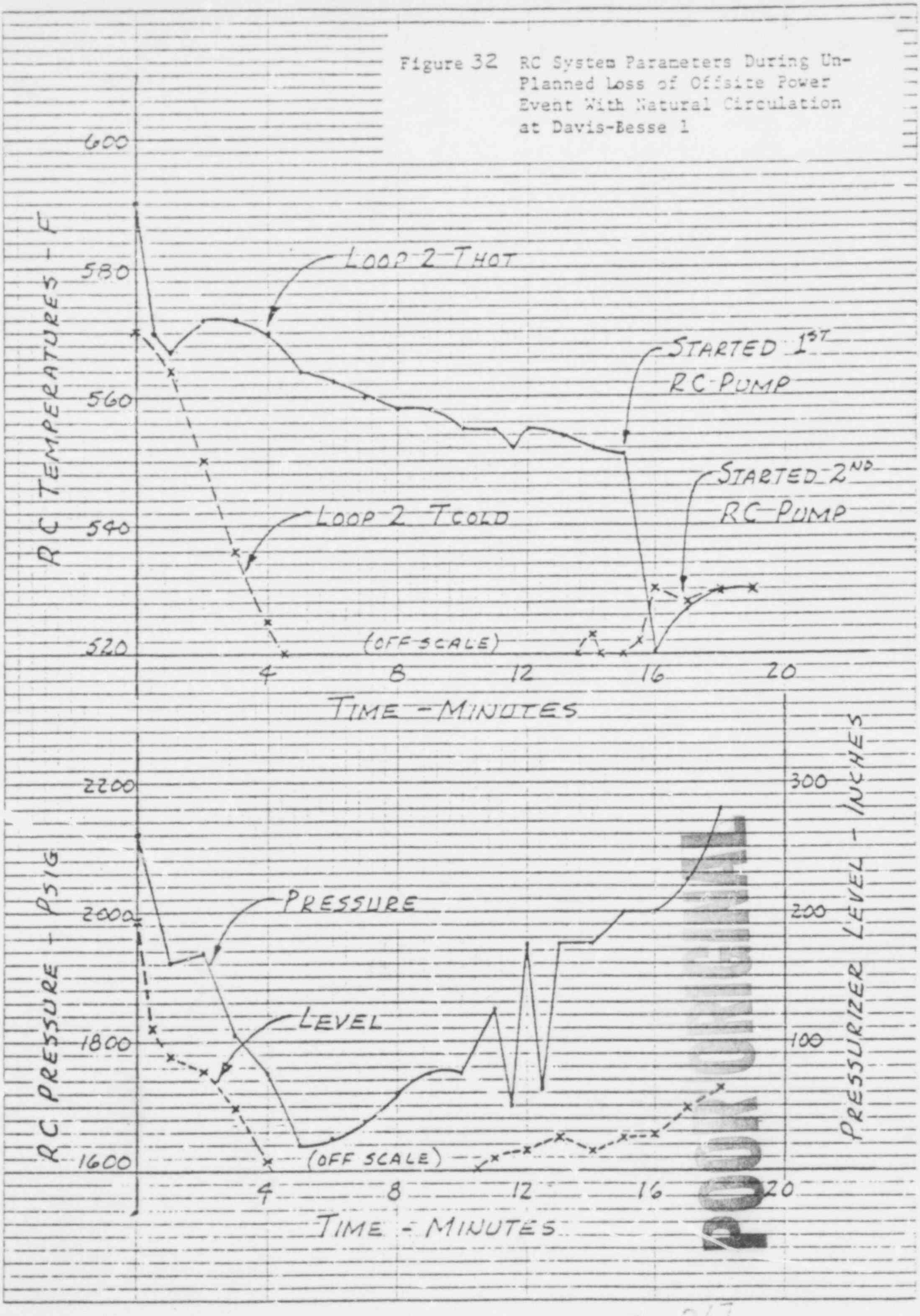


Figure 32 RC System Parameters During Unplanned Loss of Offsite Power Event With Natural Circulation at Davis-Besse 1

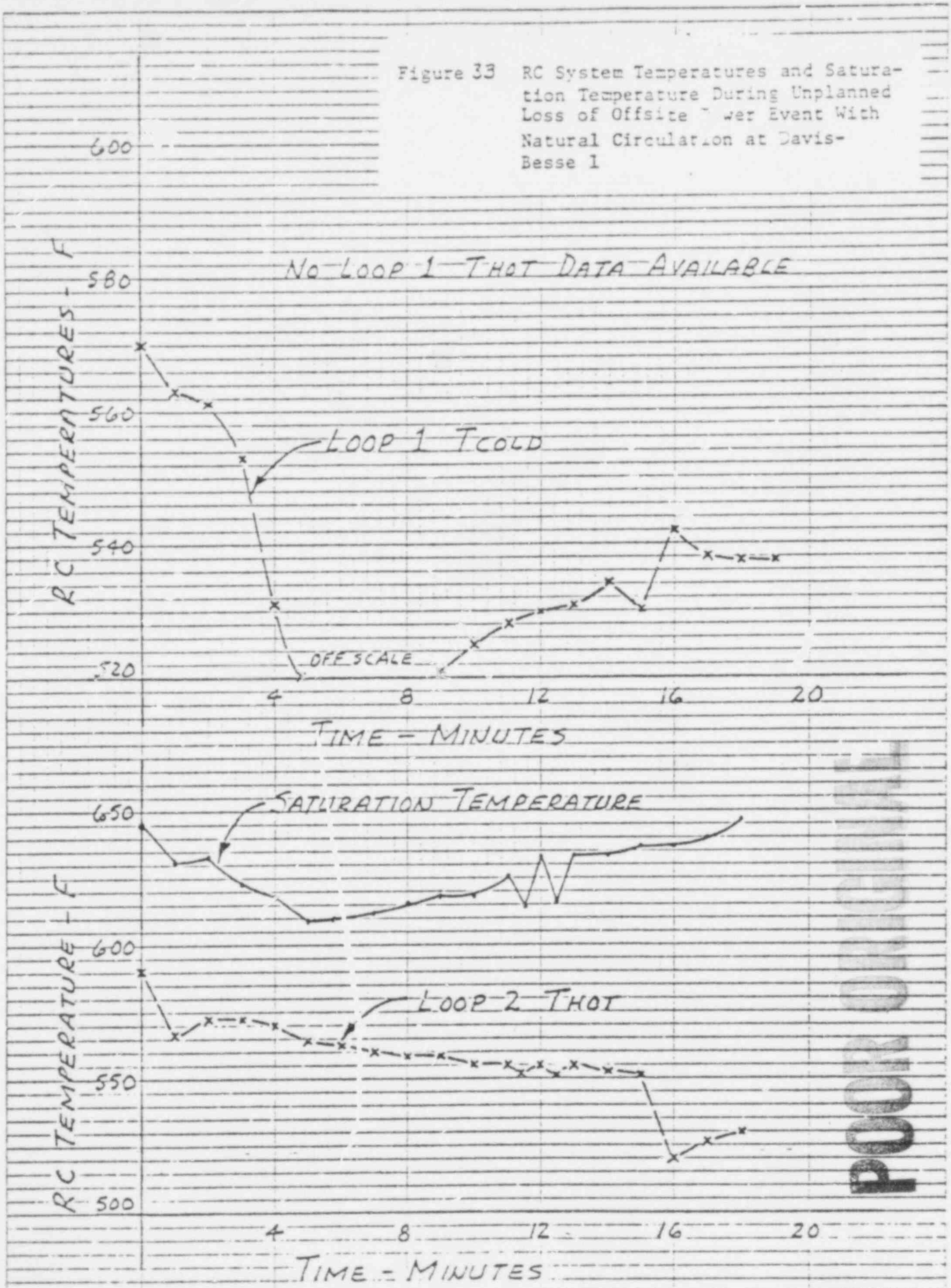


Figure 33 RC System Temperatures and Saturation Temperature During Unplanned Loss of Offsite Power Event With Natural Circulation at Davis-Besse 1

Figure 34 Steam Generator Conditions During Unplanned Loss of Offsite Power Event With Natural Circulation at Davis-Besse 1

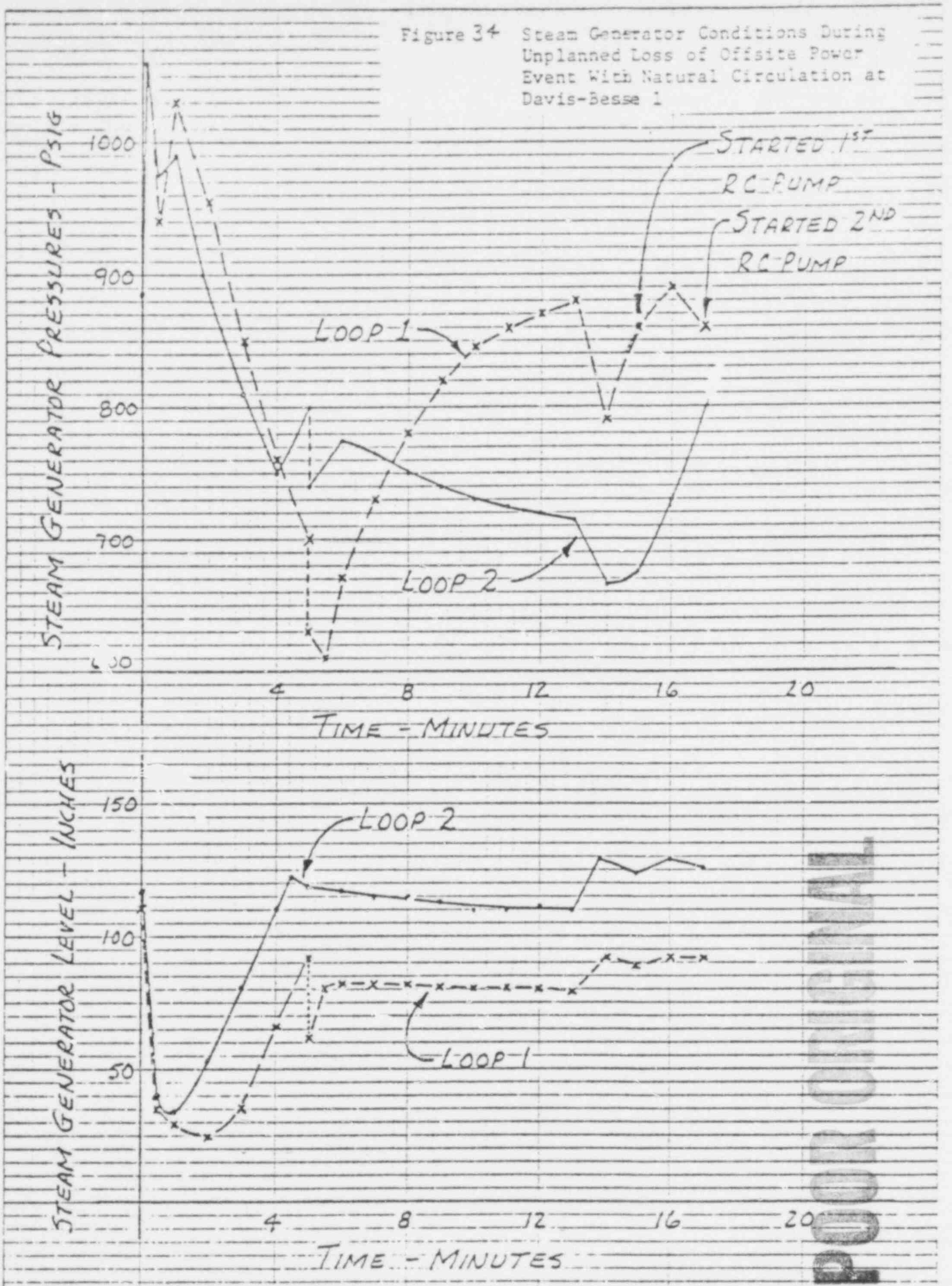
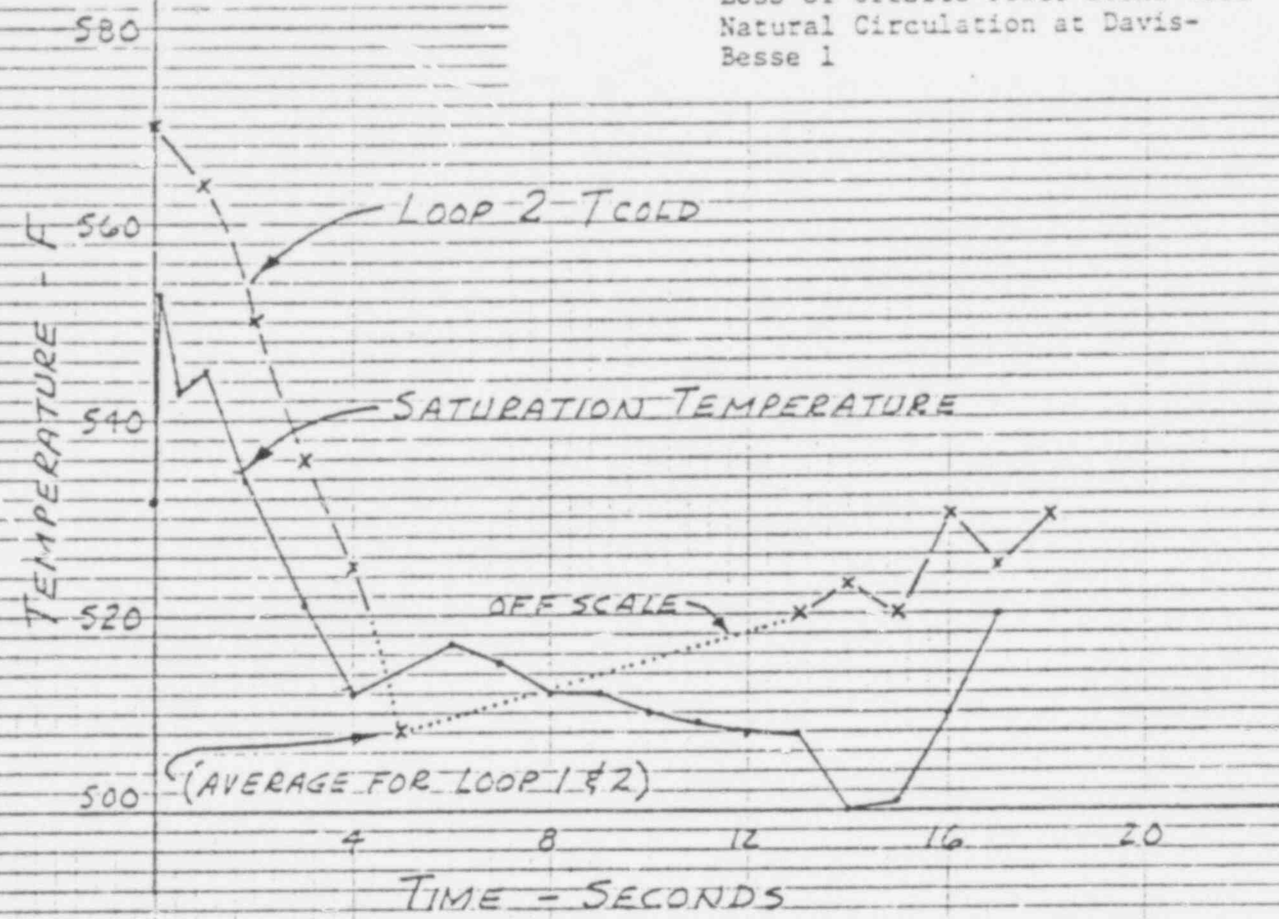


Figure 35 Steam Generator Outlet and Saturation Temperatures During Unplanned Loss of Offsite Power Event With Natural Circulation at Davis-Besse 1



533 270

Figure 36 Calculated Core Decay Heat During Unplanned Loss of Offsite Power Event With Natural Circulation at Davis-Besse 1

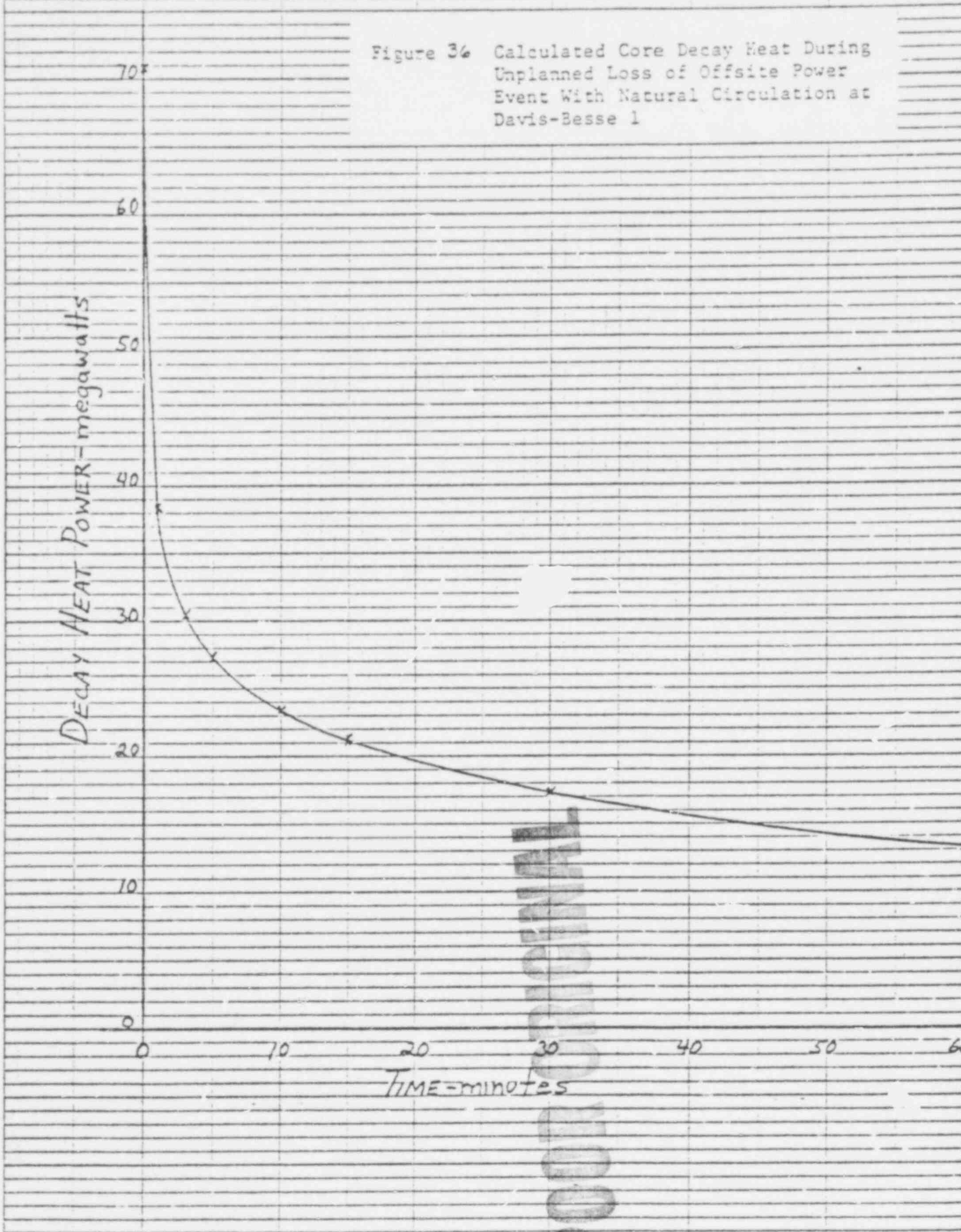
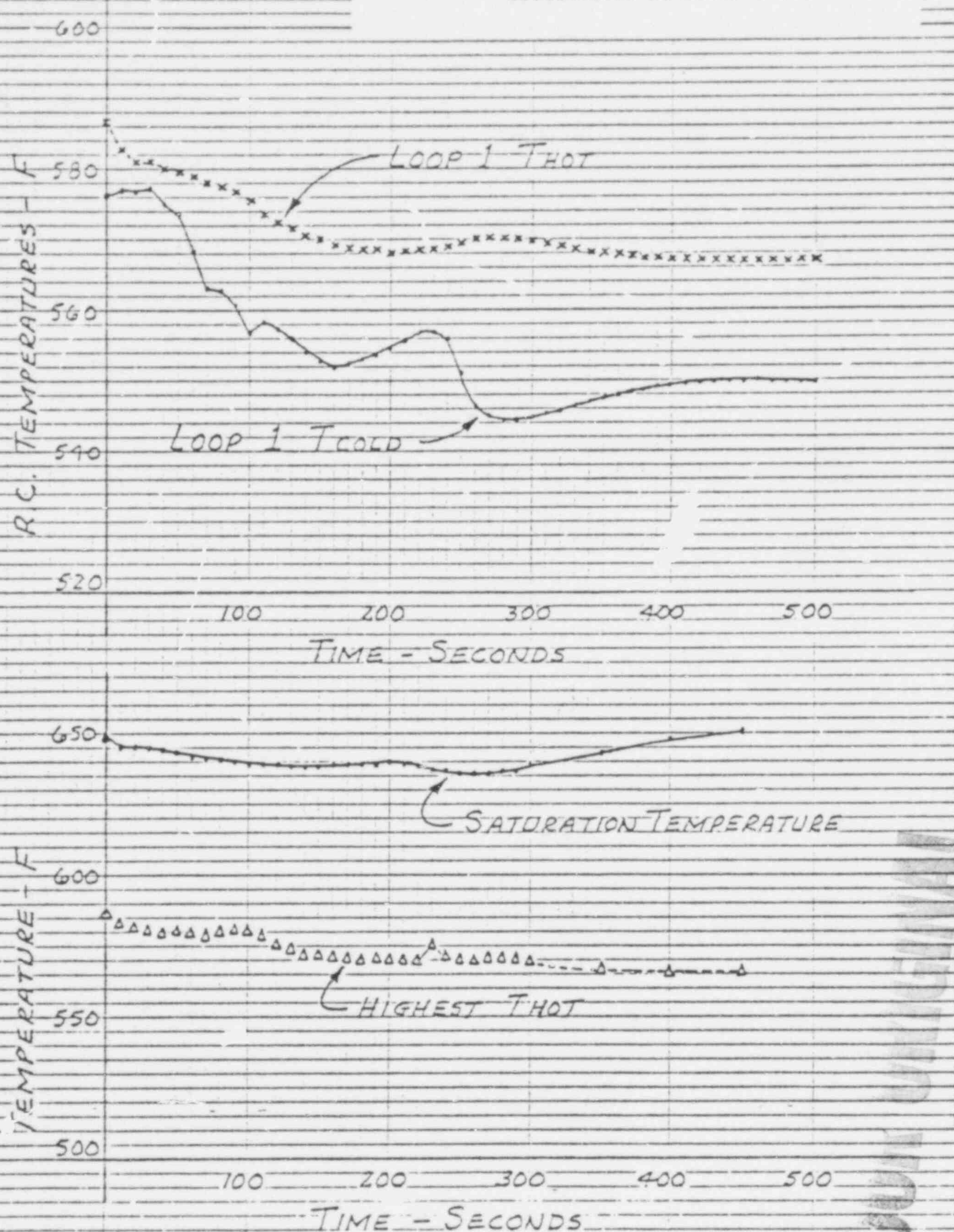


Figure 37 RC System P[illegible]eters Duirng Loss of Offsite Power Test With Natural Circulation at Davis-Besse 1

Figure 38 RC System Temperatures and Saturation Temperature During Loss of Offsite Power Test With Natural Circulation at Davis-Besse 1



533 273

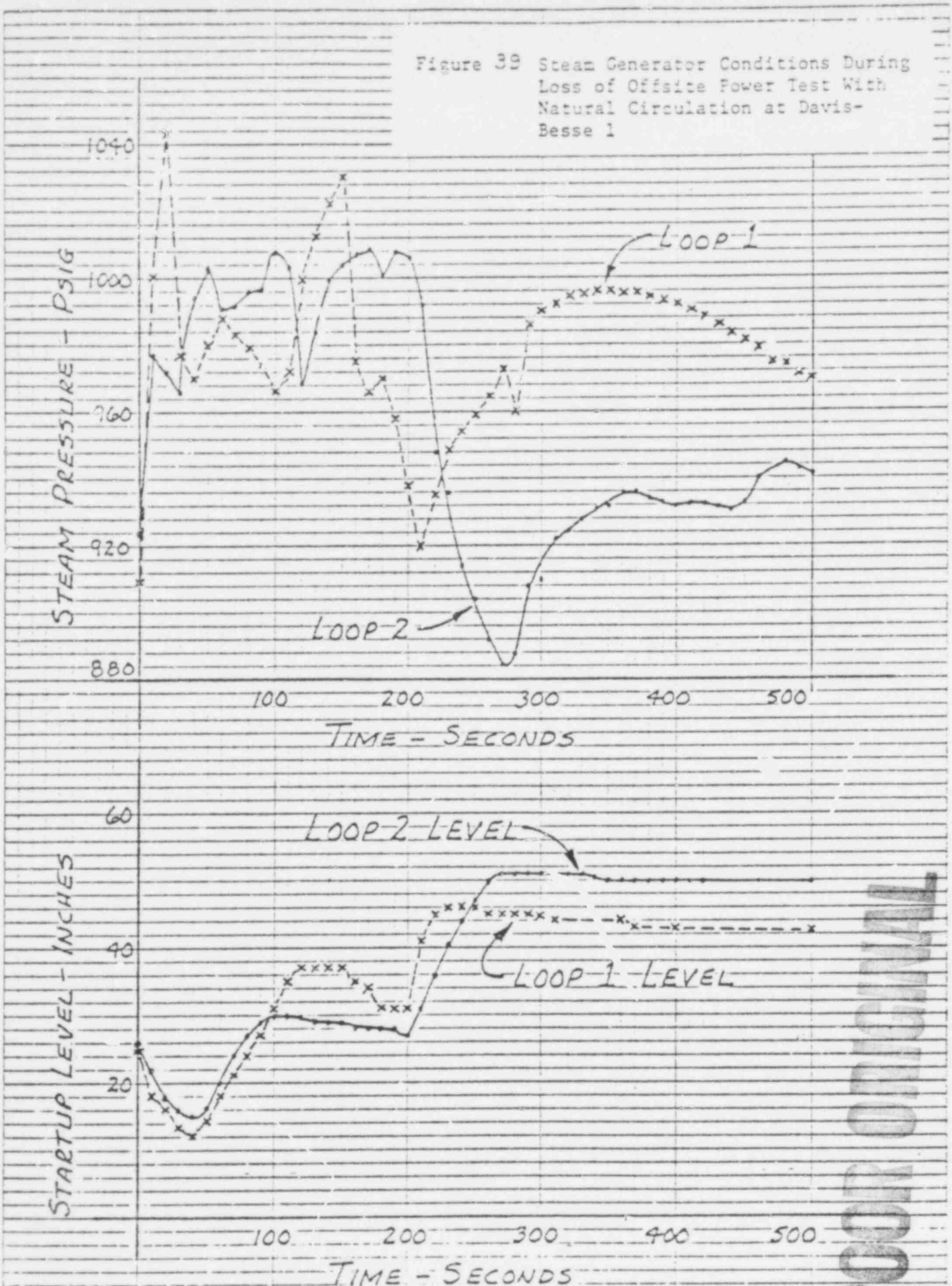


Figure 39 Steam Generator Conditions During Loss of Offsite Power Test With Natural Circulation at Davis-Besse 1

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Figure 40 Steam Generator Outlet and Saturation Temperatures During Loss of Offsite Power Test With Natural Circulation at Davis-Besse 1

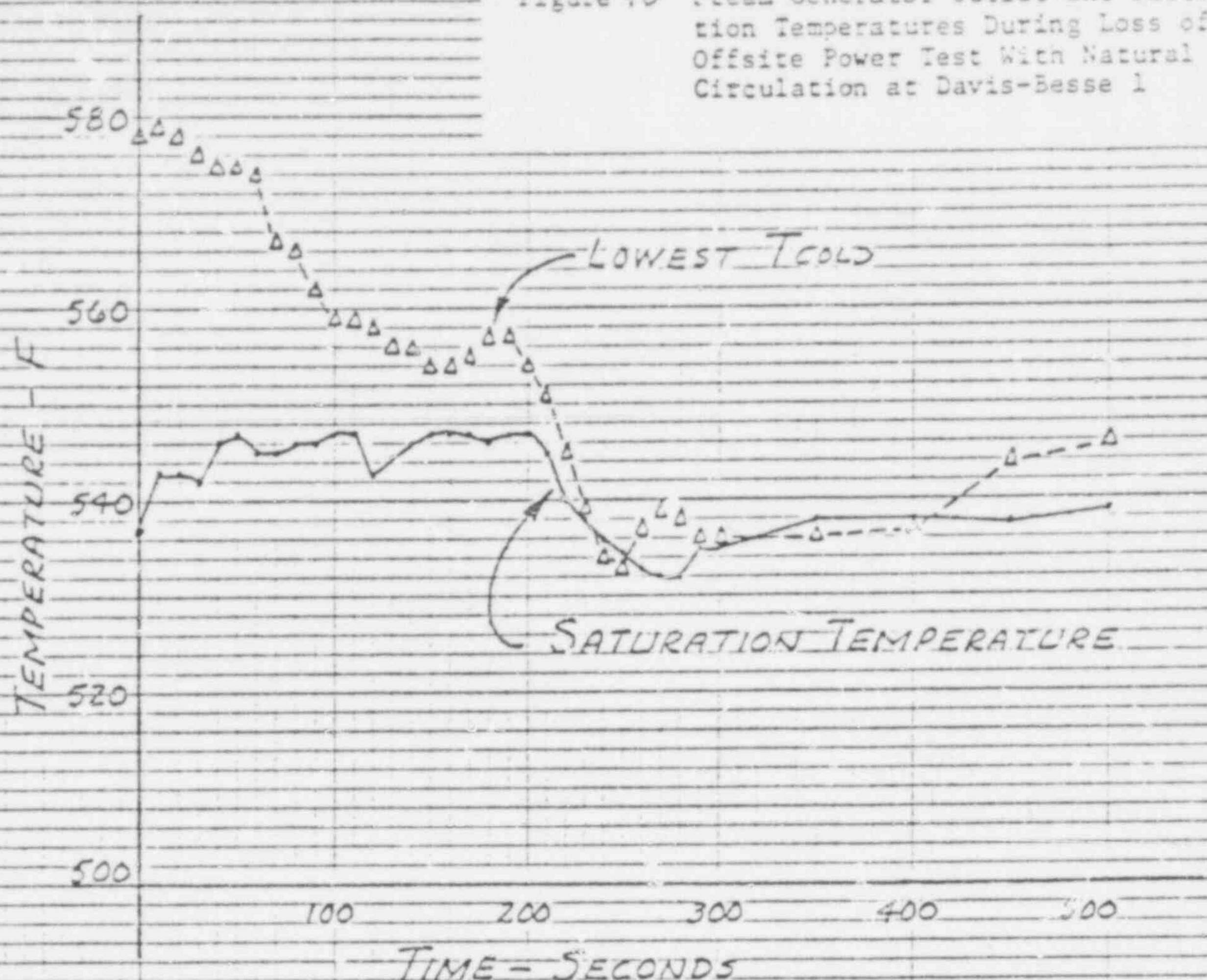

Figure 4: Calculated Core Decay Heat During Loss of Offsite Power Test With Natural Circulation at Davis-Besse 1

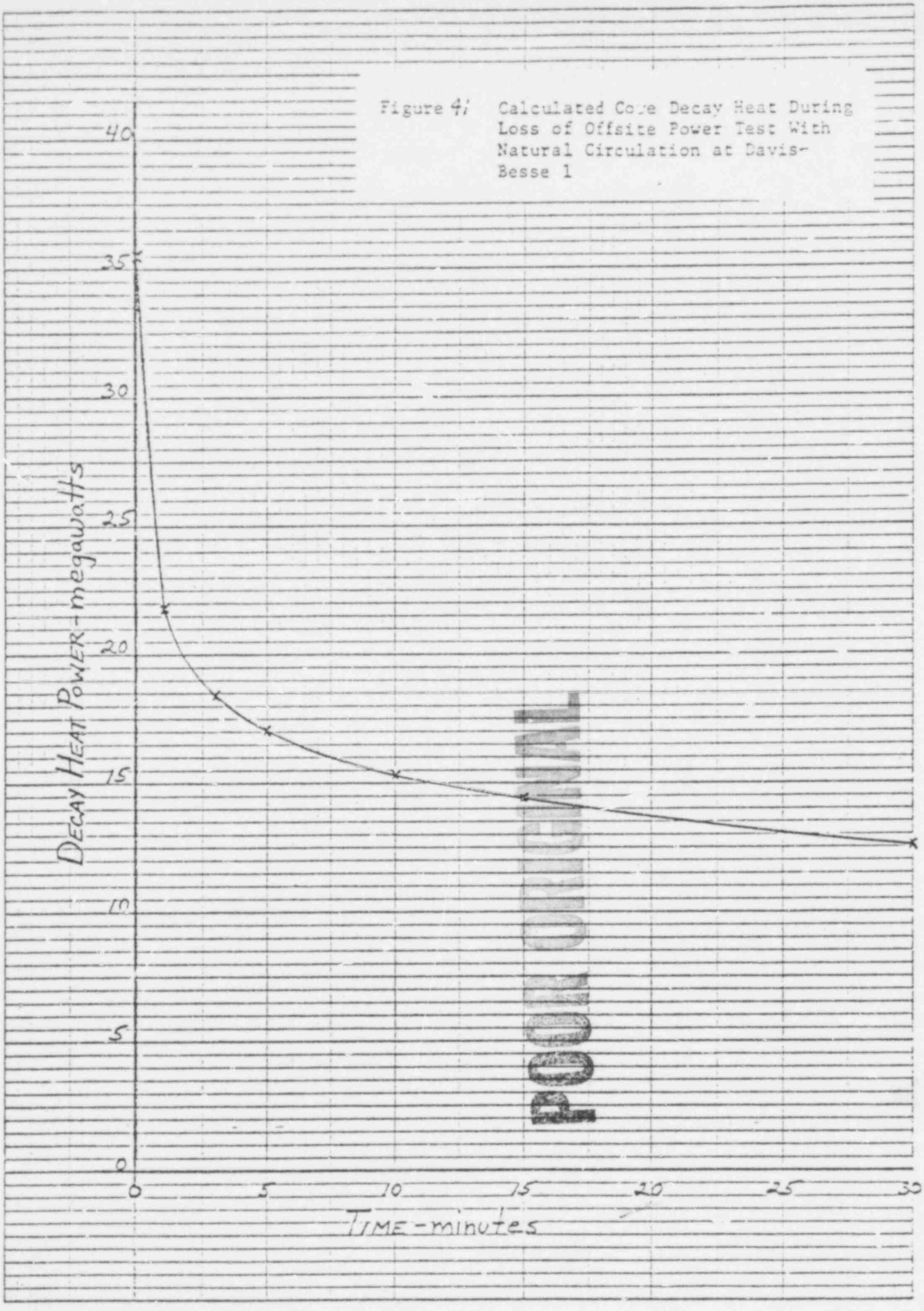

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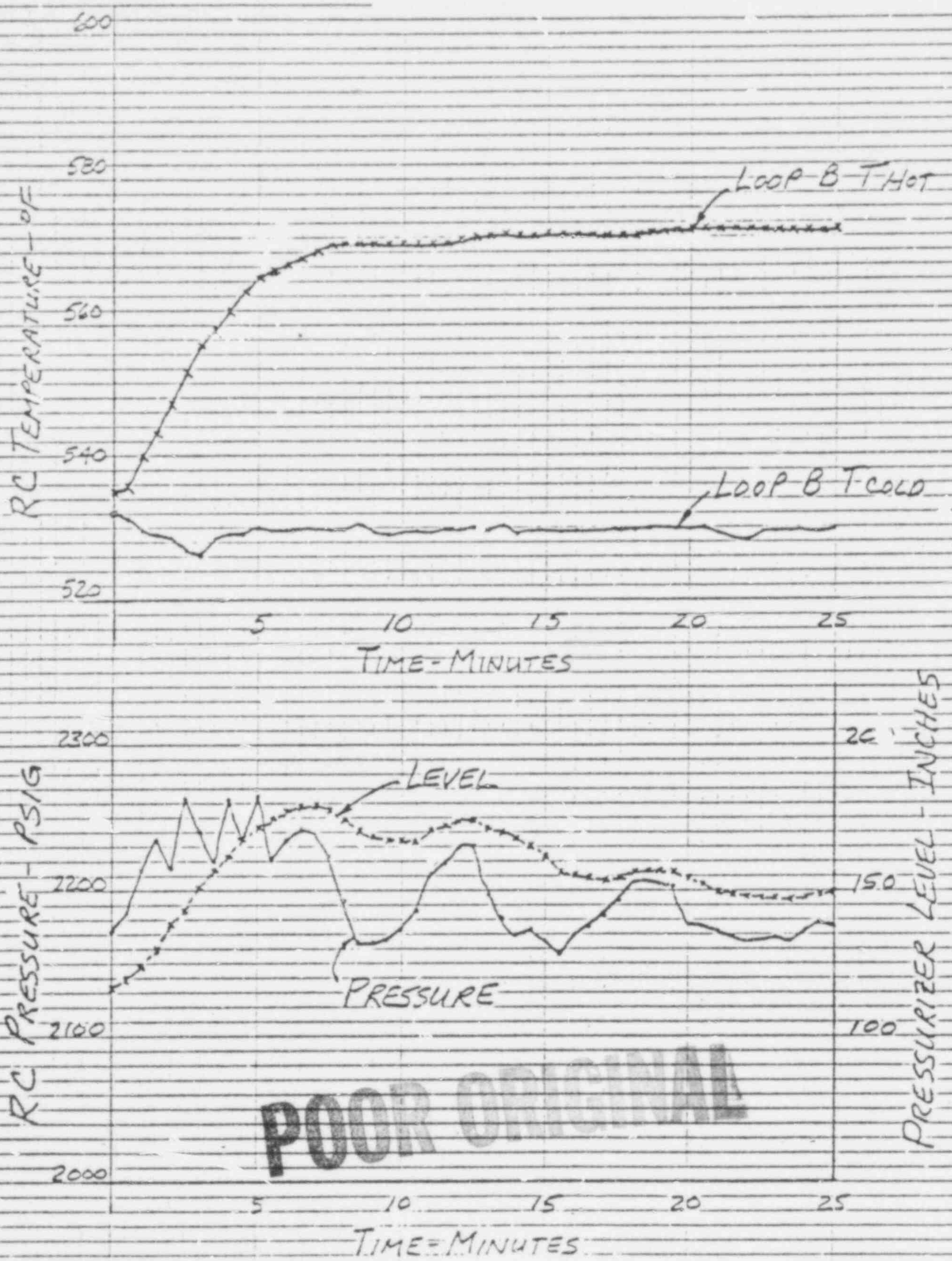
Figure 42 RC System Parameters During Natural Circulation Test at Power at Davis-Besse 1



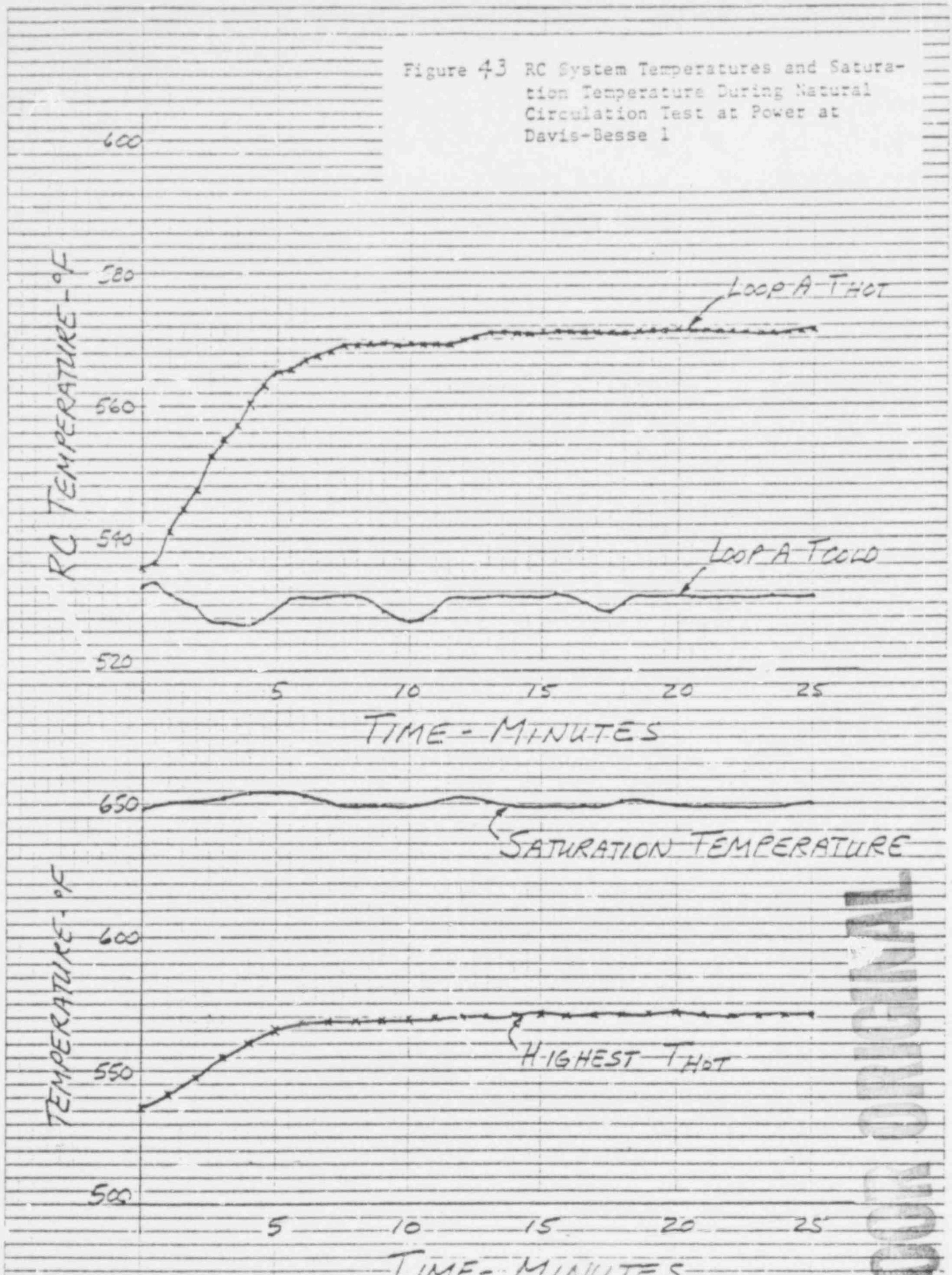


Figure 43 RC System Temperatures and Saturation Temperature During Natural Circulation Test at Power at Davis-Besse 1

Figure 44 Steam Generator Conditions During Natural Circulation Test at Power at Davis-Besse 1

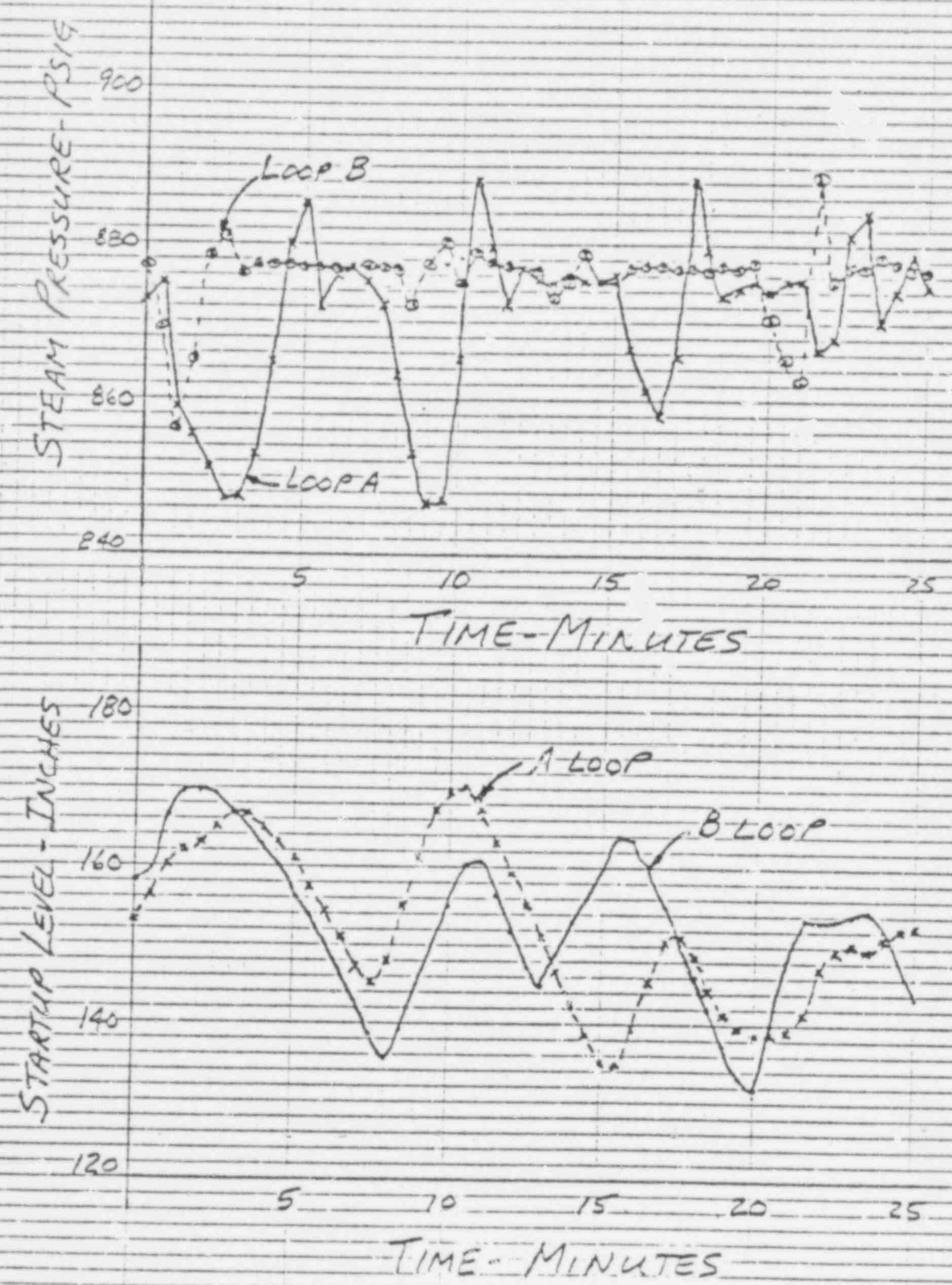


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Figure 45 Steam Generator Outlet and Saturation Temperatures During Natural Circulation Test at Power at Davis-Besse 1

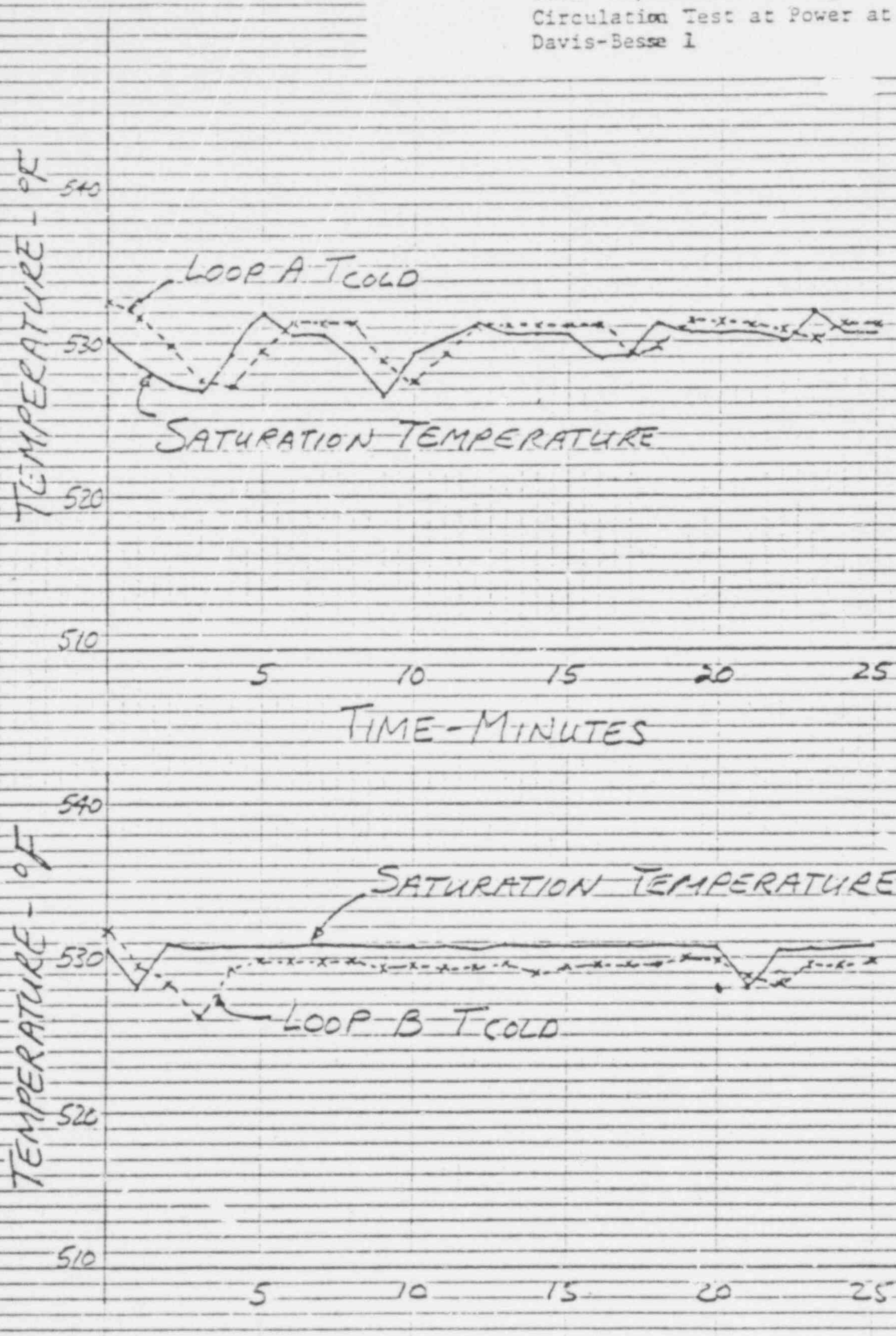
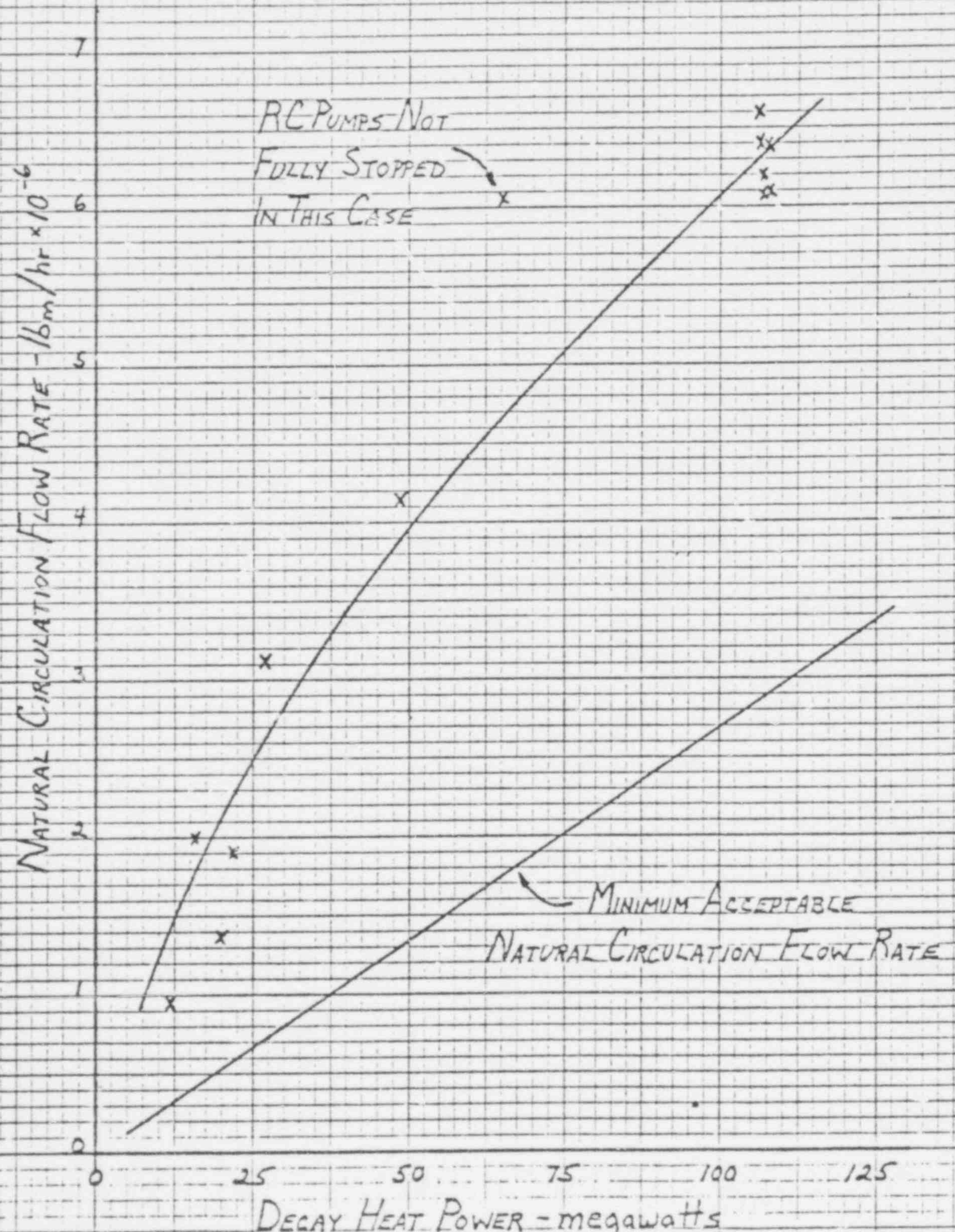


Figure 46. Calculated Natural Circulation Flow Rates vs Decay Heat Power (All documented cases to date on B&W 177-FA Plants)



SUMMARY OF NATURAL CIRCULATION FLOW
MEASURED/CALCULATED AT B&W 177-FA PLANTS

SITE	CALCULATED DECAY HEAT POWER - MW	CALCULATED NATURAL CIRCULATION FLOW - LBM/HR X 10^{-6}
Oconee I	27.0	3.11
Oconee I	48.8	4.16
Oconee II	20.0	1.38
ANO I	65.0	6.06* See Note
CR 3	12.3	0.95
DB 1	22.0	1.90
DB 1	16.2	1.99

*Note: RC Pumps Not Fully Stopped

SITE	NI MEASURED REACTOR POWER - MW	CALCULATED NATURAL CIRCULATION FLOW - LBM/HR X 10^{-6}
DB 1	105.9	6.60
DB 1	106.2	6.41
DB 1	107.8	6.39
DB 1	106.7	6.20
DB 1	106.7	6.08
DB 1	107.6	6.09

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