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**INFORMAL REPORT**

# ANALYSIS OF TMI-2 PRESSURIZER LEVEL INDICATIONS

J. L. Anderson



**Idaho National Engineering Laboratory**

*Managed by the U.S. Department of Energy*

EG&G Idaho

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# ABSTRACT

Results of a study to evaluate and understand the pressurizer level response to the reactor system thermal-hydraulic conditions during the first 1000 min of the TMI-2 accident are presented. An evaluation of the measurement system with regard to postulated problems, determined that the problems were insufficient to discount the observed pressurizer level response. It has been determined that the observed level changes can be explained in terms of response to the thermal-hydraulic conditions in the reactor coolant system. A comparison of the TMI pressurizer level response and the level response observed during integral system experiments is made. In those experiments where a TMI accident scenario was performed, the pressurizer level was observed to respond in a manner very similiar to the measured TMI response.

# SUMMARY

This report documents results from a study performed by EG&G, Idaho Inc., to resolve numerous questions regarding the validity of the pressurizer level measurement during the first 1000 min of the accident that occurred at the Three Mile Island Unit-2 nuclear reactor on March 28, 1979. The first 1000 min of the accident are of importance because this was the time period during which the core damage occurred, and forced convective cooling was reestablished. Understanding the pressurizer level response is necessary for overall understanding of the reactor system thermal-hydraulics during the accident, and the impact upon core uncovery. The pressurizer liquid level was determined using three independent differential pressure transmitters to measure the difference between hydrostatic heads in the reference legs and the liquid column within the pressurizer, and correcting for the difference in fluid densities. During the accident, operators attempted to infer the liquid inventory of the primary system, particularly the reactor vessel, from the level in the pressurizer. Unfortunately, once the primary system reached saturation conditions and steam voids existed in the reactor coolant system, the pressurizer level response was no longer coupled to the primary system in the normal manner understood by the operators. In the post-accident analysis, several questions were raised regarding whether or not the level measurement could have been correct, since the pressurizer level was indicating a full pressurizer when the remainder of the primary system was obviously in a highly voided state. Various investigators proposed failure mechanisms for the level measurement including water hammer damage, boiloff of the reference legs, heat damage to the differential pressure transmitters, and hydrogen effervescence in the reference legs. Analysis shows that none of these mechanisms could have produced the observed level response in the pressurizer.

Another argument made regards the validity of the level measurement centers upon the ground fault trips of the pressurizer heaters. It has been argued that these trips (which occurred between 270 and 595 min into the accident) could only have occurred in an empty pressurizer. Since the

pressurizer level was indicating full, the level indication must have been wrong, or so goes the argument. However, during the period when all but one of these trips occurred (270-463 min) the makeup system was injecting large quantities of liquid into the RCS, and into the pressurizer since the pilot-operated relief valve (PORV) was open. Further supporting evidence is the highly subcooled (424 K or 303°F) temperature indication in the surge line during this time period. It is therefore concluded that the heater ground fault trips were due to a source other than dryout of the heaters. Further investigation of this source is required to resolve the heater trips mechanism.

An analysis of the pressurizer level response to the known, and postulated, thermal-hydraulic conditions in the RCS is described in detail in the body of this report. From this analysis it is concluded that the pressurizer level measurement was correctly indicating the liquid level in the pressurizer, within an uncertainty band of approximately 4% of the level range, and that the pressurizer level was responding to the RCS thermal-hydraulic conditions. During the accident, the pressurizer liquid level responded to RCS pressure changes, which effected the pressurizer thermodynamic state, and to the conditions at the surge-line entrance to the hot leg. During periods when there was flow through the open PORV, and there was liquid at the surge-line entrance to the hot leg, the pressurizer level would increase until the pressurizer was full. If no liquid source was available, and the pressurizer was at saturation, the level would remain constant due to counter-current, flow-limiting phenomena in the surge line, which limited the amount of liquid draining out of the pressurizer. If the pressurizer was subcooled, either with the PORV block valve open or closed, the level would increase due to steam condensation and level swell as the pressurizer liquid was heated and density decreased. The pressurizer level decreased during periods when the PORV block valve was closed, the pressurizer was at saturation, and the RCS depressurized due to increased makeup flow or increased heat transfer in the steam generators.

The conclusions reached from this analysis are also supported by experimental data from the scaled Semiscale integral system. Further supporting evidence is provided from RELAP5 thermal-hydraulic calculations in which the pressurizer level was calculated to respond in similar fashion to that observed during the first 100 min of the accident.

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## ANALYSIS OF TMI-2 PRESSURIZER LEVEL INDICATIONS

### INTRODUCTION

During normal plant operation the function of the pressurizer is to control system pressure. This is accomplished through use of pressurizer heaters to increase fluid temperature in the saturated pressurizer, thus increasing system pressure, and by use of the spray line to inject cold liquid into the pressurizer, thus reducing temperature and pressure. The pressurizer is also equipped with a pilot-operated relief valve (PORV) to quickly relieve pressure under conditions such as a feedwater pump trip. This is the valve that stuck open and resulted in the severity of the Three Mile Island Unit-2 (TMI-2) accident. The level in the pressurizer is normally used as an indication of total system mass inventory, and is controlled through use of letdown and makeup systems. Level in the pressurizer is normally maintained between 508 and 660 cm (200 and 260 in.). The level just prior to the feedwater pump trip was 569 cm (224 in.).

Since the March 28, 1979 accident at the TMI-2 nuclear plant, there has been considerable controversy[2-4] over operability of pressurizer level measurements during the accident, as well as reasons for the pressurizer level response if those level indications were correct. This report documents results of a study performed by the TMI-2 Accident Evaluation Program of EG&G, Idaho Inc., in an attempt to clarify the pressurizer level response to the thermal-hydraulic conditions in the reactor coolant system (RCS) during the first 1000 min of the accident. The approach taken in the study consisted of:

- A description of the pressurizer level measurement system and an evaluation of the various reasons set forth for disbelieving the measurements

- An analysis of pressurizer level response to thermal-hydraulic conditions in the primary and secondary systems, assuming trends of the level measurements were correct

- An evaluation of supporting data from integral systems experiments and comparison to the measured TMI pressurizer level response

- An evaluation of results from thermal-hydraulic code calculations performed in support of accident evaluation with respect to predicted pressurizer level response.

There are two reasons for studying the pressurizer level measured during the first day of the accident. First, understanding the mechanisms causing changes in the pressurizer level can provide valuable insights into conditions existing in the reactor coolant system during major events, such as core uncovery and heatup. Unfortunately there were insufficient measurements recorded during the accident to determine RCS conditions directly; therefore, information inferred from the pressurizer level response is extremely useful. Secondly, prediction of the correct pressurizer level by thermal-hydraulic codes (such as RELAP5) is necessary for correct calculation of overall system response leading to core uncovery. Prediction of the correct level is necessary for correct calculation of mass flow rate through the open PORV. Additionally, impact of the pressurizer level response (remaining near full or draining) is especially significant during the time period that initial core uncovery and core damage occurred. RELAP5 calculations indicate that if the pressurizer did in fact drain, as is speculated by some investigators, then the additional liquid in the core would have delayed core uncovery and heatup by as much as an hour. For these reasons, this study was undertaken to determine if the measured liquid level could be used for analysis of the accident.

In this report, the measurement system is described and evaluated. The pressurizer level response to the RCS thermal-hydraulic events are then presented and discussed. Results from one integral system experiment (the Semiscale TMI simulations) are presented and compared to the TMI pressurizer response. The pressurizer liquid level response calculated by the RELAP5 analysis is compared to the measured pressurizer level. Finally, conclusions reached from the study are presented, with supporting calculations and uncertainty analyses included as Appendices.

## SYSTEM DESCRIPTION

An isometric of the TMI-2 primary system[4] is shown in Figure 1. The 10-in. schedule 140 (ID = 22.2 cm or 8.75 in.) pressurizer surge line enters the A-loop hot leg at an elevation of 98 m (321 ft 6 in.). The surge line drops down from the hot-leg entrance to an elevation of 94.3 m (309 ft 3 in.), travels approximately 10 m (34 ft) horizontally, then rises to the pressurizer entrance at an elevation of 95.1 m (312 ft 2 in.) on the inside surface of the pressurizer. This configuration acts as a loop seal for the surge line, with the entrance of the surge line to the hot leg corresponding to a measured pressurizer liquid level of 163 cm (64 in.). This level is just above the elevation of the pressurizer heaters. The 10 cm (4-in.) pressurizer spray line leaves the primary system at the discharge of the 2A reactor coolant pump, and enters the pressurizer through the top head. A control valve is installed near the entrance to the pressurizer to control the spray flow rate. The spray line does not have a check valve installed, which would prevent reverse flow from the pressurizer to the cold leg.

A schematic of the pressurizer level measurement system is shown in Figure 2. The level measurement is based upon the hydrostatic fluid head of the liquid column in the pressurizer, measured using the differential pressure between a liquid filled reference leg, external to the pressurizer, and the fluid in the pressurizer. Since the reference legs are external to the pressurizer insulation and are uninsulated, the liquid in the reference legs remains near containment temperature. As a result, there is no need for installation of condensate pots to keep the reference legs liquid-full during normal operation, and no condensate pots are installed. There are three independent measurements separated by 120° around the pressurizer. The bottom tap for each is located at an elevation of 96.4 m (316 ft 2 in.), and the top taps are at an elevation of 106.5 m (349 ft 6 in.), for a total span of 1,016 cm (400 in.). Between each of these sets of taps, a Bailey Instruments differential pressure transmitter is installed; this is setup for a -10 to +10 V output under an input head of 0-1016 cm (0-400 in.) of cold water (293 K or 68°F). These transmitters

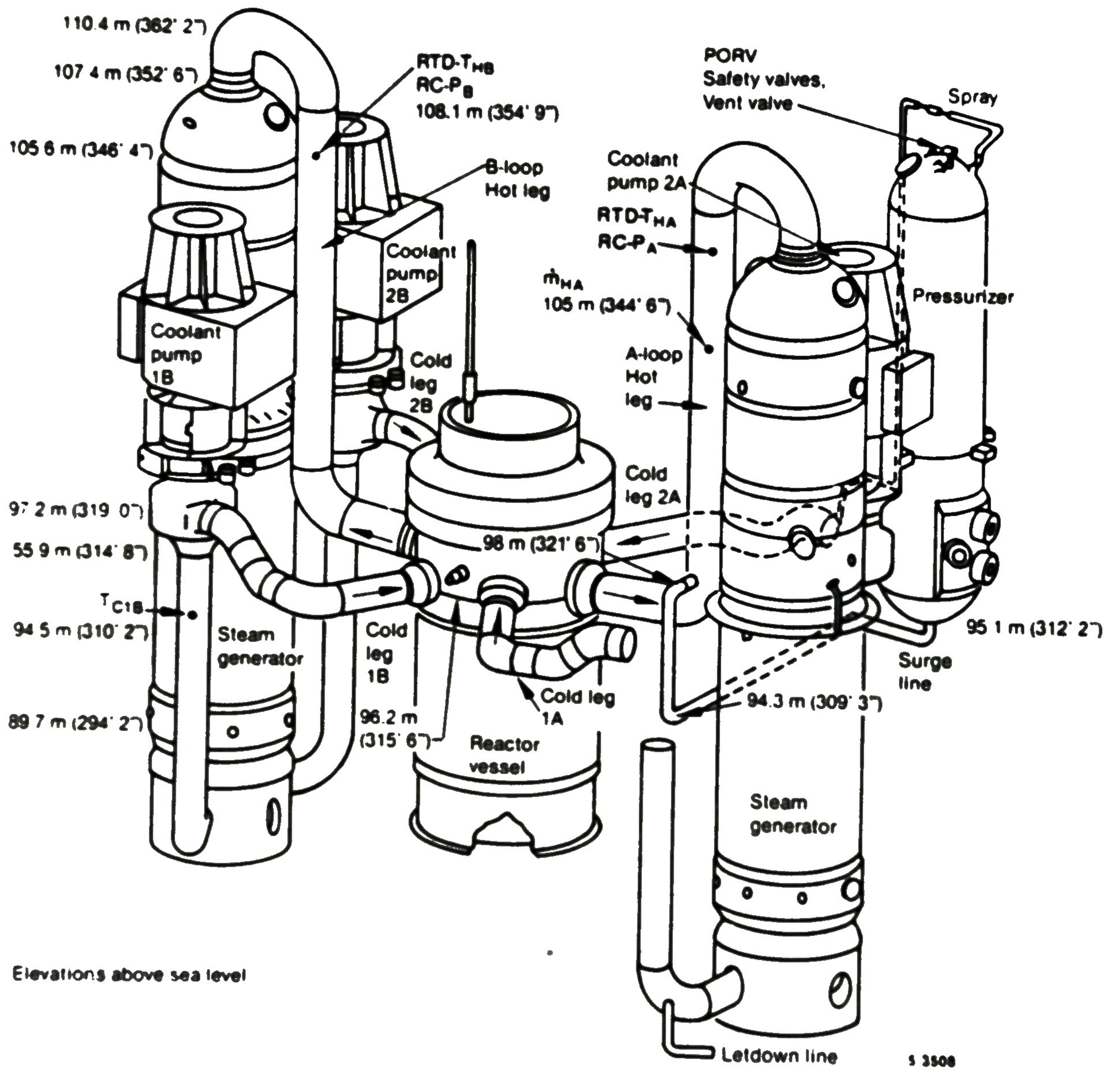


Figure 1. Isometric of the TMI-2 primary system.

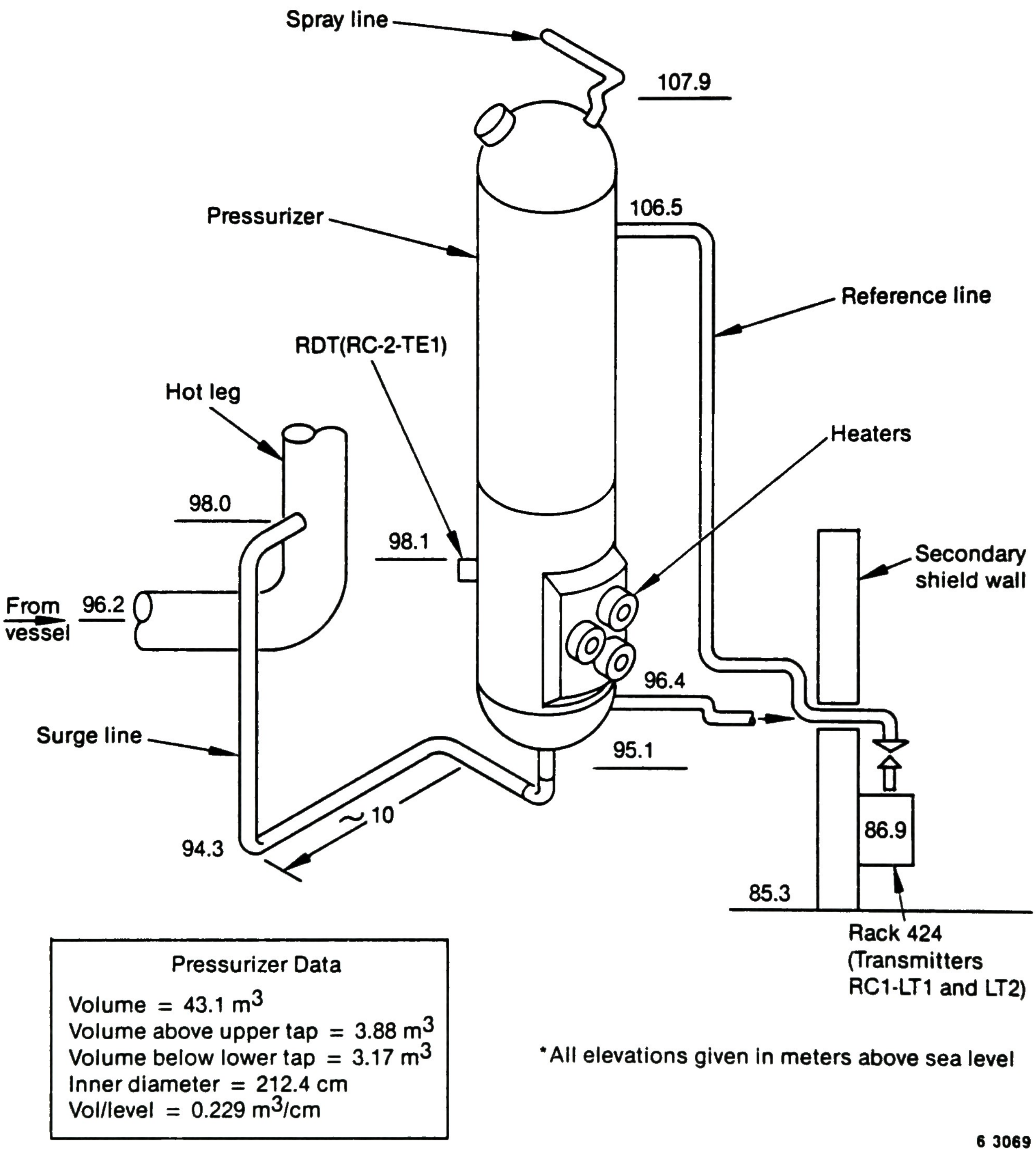


Figure 2. Pressurizer level measurement configuration.

are mounted in instrument racks 424 (RC-1-LT1 and LT2) and 426 (RC-1-LT3) which are located at an elevation of 86.9 m (285 ft) outside of the secondary shield wall in the reactor building basement. The transmitters are connected to the pressurizer taps using 2.13 cm (1/2-in.) SST tubing as sense lines. The transmitters are zeroed when valved out of the system and vented to atmosphere, i.e., with no load applied to either side. As a result, when the transmitter is valved into the system with an empty pressurizer, the transmitter measures the 1016 cm (400 in.) hydrostatic head of the reference leg. When the pressurizer is full of cold water, the transmitter measures 0 cm of differential pressure since the two hydrostatic heads balance each other.

The output from one of the three transmitters is used to calculate the "Temperature compensated" level in the pressurizer. The transmitter used for this calculation is switch selectable from the operators control panel, with no record of which transmitter is used, although for normal operation the RC-1-LT1 transmitter is used. The direct output from any of the transmitters was not recorded during the entire accident (the output from one transmitter was recorded on the utility printer starting at 570 min). Temperature compensation is performed to account for the difference in fluid densities between the reference leg and the pressurizer fluid. The level is simultaneously calculated by two methods. The first is performed using an analog circuit, which is part of the Non-Nuclear Instrumentation (NNI), the output of which goes not only to the Integrated Control System (ICS) for control of the pressurizer liquid level, using the makeup and letdown systems, but also to the control panel level indications and strip chart recorder. The second method uses the plant computer to calculate the pressurizer level. In this method, the transmitter output is combined with the specific volumes of the saturated liquid and steam which are calculated using one of two fluid temperatures measured in the pressurizer, to calculate the liquid level. The liquid level (L) is obtained in this method using the following equation (derived in Appendix A).

$$L = \frac{(\rho_r - \rho_g)\, D - \rho_c\, DP}{(\rho_f - \rho_g)} \qquad (1)$$

where

$\rho_c$ = fluid density of cold water (at 293 K or 68°F)

$\rho_r$ = fluid density for the reference leg (at 325 K or 125°F)

$\rho_f$ = fluid density of the liquid in the pressurizer ($kg/m^3$)

$\rho_g$ = fluid density of the steam in the pressurizer ($kg/m^3$)

D = distance between the pressurizer taps (= 1016 cm or 400 in.)

DP = measured differential pressure (cm of 293 K water).

Equation (1) accounts for the hydrostatic head of the steam, and is the equation used by the plant computer to obtain the level displayed on the utility printer. When the primary system temperature is above 325 K (125°F), a reference-leg temperature of 325 K (125°F) is assumed for obtaining the reference-leg fluid density. The level given by Equation (1) is the collapsed stratified level. If the liquid in the pressurizer was boiling, and thus filled with voids, the two-phase interface level would be higher than the collapsed stratified level due to level swell. Results from Equation (1) are available to operators on the utility printer upon request, and are displayed as alarms on the alarm printer when the range of 508-660 cm (200-260 in.) is exceeded.

## Heater Operation

In order to increase pressure during plant operation, the pressurizer is equipped with heaters that are controlled by the ICS in the automatic mode, based upon pressure, or manually by the operators. The heaters are divided into 13 groups of 126 kW each. These groups are divided into five banks, each bank of which is the basic control unit. The breakdown of heater banks by groups and control setpoints[5] is given in Table 1. Each

TABLE 1. TMI-2 PRESSURIZER HEATER CONFIGURATION

| Heater Bank | Corresponding Heater Group Number(s) | Total kW[a] | Low Pressure[b] On Setpoint in psig (MPa) | High Pressure[b] Off Setpoint in psig (MPa) |
|---|---|---|---|---|
| 1 | 13 | 126 | 2147 (14.904) | 2155 (14.959) |
| 2 | 12 | 126 | 2135 (14.821) | 2155 (14.959) |
| 3 | 8, 9, 10, 11 | 504 | 2135 (14.821) | 2155 (14.959) |
| 4 | 4, 5, 6, 7 | 504 | 2120 (14.718) | 2140 (14.856) |
| 5 | 1, 2, 3 | 378 | 2015 (13.995) | 2125 (14.752) |

a. Each group provides 126 kW.

b. From NSAC-80-1.[8] Pressure is the gauge pressure measure in the A-loop hot leg. Atmospheric pressure is assumed to be 14.7 psig.

bank can be controlled either manually or in automatic mode by the ICS. The control mode is switch selectable by the operators, and the setting is not recorded. During the first day of the TMI-2 accident, the operators apparently switched heater banks 4 and 5 into automatic control prior to reactor scram, and left banks 1, 2, and 3 in manual control mode. Banks 1, 2, and 3 were apparently left energized during the entire first day of the accident. Since these groups were in manual control, operation of the heaters (either on or off) was not recorded on the alarm printer, with the exception of ground fault trips which will be discussed later. Operation of heater groups 1-5, in banks 4 and 5, was recorded on the alarm printer as TRIP when each group was de-energized, and as NORM when each group was energized. Groups 6 and 7 in bank 4 were unavailable for operation during the first day of the accident.[6] Each group also showed 25 TRIP on the alarm printer when the group circuit breaker was tripped due to a ground fault. A listing of groups that tripped due to ground faults, and the times at which the trips occurred are recorded in Table 2. The heater groups are not thermostatically protected.

## HPIS/Makeup

The high-pressure injection system (HPIS) is an engineered safety (ES) system capable of injecting a total of 63 L/s (1000 gpm) of cold water into the four cold legs of the reactor system (16 L/s or 250 gpm per cold leg). The HPIS is actuated by the ES actuation signal under a number of conditions, one being a primary system pressure below 11.3 MPa (1640 psig). The HPIS uses two pumps, MU-P-1A and MU-P-1C, with an automatic valve alignment for injection into all four cold legs.

The makeup system is a high-pressure injection system which, during normal reactor operation, balances the letdown flow (normally 3-4 L/s or 45-70 gpm) and injects continuously. A single pump (MU-P-1B) is normally used, although any single makeup pump (MU-P-1A, -1B, or -1C) or combination of pumps can be used. The makeup system uses several of the same components as the HPIS. However, the flow path for normal makeup is into the reactor coolant pump seals (2 L/s or 30 gpm) and into the 1B cold-leg pump discharge.

TABLE 2. TABULATION OF PRESSURIZER HEATER GROUND FAULT TRIPS

| Accident Time (minutes) | Heater Groups that Tripped |
| --- | --- |
| 270 | 10 |
| 287 | 4 and 5 |
| 330 | 3 |
| 463 | 1 and 2[a] |
| 595 | 8 |

a. Groups 1 and 2 reset immediately after tripping.

During the accident, the operators overrode the ES signal several times, and assumed manual control of the injection. Pumps were turned off and injection flow throttled. As a result, the injection rates and injection locations during the accident are unknown. The best available estimate for injection flow rate was obtained from Reference 5. However, the locations of makeup injection (cold leg 1B or all four cold legs) are unknown. This location could have significantly effected system behavior because of steam condensation upon injection. The makeup history given in Reference 5 and the letdown history given in Reference 10 have been combined to give net injection flow into the system (makeup flow minus letdown flow). The net injection is presented later in this report, for use in analyzing RCS thermal-hydraulics.

## Postulated Measurement Problems

Since the accident, several arguments have been raised as to problems that might have resulted in an invalid measurement of the pressurizer level using the aforementioned measurement system. Twice during the first day of the accident, at 43 and 433 min after the feedwater pump trip, operators requested output of all three transmitter readings on the utility printer. Both times the three transmitters agreed within several cm. Any arguments discounting the validity of the level measurement must explain this fact. Of the four arguments examined, all were found insufficient for explaining the observed pressurizer liquid-level response during the first day of the accident. Each of the four arguments are presented and discussed below.

The first argument, raised shortly after the accident, involves possible effervescence of dissolved hydrogen in the reference legs. The argument is that prior to the accident, hydrogen was dissolved in the liquid throughout the primary system to eliminate the dissolved oxygen that would tend to increase corrosion of components. Following the reactor scram, system pressure decreased from 16 to 7 MPa (2350 to 1000 psig) during the first 30 min. The dissolved hydrogen would tend to effervesce. In the reference legs of the liquid level measurement system, such effervescence possibly occurred at a fast enough rate to force a significant amount of liquid out of the reference leg, thus invalidating the liquid level measurement. Sandia laboratories analyzed this possibility.[7] Assuming an initially hydrogen saturated reference leg at 15 MPa (2200 psi), and an instantaneous depressurization, a maximum error in the level measurement, due to liquid ejection, was calculated as 145 cm (57 in.). However, depressurization was not instantaneous, but took 30 min to reach 7 MPa (1000 psig). The conclusion was, "it is apparent that head-loss due to hydrogen effusion is too small to be responsible for the large level changes reported for the accident."

A second argument put forth involves possible boiloff of liquid from reference legs during system depressurization.[3] Since the reference legs are outside of the pressurizer insulation, it is unlikely that their fluid temperature would be much above the reactor building temperature over any significant portion of their length. For boiloff to occur, the temperature would have to be at the saturation temperature for the system pressure, which is 558 K at 7 MPa (545°F at 1000 psig). The highest recorded reactor building temperature was 354 K (175°F), which occurred at 300 min at an elevation of 101 m (330 ft). It is possible that fluid in the top few cm of the reference legs was at a sufficiently high temperature, due to heat conduction from the hot pressurizer, to boil when the system depressurized. However, this would result in a temporary error of less than 25 cm (10 in.), which would disappear as condensation refilled the reference leg. If condensation did not occur, to explain the close readings between transmitters would require that the boiloff in each of the reference legs be the same. This argument cannot be supported by thermodynamic considerations.

A third argument put forth involves damage to the reference legs by a water hammer that occurred at 174 min when the 2B reactor pump was restarted.[4] Restart of the pump forced liquid into the hot core. The liquid quickly boiled and not only produced steam, but generated hydrogen and caused rapid repressurization of the system and a surge into the pressurizer. It is postulated that the rapid pressure and level increases acted as a water hammer on the reference lines, damaging them severely enough to cause leakage. The argument continues by assuming the leaks were small, and that the reference lines would refill with condensate during periods of repressurization, resulting in false indications of a falling level in the pressurizer. No argument has been made for damage to the differential pressure transmitters by the water hammer.

There are a number of problems with the third argument. For one, a water hammer cannot occur in a vessel or line in which gas exists, as gas acts as a buffer, absorbing momentum from the liquid, thus limiting the pressure rise. Also, a water hammer results in a pressure spike, but the pressure rise in the RCS was at a rate of 1.7 MPa/min (250 psi/min). It is difficult to postulate that this pressure increase could result in equal damage to all three reference lines. These lines were hydrostatic tested at a pressure of 9000 psig prior to plant startup. (The water hammer pressure rise is analyzed in Appendix B, the result being that in-surge velocities would have had to be a factor of 2000 times larger than measured to reach the hydrostatic test pressure of the sense lines.) The final problem with this argument is the fact that 259 min after the 2B pump transient (433 min accident time), output from all three of the differential pressure transmitters were recorded on the utility printer and were in agreement within 13 cm (5-in.).

A fourth argument put forth for disbelieving the level measurement involves the environment to which the transmitters were exposed during the first day of the accident.[3] The transmitters were installed in instrument racks 424 and 426 in the reactor building basement. Rack 424 (in which transmitters RC-1-LT1 and LT2 were installed) was in the vicinity of the exhaust from the Reactor Coolant Drain Tank (RCDT) rupture disk assembly. Discharge from the PORV was routed to this tank. As such, the exhaust had the potential of raising the local temperature above the environmental specifications for the level transmitters. The maximum temperature recorded in the reactor building for this vicinity was 353 K (175°F). Specifications for the transmitter are for a maximum operational temperature of 344 K (160°F). However, Bailey Instruments performed autoclave tests on representative units in which the transmitters were maintained in a steam environment above a temperature of 383 K (230°F) for a 24 h period. During significant portions of this test, the transmitter was submerged in liquid from condensation. Periodically the transmitter was calibrated in-place to check for the environmental effects upon the transmitter calibration. The maximum calibration error experienced during the 24 h period was less than 5%, primarily a zero shift. It is unlikely that the conditions experienced by the transmitter during the first 24 h of the accident exceeded conditions created during the autoclave tests. As such, inoperable transmitters or excessive calibration shifts of the transmitters during the first 24 h of the accident are insufficient to explain the observed pressurizer level response.

## THERMAL-HYDRAULIC DESCRIPTION

A chronological description and explanation of the pressurizer response for significant time segments of the accident is presented here, with the accident phases used in Appendix TH of NSAC-80-1[8] as framework. Available data for this analysis are limited. The pressurizer level, narrow-range pressure in the B-loop (RC-3B-PT1), narrow-range temperatures in the hot legs, and wide-range temperatures in two of the cold legs were recorded on the reactimeter at a rate of 1 sample per 3 s. The wide-range system pressure was obtained by digitizing a strip chart of RC-3A-PT3 and combining it with the valid reactimeter data (that within range), and the utility printer data. During the first 570 min of the accident, limited data on the pressurizer temperature, surge-line temperature, and spray line temperature were available from the alarm and utility printers. From 570 to 1000 min, data on the pressurizer temperature, level measurement RC-1-LT1, and wide-range pressure RC-3A-PT3 were available on the utility printer as group trend data every 2 min. Timing of the PORV block valve operation is somewhat uncertain, since operation of the valve was surmised from primary system and containment building pressure changes, in conjunction with the operator interviews and PORV header temperature alarms. Many of the times for these operations may be off by several min. Some liberty was taken in adjusting the opening and closing times of the PORV block valve within this uncertainty band in order to more adequately explain the system responses. In addition, an unreported block valve cycle at 198 min is used, and is documented in Appendix D. The block valve open/close times used in this study are given in Table 3. The primary source for the sequence of events was Reference 6. A tabulation of the major pressurizer level changes, along with the physical mechanisms believed to have caused the level changes and assumptions made, is given in Table 4, to assist in clarifying the following discussion.

### Phase 1 - Initiation

The first phase of the accident is defined as the time period from the turbine trip (0.0) to the shutdown of the B-loop reactor coolant pumps at 73 min. During this phase the pressurizer liquid level first increased,

TABLE 3. PORV BLOCK VALVE OPERATION TIMES

| Time (min) | PORV Block Valve Operation |
|---|---|
| 139 | Closed |
| 191.6 | Opened |
| 194.8 | Closed |
| 197.9 | Opened |
| 198.4 | Closed |
| 220 | Opened |
| 260 | Closed |
| 276 | Opened |
| 318 | Closed |
| 343 | Opened (Valve was cycled until 458 min) |
| 458 | Opened |
| 554 | Closed |
| 560 | Opened |
| 570 | Closed |
| 601 | Opened |
| 672 | Closed |
| 754 | Opened |
| 763 | Closed |
| 772 | Opened |
| 795 | Closed |

TABLE 4. PRESSURIZER EVENTS

| Event Time | Pressurizer Event | Physcial Mechanism | Assumptions Used |
|---|---|---|---|
| 0 | Level increase | RCS fluid expansion due to decreased heat transfer in the SG's | None |
| 3 s | PORV opens | Pressure setpoint of 2255 psig reached | None |
| 8 s | Level decrease | RCS fluid contraction due to reactor scram | None |
| 54 s | Level increasing to off-scale by 6 min | RCS fluid expansion due to dryout of the SG's and voiding in the upper head as RCS depressurized | Upper head begins to void when pressure drops to saturation pressure for initial core outlet temperature |
| 5 min | Beginning of two-phase flow in the RCS loop | Energy balance between decay heat and energy removal mechanisms | Increasing SRM output is an indication of two-phase flow into the downcomer |
| 10 min | Level returns on-scale, but high | Boiling in PZR due to RCS depressurization and energy input from PZR heaters | Energy balance in the PRZ results in PZR at saturation temperature |
| 95 min | 25 in. level drop | RCS depressurization due to increased AFW in SG-A | PRZ at saturation temperature |
| 100-139 min | PRZ level decreasee from 360 to 310 in. over period | Boiling in saturated PRZ due to heater input. Mass balance of steam (inlet + generated = exit) | PRZ at saturation temperature |
| 139 min | PORV block valve closure, with no change in PRZ level | RCS is repressurizing, which maintains sufficient DP to hold liquid in PRZ due to surge line seal configuration, compounded by condensation in PRZ | PRZ is at 7-10 psi lower pressure than the hot leg due to repressurization and condensation. Hot leg level below surge line entrance |
| 144 min | PRZ level drops by 20 in. | Unknown | PRZ level drop is due to the increase of AFW TO SG-A at this time |
| 153 min | PRZ level drop of 10 in. | Unknown | PRZ level drop due to increased AFW flow to SG-B at this time |
| 174 min | PRZ level increase of 75 in. when 2B pump is restarted | Condensation in subcooled PRZ as system rapidly repressurizes. Steam enters surge line | Pressure difference large enough to drive 38 lbm/s steam from hot leg into PRZ (>20 psid) |
| 192 min | PRZ level drop when PORV block valve is opened | Flashing of saturated liquid in PRZ results in lower steam velocities in surge line allowing liquid to drain into hot leg | PRZ is at near saturation when the PORV opens |
| 195 min | PRZ level increase of 30 in. when the PORV block valve is closed | Condensation as the RCS repressurizes by about 20 psi | PRZ was subcooled as a result of the small RCS pressure increases |
| 198 min | PRZ level drop of 25 in. when block valve is opened | Response of saturated PRZ to RCS pressure decrease | PRZ is at saturation |

TABLE 4 (continued)

| Event Time | Pressurizer Event | Physical Mechanism | Assumptions Used |
|---|---|---|---|
| 199 min | PRZ level increase of 20 in. when block vlave is closed | Response of PRZ to slight RCS pressure increase | PRZ becomes slightly subcooled due to repressurization |
| 200 min | PRZ level decrease of 140 in. | Boiling of saturated liquid in the PRZ as RCS rapidly depressurizes in response to HPI injection | Steam condensation on the cold liquid drove the RCS depressurization |
| 211 min | PRZ refill | HPI repressurizes system and forces liquid in PRZ when level in hot leg reaches surge line elevation | Level in hot leg is at surge line elevation at beginning of fill |
| 270 min | Heater group 10 ground fault trip | Unknown | PRZ liquid level is above the heaters. Surge line temperature at 315 min equal 303°F |
| 287 min | Heater groups 4 and 5 trip on ground fault | Unknown | PRZ level is above the heaters |
| 318 min | RCS repressurization when the PORV block valve is closed. PRZ level shows no response | PRZ is completely full and very subcooled (surge line temperature recorded as 303°F). No vapor space to compress. | The surge line temperature is indicative of the PRZ temperature and the level is above the top level tap in the PRZ |
| 330 min | Heater group 3 trip on ground fault | Unknown | PRZ level is above the heaters |
| 458 min | RCS depressurization when PORV block valve is opened. No immediate PRZ level response, and then small drop | The PRZ is liquid full of subcooled water with little or no vapor space. As RCS depressurization gas bubble expands and gases are pulled into PRZ | Liquid level in hot leg initially above surge line. Level decreases as bubble expands and liquid leaves system through PORV |
| 458-595 min | Extended RCS depressurization results in very small level response | When level in hot leg falls to the surge line entrance, the gas flow into the subcooled PRZ limits liquid drain | A significant portion of the gas influx to the surge line is noncondensible gases. PRZ very subcooled, which holds liquid in PRZ |
| 595 min | Heater group 8 trips on ground fault | Unknown | PRZ level is above the heaters |
| 672 min | PRZ level decrease of 220 in. when PORV block vlave closed | PRZ was at saturation from three hours of hot gas flow into the PRZ. When PORV closed energy input from the heaters vaporized liquid forcing drain | None |
| 693 min | PRZ refill over 55 min | Condensation reducing PRZ pressure and drawing liquid into surge line from hot leg | Hot leg level above surge line entrance |
| 803 min | PRZ drained 110 in. when makeup inceased | Increased makeup resulted in condensation induced depressurization (30-50 psi) which caused vaporization of saturated liquid (with the heaters) and drain | None |
| 830 min | PRZ refill over next 30 min | Continued makeup resulted in primary level and pressure increse, which forced liquid into PRZ | RCS pressure increase due to compression of gases by makeup flow |

then decreased, and then increased off-scale high as the Reactor Coolant System (RCS) liquid density changed in response to the changing average fluid temperature. The level then returned on-scale, as the RCS continued to depressurize and the pressurizer liquid boiled. Once the RCS pressure stabilized, due to system voiding, the pressurizer level also stabilized at a level of approximately 950 cm (375 in.). The flow into the pressurizer surge line was probably all liquid at near saturation temperature for the first 25 min. During this time the pressurizer level fell because of boiling in the pressurizer, driven by the depressurization and pressurizer heaters. The pressurizer level response for subsets of this phase, is presented and discussed next.

0 - 10 Minutes

The pressurizer level and primary system pressure for the first 10 min of the accident are shown in Figure 3, (time zero is taken as the main feedwater pump trip). The pressure was obtained by combining the valid portions of the reactimeter narrow-range pressure data with the saturation pressure obtained from the A-loop hot-leg temperature on the reactimeter. In Figure 3 the transition point between the reactimeter pressure and saturation pressure data occurs at about 2.2 min, with the first data point from the saturation pressure at 4.8 min. The saturation pressure is used until the A-loop pumps were turned off at 100 min. The temperatures in the cold leg and hot leg of the A-loop, along with the saturation temperature based upon the system pressure, are shown in Figure 4. During the first 8 s following the feedwater pump trip, the pressurizer level increased to 650 cm (256 in.) from an initial level of 569 cm (224 in.). This was due to RCS fluid expansion with the increase in average system temperature which resulted from reduced heat removal in the steam generators as the secondary levels decreased. This initial in-surge of coolant was followed by an out-surge from the pressurizer as the RCS fluid contracted following reactor scram (with continued steaming from the steam generators and flow through the PORV). At 41 s, the operators increased makeup flow to approximately 25 L/s (400 gpm). The pressurizer level reached a minimum of 401 cm (158 in.) at 54 s, and the level began increasing at approximately

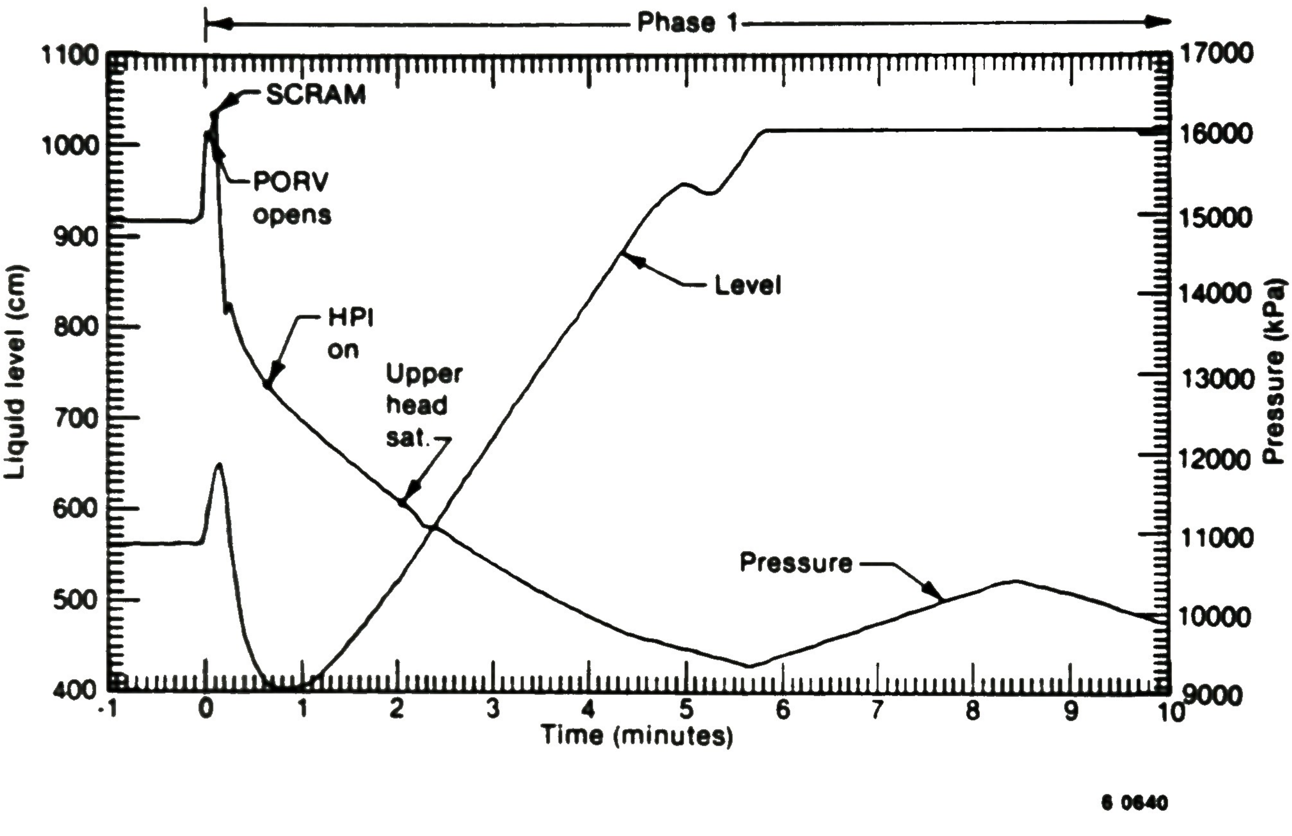


Figure 3. Comparison of pressurizer liquid level and primary system pressure.

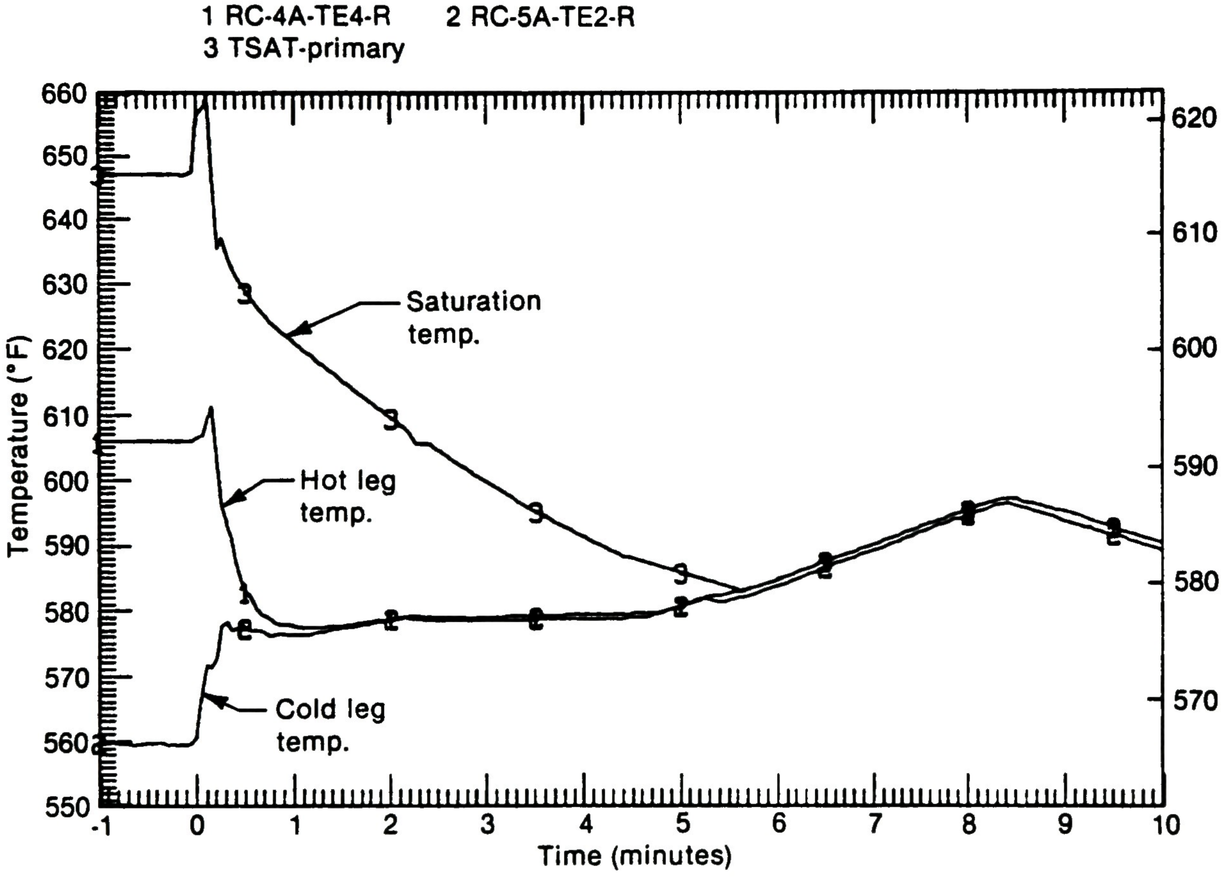


Figure 4. A-loop hot leg, cold leg, and saturation temperatures.

150 cm/min (60 in./min) as the average system temperature increased. At about 1.5 min into the transient, the steam generators' secondary pressures began falling, indicating that the steam generators were drying out. Also, at this time the A-loop hot- and cold-leg temperatures equalized, indicating that energy removal from the steam generator was near zero. At this time the reactor system was liquid-full with the exception of the pressurizer steam space. However, at approximately 2 min into the transient, the system pressure had dropped sufficiently for the fluid in the vessel upper head to reach its saturation pressure (about 11 MPa at 592 K or 1600 psig at 605°F). Also at this time (2 min), the HPIS was actuated on the ES signal when the system pressure had dropped to 11.3 MPa (1640 psig). This flow continued for 2.5 min. As the pressure continued to drop, the upper head void increased and acted as another pressurizer for the system. Indeed, the upper head fluid was probably at a higher temperature than the pressurizer fluid was. The continued decrease in system pressure resulted in reaching saturation pressure in the hot and cold legs at about 5 min. By this time, the continued PORV flow, coupled with the increasing steam void in the upper head and increasing system average temperature, resulted in the pressurizer level increasing to off-scale high (greater than 1016 cm or 400 in.).

During the final four minutes of this period, the level in the pressurizer remained off-scale high. With the PORV still open, an all-liquid or low-void fraction flow out the PORV probably resulted, with an energy loss through the PORV approximately 60 times greater than the combined power of 1.4 MW from the pressurizer heaters (see Appendix B). At 8 min, the block valves for the auxiliary feedwater (AFW) to the steam generators were opened and primary to secondary heat transfer increased dramatically. AFW was capable of removing 160 MW of energy from the RCS at the maximum AFW flowrate. The actual AFW flowrate is unknown. Primary to secondary heat transfer resulted in a continuous decrease in the average system fluid temperature over the next 20 min. With the reactor coolant pumps running, the temperature around the loops was nearly homogeneous (a calculated temperature rise across the core of 3°F). Flow into the surge line was probably mostly liquid (very low void fraction).

## 10 - 73 Minutes

The pressurizer level and primary system pressure for the first 100 min of the accident are shown in Figure 5, with the A-loop hot- and cold-leg temperatures compared to the system saturation temperature in Figure 6. The pressurizer level indication was on scale, but high (from 914 to 990 cm or 360 to 390 in.) during the 10-73 min time period. Continued system depressurization, due to flow out the open PORV, coupled with increased letdown flow and decreased makeup flow, as the operators attempted to reduce the pressurizer level, resulted in an increasingly voided RCS. (RELAP5 calculations indicate that voiding in the loops began at about 7 min.) With all four primary coolant pumps running and the steam generators essentially dry (as indicated by the secondary levels recorded on the reactimeter), flow throughout the system was predominately homogeneous two-phase flow. This condition existed until the B-loop pumps were shut off at 73 min. At approximately 25 min, output from the out of core neutron Source Range Monitor (SRM) began increasing. This coupled with the decreasing loop flow measurement on the reactimeter, indicated that the system void fraction was increasing. By 30 min the RCS pressure stabilized at approximately 7 MPa (1000 psig), where it remained throughout the remainder of this phase. During this period, one of the primary energy removal mechanisms from the primary system was the flow through the PORV. Decay power in the core was about 37 MW (at 45 min) whereas energy removal through the PORV (assuming steam flow) was approximately 17 MW. At the maximum AFW flowrate, the two SG's were capable of removing 160 MW of energy from the system. The actual AFW flowrate during the accident is unknown. (The boundary condition used in the RELAP5 calculations for the steam generators was the secondary levels recorded on the reactimeter.)

A balanced makeup and letdown flowrate of 4.7 L/s (75 gpm) would have removed about 5 MW from the primary system. At 43 min, the operators requested a printout of the values for the 3 pressurizer differential pressure (level) measurements. These values were listed on the utility printer as RC-1-LT1=269 cm =106 in., LT2=279 cm =110 in., and LT3=257 cm =101 in. (The differential pressure is given in cm of water at 293 K.)

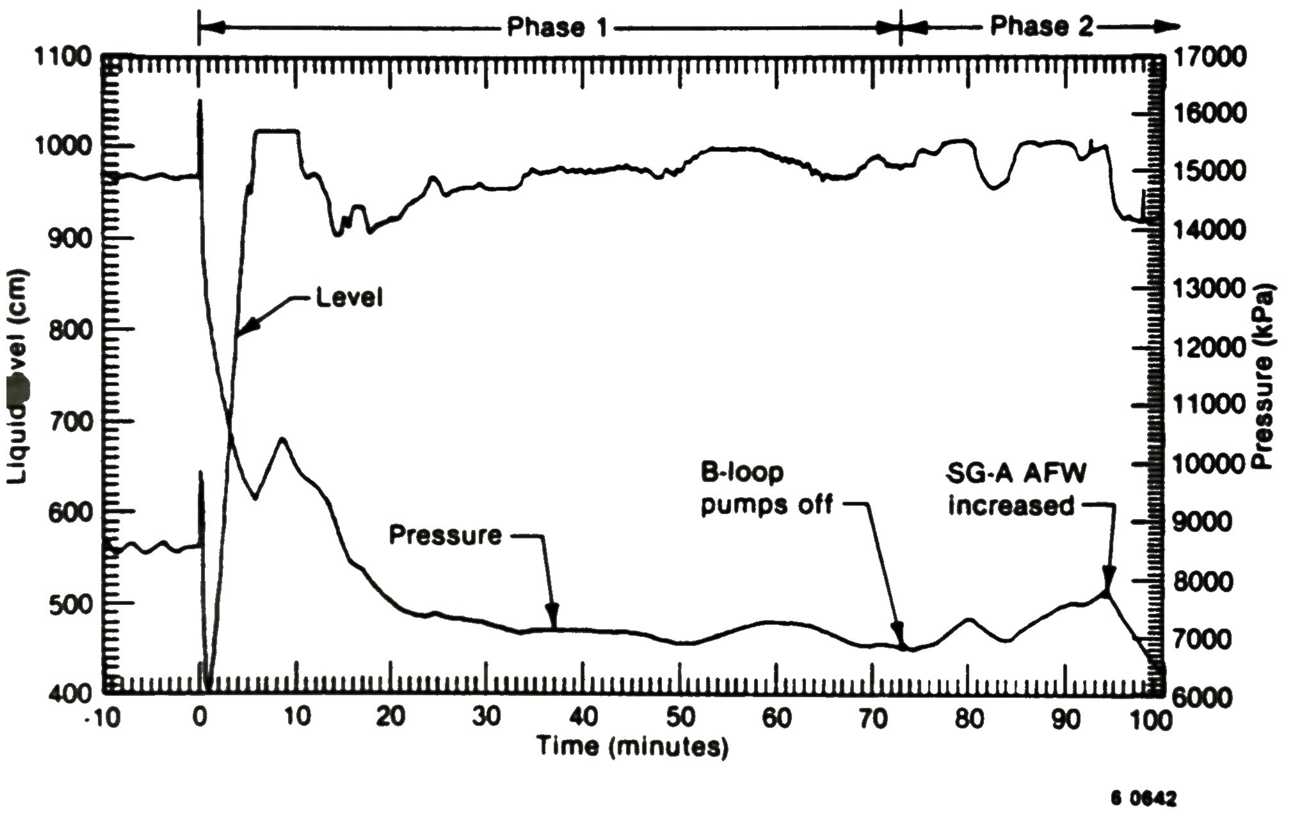


Figure 5. Comparison of pressurizer liquid level and primary system pressure.

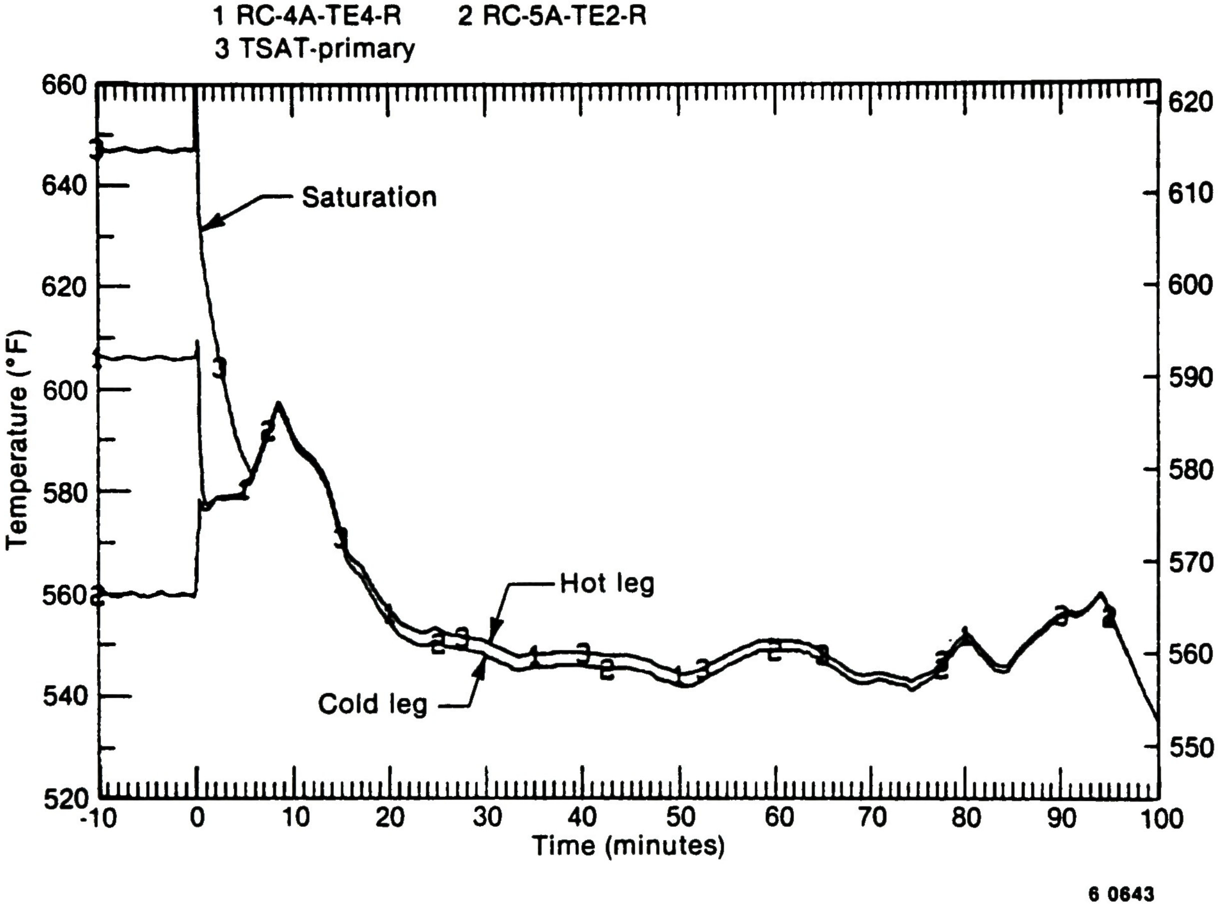


Figure 6. A-loop hot leg, cold leg, and saturation temperatures.

Due to the close agreement between the independent level measurements, it was determined that the pressurizer level indication was correct. At 73 min, the B-loop pumps were turned off due to low current and high vibration. This allowed phase separation to occur in the B-loop and flow to stagnate, with little or no communication with the A-loop. Indication of the flow stoppage was the falling secondary pressure in the B-loop SG. This reduced the energy removal from the primary system.

## Phase 2 - Continued Depressurization

The second phase of the accident is defined as the time period from the shutdown of the B-loop reactor coolant pumps at 73 min, to the closure of the PORV block valve at 139 min. During this phase, the pressurizer level steadily decreased (with exception of minor increases) to a level of 790 cm (310 in.). The decreasing pressurizer level was a direct consequence of the continued RCS depressurization, with boiling in the saturated pressurizer driven by the energy input from the heaters. Auxiliary feedwater flow was increased to the A-loop steam generator, which resulted in increased primary to secondary heat transfer and depressurization of the RCS. A major event during this phase was the shutdown of the A-loop reactor coolant pumps, which ultimately resulted in core uncovery and major damage to the core.

### 73 - 139 minutes

The pressurizer level and RCS pressure is shown in Figure 7 for the 50 to 300 min time period of the accident. The pressure was obtained by combining the valid portions of the reactimeter narrow-range pressure data with the digitized strip chart wide-range pressure data. The uncertainty involved in the digitization process has yet to be evaluated; therefore, use of these data should be with a certain amount of scepticism. The A-loop hot-[a] and cold-leg temperatures are compared to the saturation

a. The hot-leg temperature in Figure 8 was obtained from the wide-range temperature (273-700 K) recorded on a multipoint recorder once every 2.5 min.

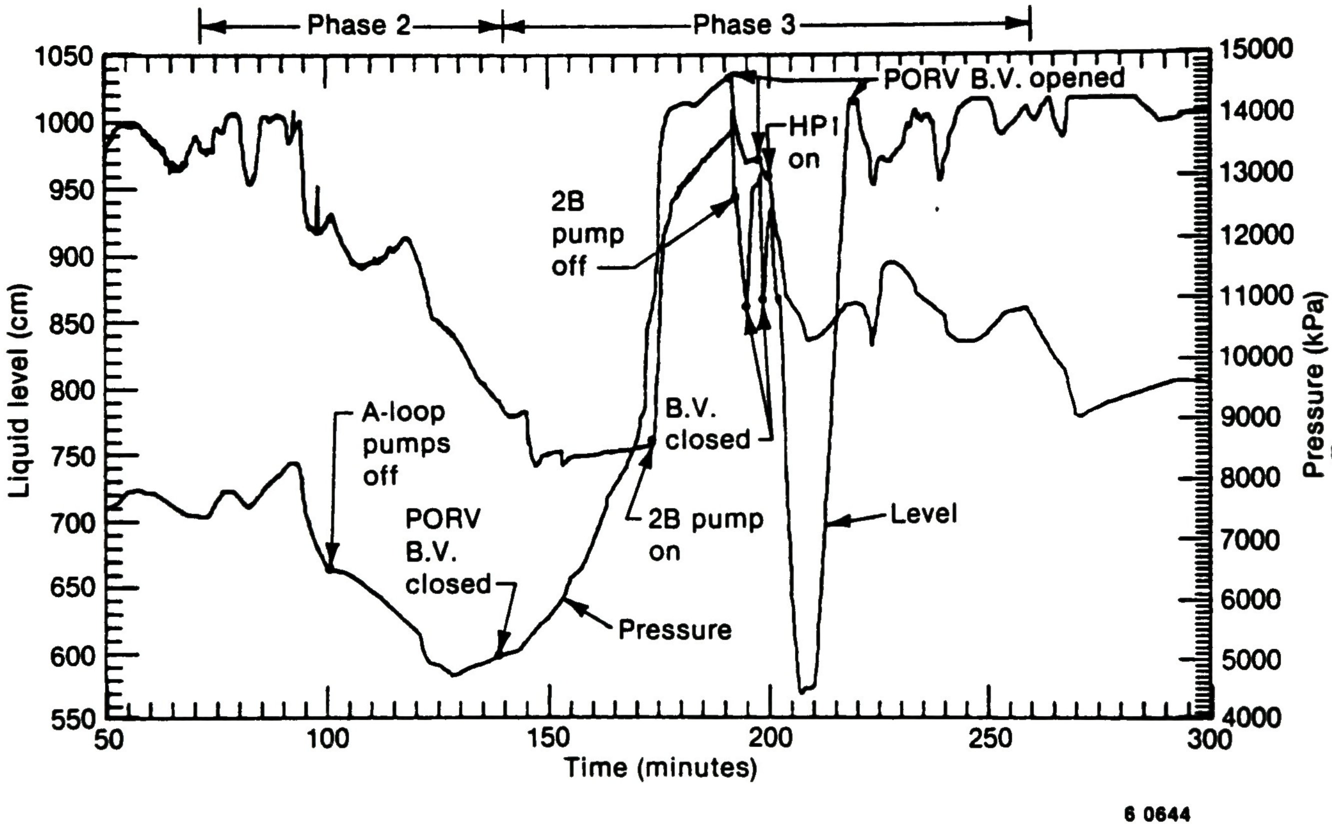


Figure 7. Comparison of pressurizer liquid level and primary system pressure.

temperature in Figure 8. The best estimate of the net liquid influx to the primary system (makeup flow minus letdown flow) is shown in Figure 9 for the same time period.[5] At 81 min, the pressurizer surge-line temperature[a] was output on the alarm printer as 541 K (514°F). The saturation temperature corresponding to the RCS pressure at that time was 562 K (552°F), indicating that the pressurizer inlet was about 20 K (38°F) subcooled at that time.

At 90 min, the output from the out-of-core neutron source and intermediate-range monitors (SRM and IRM) increased, indicating voiding in the core and/or downcomer, which allowed more neutrons to escape the vessel. At 94 min, AFW was increased to the then almost dry A-loop SG, and SG-A steaming was switched from the condenser to the atmospheric dump valves (ADV) which decreased the secondary side pressure. This resulted in increased primary to secondary heat transfer and increased condensation of steam in the primary system, producing a sharp drop in RCS pressure (about 1.4 MPa or 200 psig). This abrupt drop in pressure resulted in a drop in pressurizer level as the previously saturated liquid in the pressurizer flashed into steam. This accounts for some of the steam flow out the open PORV. Flow out the PORV was probably all steam (see Appendix B, pg B-2). At 100 min, both A-loop pumps were stopped due to excessive pump vibration. This allowed the previously homogeneous two-phase mixture in the primary system A-loop to stratify, with a level somewhere in the vicinity of the top of the core (almost certainly below the surge-line elevation in the hot leg). Starting at this time, the liquid pool in the core was boiling, with loss of system mass as steam flow into the pressurizer surge line and out the PORV. Since the indicated pressurizer level was less than 980 cm (370 in.), continued flow out the PORV was probably saturated steam (see Appendix B). While the PORV was open, steam velocities were probably high enough into the surge line that liquid flow from the pressurizer was limited by counter-current flow considerations. Flooding calculations (Appendix B) indicate that the liquid flow out of the

a. The surge-line temperature is measured with a thermocouple strapped on the outside of the surge-line pipe. As a result, the measured temperature will tend to read lower than the actual fluid temperature.

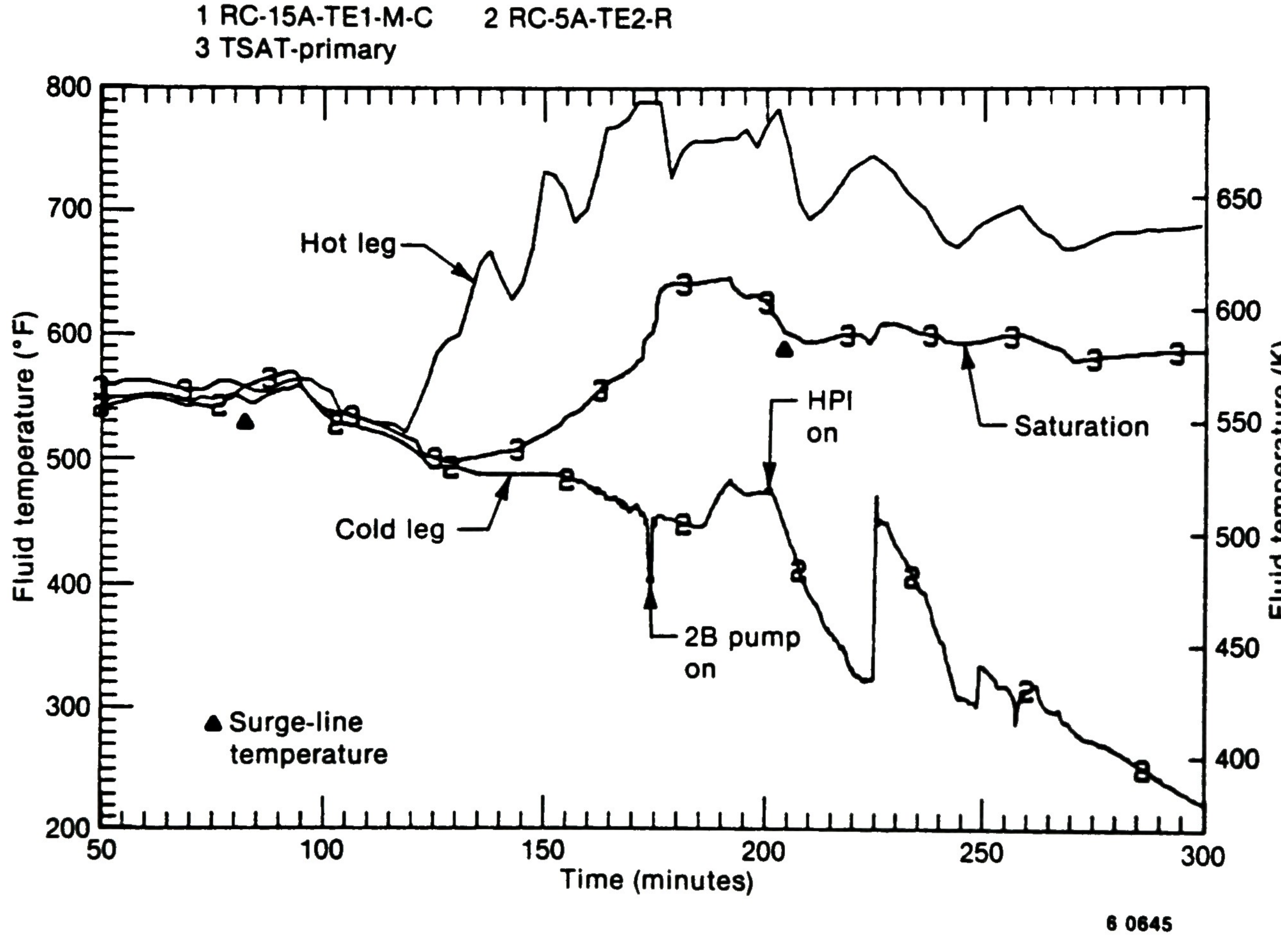


Figure 8. A-loop hot leg, cold leg, and saturation temperatures.

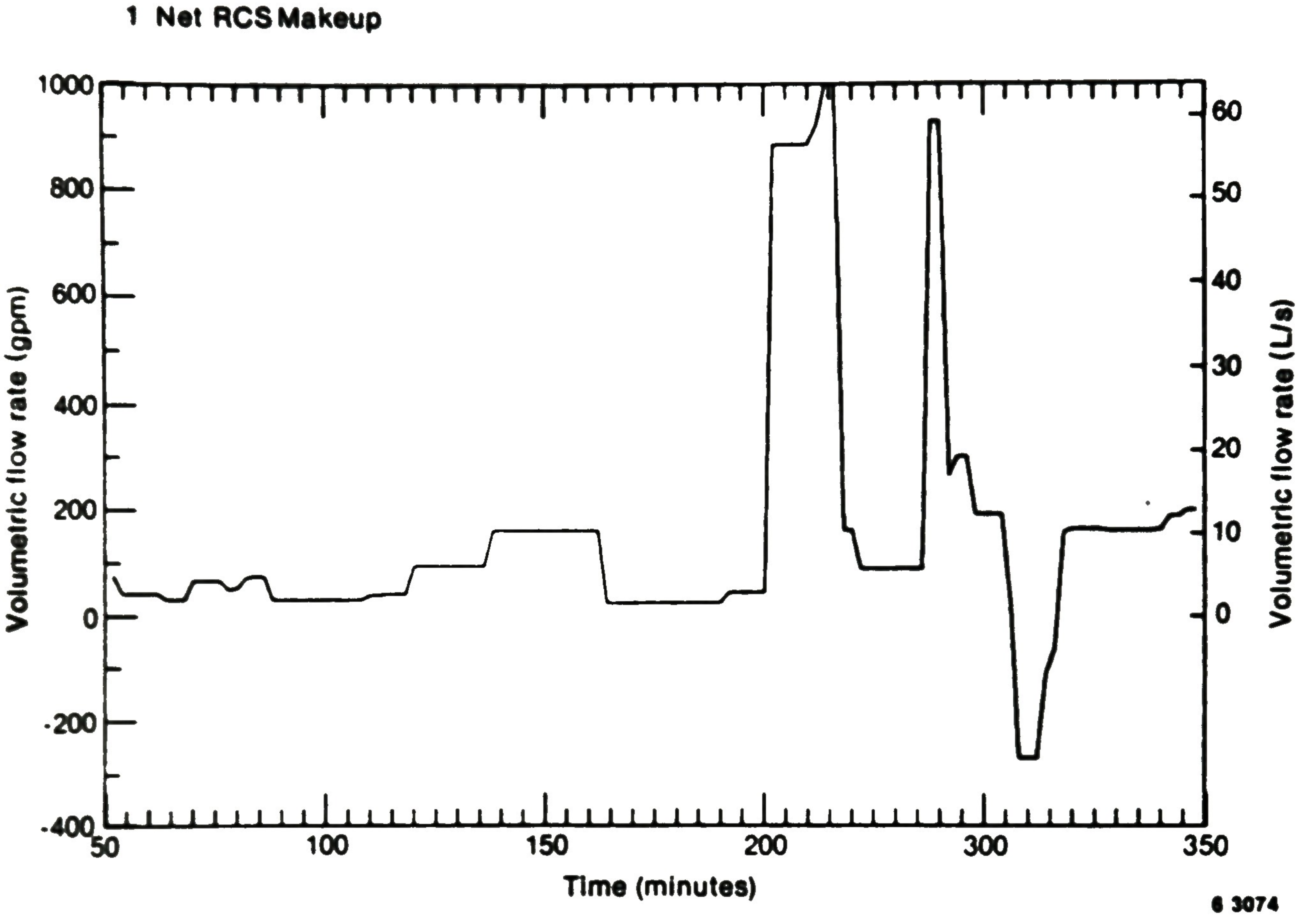


Figure 9. Estimated net liquid injection rate.

pressurizer (into the hot leg) would be zero whenever the RCS pressure was greater than about 3-6 MPa (400-800 psig) (assuming all steam flow into the surge line). This is the pressure range for which critical steam flow through the PORV would result in steam velocities in the surge line greater than the critical velocity from the Wallis flooding criteria.[15] The pressurizer liquid level continued to decrease due to steam generation in the pressurizer by the heaters. Between the time that core uncovery began (at about 125 min[9]) until the PORV block valve was closed at 139 min, the level decreased at a rate of 4.6 cm/min (1.8 in./min). Heater operation at a power of 1386 kW would account for a rate of 2.0 cm/min (0.8 in./min). The remainder of the steam flow out the PORV would have been from steam generated in the core and entering the pressurizer through the surge line. Since part of the steam flowing out the PORV was generated in the pressurizer, the reduced steam velocities in the surge line probably allowed some liquid to drain out of the pressurizer. Thus, the difference in calculated and observed level decreases.

A few minutes following shutdown of the A-loop pumps (at 100 min), output from the source range monitor (SRM) increased indicating that the downcomer level began dropping below the top of the core. (The increase in the SRM output could also be interpreted as a result of the downcomer and core void fraction both increasing.) The A-loop hot-leg temperature started a rapid increase at about 118 min, indicating that core uncovery had started and that superheated steam was being generated in the core. At about 130 min the RCS pressure began increasing, a further indication of increased vapor generation (both superheated steam and hydrogen). At 134 min, the output from the radiation monitors in the containment building began increasing, indicating that fission products were escaping the primary system through the PORV following failure of the fuel-rod cladding.

At 138 min, with saturated steam flow out the PORV, approximately 9 MW of energy was being removed from the system through the open PORV, compared to a core decay heat output of about 28 MW. The B-loop steam generator was isolated, and AFW to SG-A was interrupted. How much decay heat was removed by the SGs is unknown. However, a significant portion of the decay heat was probably going into heating the fuel rods. The cladding temperatures in the upper portion of the core may have reached a sufficient magnitude that the zircomium-water reaction had begun to generate significant energy and hydrogen. At 139 min the block valve upstream of the PORV was closed, stopping loss of coolant from the system and terminating this phase of the accident.

## Phase 3 - Initial Repressurization

This phase of the accident extends from the closure of the PORV block valve at 139 min until the beginning of sustained makeup injection at 261 min.

> "During the initial portions of this phase the primary system can be characterized as essentially static with minimal heat removal via the steam generators, even though attempts to start natural circulation were made. During this portion of the accident phase the RCS pressure continuously increased. A major thermal-hydraulic event during this period was the starting of one of the reactor coolant pumps after attempts to initiate natural circulation had failed. This resulted in a sharp increase in RCS pressure and pressurizer level. When, with one pump running, there was still no evidence of flow in the system, a series of manipulations of the relief block valve and the high pressure injection system were carried out. These manipulations apparently led to the decision to sustain high pressure injection which initiates Phase 4 of this discussion."[8]

During this phase of the accident, the pressurizer level indicated a number of large increases and decreases in response to conditions in the RCS. The first of these was a large in-surge due to the restart of the 2B pump at 174 min, and the resulting increase in RCS pressure. The pressurizer drain at 200 min was a result of a saturated pressurizer and a RCS depressurization induced by condensation on the cold HPIS liquid in the cold legs. Refill of the pressurizer at 210 min probably resulted from condensation in the pressurizer as the system pressure increased, coupled with continued HPI. This resulted in a primary system liquid level above the surge-line entrance to the hot leg.

139 - 174 minutes

Following the closure of the PORV block valve at 139 min, the system pressure began an increase of 3.8 MPa (550 psi) over the next 35 min (110 kPa/min or 16 psi/min). Prior to the valve closure, the A-loop hot-leg narrow-range temperature measurement recorded on the reactimeter went off-scale high (above 600 K or 620°F), indicating that superheated steam had already formed in the top of the core and the reactor vessel upper plenum.

At approximately 155 min, the A-loop cold-leg temperature began to decrease and the SRM output began to fall more rapidly. A possible explanation for this response is the operation of makeup pump MU-P-1C with injection of cold liquid in the A-loop cold legs, and the resulting injection of liquid into the downcomer. This operation cannot be verified by the alarm printer, since the alarm indications from the printer are unavailable for this time. Injection into the A-loop cold legs is not standard procedure for makeup flow, and this explanation is not supported by Reference 11.

During the next 21 min (until 174 min), the RCS pressure increased 2.8 MPa (400 psi), while the pressurizer level increased by 13 cm (5 in.). This small level increase may have been a result of steam condensation in a slightly subcooled and bottled up pressurizer. Because of the surge-line seal configuration and the increasing system pressure, liquid in the pressurizer failed to drain as the liquid maintained a hydrostatic balance with the system. Calculations indicate that a 50-70 kPa (7-10 psi) pressure difference between the hot leg and top of the pressurizer could maintain the pressurizer liquid-full. Since the system pressure was rising, the pressurizer was probably subcooled and a pressure difference of this magnitude is reasonable. A subcooling of 0.5 K (1°F) in the pressurizer steam space (relative to the surge-line entrance in the hot leg) would provide the 70 kPa (10 psi) pressure difference required to maintain a full pressurizer.

174 - 262 minutes

At 174 min, the reactor coolant pump RC-P-2B was successfully restarted, and ran for 19 min. Within the first minute, the pressurizer heaters were de-energized and the pressurizer spray valve opened. (Note that the spray line originates at the discharge of the 2A reactor coolant pump, and operation of the 2B pump would have resulted in minimal flow through the spray line). Restart of the pump resulted in significant liquid flow from the 2B cold leg being forced into the reactor vessel for the first 20 s of operation, perhaps reflooding the undamaged portions of the core. [One flow estimate has 28 $m^3$ (1000 $ft^3$) entering the reactor vessel.][4] Coincident with the pump restart was an approximate 28 K (50°F) drop in the A-loop cold-leg temperature over the first few min of pump operation. This is perhaps an indication of reverse flow from the B-loop into the A-loop. An indication of additional liquid in the downcomer (and perhaps the core) was the abrupt drop in output from the SRM, as neutrons were absorbed by the liquid. As liquid penetrated the core, a large amount of steam and/or hydrogen was generated, resulting in a rise in the RCS pressure of 5.5 MPa (800 psi) in 2 min, with a further 1 MPa (150 psi) increase over the next 16 min. Coincident with this large pressure increase was a sharp rise in the pressurizer level from 762 to 914 cm (300 to 360 in.), with a further slow rise to 990 cm (390 in.). It is postulated that the level rise was due to level swell in the pressurizer as steam from the hot leg entered the subcooled pressurizer, condensed, and raised the fluid temperature in the pressurizer. Assuming that the pressurizer was at saturation conditions when the pump was restarted, then mass and energy calculations (Appendix B) indicate that a 17 kg/s (38 lbm/sec) steam flow into the pressurizer, over a 2 min period, would increase the fluid temperature from 571 K (567°F) [saturation at 8 MPa (1200 psig)] to 594 K (610°F), which would be subcooled at the final pressure of 14 MPa (2000 psig). This would result in the observed 152 cm (60 in.) level increase, with 56 cm (22 in.) of this increase due to level swell as the liquid heated up. The required steam flow rate of 17 kg/s would produce an approximate 140 kPa (20 psid) pressure drop through the surge line. Since only 1 K subcooling in the pressurizer could produce this pressure difference, the required steam flow rate is a reasonable expectation. Since a flow path existed from the pressurizer steam space to the 2A cold leg through the open spray line, condensation may have occurred in the spray line which would tend to lower the pressurizer pressure. Since there is an indication of reverse flow into the 2A cold leg (the drop in fluid temperature), it is possible that some subcooled liquid may have entered the pressurizer through the spray line, which would result in further steam condensation.

At 192 min, the operators opened the PORV block valve, resulting in a drop in RCS pressure of about 1.4 MPa (200 psi), and a drop of 127 cm (50 in.) in the pressurizer level in 3 min. This was a result of the decreasing system pressure coupled with a nearly saturated pressurizer. As the pressure dropped, the saturated liquid in the pressurizer flashed into steam. A 1.4 MPa (200 psi) drop in pressure would result in formation of 1100 kg (2500 lbm) of steam from the liquid in the pressurizer. This 1100 kg (2500 lbm) of steam would have resulted from flashing 1.9 $m^3$ (66 $ft^3$) of saturated liquid, and decreased the pressurizer level by 51 cm (20 in.) as compared to the observed 127 cm (50 in.) drop in level. At a pressure of 13.8 MPa (2000 psig), the calculated steam flow rate out the PORV is approximately 15 kg/s (32 lbm/sec), for a total steam flow of 2600 kg (5,800 lbm) over the 3 min the PORV block valve was open. The steam flow out the PORV would have been a combination of steam generated in the pressurizer and steam flow through the surge line from the hot leg. The decreased steam velocities in the surge line could have permitted some liquid to drain out of the pressurizer, thus accounting for the observed 127 cm (50 in.) decrease in level.

At 195 min the PORV block valve was closed, resulting in a 200 kPa (30 psi) rise in pressure over the next 6 min. This in turn resulted in a 102 cm (40 in.) rise in the pressurizer level. As the system pressure increased, the pressurizer became increasingly subcooled relative to the hot leg. The resulting condensation in the steam space reduced the pressurizer pressure and drew steam into the surge line from the hot leg. At 198 min the PORV block valve was opened for 30 s. This resulted in another drain/fill cycle in the pressurizer. Although another cycle of the block valve has not been reported within a reasonable time period of this event, analysis of the RCDT and RCS pressures indicate that this brief valve operation did occur (see Appendix D).

At 200 min, the operators initiated the makeup pumps in the HPIS mode at an injection rate of approximately 63 L/s (1000 gpm) over the next 15 min, resulting in a sustained decrease of the RCS pressure to about 10 MPa (1500 psig). The depressurization was driven by condensation of steam due to the injection of cold makeup liquid. This event is analyzed in Appendix B. Assuming that the pressurizer was still at saturation, this decrease in RCS pressure would result in the pressurizer liquid boiling, with the resulting steam formation displacing liquid in the pressurizer, causing a decrease in the liquid level. This depressurization of saturated liquid would generate approximately 29 $m^3$ (700 $ft^3$) of steam if all of the initial liquid was available for vaporization, compared to the 13 $m^3$ (450 $ft^3$) of liquid that was displaced. If only the liquid remaining in the pressurizer after the level drop was available for vaporization, then approximately 7.6 $m^3$ (269 $ft^3$) of steam would have been generated. Thus, the observed level drop is bracketed by these two assumptions and the postulated mechanism of vaporization of saturated liquid is sufficient to explain the observed level decrease. At 204 min, the pressurizer surge-line temperature was recorded on the alarm printer as 578 K (581°F), with a system saturation temperature of 592 K (605°F). Since the thermocouple measuring the surge-line temperature is strapped on the outside of the pipe, it can be expected to measure a somewhat lower temperature than the fluid within the surge-line pipe.

At 207 min, the pressurizer level decrease stopped, and at 210 min the pressurizer level began increasing until it increased off-scale high by 218 min. Coincident with this level increase was a repressurization of the RCS by about 0.6 MPa (80 psi). The pressurizer level increased from 585 to 1015 cm (230 to 400 in.) in 8 min, which corresponds to an injection rate of 22 L/s (350 gpm) of cold water into the system (the in-surge corresponded to 32 L/s (505 gpm) of saturated liquid). It is postulated that the HPIS injection was sufficient to flood the reactor vessel and hot legs to an elevation above the surge-line entrance in the A-loop hot leg. With increasing RCS pressure and condensation in the pressurizer, liquid was drawn into the pressurizer, causing the large level increase. Calculations indicate that HPI injected approximately 57 $m^3$ (15,000 gallons) of cold liquid into the system, whereas approximately 38-53 $m^3$ (10,000-14,000 gallons) would be sufficient to fill the cold legs, vessel, and hot leg to the elevation of the surge line in the A-loop hot leg from an initially empty condition. The pressure increased slightly due to compression of the noncondensible gases by the HPI.

At 219 min, HPI was reduced to about 6 L/s (100 gpm). At 220 min, the PORV block valve was opened and the pressurizer level returned on-scale, accompanied by a 0.7 MPa (100 psi) pressure drop. The pressure decrease may have allowed the noncondensible gas bubble in the hot leg to expand down to the surge-line elevation, permitting gas flow into the pressurizer and resulting in the level decrease. At 225 min, the A-loop cold-leg temperature jumped 70 K (130°F), a probable indication of reverse flow into the A-loop cold leg. This may have been caused by molten fuel falling into the liquid pool in the lower plenum, forcing the hot liquid in the downcomer back into the cold legs. At the same time the RCS pressure rapidly increased by 1.4 MPa (200 psi). This could have been a result of steam generation from molten fuel. An event at 225 min which may have contributed to the A-loop cold-leg temperature rise was the opening of the pressurizer sprayline valve. This may have equalized pressures and resulted in a shift in fluid levels due to hydrostatic head balances. It should be noted that the cold-leg temperature shown in Figure 8 is from an RTD installed in the RC-P-1A pump suction, whereas the sprayline enters the RC-P-2A pump discharge.

## Phase 4 - Sustained Injection

This phase of the accident is characterized by sustained injection of liquid into the primary system at approximately 16-19 L/s (250-300 gpm) of makeup flow in an attempt to refill the system. The period started with increased liquid injection at 262 min, and ended with opening of the PORV block valve at 458 min for depressurization of the RCS. Large quantities of steam and noncondensibles existed in the high points of the system (the upper head, hot legs, and upper portions of the steam generators) during the entire phase.

During this phase, the pressurizer level measurement was indicating a full, or nearly full, pressurizer. The level did not respond to pressure changes in the RCS. This was probably due to a liquid level in the A-loop hot leg above the surge-line entrance, and a highly subcooled pressurizer.

<u>262 - 458 minutes</u>

The pressurizer liquid level is compared with the RCS pressure in Figure 10 for the 250-500 min time frame. The A-loop hot- and cold-leg temperatures are compared with the system-saturation temperature in Figure 11. Also shown in Figure 11 are temperatures recorded on the alarm and utility printers for the pressurizer, surge line, and spray line.

During this phase approximately 43% of the core decay heat (21 MW) could have been removed by heatup of the cold liquid injected by the makeup system. The B-loop steam generator was isolated. The A-loop steam generator operating level was increased to 100% between 360 and 420 min. However, heat transfer in the SG-A was severely limited by blockage of steam flow into the SG by the collection of noncondensibles in the RCS high points. This resulted in a slight RCS repressurization until 300 min. During the initial 50 min of the period, when the PORV block valve was open, the flow path was from the injection ports in the cold legs, through the cold legs and downcomer, up through the core where a portion of the decay heat was removed, and out of the RCS through the pressurizer and PORV. The core was probably covered, although portions of the core may have been molten and not quenched, with a liquid level in the hot legs above the surge-line entrance elevation. Since liquid was apparently available at the surge-line entrance, with flow out the PORV, the pressurizer stayed full even though the pressurizer heaters were on at an

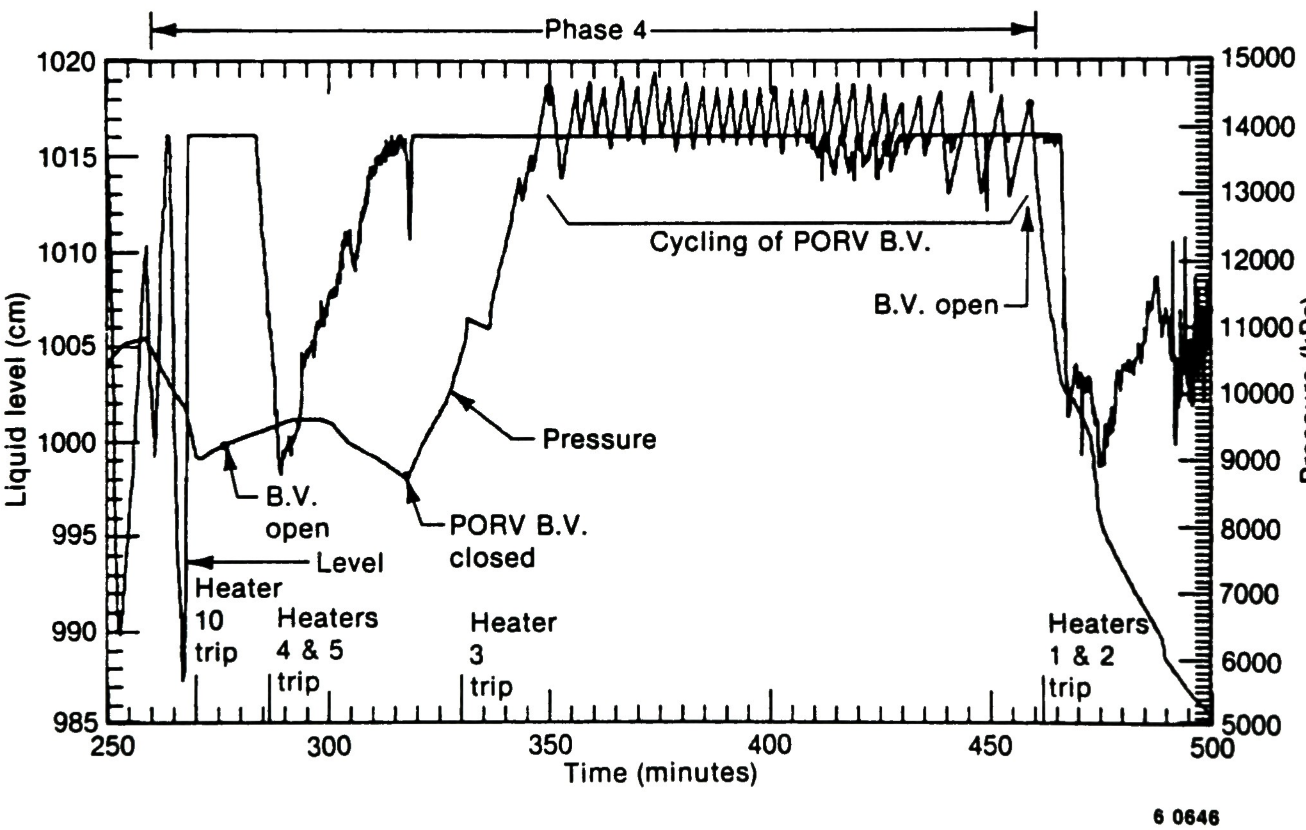


Figure 10. Comparison of pressurizer liquid level and primary system pressure.

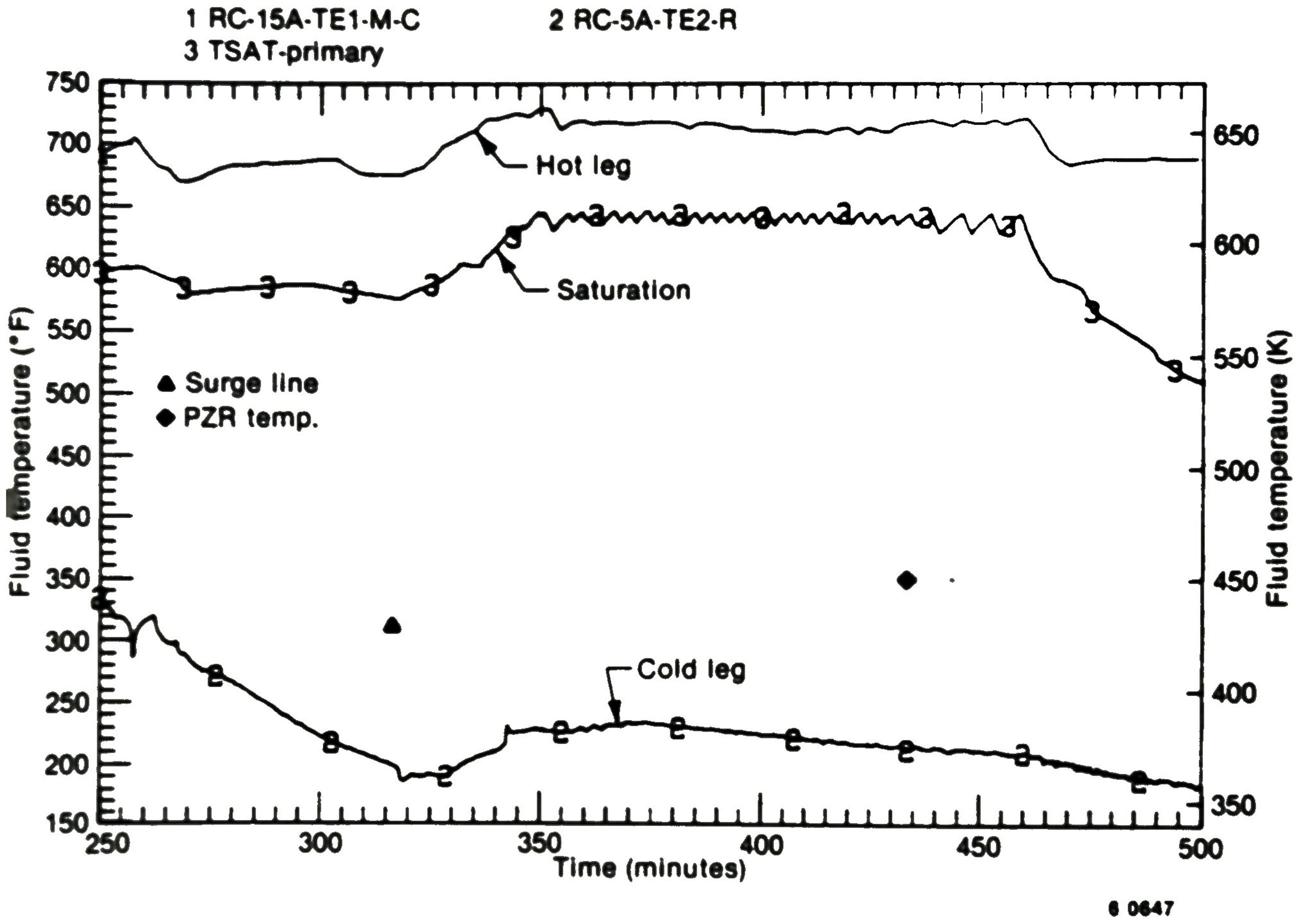


Figure 11. A-loop hot leg, cold leg, and saturation temperatures.

estimated power level of 1386 kW. If the liquid level in the A-loop hot leg had been below the surge-line entrance a flow path would have been available for the noncondensible gases to escape the RCS and the system would have been expected to depressurize, rather than the observed repressurization.

At 270 min, the pressurizer heater group 10 tripped due to a ground fault. A ground fault is a condition in which the current flow in a circuit becomes unbalanced due to breakdown in the insulation, such as would occur if the heaters were shorting or the cabling was wet. Heater groups 4 and 5 tripped due to ground faults at 286 min, and heater group 3 tripped due to ground fault at 330 min. At 315 min, the surge-line temperature was recorded on the utility printer as 424 K (303 °F), which was about 150 K (275 °F) subcooled. There has been speculation that these heater trips were due to a dry pressurizer. However, by the time group 3 tripped, more than 45 $m^3$ (12,000 gallons) of liquid had been injected into the system. All indications are that the core was covered prior to this time period, and this amount of injected liquid was sufficient to fill the outlet plenum and hot legs to above the surge-line elevation, and to fill the pressurizer. Therefore, another mechanism for the heater trips needs to be investigated.

At 318 min, the PORV block valve was closed in order to repressurize the system, in an attempt to compress and eliminate the noncondensible gases which existed in the RCS high points. With the block valve closed, and continued makeup, the pressure increased from 8.7 to 14.7 MPa (1260 to 2130 psig) in 30 min. Analysis of this repressurization indicates a compression of the noncondensible gas corresponding to an injection rate of 14 L/s (220 gpm), and a increase in hot-leg level of 4 m (12 ft). Over the next hour, until 458 min, the PORV block valve was cycled open and closed to maintain the RCS pressure between 13.1 to 14.5 MPa (1900 and 2100 psig). During periods when the PORV block valve was open, flow out the PORV was probably all liquid (the pressurizer level measurement indicated a full pressurizer). This implies that the surge-line entrance into the hot leg was covered with liquid and the flow of noncondensible gases into the pressurizer was limited to those gases in solution, and perhaps some gases entering the hot leg as small bubbles. As a result, during this time period very little of the hydrogen which had been previously generated in the core could have been escaping the primary system into the reactor building. Net liquid injection into the system was maintained at about 13 L/s (200 gpm). At 315 min (just before closure of the PORV block valve) the pressurizer surge-line temperature was recorded on the alarm printer as 424 K (303°F), compared to a cold-leg temperature of 361 K (190°F) and a saturation temperature of 577 K (578°F). Obviously the flow into the pressurizer was very subcooled while the PORV block valve was open. Also, the liquid level in the hot legs was below the RTD at an elevation of 107 m (353 ft) (a higher level would have cooled the RTD which was still indicating above 600 K or 620°F). This compares to an elevation of 98 m (321 ft 6 in.) for the surge-line entrance into the hot leg. Calculations indicate that an injection rate of approximately 37 L/s (580 gpm) would have been required to remove the core decay heat of 21 MW (assuming an injection temperature of 311 K (100°F) and a core-exit temperature at the recorded pressurizer temperature with no steam generation). The energy removal mechanism for the excess decay heat is unknown.

Following closure of the PORV block valve at 318 min, the system quickly repressurized to 14.5 MPa (2100 psig). The pressurizer level was off-scale high, and remained in this condition throughout the remainder of this phase of the accident. With the liquid level in the hot leg above the surge-line entrance (a postulate), and increasing system pressure, no mechanism existed for draining the pressurizer as long as the pressurizer was subcooled. Calculations (Appendix B) indicate that if the system high points were filled with noncondensible gas down to a level just above the elevation of the surge-line entrance in the hot leg (98 m or 321 ft 6 in.), when the PORV block valve was closed, then a 14 L/s (220 gpm) net makeup injection rate would produce the observed repressurization rate. This compares to the estimated net injection rate (makeup minus letdown) of 10-16 L/s (160-250 gpm) at this time.

At 433 min, the pressurizer temperature, from the RTD located above the heaters, was recorded on the utility printer as 446 K (342°F), very subcooled when compared to a system saturation temperature of 620 K (650°F). During periods in which the PORV block valve was cycled open, flow was into the pressurizer which was probably completely full with liquid, with flow out the PORV. Calculations indicate that the liquid injection into the system (at a 16 L/s or 250 gpm injection rate) and out through the PORV was removing approximately 9 MW of the 21 MW decay heat. At 433 min, the levels from the individual pressurizer differential pressure transmitters were recorded on the utility printer as (LT1=94 cm =37.1 in., LT2=90 cm =35.5 in., LT3=83 cm =32.5 in.), with a recorded temperature compensated level of 1017 cm (400.5 in.). (Remember that the measured differential pressure approaches 0 as the pressurizer level approaches 1016 cm (400 in.) of cold water.)

## Phase 5 - Extended Depressurization

This phase of the accident, covering the period of 458-672 min, is characterized by an extended depressurization of the RCS in an attempt to reflood the system using the core flood tanks, which are pressurized with nitrogen at 4.1 MPa (600 psig). Conditions at the beginning of this phase consisted of core cooling via makeup injection with flow out through the highly subcooled pressurizer. Noncondensible gases filled the system high points and blocked flow to the steam generators.

During most of the initial portion of this phase, until 650 min, the pressurizer level showed minimal response to the RCS depressurization, remaining at a level of 990-1016 cm (390-400 in.). The depressurization probably resulted in the expansion of the noncondensible bubble down to the surge-line elevation, with flow of hot noncondensible gases through the pressurizer and out the PORV. This resulted in noncondensible gases bubbling through the pressurizer, with the level measurement indicating slightly less than 1016 cm (400 in.) of collapsed level, and a slow rise in the pressurizer fluid temperature.

The pressurizer level and RCS pressure are shown in Figure 12 for the 400-1000 min time segment. The hot- and cold-leg temperatures are compared to the RCS saturation temperature in Figure 13. Also shown in Figure 13 is the pressurizer temperature which was recorded on the utility printer beginning at 570 min. The estimated net liquid injection rate is given in Figure 14. This phase of the accident was initiated when the PORV block valve was opened at 458 min, which resulted in a continuous pressure decrease of 12.4 MPa (1800 psi) over the next hour. As the pressure decreased, gas bubbles in the system expanded downward, probably resulting in increased noncondensible gas flow into and through the pressurizer. However, the net makeup rate was still about 13 L/s (200 gpm), and most of the flow into the pressurizer would have been subcooled liquid. At 460 min, the pressurizer level indication returned on-scale, but remained very high (above 990 cm or 390 in.). This was probably due to gases entering the pressurizer, displacing liquid, and resulting in a two-phase interface level above the upper level measurement tap. At 480 min the pressurizer surge-line and spray-line temperatures were recorded on the utility printer as 431 K and 352 K (316°F and 173°F), respectively. Saturation temperature was approximately 560 K (550°F). This was at the time when the spray valve, and probably the pressurizer vent valve were opened. At 492 min, the spray-line temperature was recorded as 347 K (165°F). At 499 min, the surge-line temperature was recorded as 439 K (331°F), and at 505 min, the pressurizer temperature was recorded as 451 K (351°F). These temperatures indicate that the pressurizer was slowly heating up at a rate of about 0.5 K/min (0.8°F/min). The pressurizer heaters, operating at an estimated power output of 600 kW, would have been raising the temperature at a rate of 0.3 K/min (0.5°F/min). The difference between the observed and calculated heatup rates is probably due to the flow of hot noncondensible gases into the pressurizer. Concurrently, there was a slow temperature decrease in the A-loop cold leg of approximately 0.3 K/min (0.5°F/min) resulting from continued injection of cold makeup. By 510 min, the system pressure had reduced to the core flood tanks pressure of 4.1 MPa (600 psig). It has been calculated (Reference 8, pg TH-4) that only about 2.8 $m^3$ (100 $ft^3$) of coolant was injected from the core flood tanks over the next 40 min.

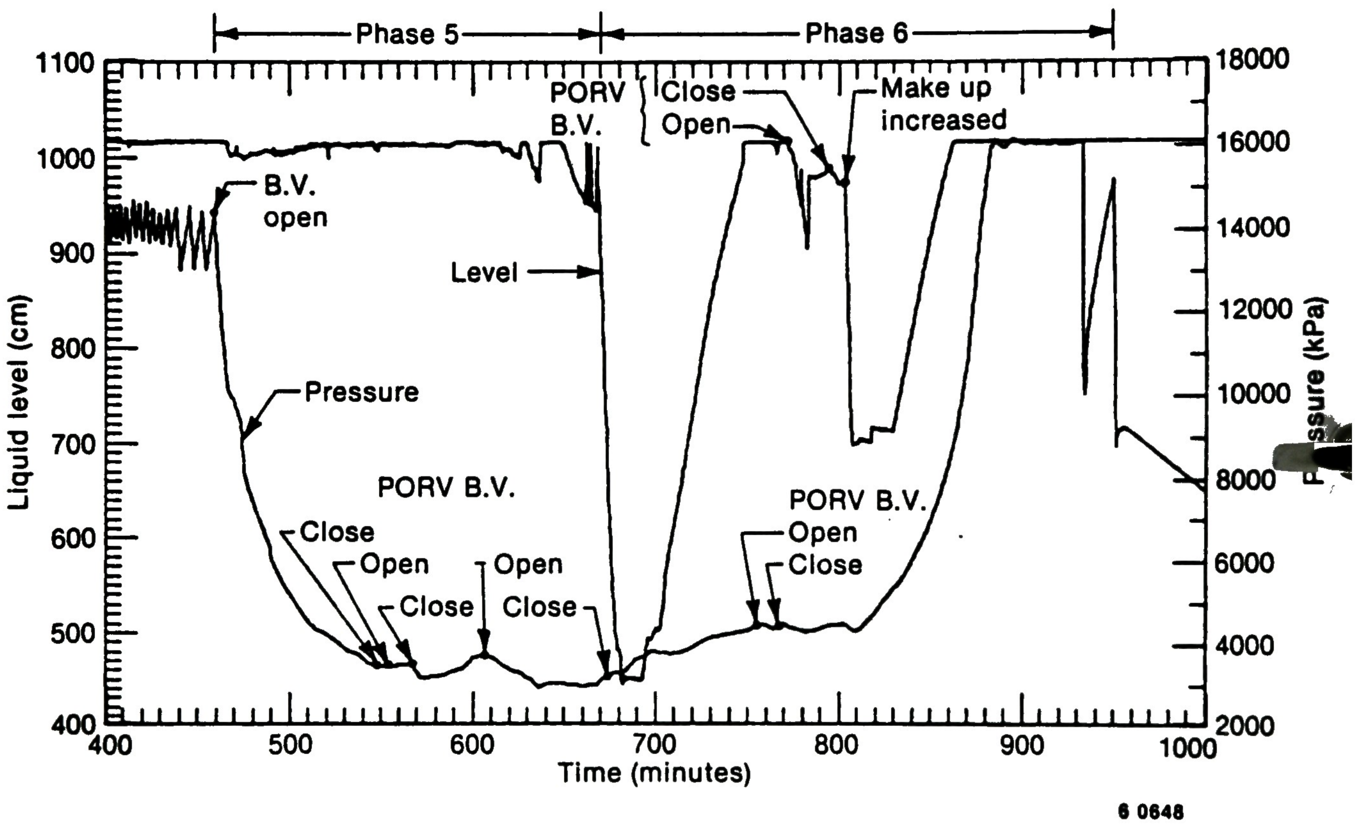


Figure 12. Comparison of pressurizer liquid level and primary system pressure.

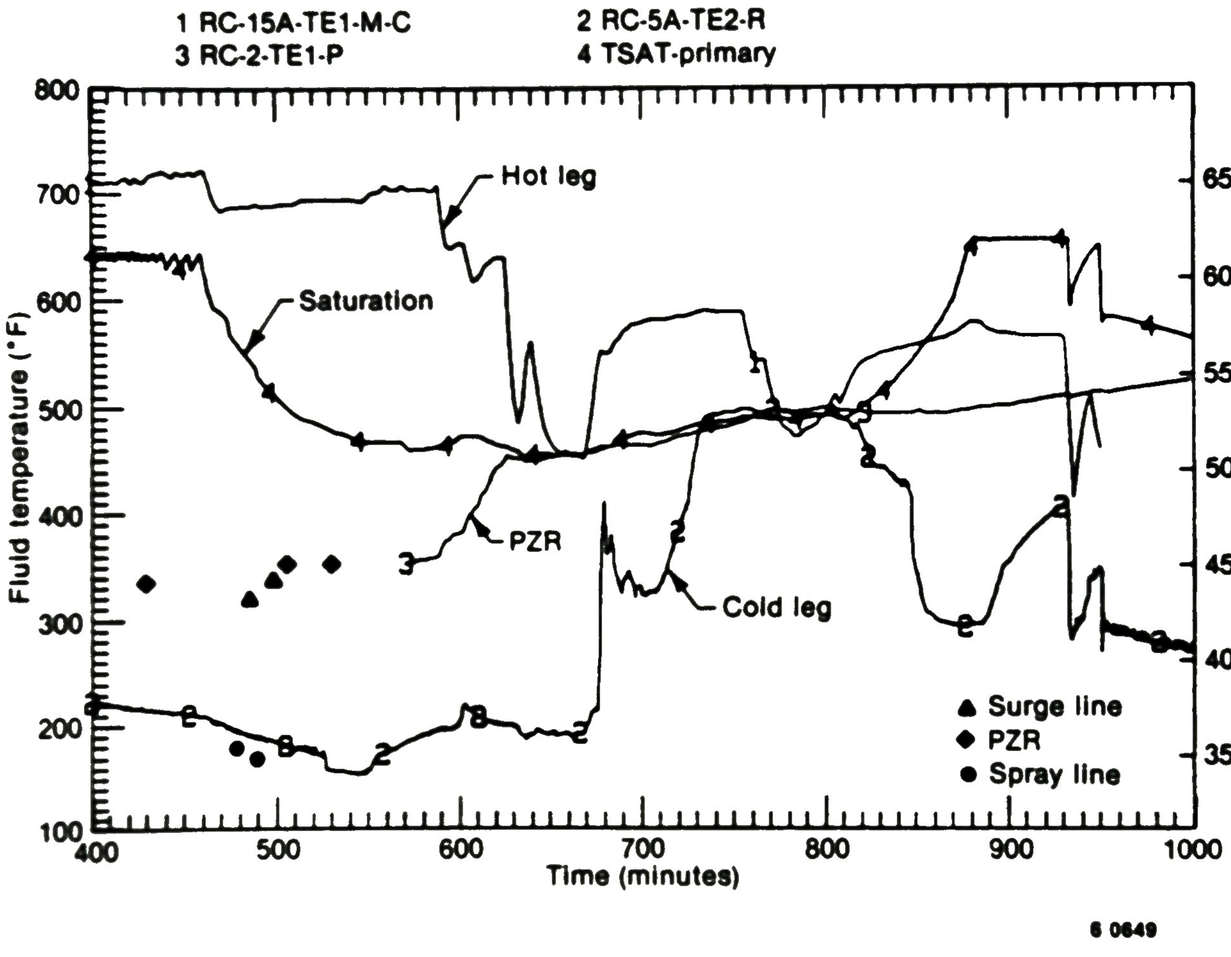


Figure 13. Comparison of system temperatures A-loop hot and cold and PZR with saturation.

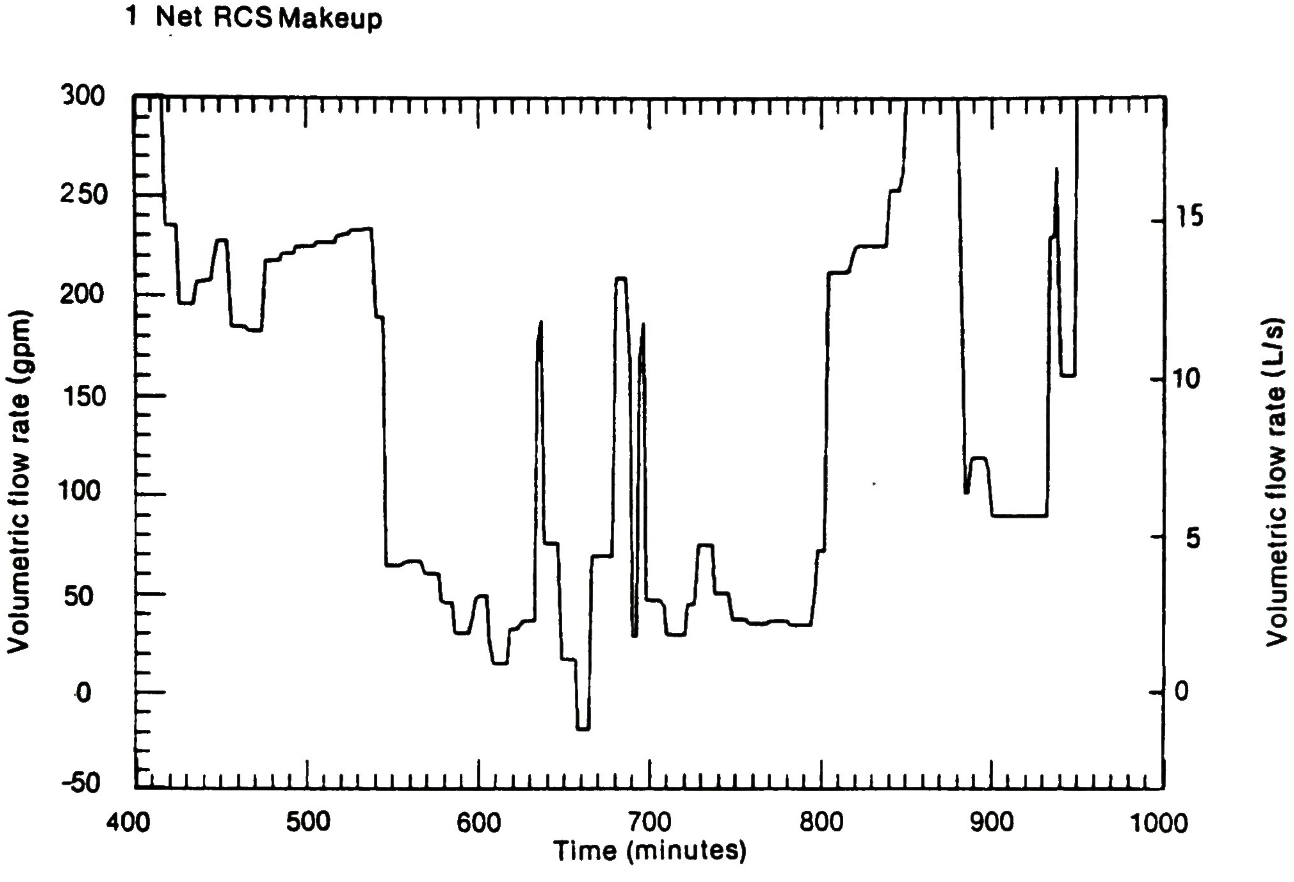


Figure 14. Estimated net liquid injection rate.

At 544 min, the net makeup rate was reduced from about 15 L/s (230 gpm) to about 4 L/s (60 gpm). At 547 min, the spray valve was closed. At 554 min, the PORV block valve was briefly closed, with no significant effect upon the system conditions. The RCS pressure at this time was 3 MPa (435 psig), and floating on the core flood tanks pressure.

Beginning at 570 min, the operators requested output of group trend data on the utility printer at the rate of once every 2 min. This output was continued throughout the remainder of the day and contains the best available data on pressurizer temperature. These temperature data are compared with the hot- and cold-leg temperatures and the system saturation temperature in Figure 13. Also at this time, the PORV block valve was closed, resulting in a slow increase in RCS pressure. No significant change in the pressurizer level occurred. Since the pressurizer was approximately 65 K (120°F) subcooled, the level remained at the upper limits of the measurement. This was probably due to a hydrostatic balance between the liquid in the pressurizer and the increasing system pressure. The pressurizer vent valve may have been open which would have contributed to maintaining the pressurizer at a lower pressure. Again, only a 50-70 kPa (7-10 psi) pressure difference is required to maintain the pressurizer liquid-full.

At 589 min, the hydrogen burn occurred in the containment building. This was a result of generation and discharge of hydrogen from the primary system into the containment building. There is no indication that the burn damaged any of the measurements used in this analysis, including the pressurizer level transmitters or cables. At 595 min, heater group 8 tripped due to a ground fault. The pressurizer level indicated full.

From the beginning of the group trend data at 570 min until 625 min, the pressurizer temperature showed a continuous increase at a rate of 1.5 K/min (2.7°F/min) until saturation temperature was reached at 625 min (Figure 13). The pressurizer remained at saturation until 815 min. At 601 min, the PORV block valve was again opened and the pressurizer level responded with a slow decrease of about 25 cm (10 in.). This was probably due to hydrogen being pulled into the pressurizer from the hot leg, and displacing liquid in the pressurizer coupled with liquid flashing into steam. At 604 min, the spray valve was again opened, with no indicated effect. However, the hot-leg temperature responded with a rapid 30 K (50°F) decrease, ultimately reaching RCS saturation temperature by 655 min. The pressurizer level responded by increasing off-scale high, and then decreasing slightly (13 cm or 5 in.) as the pressurizer liquid reached saturation and vapor generation in the pressurizer displaced liquid, forcing it back into the hot leg. At 631 min, the makeup pump MU-P-1C was turned on for about 14 min at a net injection rate of approximately 13 L/s (200 gpm). This stopped the system depressurization and probably refilled the hot leg to above the surge-line entrance. The pressurizer responded by increasing off-scale high, where it remained until 650 min, after which the level decreased to 953 cm (375 in.) in 12 min, and then cycled for another 12 min. The reason for this behavior may be continued heater operation in the saturated pressurizer, resulting in boiling and displacing liquid back into the hot leg.

## Phase 6 - Repressurization and Recovery

System repressurization and recovery comprises this phase of the accident, which begins with the closure of the PORV block valve at 672 min, and ends with the restart of one of the reactor coolant pumps at 950 min, reestablishing long-term forced convection cooling of the core. The pressurizer spray valve was open prior to this phase, and remained open until 726 min.

During this phase of the accident, the pressurizer experienced two major drain/refill cycles. Both drains were driven by vaporization in the saturated pressurizer by the energy input from the heaters. Both drains resulted in repressurization of the RCS. This in turn stopped the drain and resulted in refill of the pressurizer.

672 - 803 minutes

The first portion of this phase is from 672 to 803 min, during which there was minimal net makeup into the system (about 3 L/s or 50 gpm), with little or no primary to secondary heat transfer through either steam generator due to isolation of the secondaries. With the PORV block valve closed and little makeup flow, the core was being cooled by pool boiling, with the level gradually dropping in the downcomer. It is possible that uncovery of the upper region of the core occurred. At 672 min, when the PORV block valve was closed, the pressurizer responded by beginning a rapid level decrease of 569 cm (224 in.) in 13 min. This dump occurred because the pressurizer was at saturation temperature prior to the valve closure, with the pressurizer heaters supplying 756 kW of energy to the fluid. The flow through the PORV had been maintaining the pressurizer at a lower pressure then the rest of the RCS, thus holding the level up. With the block valve closed the fluid in the pressurizer continued to boil and the steam displaced the liquid (19.5 $m^3$ or 690 $ft^3$ of steam is calculated to have been generated by the heaters, compared to 20.4 $m^3$ or 720 $ft^3$ from the level change). The pressurizer level reached a minimum level of 445 cm (175 in.), which resulted in approximately 20 $m^3$ (720 $ft^3$) of liquid leaving the pressurizer and draining into the hot leg and probably into the core. Concurrent with this drain, the hot-leg temperature increased from saturation to 570 K (560°F), where it remained for the next 90 min. At 675 min the A-loop cold-leg temperature suddenly increased from 372 K to 483 K (210°F to 410°F) in a 2-minute period, and then gradually decreased to 439 K (330°F) in the next 30 min. It has been speculated that this sudden cold-leg temperature increase was due to establishing natural circulation flow in the A-loop.[8] If natural circulation had been established in the normal flow direction, then the hot-leg temperature would be expected to significantly decrease, which it did not do. Also, with the steam generators isolated, no driving force existed for natural circulation in the normal direction. The possibility exists that the fluid in the exposed portions of the core was colder than the steam in the hot leg, with the possibility that this could result in a natural circulation reverse flow. However, the major composition of the gas in the hot leg was almost certainly noncondensible gas, which would tend to block the establishment of a natural circulation flow. It is more likely that the drain of 720 $ft^3$ of liquid from the pressurizer resulted in reverse flow through the core and into the cold legs.

At 678 min, the makeup flow was increased for 10 min, and the pressurizer level decrease stopped and remained constant. This was concurrent with a slight repressurization. At 689 min, the operators de-energized heater groups 1 and 2. At 693 min, the pressurizer began to refill and reached a level of 1016 cm (400 in.) at 747 min, a refill rate of 11 cm/min (4.2 in./min). The pressurizer refill may have been due to the increasing RCS pressure coupled with steam condensation in the slightly subcooled pressurizer. The level in the hot leg would have had to have been above the surge line during this refill. The spray valve was still open during most of this refill (until 726 min), and condensation through this open path may have been the major mechanism for the fill. This argument is supported by the reaction of the cold-leg temperature. At 702 min, the A-loop cold-leg temperature began to increase and reached system saturation temperature at 732 min, where it remained until after 800 min.

At 754 min, the PORV block valve was opened for 9 min. Just prior to its closure, the pressurizer level briefly came back on-scale. This is a possible indication that steam and/or hydrogen was entering the surge line and displacing liquid in the pressurizer. At 772 min, the PORV block valve was again opened. The pressurizer level responded by decreasing 114 cm (45 in.) in less than 10 min, and then sharply increasing by 89 cm (35 in.) when the block valve was closed. The pressure decreased slightly upon opening the block valve and increased slightly upon the block valve closure, with both the pressurizer and cold-leg temperatures remaining at saturation. The hot-leg temperature responded to the opening of the block valve by decreasing to saturation for about 30 min. It is possible that natural circulation was established during this period. At 786 min, the condenser steaming mode was reestablished for SG-A. The SG-A secondary pressure also increased, which is another indication of natural circulation flow.

803 - 950 minutes

At 803 min, the operators started MU-P-1C (increasing net makeup to more than 13 L/s (200 gpm), and decreased heater power to the pressurizer. The RCS pressure dropped 200-350 kPa (30-50 psi) (probably due to condensation effects) and the pressurizer level responded by decreasing 280 cm (110 in.) in about 4 min. The pressurizer level drop was a result of liquid evaporation in the saturated pressurizer, where generated steam displaced liquid. The hot-leg temperature sharply dropped by 10 K (20°F), and the A-loop cold-leg temperature began a sustained decrease. The addition of 10 $m^3$ (350 $ft^3$) of near saturated liquid from the pressurizer into the core and continued makeup resulted in a continuous RCS pressure increase from 4 to 16 MPa (600 to 2300 psig) over the next 70 min. Although the pressurizer temperature remained fairly constant throughout the remainder of this phase, it was increasingly subcooled due to the increasing RCS pressure. The pressurizer level responded to the pressure increase by refilling at a linear rate of 10 cm/min (3.8 in./min) until going off-scale high at 860 min. This refill was probably condensation-induced with liquid available at the surge-line entrance. Once the pressurizer had refilled, the RCS repressurization rate increased. At this point, the pressurizer was probably liquid-full, unless a small bubble of noncondensible gases existed above the top level tap.

At 932 min the operators ran the RC-P-1A pump for 10 sec. This resulted in a brief flow of coolant in the A-loop, which caused a sharp drop in RCS pressure and the temperatures in both the hot and cold legs. Pressure in the secondary of the A-loop steam generator sharply increased, indicating that primary to secondary heat transfer increased due to the start of forced convection. Neither the pressurizer level or temperature responded to the pump operation. Because the pressurizer was liquid-full and very subcooled (about 50 K), there was no mechanism for a pressurizer drain at this point. At 950 min, the operator successfully restarted the RC-P-1A pump and reestablished forced convection in the system. This action established long-term cooling of the core and essentially recovered control of the plant, although the large quantities of noncondensible gases were not successfully eliminated from the upper head for another 3-5 days.

## THERMAL-HYDRAULIC EXPERIMENTAL RESULTS

There have been a number of integral systems experiments studying the RCS thermal-hydraulic behavior (particularly the pressurizer level response) during a TMI-type accident scenario. The major experimental facilities that have been used for this type of research, in which the experimental results are significant to the current analysis effort, are the Semiscale, Loss of Fluid Test (LOFT), and Rig of Safety Assessment (ROSA-IV) facilities. In the following section, results from the Semiscale experiments will be presented and discussed with regard to the pressurizer level response to the systems thermal-hydraulic phenomena. The Semiscale experiments are the most significant in terms of the current analysis effort because of the mockup of the TMI-2 surge-line configuration. The LOFT and ROSA-IV experiments demonstrated very similar behavior to the Semiscale experiments, even though exact TMI-accident scenarios were not performed in these facilities.

### Semiscale Experimental Results

A total of ten Semiscale simulations were performed with the objective of gaining a more fundamental understanding of the thermal-hydraulic phenomena which occurred in the TMI reactor.[12] These simulations used a scaled mock-up of the TMI surge line, including hydraulic resistances, elevations and point of connection to the loop hot leg in the TMI plant. Several unknown aspects relative to the actual TMI plant transient required a certain amount of educated speculation in order to complete the tests, such as the value of the actual HPIS flow rate as a function of time and the letdown/makeup flow histories. The primary result of the simulations, with respect to the pressurizer, is that core uncovery and core heatup occurred in the Semiscale simulations even though the pressurizer remained liquid-full. The pressurizer level response was noted to be generally similar in trend to the measured plant pressurizer level behavior. Although there were shifts in the timing, the Semiscale level basically showed filling trends as the transient progressed. It was clearly demonstrated that the pressurizer level was an inappropriate reflection of system mass inventory when the system was in a saturated two-phase state.

A comparison of the TMI and Semiscale pressurizer level variations is illustrated in Figure 15. Two parameters found important in the level variations were the HPIS injection rate and the average system temperature resulting from the auxiliary feedwater flow. During the Semiscale tests, the pumps remained running for the first 100 min, at which time the pumps were shut off. At 100 min, the Semiscale collapsed liquid level was near the top of the core.

"The pressurizer was nearly full during the entire period of core uncovery, even though mass was leaving through the PORV. Thus, an equivalent amount of mass was entering the surge line from the hot leg. The most likely source for the mass entering the surge line was steam produced in the core that eventually condensed in the pressurizer surge line or the pressurizer."[12]

Figure 16 compares the pressurizer level, core collapsed liquid level and core rod thermocouple response for the Semiscale simulation. When core power was terminated at 114 min (due to high core temperatures), the pressurizer drained.

An indication of core uncovery and core heatup at TMI is the observance of superheated fluid temperature in the hot leg. The Semiscale simulation indicated the presence of superheated fluid in the hot legs at about the same time as occurred for the TMI transient, as shown in Figure 17.

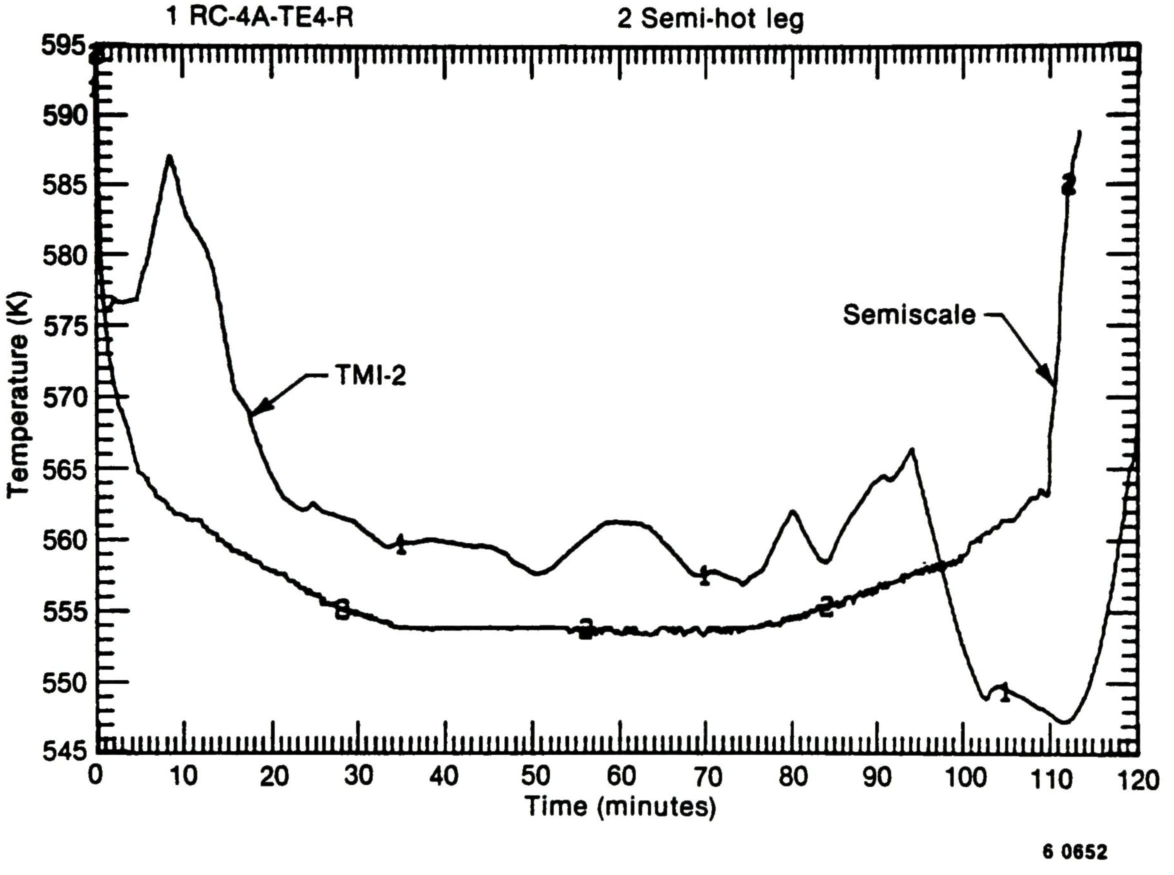


Figure 17. Comparison of semiscale and TMI hot leg fluid temperatures.

## RESULTS OF RELAP5 CALCULATIONS

In the following discussion, results for the pressurizer level response of a preliminary RELAP5 calculation for the TMI-2 accident will be compared to the measured pressurizer level response during the TMI-2 accident. These results are preliminary and may change as the RELAP5 analysis is refined and extended. The RELAP5 analysis has currently progressed beyond the point where all pumps were turned off, at 100 min.

The measured pressurizer level is compared to the response calculated by RELAP5 in Figure 18. The calculated response demonstrates the same initial in-surge and out-surge as was measured. The reason for these level changes was the RCS fluid expansion/contraction as energy removal and input changed due to dryout of the SG's and SCRAM of the reactor core. The RELAP5 calculated level showed the pressurizer full by 7 min, and then beginning to void 5 min later. The calculated level decreases to a minimum of about 825 cm (325 in.) by 15 min, and slowly increases.

Up until the A-loop pumps were turned off at 100 min, the calculated level in the pressurizer was very close to the measured level. The close comparison between the measured level and the RELAP5 analysis results gives confidence in both the measurement and the analysis. At 100 min, when the A-loop pumps were turned off, the calculated level dropped by approximately 200 cm (80 in.) as compared to a measured level decrease of about 25 cm (10 in.). The code may not be capable of correctly calculating the pressurizer response when the pumps are turned off. This is a result of counter-current flow-limiting (CCFL) phenomena being the mechanism for holding the level up, and the fact that RELAP5 uses an interfacial drag model to simulate the CCFL phenomena.

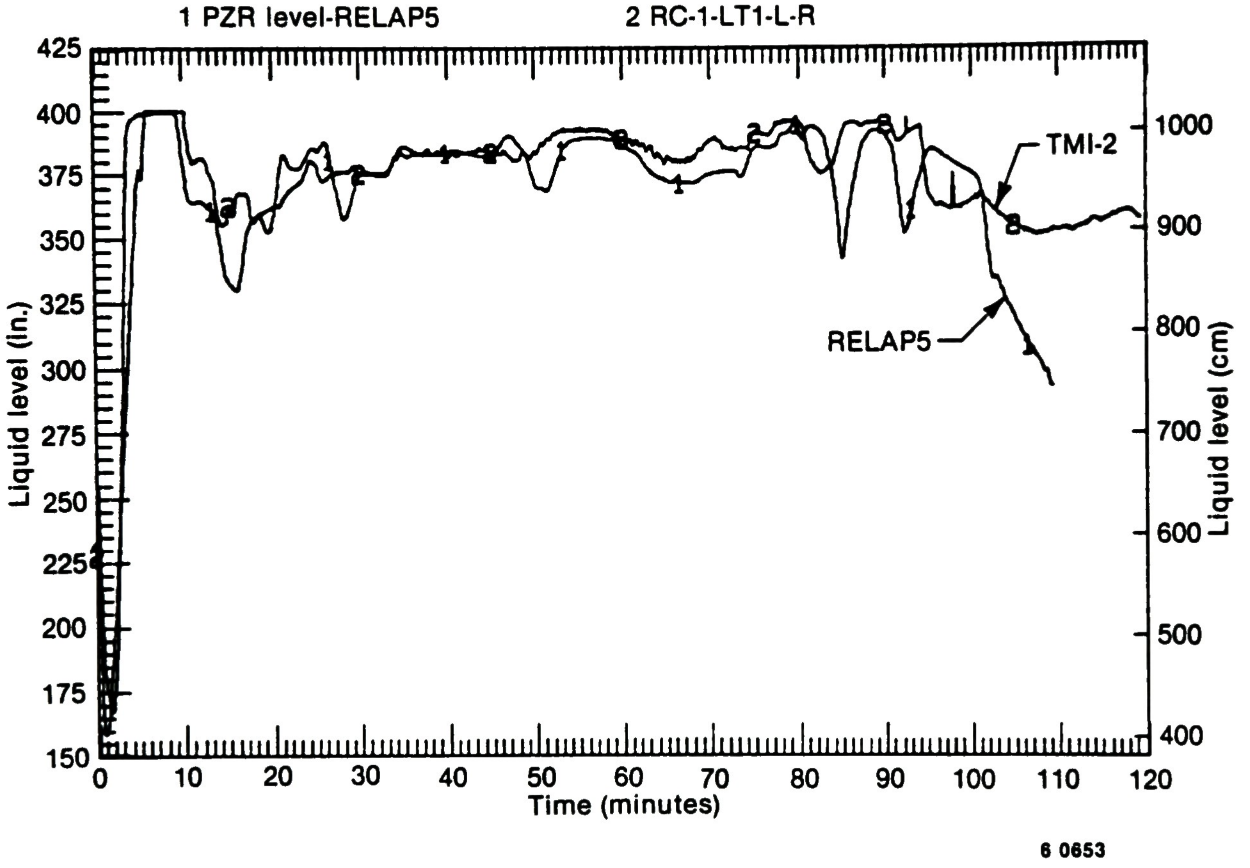


Figure 18. RELAP5 analysis results compared to TMI-2 reactimeter data.

## CONCLUSIONS

As a result of this study, the following conclusions have been reached.

- The pressurizer liquid level measurement indicated the correct level in the pressurizer, within an uncertainty of approximately ±43 cm (±17 in.).

- Most pressurizer liquid level changes have been explained in terms of response to the thermal-hydraulic conditions in the pressurizer and the remainder of the reactor coolant system. Unsatisfactory explanations for the minor level changes are due to insufficient understanding of conditions that existed in the reactor system.

- No supporting evidence of damage to the pressurizer liquid level measurement system has been discovered. Neither measured data from the accident, nor thermodynamic considerations and calculations of water hammer pressure increases support the argument for damage to the level measurement.

- During periods when the pressurizer heaters were undergoing ground fault trips, available evidence indicates that the pressurizer was full of very subcooled liquid [as much as 150 K (275°F) subcooled]. Further investigation of the heater trips is required to resolve the mechanism causing the ground fault. This investigation should include removal and physical examination of the heaters.

- All analyses performed in support of this study have been simplified hand calculations utilizing basic engineering knowledge. More complete understanding of the pressurizer response may be gained as the more detailed RELAP5 analysis progresses, although RELAP5 may be incapable of correctly calculating the CCFL phenomena.

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# APPENDIX A

# PRESSURIZER LIQUID LEVEL CALCULATED FROM MEASURED DIFFERENTIAL PRESSURE

## APPENDIX A

## PRESSURIZER LIQUID LEVEL CALCULATED FROM MEASURED DIFFERENTIAL PRESSURE

The basic pressurizer liquid level measurement configuration is shown in Figure A-1. The differential pressure across the transmitter is the difference between the hydrostatic heads of the fluid columns in the reference leg and the pressurizer. This differential pressure, accounting for the hydrostatic head of the steam column, can be writen as,

$$\Delta P = [\rho_r D - \rho_f L - \rho_g (D-L)]g \qquad \text{(A-1)}$$

where

$\Delta P$ = the measured differential pressure

$D$ = the vertical distance between taps (= 400 in.)

$L$ = the level of the stratified liquid interface (inches)

$\rho_r$ = the liquid density in the reference leg

$\rho_f$ = the liquid density in the pressurizer

$\rho_g$ = the steam density in the pressurizer

$g$ = the gravitational acceleration.

The liquid and steam densities are obtained assuming saturation conditions in the pressurizer and using the average of two of the RTD temperature measurements in the pressurizer in conjunction with the steam tables. The reference leg is assumed to be at a temperature of 125°F whenever the reactor system is above a temperature of 125°F. Therefore a

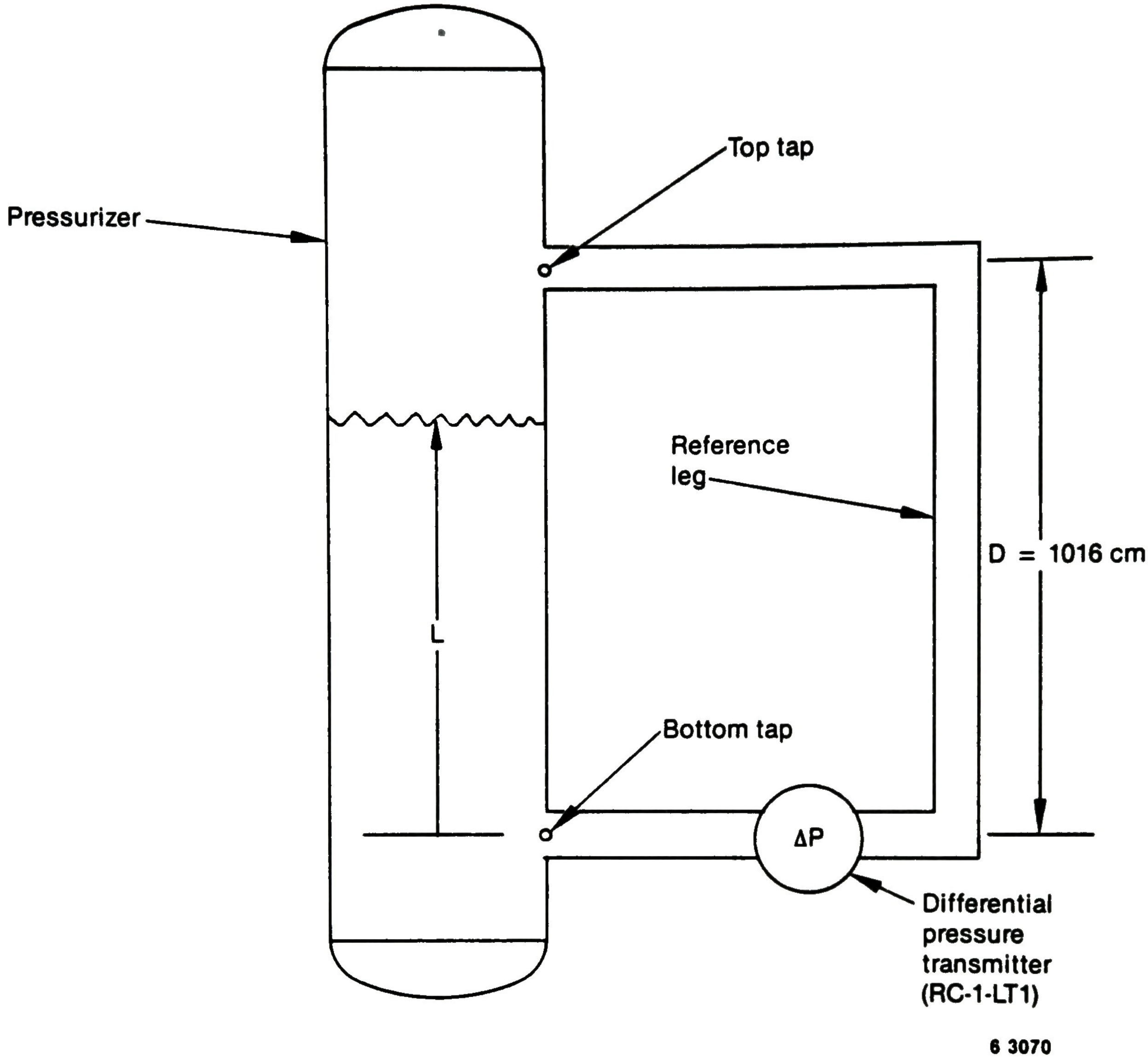


Figure A-1. TMI-2 Pressurizer Liquid Level Measurement Schematic.

constant reference leg density of 61.7 $lbm/ft^3$ is used in Equation (A-1). The differential pressure transmitter is calibrated in terms of inches of cold water (@ 68°F), therefore the transmitter output, DP, is related to the $\Delta P$ in Equation (A-1) by,

$$DP = [\rho_r D - \rho_f L - \rho_g (D-L)]/\rho_c \qquad \text{(A-2)}$$

where

$\rho_c$ = the density of cold water at 68°F

= 62.3 $lbm/ft^3$

Equation (A-2) can be solved for the stratified liquid level, L, in the pressurizer resulting in,

$$L = \frac{(\rho_r - \rho_g)D - \rho_c\, DP}{(\rho_f - \rho_g)} \qquad \text{(A-3)}$$

The output from one of the three independent differential pressure transmitters on the pressurizer, is used as input to the analog circuit which calculates the liquid level. This signal is combined with the temperature from one of the RTD's to calculate the temperature compensated liquid level. The output from the analog circuit is used for input to the Integrated Control System, which controls the liquid level in the pressurizer using the makeup and letdown systems. This output also goes to an operator's control panel for indication of the liquid level (this panel includes a strip chart recorder). For Unit-2, the analog output also was used as input to the reactimeter channel 7. The output from each of the three transmitters is also input to the plant computer, where the liquid level is calculated using Equation (A-3), and one of the transmitters, usually RC-1-LT1. Results from this calculation are available to the operators on request, and are used for the alarm setpoints, which are output on the alarm printer.

From 570-1000 min following the feedwater pump trip, the operators requested output of group trend data on the utility printer at a frequency of one sample every 2 min. Included in these data was the pressurizer temperature from one of two RTDs, and the differential pressure measured using transmitter RC-1-LT1. A comparison of the measured DP (in inches of water) and the pressurizer liquid level, recorded on the reactimeter, is shown in Figure A-2. The DP transmitter is zereod when valved out of the system and vented to atmosphere, therefore measures the hydrostatic head of the reference leg when the pressurizer is empty. As a result, the DP respondes in the inverse of the liquid level, as shown in Figure A-2.

Using the data recorded on the utility printer, and Equation (A-3), the liquid level in the pressurizer can be calculated. The results of this calculation are compared to the liquid level recorded on the reactimeter in Figure A-3. In most cases the comparison is quite good. An exception is the step in the reactimeter data at 13.7 h, which is not shown in the utility printer data. The reason for this descrepancy is unknown.

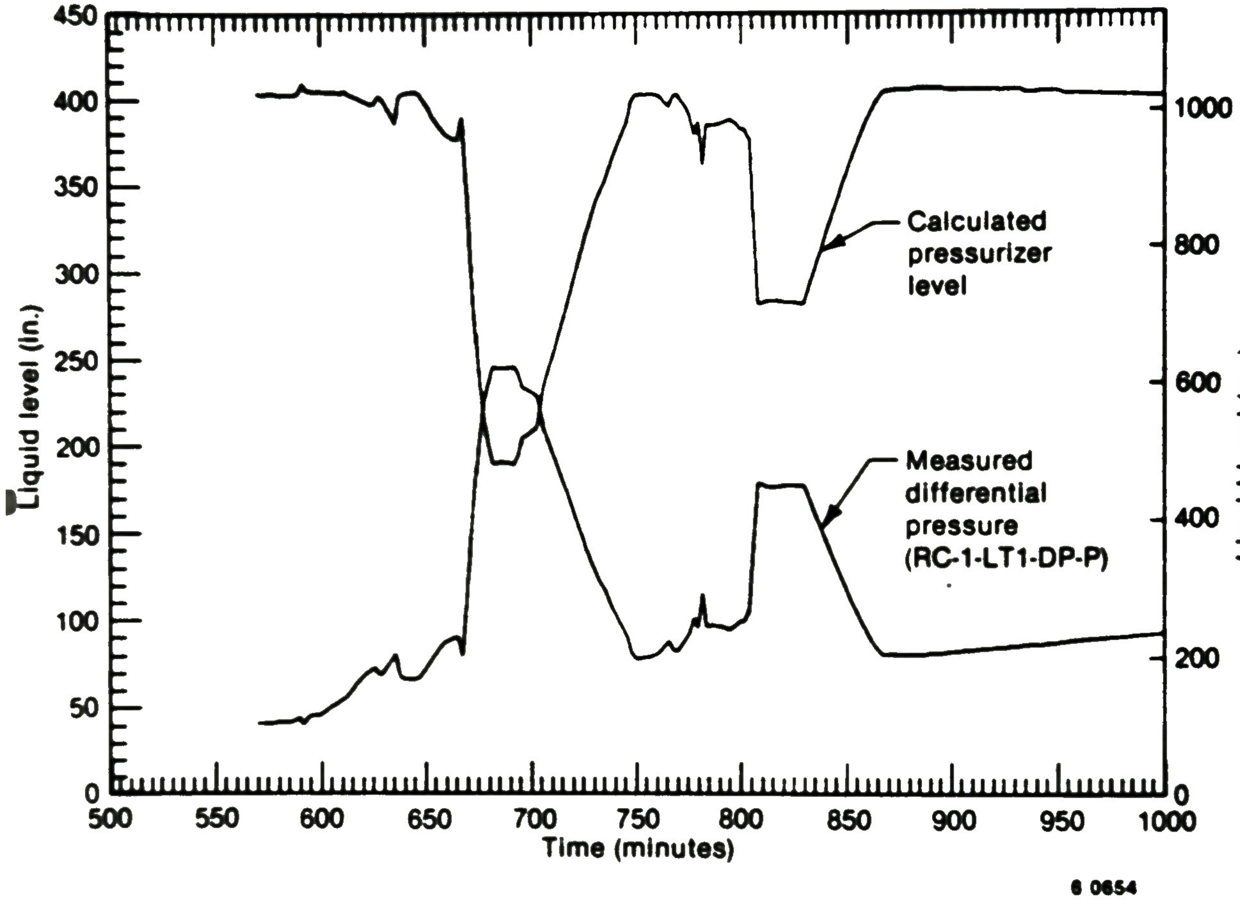


Figure A-2. Comparison of the measured differential pressure RC-1-LT1-DP-P, recorded on utility printer, and the pressurizer liquid level calculated using RC-1-LT1-DP-P.

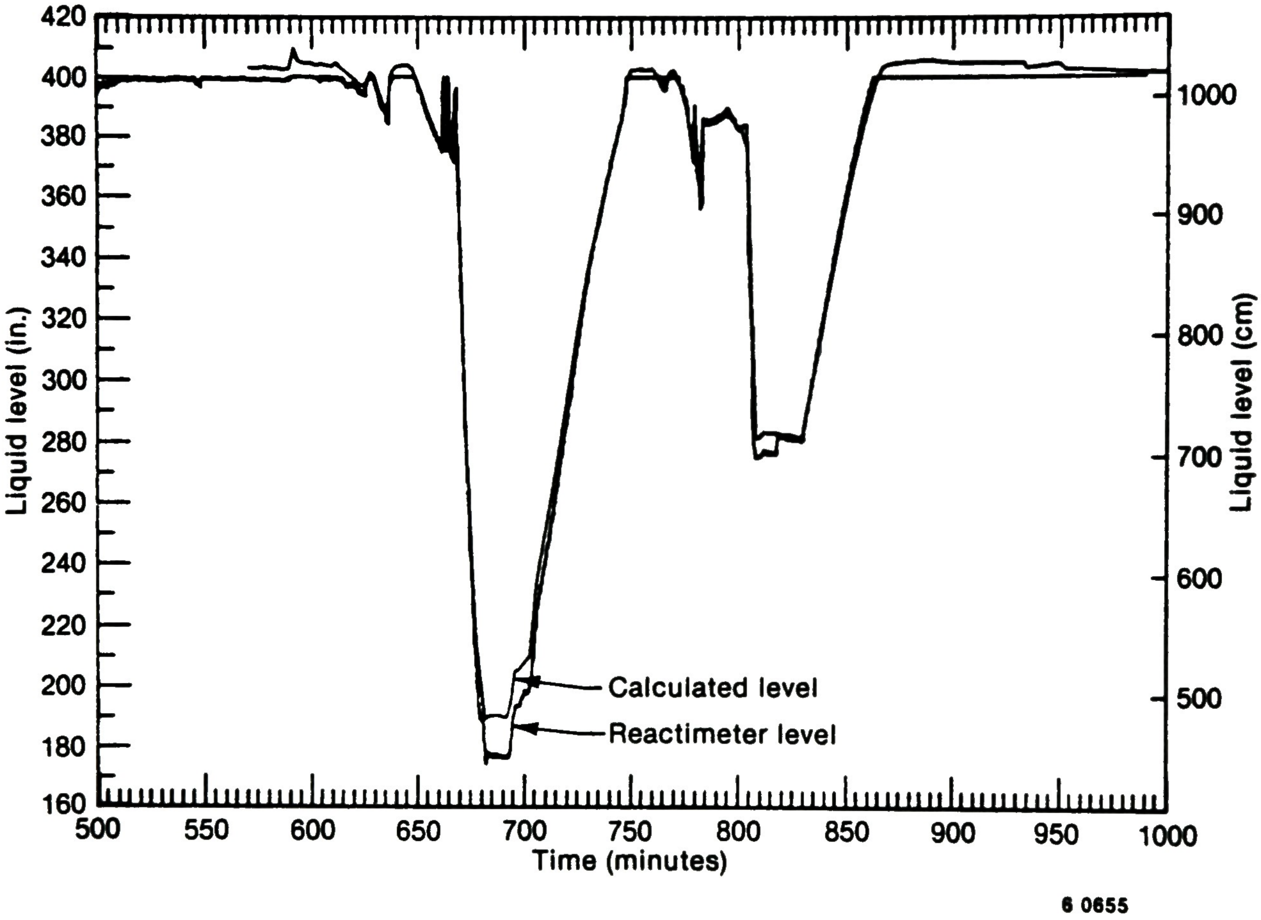


Figure A-3. Comparison of pressurizer liquid levels. One recorded on the reactimeter and the others calculated using RC-1-LT1-DP-P.

# APPENDIX B

# SIMPLIFIED CALCULATIONS IN SUPPORT OF PRESSURIZER STUDY

## APPENDIX B
## SIMPLIFIED CALCULATIONS IN SUPPORT OF PRESSURIZER STUDY

### PORV CRITICAL STEAM FLOW RATES

The critical mass flow rate of steam out the open PORV is a function of the pressurizer pressure and steam enthalpy. Assuming saturation conditions in the pressurizer, the mass flow rate is solely a function of the pressure upstream of the PORV. The PORV orifice diameter is 2.94 cm (1-5/32 in.)[B-1] for a flow area of $6.774 \times 10^{-4}$ $m^2$ (0.00792 $ft^2$). A simillar valve has been flow tested[B-2] with a flow of 17.3 kg/s (38.1 lbm/s) for pure steam at 16.24 MPa (2355 psia). The critical flow rate can also be obtained from the ASME steam tables[B-3] Figure 14, with the mass flow rate for enthalpies at 1000 and 2000 psia, given by;

$$\dot{m}\ (\text{lbm/s}) = 0.0147 \times P\ (\text{psia}) \qquad \text{at 1000 psia} \tag{B-1}$$

$$\dot{m}\ (\text{lbm/s}) = 0.0155 \times P\ (\text{psia}) \qquad \text{at 2000 psia} \tag{B-2}$$

Equation B-2 results in a steam mass flow rate of 36.5 lbm/s at a pressure of 2355 psia. The constant in Equation (B-2) can be adjusted to give the flow rate obtained from the flow tests resulting in,

$$\dot{m}\ (\text{lbm/s}) = 0.0162 \times P\ (\text{psia}) \tag{B-3}$$

This equation will be used to obtain the PORV steam flow rates used in the simplified calculations in this appendix.

### PRESSURIZER SURGE LINE CCFL

During conditions of steam flow into the surge line, through the pressurizer, and out the PORV, liquid flow out of the pressurizer and into the hot leg can be limited by counter-current flow-limiting (CCFL) phenomena. The amount of liquid flow out of the pressurizer can be estimated using a correlation given in Wallis[B-4] for CCFL in vertical tubes with upflowing gas. This correlation is,

$$m\, j_f^{*1/2} + j_g^{*1/2} = C \qquad \text{(B-4)}$$

where

| | | |
|---|---|---|
| $m$ | = | a constant, a value of 1 is used in this analysis |
| $C$ | = | a constant, = 0.725-1.0 |
| $j_g^*, j_f^*$ | = | the gas and liquid nondimensional velocities, given by |

$$j_g^* = \frac{\dot{m}_g}{\pi/4\,[g\,D^5\,\rho_g(\rho_f - \rho_g)]^{1/2}} \qquad \text{(B-5)}$$

$$j_f^* = \frac{\dot{m}_f}{\pi/4\,[g\,D^5\,\rho_f(\rho_f - \rho_g)]^{1/2}} \qquad \text{(B-5)}$$

Results from the Wallis correlation for complete flooding ($j_f^*=0$) are that no liquid outflow from the pressurizer could occur at RCS pressures above 400-800 psia. Since the minimum RCS pressure reached during the first 8 h of the accident was 675 psig (the pressure was only below 800 psig for 30 min), it can therefore be concluded that draining of liquid out of the pressurizer would not occur during most periods in which the PORV block valve was open and the liquid level in the hot leg was below the surge line entrance to the hot leg. Flooding would occur at the surge line entrance into the bottom of the pressurizer.

## MINIMUM PRESSURIZER LEVEL FOR TWO-PHASE FLOW OUT PORV

For conditions where there was steam flow into the surge line and out of the PORV, the minimum measured pressurizer level at which the two-phase interface level reached the PORV can be calculated. The pressurizer average void fraction can be calculated using the drift flux model[B-4] as,

$$\bar{\alpha} = \frac{u_g}{2 + C_0 u_g} \tag{B-7}$$

where

$u_g$ = the superficial steam velocity through the pressurizer

$C_0$ = a constant, usually taken as equal to 1.2

The pressurizer average void fraction required for the interface level to reach the PORV can also be calculated from the measured, or collapsed, liquid level as,

$$\bar{\alpha} = \frac{(h_s - h_0)}{h_s} \tag{B-8}$$

where

$h_0$ = the measured level

$h_s$ = the swelled interface level (=455 in. at the PORV)

Equating Equation (B-7) and Equation (B-8) results in the minimum measured level at which the two-phase interface level would reach the PORV and two-phase flow out the PORV would result.

$$h_0 = h_s \left(1 - \frac{u_g}{2 + C_0 u_g}\right) \tag{B-9}$$

Evaluating Equation (B-9) results in a minimum level of 416 in. at a RCS pressure of 1000 psia (the minimum level increases slightly with increasing pressure to a value of 426 in. at 2500 psia). The result of this calculation is that during periods in which the measured level was on scale (less than 400 in.) flow out the PORV would have been all or near all steam. (The effects of droplet entrainment are not included in this analysis).

## WATER HAMMER EFFECTS

It has been speculated that the sense lines leading from the pressurizer to the differential pressure transmitters were damaged due to water hammer at 174 min when the 2B pump was restarted and there was a large level increase in the pressurizer indicating a large in-surge. Water hammer is an effect of the rapid acceleration of liquid. Typically the phenomenon occurs at the opening or closing of a valve, resulting in a large pressure spike. This pressure increase (above the static pressure) can be calculated by,[B-5]

$$\Delta P = - \rho \; c \; \Delta V \tag{B-10}$$

where

$\rho$ = the liquid density

$c$ = the velocity of sound (=4720 ft/s)

$\Delta V$ = the change in liquid velocity

During the repressurization event at 174 min, the pressure increased from an initial pressure of 1300 psig to 2100 psig over a 2 min period. Since the sense lines had been hydrostatically tested during plant startup to a pressure of 6000 psi, it seems reasonable that for damage to occur the pressure must have been greater than this value. Using a pressure increase of 3900 psi (= 6000-2100 psig) in Equation (B-10) results in a required velocity change of 80 ft/s. (Note that this velocity change would need to occur parallel to the sense line entrance to the pressurizer, which is 90° from the direction of the liquid velocity resulting from an in-surge into the pressurizer. Therefore, there is really no physical mechanism for water hammer damage to occur.) Using the measured level to calculate the in-surge liquid velocity (a 57.8 in. increase over 120 s) results in a velocity of 0.04 ft/s, which is a factor of 2000 too small to increase the pressure up to the hydrostatic test pressure. Therefore, the likelihood of water hammer damage occuring to the sense lines is negligible.

## PRESSURIZER IN-SURGE AT 174 MINUTES

At 174 min, the 2B pump was restarted. This resulted in a large pressure increase and a large increase in the measured pressurizer liquid level. The following analysis will show that the level increase could have occured solely due to condensation effects in the pressurizer, as the system pressure increased due to steam generation in the core. It will be assumed that the pressurizer was initially at saturation; that the liquid level in the hot leg was always below the surge line entrance to the hot leg; and that the gas flow into the pressurizer was all steam. The liquid mass in the pressurizer at a given time after the start of the 2B pump, M(t), can be given by,

$$M(t) = M_0 + \dot{m}_g\, t = V(t)\rho_f \qquad \text{(B-11)}$$

where

$M_0$ = the initial liquid mass (lbm)

$\dot{m}_g$ = the steam mass flow rate into the pressurizer (lbm/s)

$V(t)$ = the liquid volume in the pressurizer as a function of time

= $V_0 + (3.18\ ft^3/in) * L(t)$

$V_0$ = the pressurizer volume below the bottom tap (+ 112 $ft^3$)

$L(t)$ = the measured liquid level as a function of time (in.)

$t$ = time after start of 2B pump (seconds)

The enthalpy of the liquid in the pressurizer as a function of time, $H(t)$, is given by,

$$H(t) = h_{f0}M_0 + h_g\dot{m}_g t = V(t)\ \rho_f h_f \tag{B-12}$$

where

$h_{f0}$ = the initial liquid enthalpy (btu/lbm)

$h_g$ = the enthalpy of the inlet steam (btu/lbm)

$h_f$ = the liquid enthalpy at time t (btu/lbm)

Equations (B-11 and B-12) can each be solved for $\dot{m}_g t$ and equated giving,

$$V(t)\rho_f - M_0 = [V(t)\ \rho_f h_f - M_0 h_{f0}]/h_g \tag{B-13}$$

Over the temperature and pressure range of the repressurization event, the liquid density, enthalpy, and level can be written in linearized forms as,

$$L(t) = L_0 + a\ t \tag{B-14a}$$

$$\rho_f = b + c\ T \qquad \text{(B-14b)}$$

$$h_f = d + e\ T \qquad \text{(B-14c)}$$

where

a,b,c,d,e = constants given in Table B-1

T = the pressurizer liquid temperature at time t (°F)

Subsituting the linearized forms of Equation (B-14) into Equation(B-13) and rearanging results in,

$$A\ T^2 + B\ T + C = 0 \qquad \text{(B-15)}$$

where

$A = c\ e$

$B = c\ d + b\ e - c\ h_g$

$C = [M_0(h_g - h_{f0})]/V(t) + d\ b - b\ h_g$

The one valid solution of Equation (B-15) is given by,

$$T = [-B + (B^2 - 4AC)^{1/2}]/2B \qquad \text{(B-16)}$$

The mass flow rate at a time t can then be obtained from,

$$\dot{m}_g(t) = [M(t) - M_0]/t \qquad \text{(B-17)}$$

TABLE B-1. VALUES USED IN THE ANALYSIS AT 174 MINUTES

| | |
|---|---|
| $a$ = 0.482 in./s | |
| $b$ = 94.3 lbm/ft$^3$ | $c$ = -0.087 lbm/ft$^3$-°F |
| $d$ = -269.6 btu/lbm | $e$ = 1.481 btu/lbm-°F |
| $A$ = -0.1421 | $B$ = 298.7 |
| **Initial Conditions** | **Final Conditions** |
| $L_0$ = 298.6 in. | $L$ = 356.4 in. |
| $V_0$ = 1061.5 ft$^3$ | |
| $M_0$ = 46,810 lbm | |
| $P$ = 1300 psia | $P$ = 2100 psia |
| $T = T_{sat}$ = 577.4°F | $T_{sat}$ = 635°F |

RESULTS

| | | |
|---|---|---|
| $h_g$ = 1150 btu/lbm | T(120 s) = 610.8 °F | M(120) = 51,300 lbm |
| | | $\dot{m}_g$ = 37.4 lbm/s |
| $h_g$ = 1335 btu/lbm | T(120 s) = 616.3 °F | M(120) = 50,704 lbm |
| | | $\dot{m}_g$ = 32.5 lbm/s |

The calculational procedure is to assume an enthalpy of the inlet steam and solve Equation (B-16) for the temperature, T, at time, t, and then use this temperature to obtain the liquid density from Equation (B-14b). This allows the liquid mass, M(t), and required steam mass flow rate, $\dot{m}_g$, to be calculated. If this steam mass flow rate is a reasonable value, then it can be concluded that the pressurizer level increase could have been a direct result of condensation effects without the requirement that a liquid source was available at the surge line entrance to the hot leg. This procedure was performed and the values used, along with the results, are presented in Table B-1 for two different values of the inlet steam enthalpy. One value for saturated steam at 2100 psia, and a second for superheated steam at 2000 psia and 800°F. The results for both cases are very similar. Basically a 32-37 lbm/s steam flow rate into the pressurizer is required to result in the measured level increase due to steam condensation in the subcooled pressurizer as the system pressure increases. This is a reasonable steam flow rate into the pressurizer through the 8.75-in. diameter surge line. The resulting condition at the end of the 2 min period is an approximately 20-25 °F subcooled pressurizer.

## PRESSURIZER OUT-SURGE AT 200 MINUTES

At 200 min, the operators manually initiated HPI. This resulted in a rapid depressurization of the system due to steam condensing on the cold HPI water. Simulataneous with the depressurization, the level in the pressurizer dropped by 145.9 in. in 7 min. The following analysis shows that this level drop was a result of flashing of liquid into steam in the pressurizer once the RCS pressure reached the saturation pressure in the pressurizer. The steam generated by vaporization of the liquid displaced the liquid in the pressurizer, thus resulting in a falling liquid level.

A mass and energy balance in the pressurizer (assuming an isenthalpic state change) for the initial and final states results in the final steam mass in the pressurizer as given by,

$$M_{gF} = \frac{M_{fI}h_{fI} + M_{gI}h_{gI} - M_T h_{fF}}{(h_{gF} - h_{fF})} \quad . \tag{B-18}$$

where

$M_{gI}, M_{fI}$ = the initial steam and liquid masses

$M_{gF}, M_{fF}$ = the final steam and liquid masses

$M_T$ = the total initial mass in the pressurizer

$h_{gI}, h_{fI}$ = the initial steam and liquid enthalpies

$h_{gF}, h_{fF}$ = the final steam and liquid enthalpies

If it is assumed that the pressurizer was initially at saturation, and that all of the liquid initially in the pressurizer was available for generation of steam as the depressurization proceeded, then the maximum steam generation will be calculated. If it is assumed that only the liquid remaining in the pressurizer at the end of the depressurization was available for steam generation, then the minimum steam generation will be calculated. The actual steam generation should be between these values. The values of parameters used and the results are tabulated in Table B-2. Summarizing the results, the maximum steam generation accounts for a 1028 $ft^3$ gas volume change, or a level decrease of 323 in. (compared to the measured level decrease of 146 in.). The minimum steam generation accounts for a 269 $ft^3$ gas volume change, or a level decrease of 85 in. Thus, the measured level change does indeed lie between the minimum and maximum values calculated.

## PRESSURIZER LEVEL INCREASE AT 210 MINUTES

At 210.2 min the measured level in pressurizer began a linear increase of 22.1 in./min over the next 8 min, finally increasing off-scale high (>400 in.). This increase was concurrent with an 80 psi increase in

TABLE B-2. CALCULATIONS FOR PRESSURIZER EVENT AT 200 MINUTES

| Parameter | Initial Conditions | Final Conditions | Calculated[a] | Calculated[b] |
|---|---|---|---|---|
| Time | 200.25 min | 207.25 | | |
| P | 1870 psia | 1547 | | |
| $T_{sat}$ | 626°F | 599 | | |
| $\rho_f$ | 39.9 lbm/ft$^3$ | 42.3 | | |
| $\rho_g$ | 4.82 lbm/ft$^3$ | 3.74 | | |
| $h_f$ | 656.8 btu/lbm | 617.9 | | |
| $h_g$ | 1147.7 btu/lbm | 1167.5 | | |
| $M_f$ | 51,340 lbm | 34,800 | | |
| $M_g$ | 1,123 lbm | 2,607 | 4,716 | 1,877 |
| $M_T$ | 52,463 lbm | 37,407 | | |
| $V_g$ | 233 ft$^3$ | 697 | 1,261 | 502 |
| $\Delta V_g$ | | 464 | 1,028 | 269 |
| L | 369.4 in. | 223.5 | 46.5 | 284.8 |

a. Assuming that all of the initial liquid in the pressurizer was available for vaporization.

b. Assuming that only the liquid measured in the pressurizer at the end of the depressurization was available for vaporization.

system pressure. A level increase of 22.1 in./min correspondes to a liquid flow rate into the pressurizer of 525 gpm. One explaination for this level increase is that the liquid level in the hot leg rose up to the surge-line entrance, due to HPI. Continued injection resulted in a slight system repressurization, which refilled the pressurizer. The calculated flow rate of 525 gpm would correspond to an HPI injection rate of 360 gpm of cold water (assuming a liquid density of 42.6 $lbm/ft^3$ in the pressurizer and a HPI liquid density of 62.0 $lbm/ft^3$).

## REPRESSURIZATION OF NON-CONDENSIBLE GASES AT 318 MINUTES

At 318 min, the PORV block valve was closed in order to repressurize the RCS, in an attempt to compress and eliminate the noncondensible gas which filled the RCS high points. With the block valve closed, and makeup continuing at a rate of approximately 250-300 gpm[B-6], The pressure increased from 1275 to 2050 psia over the next 30 min. The following analysis looks at the compression of the noncondensible gas in an attempt to obtain a verification of the injection rate. Since the hot-leg RTD temperature never decreased from its superheated output during this time frame, we know that the final liquid level in the hot leg was below 353 ft. For this analysis, it is assumed that the initial level was just above the surge-line entrance elevation at 321 ft 6 in., and that the noncondensible gas follows the ideal gas law for a reversible adiabatic compression. Thus,

$$P_1V_1^k = P_2V_2^k \tag{B-19}$$

where

P = the absolute pressure

V = the gas volume

k = the ratio of specific heats (= 1.41 for hydrogen).

Equation (B-19) can be solved for the ratio of final to initial gas volumes giving, using the pressures given above, $V_2/V_1 = 0.71$. The total primary side volume in the hot legs and SGs above the surge line entrance elevation (and including the vessel upper head) is about 3067 $ft^3$. Thus, the volume change is $\Delta V = (1-0.71) * 3067\ ft^3 = 890\ ft^3$. This volume change corresponds to a injection rate of 220 gpm over the 30 min period, and to an increase in hot-leg level of about 12 ft (assuming the above compression of the upper head and using the hot-leg and SG-tube areas).

## ENERGY FLOW OUT PORV

At 6 min, the pressurizer was probably liquid full, with liquid flow out the PORV. Assuming that the temperature of the liquid leaving the PORV was at the hot-leg temperature, and that liquid leaving the system was being replaced by the cold makeup liquid at 100°F, the energy being removed from the RCS through the PORV, ΔE, can be calculated as,

$$\Delta E = \dot{m}_f \Delta h \quad \text{(B-20)}$$

where

$\Delta h$ = the change in liquid enthalpy

$\dot{m}_f$ = the liquid mass flow out the PORV [$= C\ (\rho \Delta P)^{1/2}$]

$C$ = constant for the PORV (= $8.32 \times 10^{-4}$) (= the liquid density).

Using the follwing values, the energy removal from the RCS through the PORV can be calculated from Equation (B-20) as 83 MW.

$h_f$ = 71.7 btu/lbm @ 100°F

$h_f$ = 598.8 btu/lbm @ 582°F

$\rho_f$ = 43.3 $lbm/ft^3$ @ 582°F

$\dot{m}_f$ = 150 lbm/s

$\Delta E$ = 79,100 btu/s (= 83 MW).

At 138 min (just prior to the initial closure of the PORV block valve), the flow out the PORV can be assumed to have been saturated steam. The energy removal from the system can be calculated as the product of the steam mass flow rate and the heat of vaporization at the pressure of 725 psia. For this pressure, using Equation (B-3), the steam mass flow rate was 11.7 lbm/s. From the steam tables at 725 psia, the heat of vaporization is 705 btu/lbm. Thus, the energy removal out the PORV was 8250 btu/s or 8.7 MW.

During the time period of 348-458 min, the operators were cycling the PORV block valve, with a makeup injection rate of about 250-300 gpm.[B-6] At 433 min, the pressurizer temperature was recorded on the utility printer as 343°F. Assuming an injection temperature of 100°F, an exit temperature at the recorded pressurizer temperature, and a mass flow rate out the PORV equal to the injection mass flow rate (250 gpm is 35 lbm/s), the energy removal can be obtained from Eq(B-19) as 9 MW (8500 Btu/s). This compares to the calculated decay heat of 21 MW.

## REFERENCES


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# APPENDIX C

# UNCERTAINTY ANALYSIS OF PRESSURIZER LEVEL MEASUREMENT

## APPENDIX C

## UNCERTAINTY ANALYSIS OF PRESSURIZER LEVEL MEASUREMENT

The liquid level measurement system for the pressurizer has been described in Appendix A, where the equation for obtaining the liquid level from the measured differential pressure was derived, Equation (A-3). The usefulness of data is a direct function of how accurate the data are and how well that accuracy (or inversely the uncertainty) is known. The uncertainty in the calculated level is a function of a number of possible error sources. In this appendix, potential error sources will be evaluated and combined to obtain the total estimate of the uncertainty in the recorded level. The method used for combining individual uncertainties for the calculated liquid level is the root-sum square (RSS) method. All quoted uncertainties are at the 95% confidence level. The document which forms the basis for uncertainty analysis in the TMI-2 Accident Evaluation Program is Reference C-1.

Possible error sources in need of evaluation include:

- The measurement mechanism (The potential error source identified in this category is a partially voided reference leg.)

- The differential pressure measurement introduced by the Bailey differential pressure transmitter (Possible error sources include basic transmitter accuracy, amplifier adjustment, pressure sensitivity, and environmental effects, predominately temperature).

- The level calculation circuitry from the electronic setup, and the assumption of saturation conditions in the pressurizer

- The recording system, in this case the reactimeter.

Sources of information for evaluation of uncertainty components are the Bailey transmitter instruction manual (Reference C-2), the Bailey elevated environment qualification report (Reference C-3), and the Pressurizer Temperature and Level Channel Calibration procedure (Reference C-4).

A block diagram of the liquid level measurement system is shown in Figure C-1. The output from each of the three independent differential pressure transmitters (output range of -10 to + 10 volts) is split, with one output going to the plant computer and the other going to a manual switch, which is located on the operators control panel. Output from this switch is routed to the pressurizer liquid level calculation circuit. The differential pressure transmitter output which goes to the circuit is not recorded. The output from each of the elements of the dual element RTD, located in the pressurizer at an elevation of 322 ft, is input to a manual switch, output of which is split. One output goes to the plant computer, and the other output goes to the level calculation circuit. Output from the manual switch is not recorded as to which element is the source. Output from the liquid level calculation analog circuit is split and routed to a number of locations. These include the reactimeter, the operators control indication, strip charts, the integrated control system, and the plant computer (this latter is not certain).

Each of the aforementioned potential error sources are listed in Table C-1 and estimates of the resulting uncertainty given. Since no statistically valid test data exist, all estimates are given as bias components.

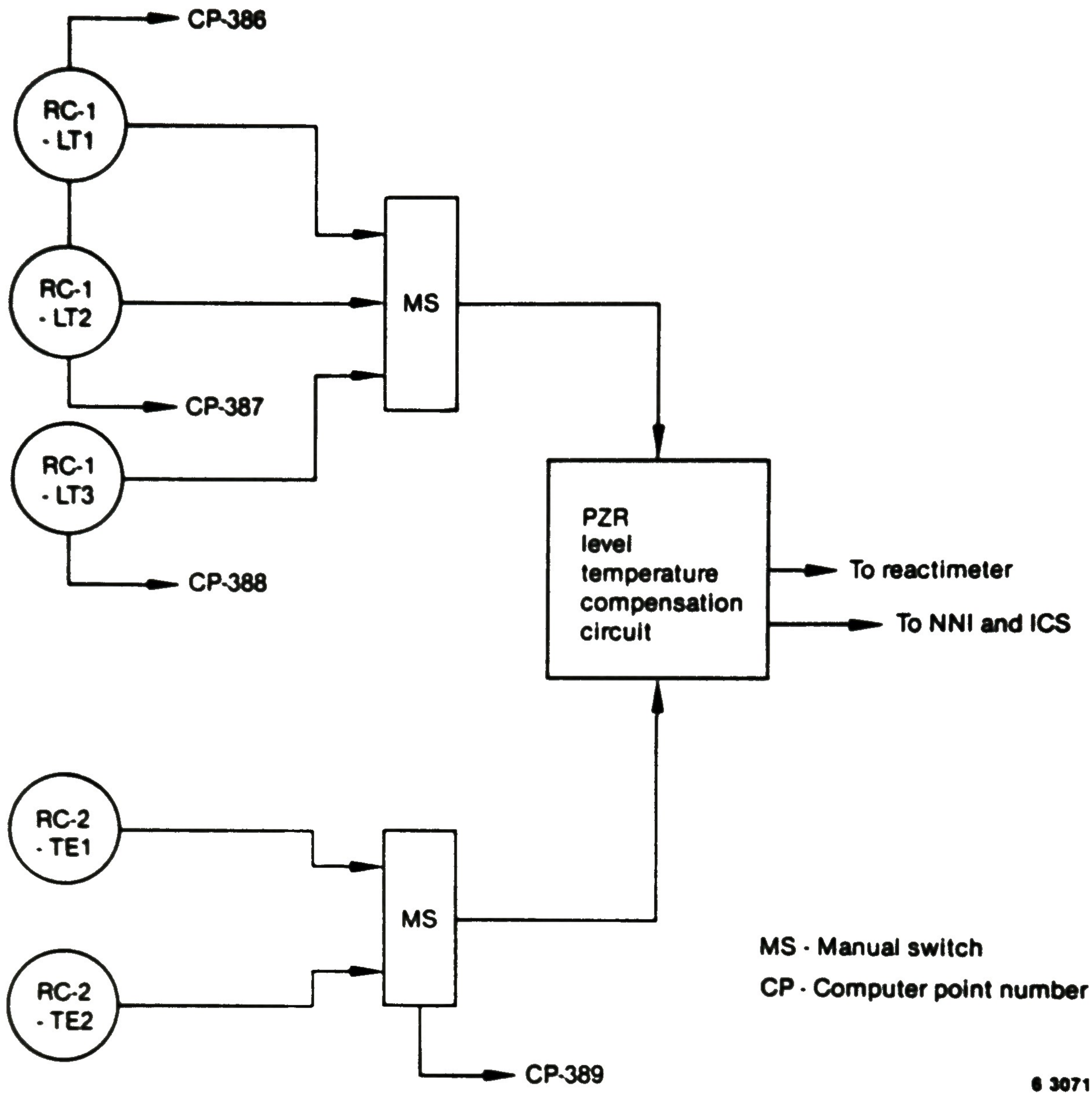


Figure C-1. Block diagram of electronics for the pressurizer liquid level measurement.

TABLE C-1. PRESSURIZER LIQUID LEVEL UNCERTAINTY ESTIMATES

| Uncertainty Component | Uncertainty Estimates (Bias) |
|---|---|
| Transmitter | |
| Accuracy[a] | 0.5% FS (of range span) |
| Amplifier Adjustment[b] | 0.5% FS |
| Temperature Effects [c] | 2.0% FS |
| Pressure Sensitivity[d] | 0.2% FS |
| Transmitter Drift[h] | 0.6% FS |
| Reference Leg Level[e] | 0 |
| Liquid Level Calc. Circuit | |
| Set-up[b] | 1.5% FS |
| Temperature effect for a subcooled pressurizer[f] | 0.8% FS |
| Temperature measurement[i] | 1.2% FS |
| Reference Leg Temperature[j] | 2.1 % FS |
| Recording on Reactimeter[g] | 2.0% FS. |
| TOTAL UNCERTAINTY | 4.2% FS (17 in.) |

a. Given in Bailey instruction manual, Reference C2.

b. Given in Pressurizer Temperature and Level Channel Calibration Procedure, Reference C3.

c. This estimate is a combination of the stated temperature effects within the operational range of -20 - 160°F (0.01% FS/°F) and the maximum reported[C3] error for elevated temperatures (270°F), under postulated accident conditions, of 5% observed zero offset. Since the maximum observed reactor building temperature was 175°F, the 5% value is probably much too large; therfore, a value of twice the stated temperature effect is used [2.0% FS = (.01% FS/°F * (175-75°F)) * 2].

d. The pressure sensitivity is calculated using the value given in Reference C2 as, ( $1.05 \times 10^{-4}$ % FS/psi * 2250 psi ) = 0.24% FS.

e. Using the arguments in the main body of this report, any reduction in the reference leg level due to the initial boiloff or hydrogen effervesence would have been a temporary condition which would have been corrected by condensation in the reference leg as the accident progressed. Therefore, no uncertainty estimate for this effect is included in this analysis.

TABLE C-1. (continued)

| Uncertainty Component | Uncertainty Estimates (Bias) |
| --- | --- |

f. This uncertainty component is based upon an assumption of a subcooled pressurizer at 300°F and 2250 psia. The gas space is assumed to be all hydrogen (any steam would have condensed). A comparison of the results from Equation (A-3) for a saturated pressurizer at 300°F and the above assumptions results in an error of 3.2 in. (0.8% FS) for a DP=200 in. This appears to be the worst case assumption.

g. The information to obtain a good estimate of the uncertainty due to the recording system is currently unavailable. Therefore an estimate of 2% FS is used, which is based upon the tolerance for the plant computer given in Reference C4.

h. Drift is based upon 0.15%/3 months given in Reference C2.

i. Based upon an assumed uncertainty in the RTD temperature measurement of ±2°F. The uncertainty is a function of temperature; however, at 650°F the uncertainty is 1.2% of reading.

j. Assumed reference-leg temperature is 125°F. Maximum recorded reactor building temperature was 175°F. Assuming this temperature was the actual reference-leg temperature during portions of the accident, and a RCS pressure of 1000 psia, results in the tabulated uncertainty.

## REFERENCES

C-1. Data Qualification and Uncertainty Analysis, Appendix A, "TMI-2 Data Analysis and Data Base Development Program," TMI-2 Accident Evaluation Program, 1986.

C-2. Process Computer Transmitter Type BY Series 11, Product Instruction Manual, E21-17, Bailey Meter Co., 1971.

C-3. Elevated Environment Qualification of Bailey BY Differential Pressure Transmitter, Report # 2482, Bailey Meter Co., Engineering Division, January 15, 1972.

C-4. Pressurizer Temperature and Level Channel Calibration, TMI-1 Surveillance Procedure 1302-5.12, Rev. 8, GPU, October 1, 1982.

# APPENDIX D

# PORV BLOCK VALVE OPERATIONS

# APPENDIX D
# PORV BLOCK VALVE OPERATIONS

The operation of the PORV block valve has been surmised from a combination of reactimeter data, reactor building temperatures, and pressures obtained from strip charts, and operator interviews. Timing information obtained from the latter two sources must be suspect. Timing obtained from the strip charts is perhaps within 2-6 min. In the GPU sequence of events (SOE), the times of the block valve operations are given as approximate (although they are given to the second). During the time period of the 2B pump transient, and immediately thereafter, the open and closed times given in the GPU SOE (open at 192.5 min and closed at 210 min) do not correspond to the RCS pressure, RCDT pressure, or pressurizer level responses recorded on the reactimeter.

The RCS pressure and pressurizer level are compared in Figure D-1 for the time period of 192-202 min. The pressurizer level responds significantly to the RCS pressure changes, with dramatic level decreases as the pressure drops, and level increases in response to pressure increases. This response is probably due to the pressurizer being at saturation temperature, and liquid boiling off during pressure decreases, which would force liquid out of the pressurizer surge line along with steam flow out the PORV; this would happen if the block valve is open and causing the pressure decrease. During pressure increases, the pressurizer would be slightly subcooled and condensation effects would result in level increase. The primary temperatures and secondary pressures and levels that were recorded on the reactimeter reveal no changes that would explain the RCS pressure response.

In Figure D-2, the reactor coolant drain tank (RCDT) pressure is compared to the RCS pressure. Note that the RCDT rupture disk had burst at 15 min, and that pressure increases in the RCDT can only be a result of significant steam flow into the RCDT. The RCDT pressure began to increase

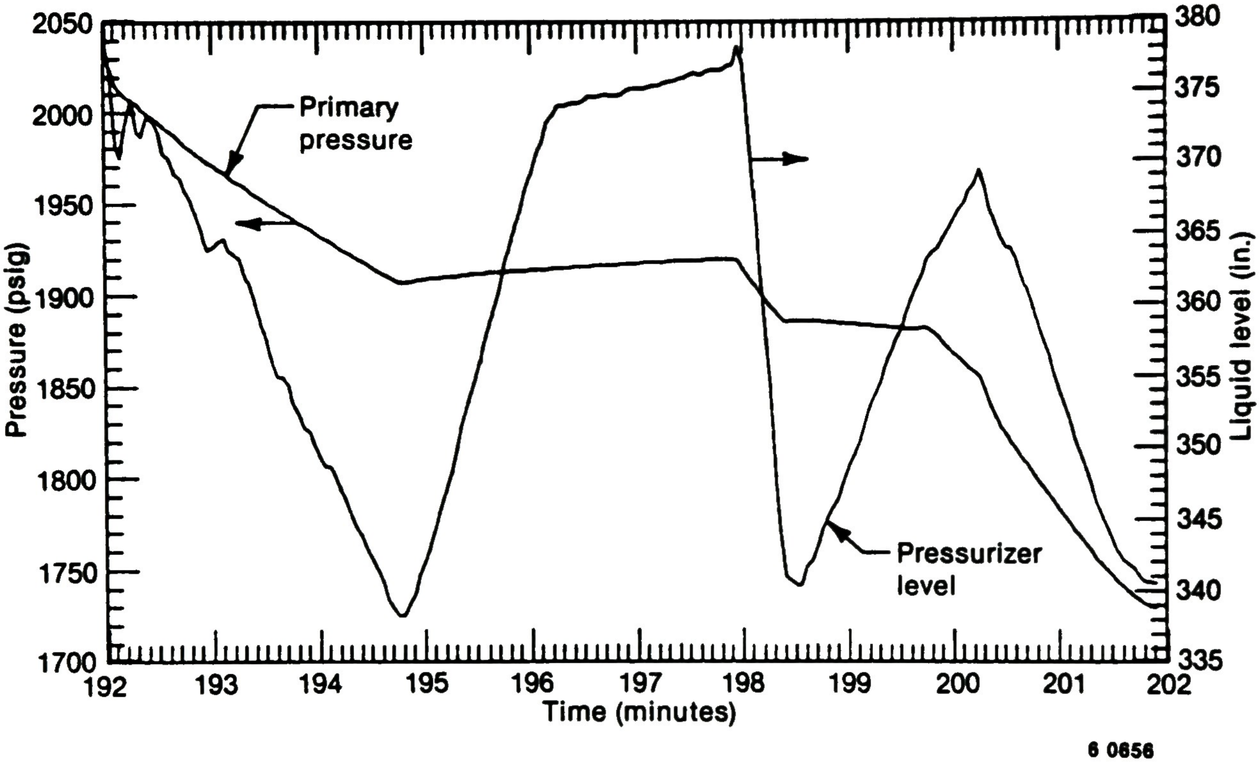


Figure D-1. Comparison of PZR level and pressure from the reactimeter.

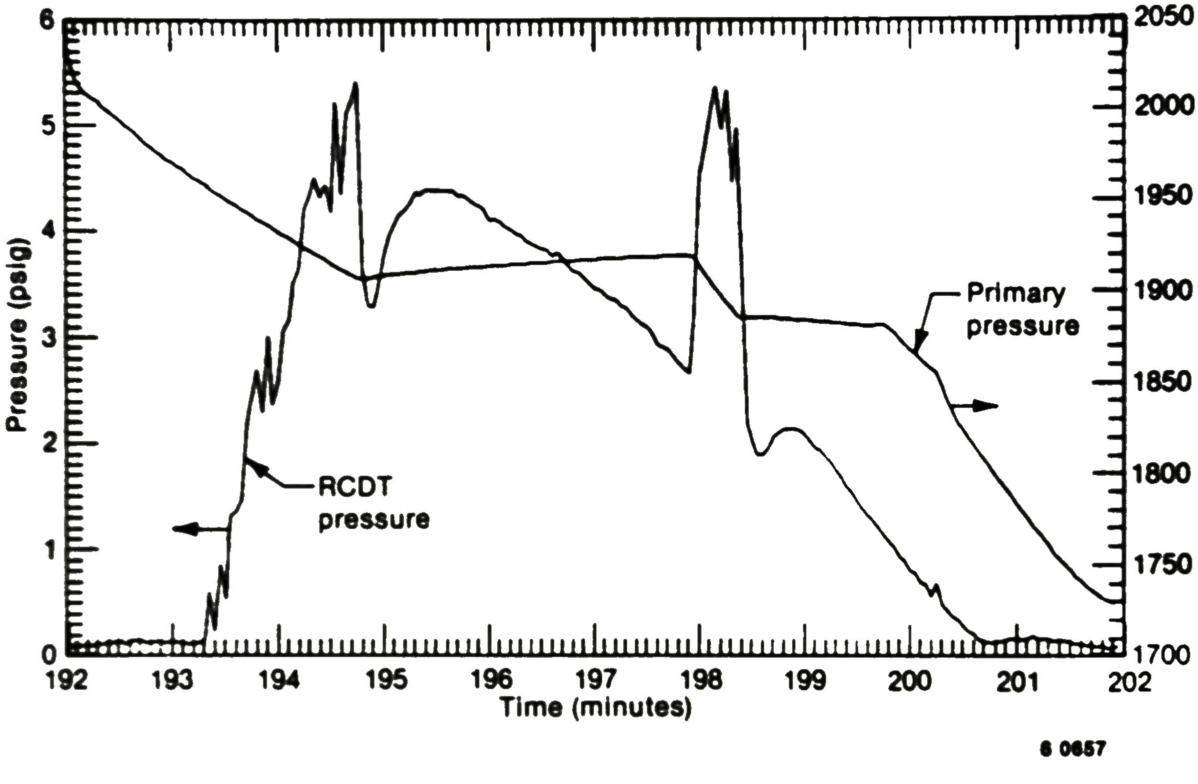


Figure D-2. Comparison of RCDT and primary pressures from reactimeter.

at 193.3 min, increasing to a maximum at 194.75 min, and then abruptly decreasing, coincident with an increase in RCS pressure. Although the RCS pressure continued to increase, the RCDT pressure reached a minimum, then increased, and then decreased until 197.9 min. At this time, no explanation is available for the RCDT pressure increase while the RCS pressure increased. The increase at 197.9 min is coincident with the RCS pressure decrease, until 198.4 min when the RCDT pressure abruptly dropped coincident with the RCS pressure increasing. Finally, at 199.8 min the RCS pressure began a significant decrease, which corresponds to the time given in the GPU SOE for initiation of HPI, obtained from the alarm summary. It is postulated that the RCDT and RCS pressures were responding to unreported operations of the PORV block valve, and that the RCDT was functioning as a surge tank; this resulted in pressure increases when the steam flow through the PORV exceeded the flow capacity out the rupture disk, and then slow decrease as pressurized steam flowed out the rupture disk following closure of the PORV block valve. The PORV block valve operational times given in Table D-1 are therefore proposed.

TABLE D-1. PORV BLOCK VALVE OPERATION TIMES

| Time (min) | PORV Block Valve Operation |
| --- | --- |
| 139 | Closed |
| 191.6 | Opened |
| 194.8 | Closed |
| 197.9 | Opened |
| 198.4 | Closed |
| 220 | Opened |
| 260 | Closed |
| 276 | Opened |
| 318 | Closed |
| 343 | Opened (Valve was cycled until 458 min) |
| 458 | Opened |
| 554 | Closed |
| 560 | Opened |
| 570 | Closed |
| 601 | Opened |
| 672 | Closed |
| 754 | Opened |
| 763 | Closed |
| 772 | Opened |
| 795 | Closed |