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SITE INSTRUCTION NO. 6/202

DISTRIBUTION 6

RESPONSE REQ'D FROM 6

APPROVED [signature] for W.H. Spangler 5/9/78

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

To	L. C. Rogers - Site Operations Manager, Three Mile Island 2
From	G. K. Wandling - Startup Task Engineer (Ext. 2034) [signature]
Cust.	JCP&L
Subj.	Rapid Cooldown Transient of 4/23/78

BDS 663-5

File No. NSS-6
or Ref. T3.5

Date May 8, 1978

This letter to cover one customer and one subject only.

The following is a preliminary evaluation of the subject transient as requested by the customer. Specific areas of concern are individually addressed with recommendations made where necessary. The evaluations made are based upon data received from the site in the form of plant parameter graphs, reactimeter data, RCS chemistry analysis results, and the sequence of events during the transient.

I. Reactor Coolant Pumps

During the first 22 minutes of this transient, three pumps were operating, two in "B" Loop and one in "A" Loop. The worst condition, from an NPSH standpoint, would have affected the single pump in the "A" Loop. This pump was flowing approximately 117,000 GPM at the time of the incident and would have required a minimum system pressure of about 560 PSIG to prevent cavitation in the impeller at a cold leg temperature of 464 F. As the indication is that the system pressure only dropped to 752 PSIG, we do not believe the pumps operated under cavitating conditions. We note that after 22 minutes, single pump operation in each loop was initiated, however, by this time the RCS pressure was back up to 2140 PSIG, which would provide adequate NPSH available to the pumps.

From verbal observations, we understand that injection water was maintained and there was no noted change in shaft vibration. We would like to get additional information, such as seal cavity pressure response and seal leakage during this transient to further our knowledge as to how these seals respond to such plant transients.

As this was a very quick temperature and pressure ramp, 125 F in 3 minutes, 1448 PSI in 3 minutes, we cannot make any statement as to the affect of long-term fatigue life on the pump casing and cover. These transients would have to be evaluated by Mechanics Research Institute, the consultant that performed the stress analysis on these pumps. We recently obtained a quotation for $8K for a similar analysis on the SMUD pumps. If we are advised to proceed with this analysis, we will pursue obtaining a new quotation for the TMI-2 pumps.

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Recommendations

B&W recommends continued operation of the reactor coolant pumps. Startup and power escalation data pertaining to RCP seals and pump vibration data should be obtained and compared with baseline reference data. This data should be forwarded to B&W for final recommendations and confirmation of our assessment. We would recommend pursuing the analysis described above, however, this would not delay the present operation of these pumps.

II. CRDM's

Confirmation was received from the site office that the safety rods which were withdrawn during the transient were driven, not tripped, back into place.

Recommendations

Based on the above, and the similarities of this transient to the recently analyzed SMUD transient (3/20/78), we do not feel there are any significant concerns regarding long-term damage to the CRDM's or their ability to continue to perform as designed. Final calculations to support these findings are anticipated by 6/1/78.

In addition, the normal drive venting procedure must be followed prior to returning the CRDM's to service.

III. Fuel

It appears that the cooldown limit for BOL clad compression found in Limits/Precautions Curve 1.0-05.2 was violated by as much as 250 PSIG during the accidental depressurization. However, this limit represents a worst case envelope and does not realistically reflect the conditions encountered in the TMI-2 transient. A specific analysis of the TMI-2 conditions using the supplied RC temperature/pressure information and the TACO code (Version 18) indicates that in actuality the fuel rod cladding never experienced a tensile force. This conclusion is based on a conservative analysis and all information available to date. Further analysis details can be found in calculational file 32-9072-00.

Recommendations

Although the limits/precautions curve was violated, a transient specific analysis indicates the fuel design criteria was not, therefore, B&W does not feel there is any concern regarding the ability of the fuel to continue to perform as designed.

IV. OTSG's

OTSG Engineering Unit, Mt. Vernon, has reviewed the data sent concerning the subject rapid cooldown. Based on the information received, it appears the hot leg RC temperature dropped from 592 F to 460 F in 3-1/2 minutes and the minimum temperature of 454 F was reached 6-1/2 minutes after the trip. We understand the RC temperature remained relatively constant at this temperature for at least three hours before a gradual cooldown was initiated. It is conservative to assume that the maximum tube to shell temperature difference is equal to the maximum temperature drop of 138 F.

This is less than the 170 F tube to shell temperature difference that was analyzed for the rapid cooldown at SMUD in March and based on that analysis is acceptable.

The rapid temperature decrease will cause local thermal stresses which will have to be evaluated in order to determine the affect of this transient on the fatigue life of the vessel. Even though this transient was more rapid than the SMUD cooldown in March, the total temperature drop was significantly less and we feel the net affect of the TMI-2 cooldown will be less severe than the SMUD cooldown currently being analyzed.

Attached (See Attachment 1) is a graph showing the temperature distribution in the support skirt during the critical transient times considered in the 620-0006-55 Stress Report. Since the RC temperature did not drop below 454 F during the initial cooldown, this transient should not produce a temperature gradient anymore severe than the condition analyzed in the stress report. Therefore, the affect of this transient on the support skirt would be the same as a 100 F/hr cooldown.

Based on the above items, B&W feels that the structural integrity of the TMI-2 generators is acceptable for resuming normal plant operation.

V. Reactor Vessel

A preliminary review of the rapid cooldown transient telecopied to B&W, Mt. Vernon, has been reviewed and found to be acceptable. The telecopied curves of this transient were difficult to read; therefore, the transient analyzed from a fracture mechanics standpoint is described below:

Thus, there are no known consequences of this abnormal transient on the RC piping and pressurizer which would prevent startup of the plant at this time.

Time	Temperature Outlet	Temperature Inlet	Pressure
0.0	592	575	2200
0.2	592	575	--
0.8	520	455	--
1.0	-	-	1750
2.5	-	-	820
2.7	455	-	--
3.0	-	-	760
5.0	-	-	980
8.0	-	-	1570
11.0	-	-	2220
13.0	-	-	2160
16.0	-	-	2200
28.0	458	458	--
115.0	480	480	2200

VII. Conclusions Concerning Voiding of Pressurizer

There are four observations or calculations which can be drawn to ascertain whether or not the pressurizer-surge line emptied during the transient. Each of these suggests that, in fact, no steam bubble was drawn into the RCS proper from the pressurizer or formed spontaneously during the depressurization.

Calculations were made based on the measured RCS parameters to determine the rate of and amount of RCS coolant shrinkage during the first two minutes of the transient which includes most of the period when pressurizer level was low-off-scale. These calculations considered the effects of the Makeup System and the operator initiation of HPI, estimations of the effects of temperature measurement errors, and flashing and subsequent cooling of liquid in the pressurizer as the pressure rapidly dropped. Using conservative estimates for these effects, the calculation showed that the pressurizer steam bubble reached the upper section of the surge line, i.e., the pressurizer did empty, but the bubble did not reach the hot leg.

The second item of note is that the RCS pressure was a smoothly varying parameter throughout the transient. In-house codes (e.g., TRAP2) have shown that following pressurizer-surge line emptying, RCS pressure promptly flattens as hot legs flash to steam, thus maintaining the RCS at saturation pressure for the highest temperature in the RCS. This sort of phenomenon did not occur at TMI-2 as can be seen from the loop RCS pressure traces.

The third indication that no bubble was drawn in the RCS is that the hotter of the two RCS hot legs (Loop "A") never got within 20 F of saturation temperature as the RCS pressure fell. At the 3 minute mark after reactor trip, the pressure turned sharply upward and this aspect of the problem ceased to be a concern.

The fourth item is a qualitative agreement based on the reactimeter data from the site. For purposes of explanation, the plot of pressurizer level, RCS pressure and pressurizer temperature are attached (See Attachment 2). According to site sources, 43 seconds after reactor trip, the HPI isolation valves were opened and the second makeup pump was started. This point is marked with an "X" on the pressurizer level plot. Design stroke time for the HPI valves is 10 seconds; design startup time for the makeup pumps is 6 seconds. Therefore, it seems reasonable to assume that full HPI flow was achieved well before the 1 minute mark. Since the makeup pumps are centrifugal pumps, their flow rate increases as RCS pressure decreases. Notice on the pressurizer level trace that when it comes back on scale (t = 2.2 minutes), its slope is relatively linear for about half a minute. At t = 2.5 minutes, the operators throttled down HPI flow. Notice that, at that time, the slope decreased as one would expect. Now look at the dashed curve which covers that range in time when level was off-scale. For the level to have behaved in this way, makeup flow rate would have had to decrease sometime before level came back on scale. Since RCS pressure was continuously decreasing throughout this time, and the HPI valves were at a fixed position, this conclusion does not seem physically realizable. The only curve shape which meets the physical conditions is a concave curve, or straight line. This alternative is shown as a dotted line. According to the HPI flow vs. RCS pressure curves put out by Component Engineering, the HPI flow rate should be continuously increasing as the RCS pressure falls, thus a concave curve shape seems most logical. The conclusion from this line of reasoning is that pressurizer level could not have dropped very far below the lower level tap when HPI turned the level decrease around. The curves suggest that the level stayed within two feet of the lower level tap.

Based on the arguments and observations outlined above, the B&W conclusion is that the pressurizer was never emptied. It appears that only the operator's timely initiation of HPI prevented this from occurring, but the data seems to support our contention that the pressurizer was never entirely drained.

VIII. Technical Specification Summary

The preliminary evaluations performed to date have specifically addressed violations of Plant Technical Specification Items 3.4.9.1 (b) and 3.4.9.2 (a). These preliminary evaluations indicate no deleterious effects on the RCS loop pressure boundary (including pressurizer and surge line) that would prevent continued plant operation. Formal evaluation of the above noted transient will be performed to support the above statements in the near future.

IX. RCS Chemistry

A review has been made of the operating and chemistry data associated with the chloride contamination of the RCS on 4/23 and it is our opinion that the high chlorides will have no deleterious effect on the structural integrity of the RCS or associated systems and equipment. Therefore, the RCS remains acceptable for continued operation (heatup and startup). This constitutes the required engineering evaluation in accordance with Plant Technical Specification (3/4 4-17).

POOR ORIGINAL

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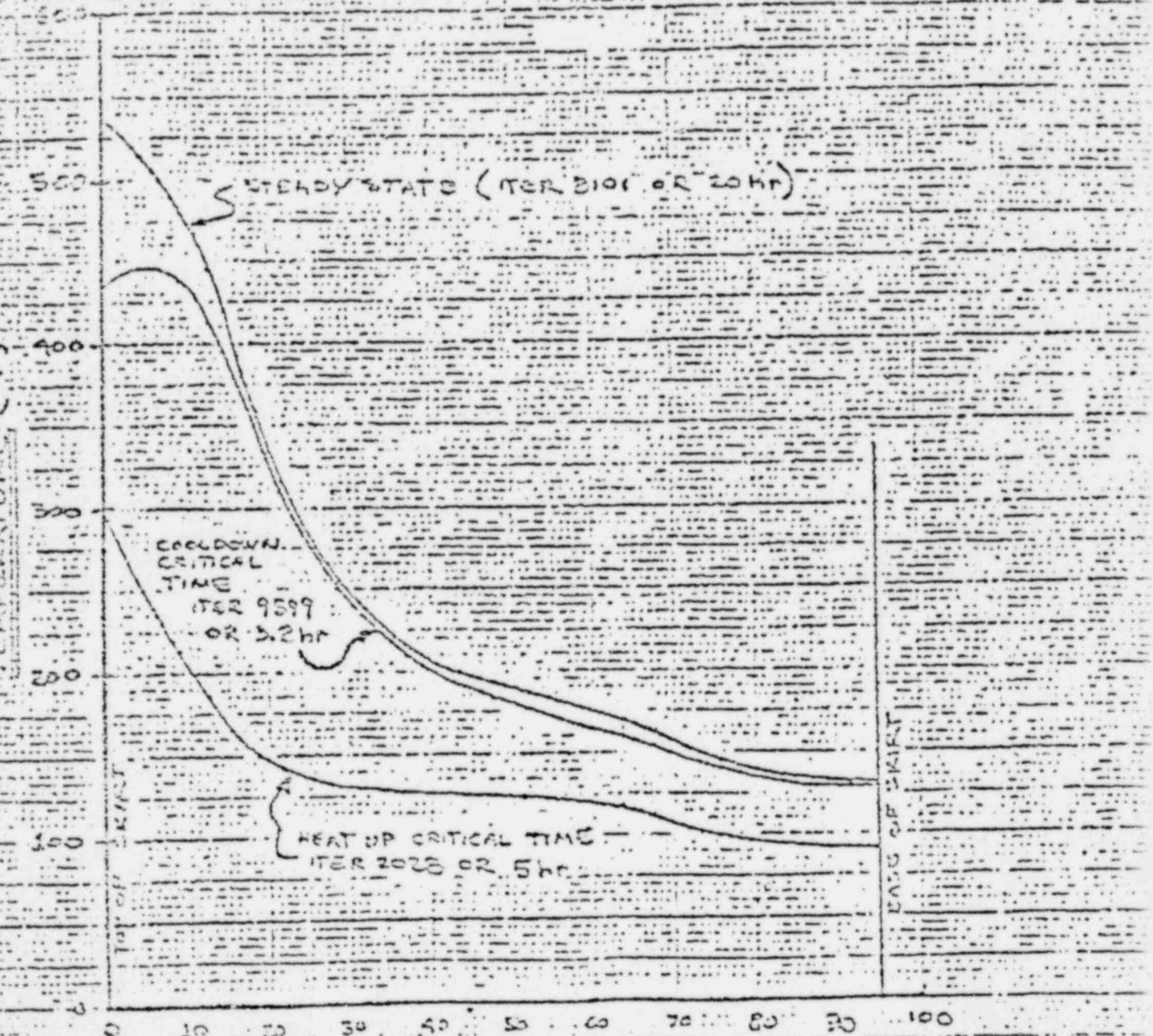
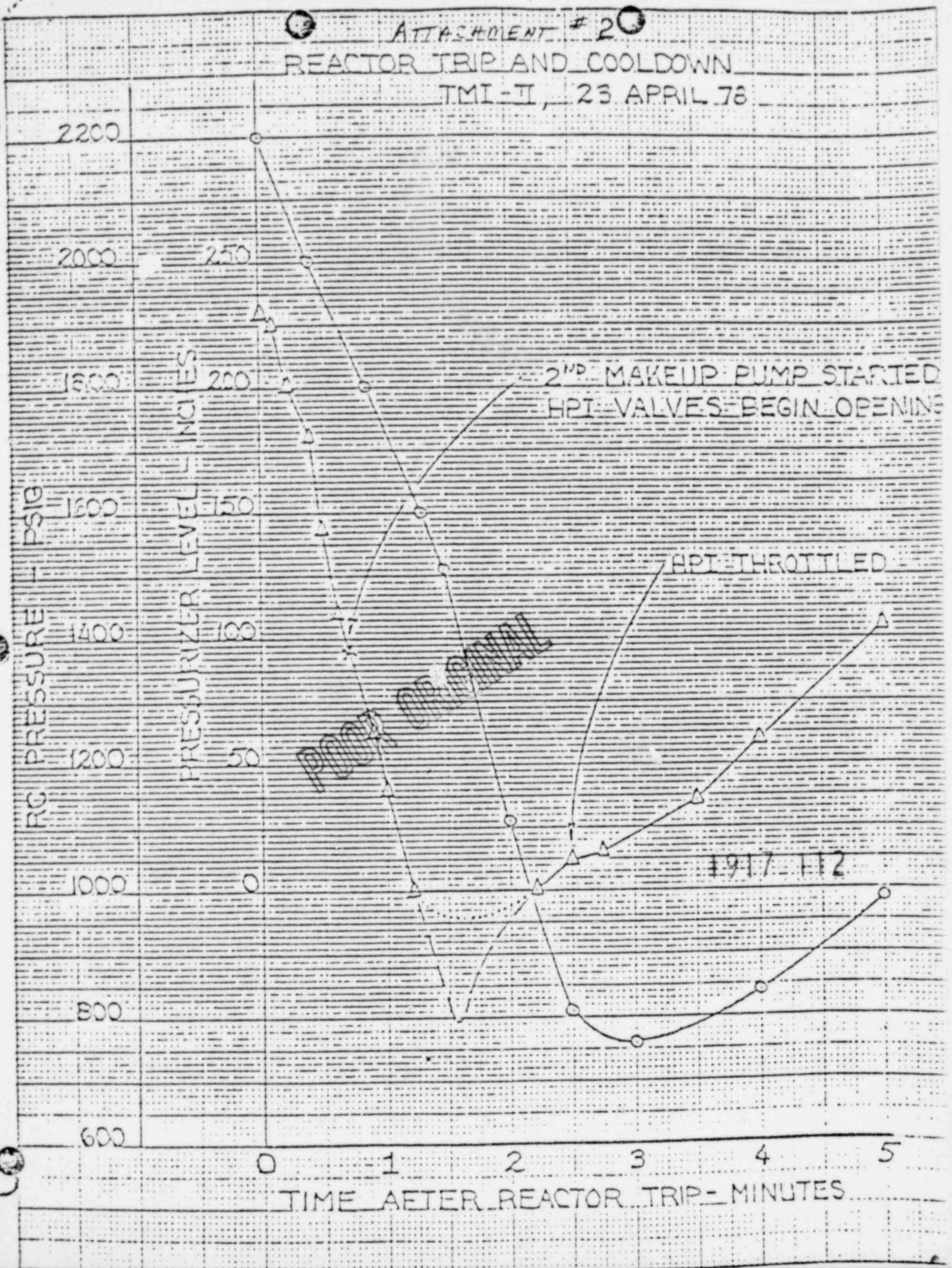
Attachment # 1





ATTACHMENT # 2
REACTOR TRIP AND COOLDOWN
TMI-II, 23 APRIL 78
RC PRESSURE - PSIG
PRESSURIZER LEVEL - INCHES
2200
2000
1800
1600
1400
1200
1000
800
600
250
200
150
100
50
0
2ND MAKEUP PUMP STARTED
HPI VALVES BEGIN OPENING
HPI THROTTLED
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0
1
2
3
4
5
TIME AFTER REACTOR TRIP - MINUTES
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Time (Minutes)	Inlet Temperature (F)
0.0	575
0.2	575
2.8	455
28.0	458
115.0	480

Time (Minutes)	Pressure (PSI)
0.0	2200
1.0	1750
2.5	820
3.0	760
5.0	980
8.0	1520
11.0	2220
13.0	2160
16.0	2200
115.0	2200

Component Engineering has received the computer printout of the fracture mechanics analysis for the core region and the outlet nozzle-nozzle belt region. The stress intensity factors for this transient were less than those experienced by SMUD during a rapid cooldown. Preliminary evaluation of the transient from a fracture mechanics standpoint indicates that this is not an area of concern and the transient is acceptable.

A detailed primary plus secondary and subsequent fatigue analysis was not performed. The short time duration of the transient results in peak skin stresses on the inside surfaces and small discontinuity stresses (primary and secondary). Using an upper bound peak skin stress equation which assumes an infinite film coefficient and a step down in fluid temperature, the resulting skin stress for this transient is approximately 26,000 PSI. One cycle of this 26,000 PSI thermal stress and pressure stress lumped with the maximum stress resulting from normal operating transients will have an insignificant affect on the cumulative fatigue usage factors of the reactor vessel components.

A further fatigue justification of this transient is that it is less severe than the SMUD rapid cooldown. The SMUD rapid cooldown transient is in the process of being analyzed by utilizing simplified conservative analysis methods. This analysis is in the final stages and the increase in cumulative factors at the more critical locations are small.

VI. Reactor Coolant Piping and Pressurizer

The transient data summarized in the table below has been reviewed for preliminary impact on the operating life of the TMI-2 reactor coolant piping and pressurizer. The results of this review show there would be no significant increase in the cumulative usage factor for any portion of the RC piping or pressurizer. However, a detailed analysis would need to be performed at a later date to document this transient's actual effects for purposes of the plant's life history CUF.

Most of the RC piping has, at present, a low CUF as does the pressurizer. Therefore, the increase in CUF due to this one time occurring transient would not cause any portion of the RC piping or pressurizer to have a CUF greater than the allowable for the life of the plant.