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ANALYSIS OF THREE MILE ISLAND UNIT 2 REACTOR COOLING SYSTEM TRANSIENTS

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Prepared by (Name and Dept. No.): J. O. Henrie 65600 A. K. Postma

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Abstract

The reactimeter data recorded on 3/28/79 at Three Mile Island (TMI-2) indicates a number of abrupt transients starting at 13:52. These transients appeared to be the result of very rapid energy releases in the reactor cooling system. A study was initiated by the U. S. Department of Energy to determine the causes and consequences of these transients.

The study shows that the transients were not caused by energy releases in the reactor cooling system. They were probably caused by malfunctions in the reactimeter power supply or by reactimeter ground loop faults.

Information obtained and observations derived from the study of real energy release transients which occurred during the first day of the TMI-2 loss of coolant accident are presented.

* Distribution Name Mail Address

* D. J. Cockeram, 2750E, 200 E
* J. O. Henrie, 2750E, 200 E
* G. A. Huff, 2750E, 200E
* P. G. Lorenzini, 2750E, 200 E
* J. H. Roecker, 2750E, 200 E
* R. E. Smith, 2750E, 200 E, D264
* R. O. Williams,
* D. D. Wodrich, 2750E, 200 E

Nuclear Safety Analysis Center

* J. Haugh

Department of Energy

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EG&G, Idaho

* J. Jacoby
* R. Minenger

Benton City Technology

* A. K. Postma

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Approvals

☐ [signature] P. E. Smith
Program Office

☐ [signature] 5/16/83 E. M. Huff
Research and Engineering

☐ Plant Operations

☐ Health, Safety and Environment

☐ Quality Assurance

☐ Training

☐ End Function

☐ End Function

☐

☐

☐

☐

☒ [signature] D. D. Wodrich
Approval Authority

* Distribution	Name	Mail Address

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1.0 INTRODUCTION

To augment the standard instrumentation of the Three Mile Island-2 (TMI-2) nuclear steam supply system during its startup and initial operation, a data recording system called a reactimeter was installed. It electronically recorded data from 24 instrument channels at 3 sec intervals. This reactimeter system was still in operation on March 28, 1979, when the TMI-2 loss of coolant accident (LOCA) occurred. The reactimeter record (magnetic tapes) contributed greatly to the data base used for the analysis of that accident.

Early reviews of the reactimeter data by one of the authors (Henrie) identified a sharp rise in pressure in the Once Through Steam Generators (OTSG) A and B about 2 min after the initiation of the containment hydrogen burn. The indicated pressure rises occurred in both steam generators during the same 3 sec period, and maintained the same or higher indicated pressures for longer than 1/2 min. Since no satisfactory explanation for these data was apparent, reactimeter data for other channels and other times were evaluated. Eight sharp spikes or steps were identified: two in a 1 min period starting at 08:17 and six in the 6 min period starting at 13:52. Each of these pulses or steps were indicated on many, but not all, of the 24 channels monitored by the reactimeter. The pulses at the various times had greatly different shapes. The pulses that occurred at the same time on the various instrument channels had similar but not identical shapes, and the latest pulses differed significantly in shape. See Figures 1A to 2C. The temperature, pressure, and flow sensing instruments that indicated those rapid transients were at widely differing locations--some inside and some outside of the containment building.

It was tentatively concluded that these rapid transients indicated a significant energy release and were not simply the result of electrical noise. Therefore, a study sponsored by the U.S. Department of Energy was initiated to determine the causes and consequences of these transients. This report presents the findings of that study.

2.0 INVESTIGATION

The investigation consisted of collecting data, standardizing timing, and evaluating the temperature, pressure, and flow data available from the computer alarm printer, auxiliary printer, and various recorder strip charts. It also included plotting and evaluating reactimeter data during the early spike period (08:15 to 08:25) and during the times of known energy releases such as the metal-water reaction which started at ~06:15 and was quenched at ~6:55, the core transient at ~07:45, and the emergency feed-water transients at ~07:43 and ~15:39. The timing of all data referred to in this document has been corrected to correspond with plant computer time.

The locations and characteristics of many affected instruments, pumps, and valves, and information concerning their on-off timing and interrelated effects were determined. An evaluation of all available information was then conducted.

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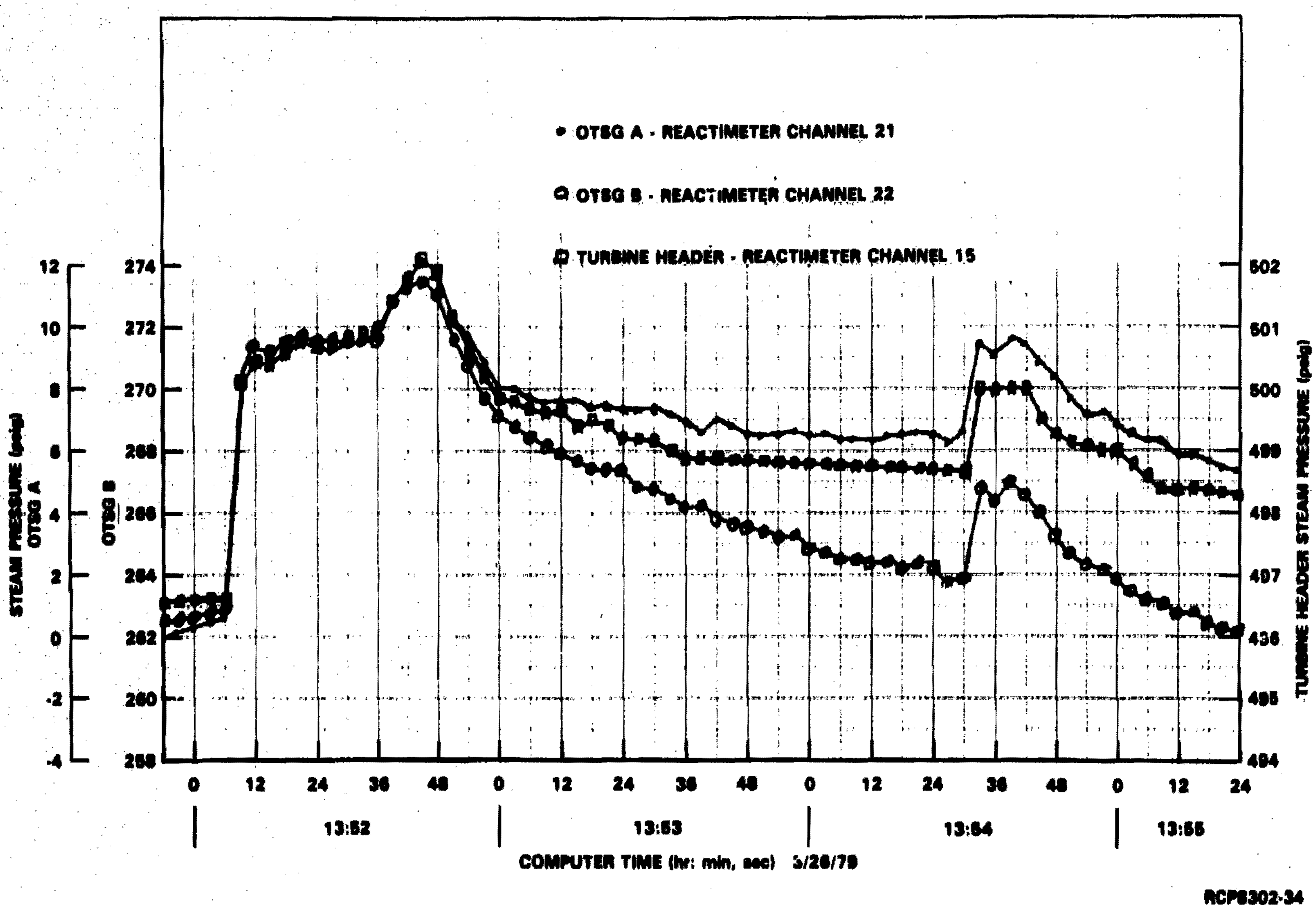

FIGURE 1A. Steam Pressure as Indicated by Reactimeter, 13:52 to 13:55.

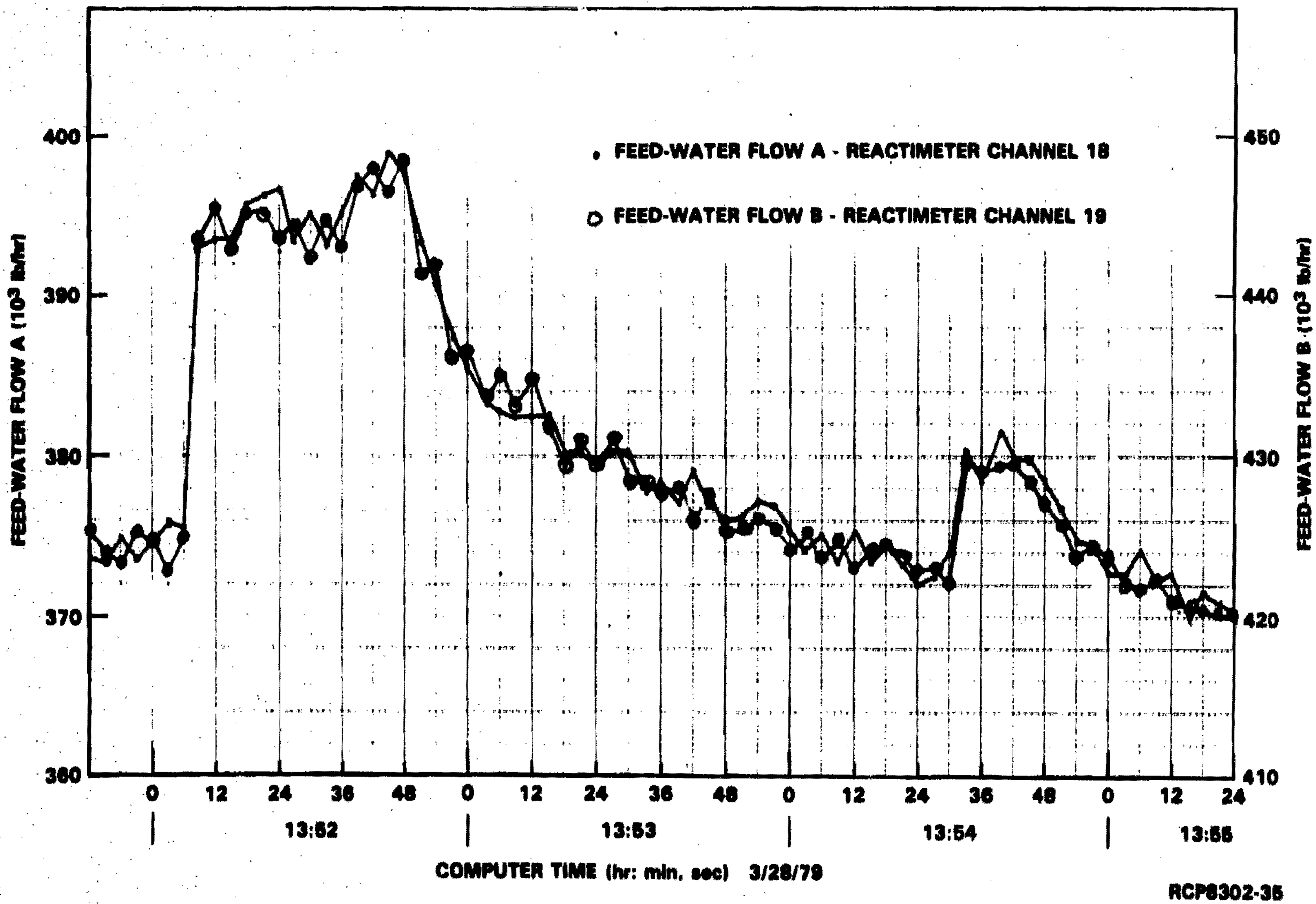

FIGURE 1B. Feed-Water Flow as Indicated by Reactimeter, 13:52 to 13:55.

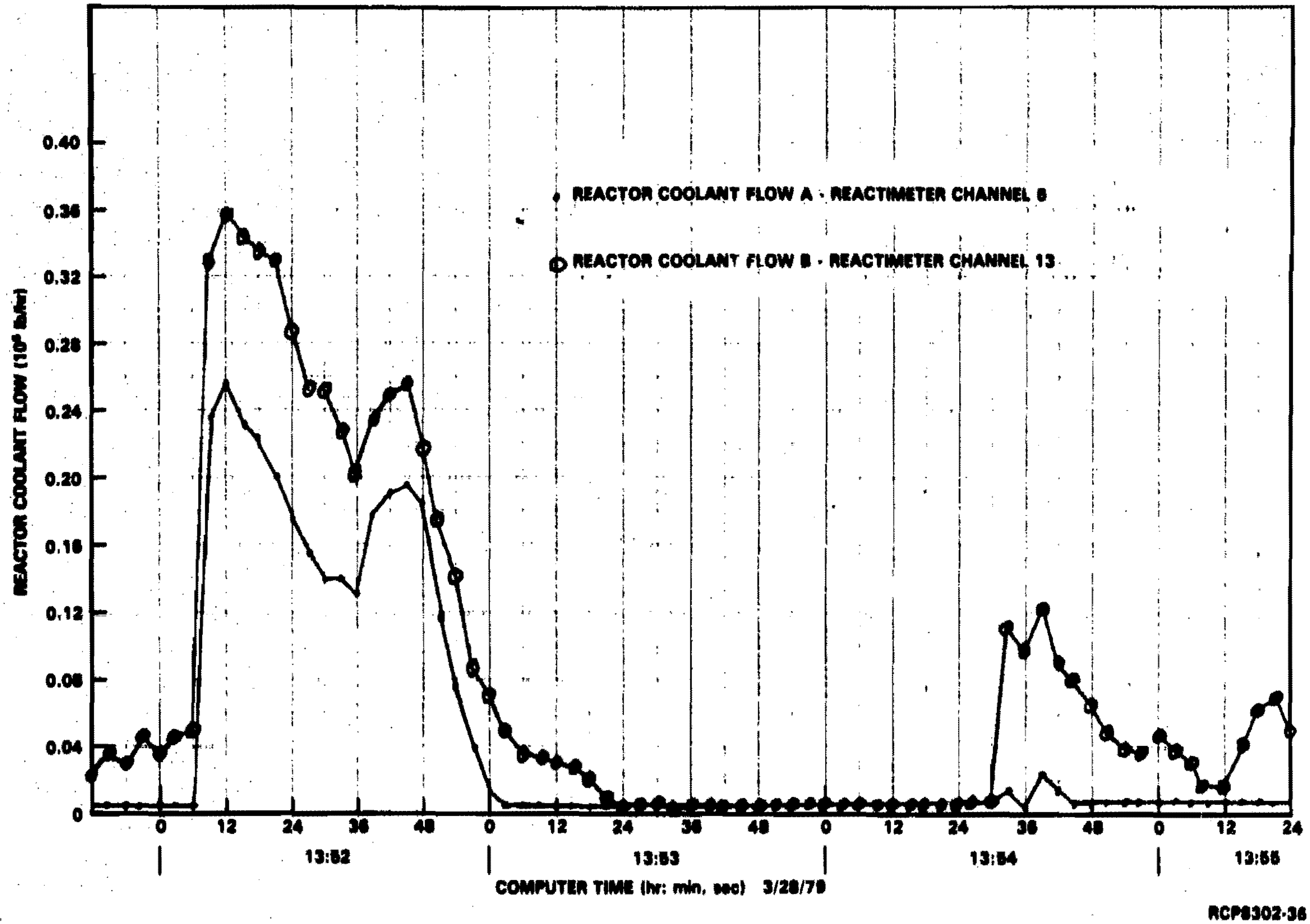

FIGURE 1C. Reactor Coolant Flow as Indicated by Reactimeter, 13:52 to 13:55.

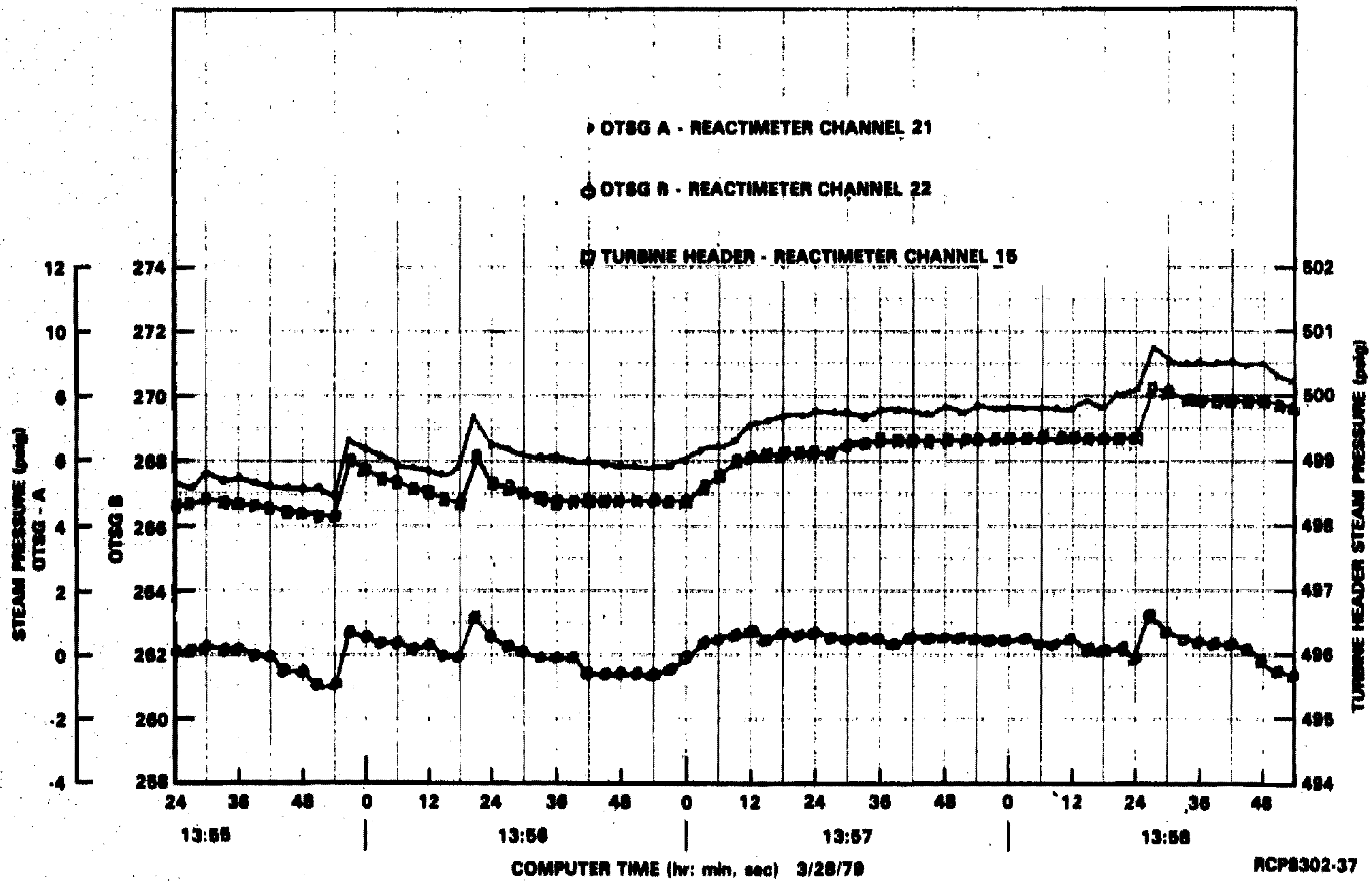

FIGURE 2A. Steam Pressure as Indicated by Reactimeter, 13:55 to 13:58.

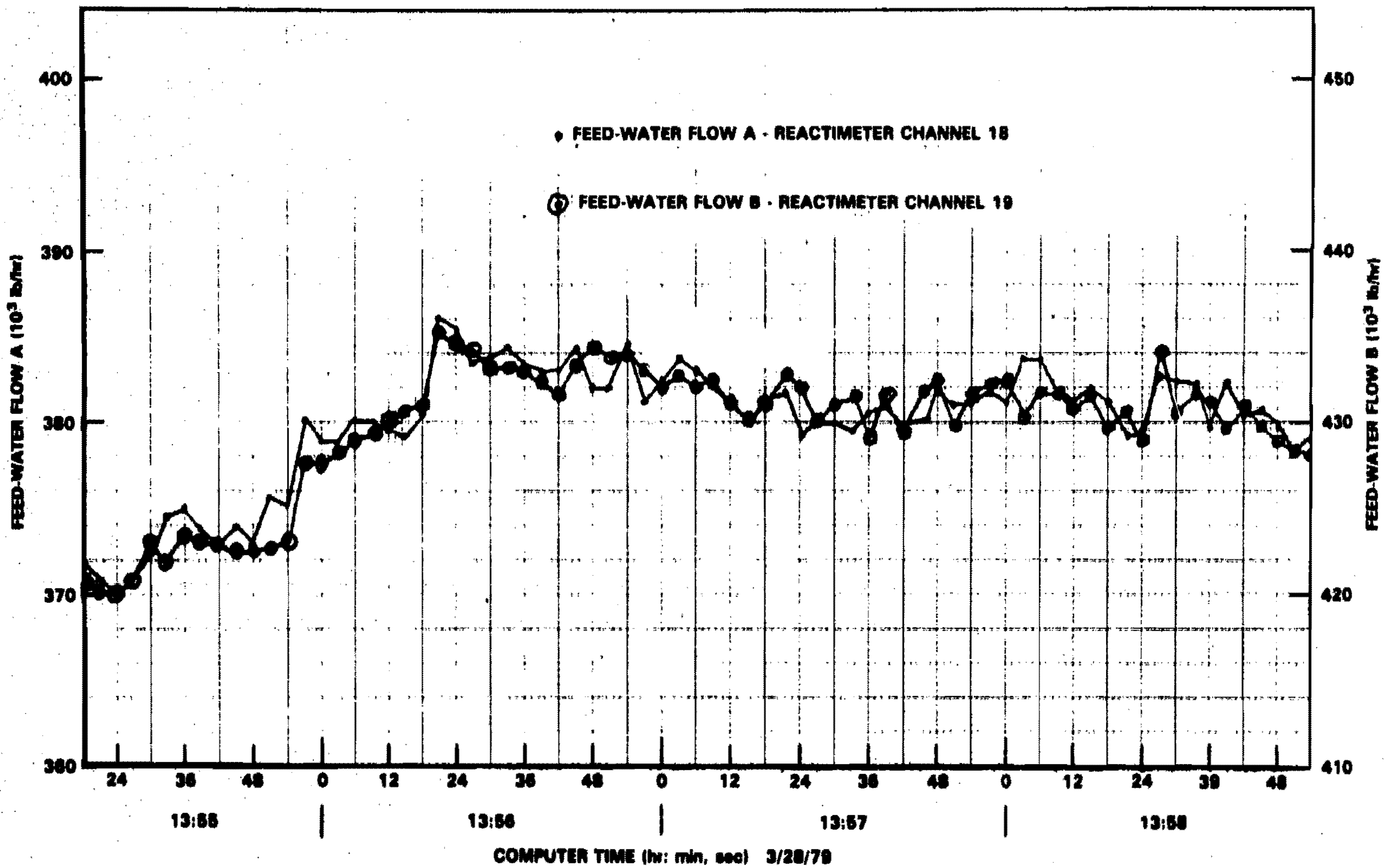

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FIGURE 2B. Feed-Water Flow as Indicated by Reactimeter, 13:55 to 13:58.

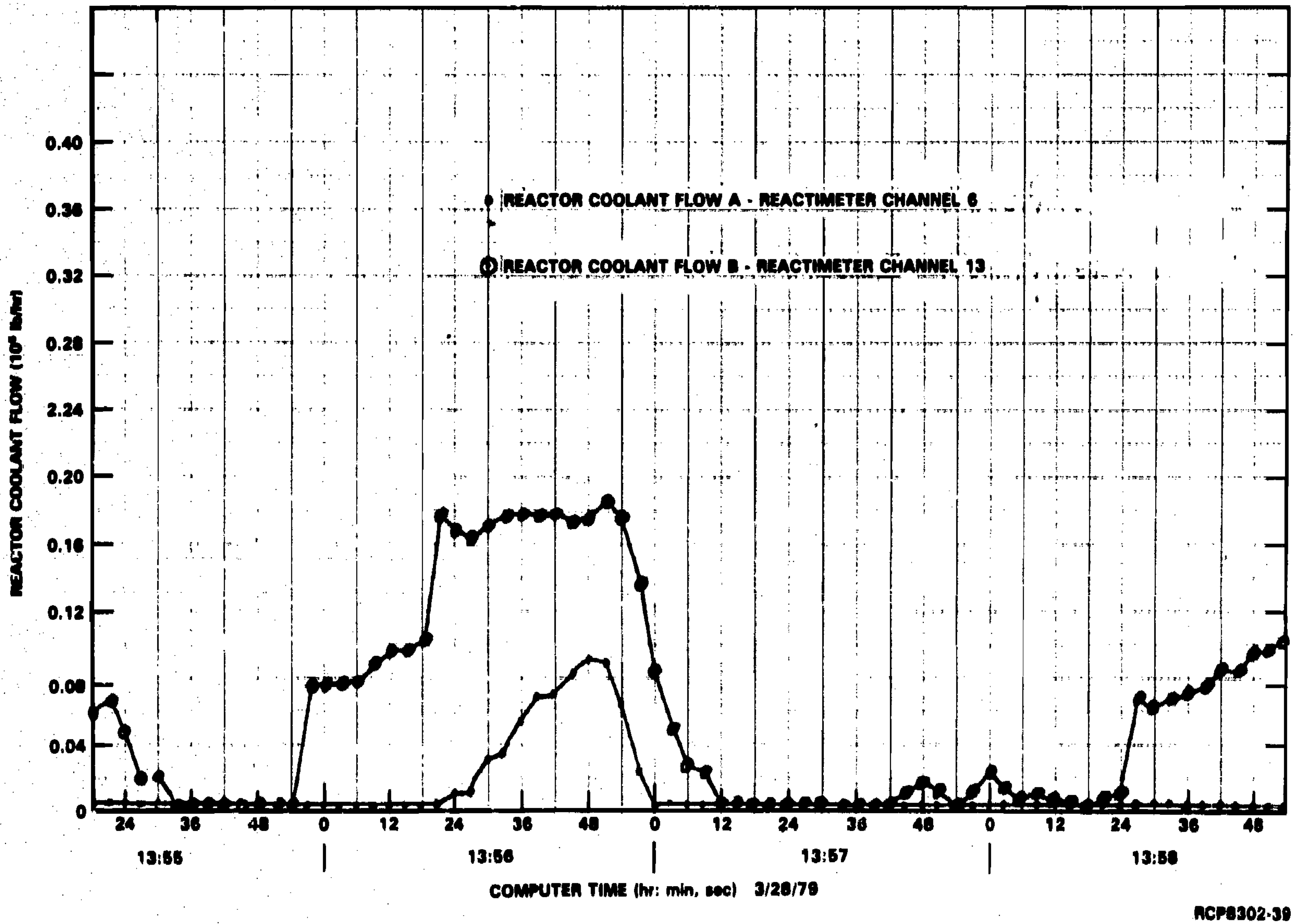

FIGURE 2C. Reactor Coolant Flow as Indicated by Reactimeter, 13:55 to 13:58.

3.0 CONCLUSIONS

The evaluation resulted in the following conclusions and summary statements:

1. *The steam pressure in steam generator A, as indicated by computer data recorded on the auxiliary printer, did not show any of the pressure pulses that were recorded by the reactimeter between* 13:52 and 13:58 (see Figures 1A, 2A, and 3). *The computer data is considered to be the more reliable. The computer data shows that the shell of steam generator A, near its upper end, was much hotter than the steam saturation temperature and that it cooled rapidly* at the same time as the reactimeter-indicated transient. This rapid cooling was apparently caused by the containment spray wetting the temperature sensor rather than the cooling effect of large quantities of saturated steam in the steam generator. That this temperature sensor could have been cooled by the containment spray should be verified by careful examination including physical arrangement, water marks, water deposited fission products, etc. If the sensor could not have been wetted, further study would be warranted.

2. A detailed evaluation of the feed-water flowmeters indicates that their output would not have responded to a pressure pulse. In a differential pressure sensing device of this type, both sides would be exposed to essentially the same pressure at any given time and the net differential would be essentially zero at all times during a pressure pulse. Further, records indicate that the feed-water pumps were not operating during the period of the pulses. Also, feed-water temperature data show less than 1°F change during the period of the pulses, indicating that there was probably no flow surge at that time. Therefore, the increased feed-water flow signal recorded by the reactimeter data (see Fig. 1B and 2B) appears to have been caused by something other than a change in feed-water flow or pressure.

3. An evaluation of the simultaneous increase in reactor coolant flow in loops A and B (see Fig. 1C and 2C) shows that the zero flow indication for loop A is more suppressed than for loop B. This gives an apparently false indication that loop B flow is typically higher than loop A flow during the transients. The significant difference in the shape of the A and B flow signals during the 13:56 transient (see Fig. 2C) could have been caused by an electrical noise signal superimposed on the two flowmeter outputs which were varying quite differently due to some other influence at that time.

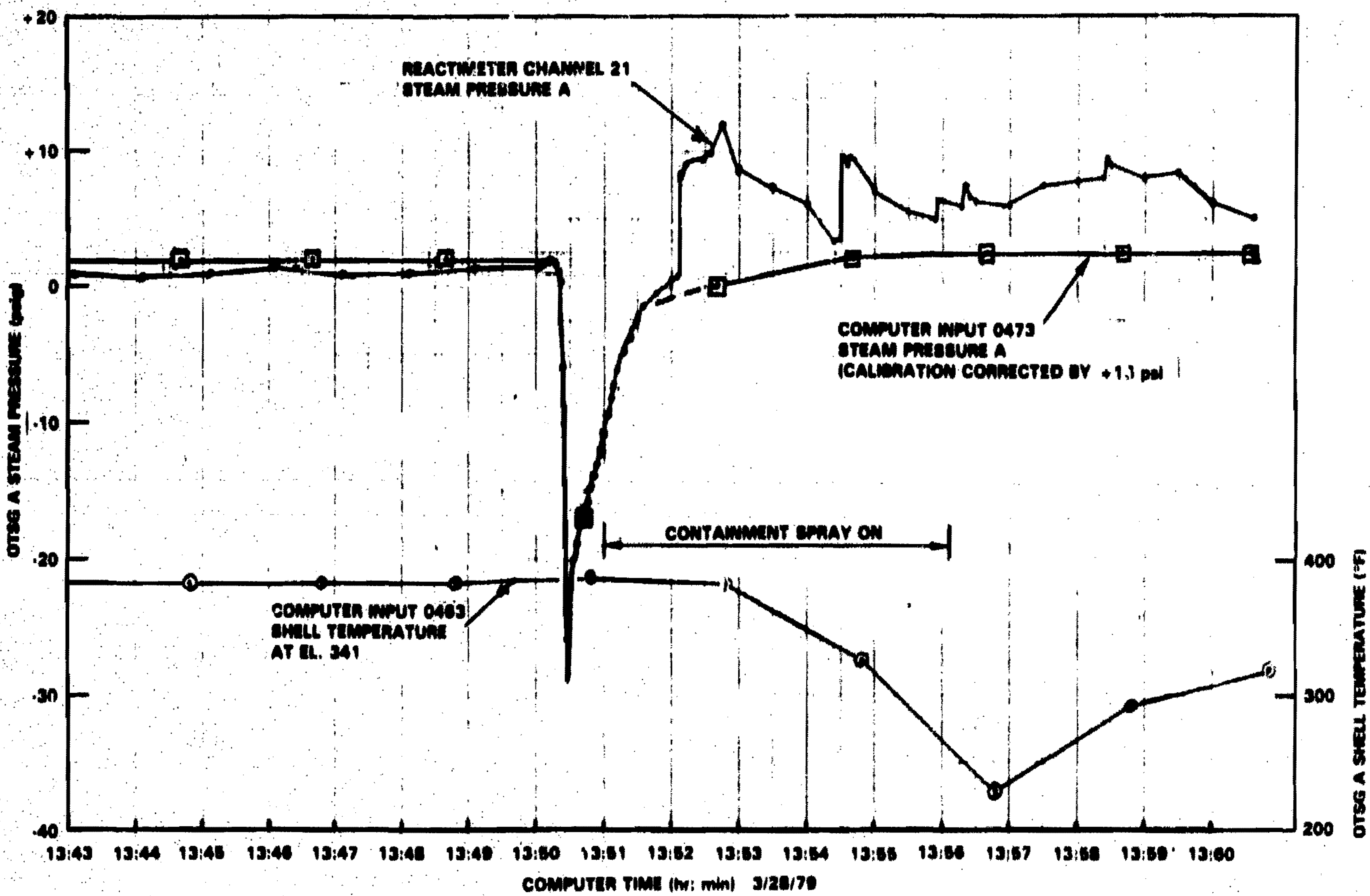

FIGURE 3. Comparison of Reactimeter and Computer-Indicated Steam Pressures, and Computer-Indicated Shell Temperature for Steam Generator A, 13:43 to 13:60.

4. The simultaneous increases in reactimeter-indicated steam generator liquid levels appeared to be supported by similar changes shown on strip chart recorders. However, it was determined that the transient indicated by the strip charts occurred earlier starting at 13:50. Therefore, the 13:52 to 13:58 increases in steam generator level indicated by the reactimeter data were not indicated on the strip chart recorders and were apparently caused by something other than actual level increases.

5. Recent tests conducted at Three Mile Island on the actual turbine header pressure sensor show that even though it is functioning properly, it does not respond to pressures below about 500 psi. Since OTSG A was open to the steam header and its pressure was actually between 0 and 10 psig (see Fig. 1A and 2A), the indicated turbine header pressure changes (ranging off-scale-low between 496 and 502 psig) were probably caused by some type of signal cross coupling.

6. The analysis of the reactimeter pulses that occurred at 08:17 and 08:18 showed that they were all short-term spikes with quarter cycle periods of 3 sec or less. Positive and negative spikes from most of the reactimeter channels correlated well with each other during this period. See Figure 4. The asterisks (*) indicate spikes that correlate in their timing with other spikes. Unlike the broader pulses recorded during the 13:52 to 13:58 period, these spikes are typical of superimposed electrical noise. Note that there are longer repeating cycles, most of which are in phase, with some about 120 degrees out of phase. Further, the spikes indicated on channel 7, pressurizer level, were noted to be 180 degrees out of phase with those of all other channels that were responding to the spikes. It was determined that this channel was established to have a reverse input signal. Its output reads from 0 to 400 in. as its input signal varies from 10 V positive to 10 V negative. All other channels that did not respond to the spikes were off-scale, except for channel 14, feed-water temperature. The feed-water temperature channel may have signal conditioning that is different from that reported--which is to indicate a 0 to 500°F output as its input signal varies from 10 V positive to 10 V negative. (This is not typical of temperature-indicating channels.) Since the spikes all appear to be caused by superimposed noise, and since essentially all of the reactimeter channels that were on scale were responding to the noise, the noise signal may have originated in the reactimeter power supply or grounding system which is common to the signal conditioning circuitry for all channels.

A correlation of the reactimeter channels affected by the noise at 08:17 and 08:18 shows that the same channels were affected by the longer pulses that occurred between 13:52 to 13:58 (see

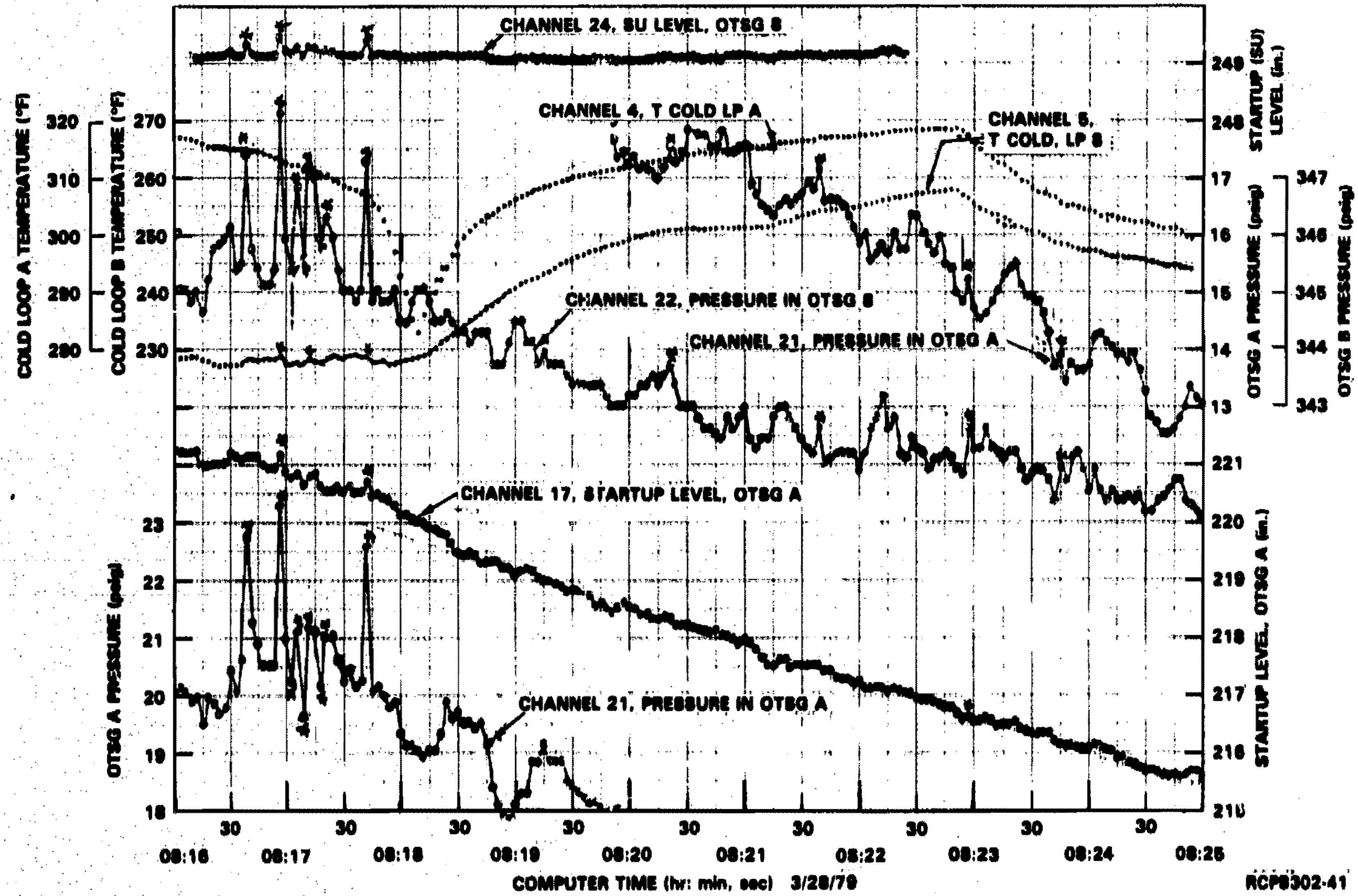

FIGURE 4. System Response as Indicated by Reactimeter. (Sheet 1 of 3.)

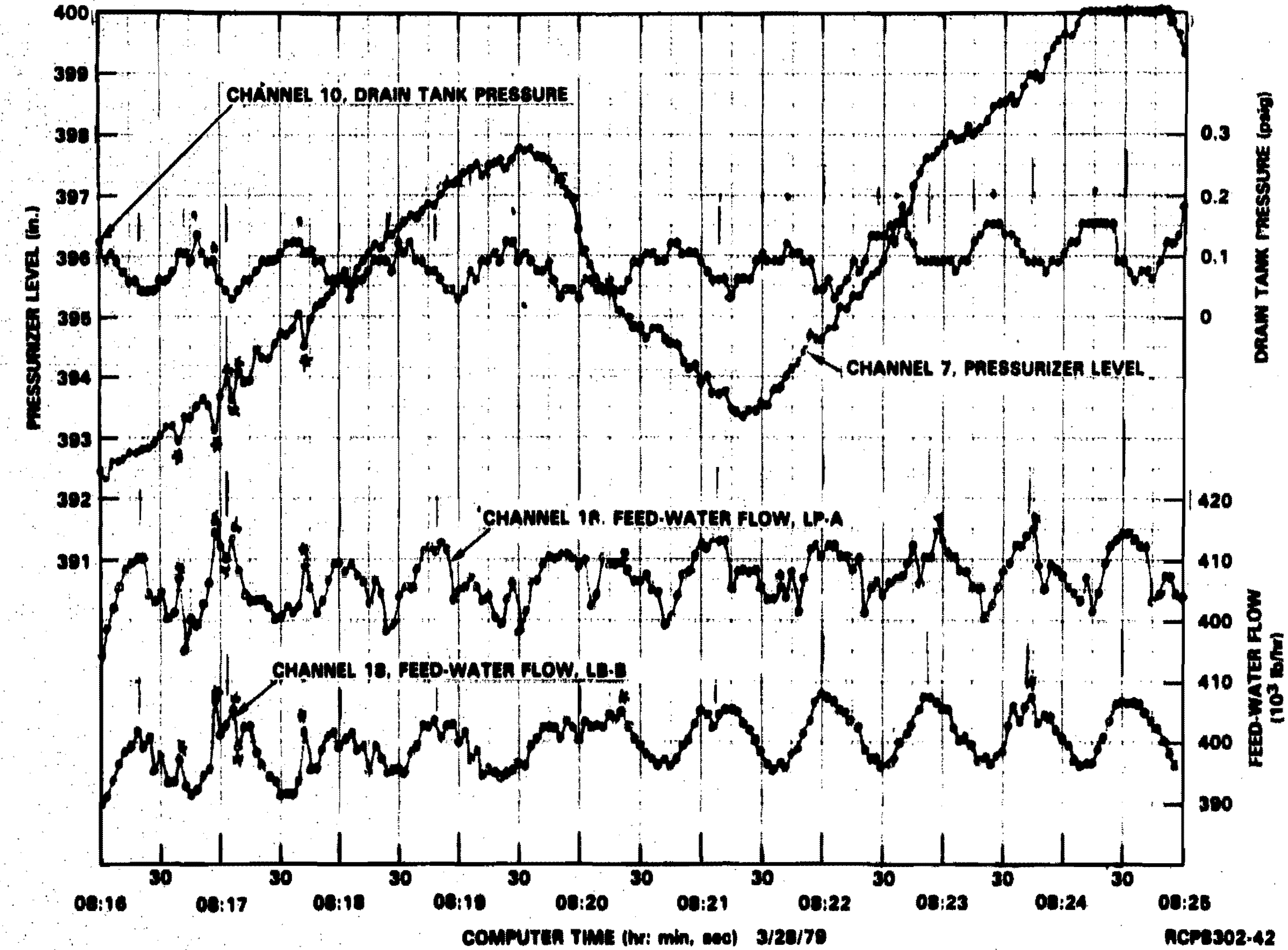

FIGURE 4. System Response as Indicated by Reactimeter. (Sheet 2 of 3.)

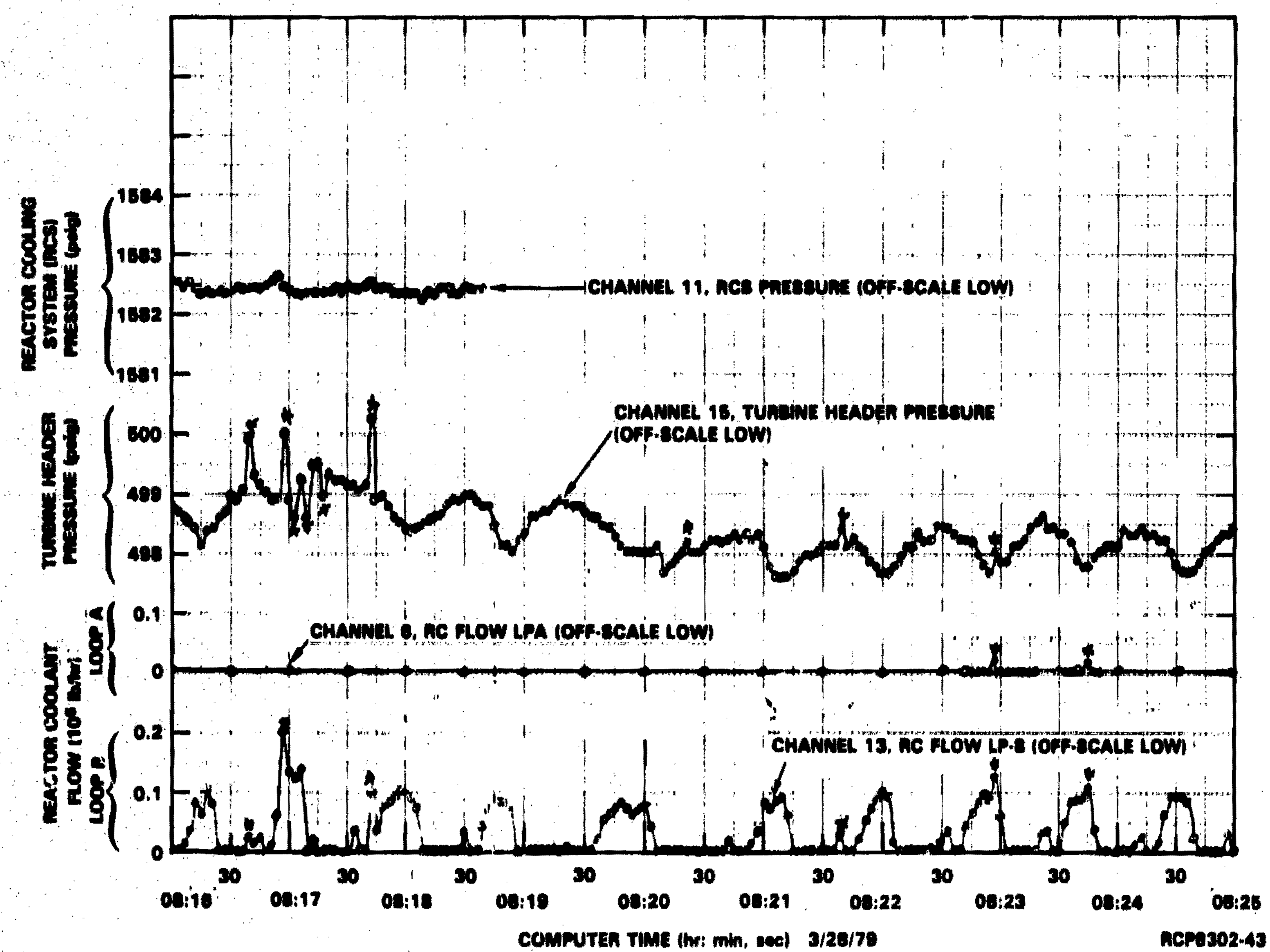

FIGURE 4. System Response as Indicated by Reactimeter. (Sheet 3 of 3.)

Table 1). This correlation indicates that these later, longer pulses may also have been caused by superimposed electrical noise, even though the noise characteristics were drastically different.

7. A statistical study of the reactimeter data by EG&G Idaho personnel assigned to assist in this study shows good correlation between signal outputs on various channels many times during the day (3/28/79). This finding supports the conclusion that the transients had a single common cause.

8. It is concluded from the preceeding evaluations, that the two sets of spikes and steps were false indications of parameter changes, possibly caused by reactimeter power supply malfunctions, or ground loop faults. Studies of energy releases that occurred during the first day of the TMI-2 LOCA indicate that energy was not released as quickly as would have been implied by the 13:52 to 13:58 reactimeter pulses, if these pulses had resulted from real parameter changes. However, since much was learned about these energy releases, some of the information is documented in Sections 4.0 and 5.0. Explanations were developed for many of the observed temperature, pressure, and liquid-level anomalies, which help to better understand the accident. These are summarized in Section 6.0.

4.0 EMERGENCY FEED-WATER TRANSIENTS

Operation of the emergency feed-water (EFW) system would be expected to produce transient responses in various plant systems. Therefore, feed-water transients were analyzed to determine their signatures and, thereby, determine whether the transients observed at 13:52 could have been produced by an unrecognized EFW operation.

Two well characterized EFW operations, one in each steam generator, have been analyzed in detail; the more pertinent results are described in the following paragraphs.

4.1 KEY PROPERTIES OF EMERGENCY FEED-WATER SYSTEM

Emergency feed water can be supplied to either steam generator through supply systems that are independent from the normal feed-water systems. Emergency feed water enters the shell side of the steam generator at a high elevation where it contacts the tubes as spray. Because fluid from the primary coolant system enters the steam generator from the top, the EFW initially contacts the hottest regions of the tubes and flashes the water to steam.

Startup of EFW pumps is recorded as an event on the alarm printer. The time that a threshold pressure is reached on the pump discharge is also recorded on the alarm printer. However, feed-water flow cannot begin until

TABLE 1. Response of Reactimeter Channels to Pulses at 08:16 and 13:52.

Reactimeter channel	Input voltage	Description	Readings at stated time[a]							
			08:16			13:52				
			54 sec	57 sec	60 sec	6 sec	9 sec	12 sec	15 sec	18 sec
1	0 to 10	POWER RANGE[b]	0.144967	c	c	0.213636	c	0.206006	c	0.213636
2	-10 to 10	T/HOT A[b]	620	c	c	620	c	c	c	c
3	-10 to 10	T/HOT B[b]	620	c	c	620	c	c	c	c
4	-10 to 10	T/COLD A (small)	313.706	c	312.772	194.406	195.633	195.871	195.962	196.182
5	-10 to 10	T/COLD B	228.520	228.923	227.513	145.788	147.015	147.253	147.235	147.363
6	-10 to 10	RC/FLOW A[b]	0.00274658	c	c	0.002747	0.238983	0.258209	0.233490	0.222504
7	+10 to 10[d]	PZR LVL	393.554	393.139	393.1688[d]	400.012[b]	c	c	c	c
8	-10 to 10	MUT LVL	84.5755	84.9509	85.1157	65.6320	65.8182	65.9128	65.8671	65.8457
10	0 to 10	DR. TK. PRESS[b]	0.091558	c	0.061039	0.091558	0.106818	0.106818	0.137337	0.122078
11	0 to 10	RC PRESS[b]	1582.61	1582.41	1582.37	1577.19	1577.24	1577.29	1577.29	1577.29
13	-10 to 10	RC FLOW B	0.065933	0.206009	0.134590	0.052200	0.324127	0.359833	0.340607	0.335114
14	+10 to -10[d]	F.W. TEMP	437.389	437.374	437.359	288.149	288.119	288.119	288.119	288.103
15	+2 to +10[e]	TURB HDR PRES	498.905	500.148	498.905	496.604	500.194	500.654	500.562	500.654
16	-10 to 10	OTSG A OP LVL	47.9949	48.0956	48.0437	100[b]	c	c	c	c
17	-10 to 10	OTSG A SU LVL	220.953	221.167	220.892	250[b]	c	c	c	c
18	-10 to 10	F.W. FLOW A	406.275	414.805	412.623	375.527	392.984	393.381	393.381	395.761
19	-10 to 10	F.W. FLOW B	395.563	406.870	401.316	425.121	443.570	445.553	442.975	445.355
21	+2 to +10	OTSG A ST PRESS	20.5493	23.3462	21.0154	0.6897	8.1487	9.0811	8.8945	9.1743
22	+2 to +10	OTSG B ST PRESS	345.410	348.151	345.958	262.835	270.143	271.421	271.239	271.421
23	-10 to 10	OTSG B OP LVL	65.8213	65.9586	65.9189	61.0297	61.2495	61.3166	61.2769	61.2128
24	-10 to 10	OTSG B SU LVL	248.169	248.482	248.253	243.843	244.575	244.606	244.529	244.499

NOTE: Underline indicates response to pulses.

[a] Reactimeter time corrected to plant computer time by adding 1 min 12 sec from 04:00 to ~10:00 and 1 min 6 sec from ~10:00 to 15:00.

[b] Off-scale.

[c] No change from previous reading.

[d] Channel 7 has a negative spike (dip) signal condition noted to be 180 degrees out of phase. Same notation for channel 14, which did not respond.

[e] Tests of turbine header pressure orally reported by EG&G Idaho personnel on 12/10/82 show no response below 500 psi.

valves are opened; valve opening is not recorded on the alarm printer, and the time at which flow started had to be inferred by changes in steam pressure and water level. The latter two parameters were recorded by the reactimeter and were readily available.

The EFW system was operated at 07:43 and at 15:39 on March 28, 1979. For these two periods the data record is available so that system response can be analyzed in detail. The second operation (at 15:39) will be described first because it was a dominant event at that time. The earlier EFW operation (at 07:43) was complicated by a core transient that occurred 2 min later. By comparison with the "clean" EFW operation at 15:39, the effects of EFW operation at 07:43 can be separated from those of the 07:45 core transient.

4.2 SYSTEM RESPONSE TO EFW OPERATION AT 15:39

Emergency feed-water pump 2B (EF-P-2B) was started at 15:34:21; pump outlet pressure reached a switch trip level (1,555 psi) a few seconds later at 15:34:26. These events are recorded by the alarm printer.

Feed water began spraying into OTSG B at ~15:39:06. This is indicated by the beginning of a sharp rise in steam pressure in OSTG B that began at that time. This timing is illustrated in Figure 5. Also plotted on Figure 5 is the water level in OTSG B. As indicated, water level begins to increase at ~15:39:27, 21 sec after the pressure spike began. Essentially all of the water entering the steam generator during that lag period apparently flashed into steam or was held up in the steam space.

Feed-water flow continued until ~15:52:41. This is indicated by the time that the feed-water pump tripped (alarm printer record). Feed-water termination is also indicated by the graph of water level versus time (see Fig. 6). The water level stops increasing at ~15:53:03, some 20 sec after pump trip. This short delay between pump trip and the time for maximum water level is reasonably accounted for by pump coast-down, by the draining of water held on the surfaces of tubes, and by other inertias in the systems.

The thermal response of OTSG B was obtained from data recorded by the utility printer, and is plotted on Figure 7. Steam pressure, obtained from the reactimeter, is also shown on Figure 7.

The data given in Figure 7 indicate that steam pressure first increased, then decreased, until feed-water flow was terminated. Upper downcomer temperature cooled noticeably, from 412°F to 375°F, as a result of feed-water flow. Shell temperature and steam temperature began to cool slowly after feed-water flow had started.

The cooling of the exposed steam generator tubes could be expected to condense steam on the primary side and to possibly affect pressure in the reactor coolant system (RCS). A plot of RCS pressure versus time (Figure 8) shows that RCS pressure was mildly affected; the EFW operation appears to have halted a ramp in pressure. After feed-water flow stopped, the pressure ramp reappeared.

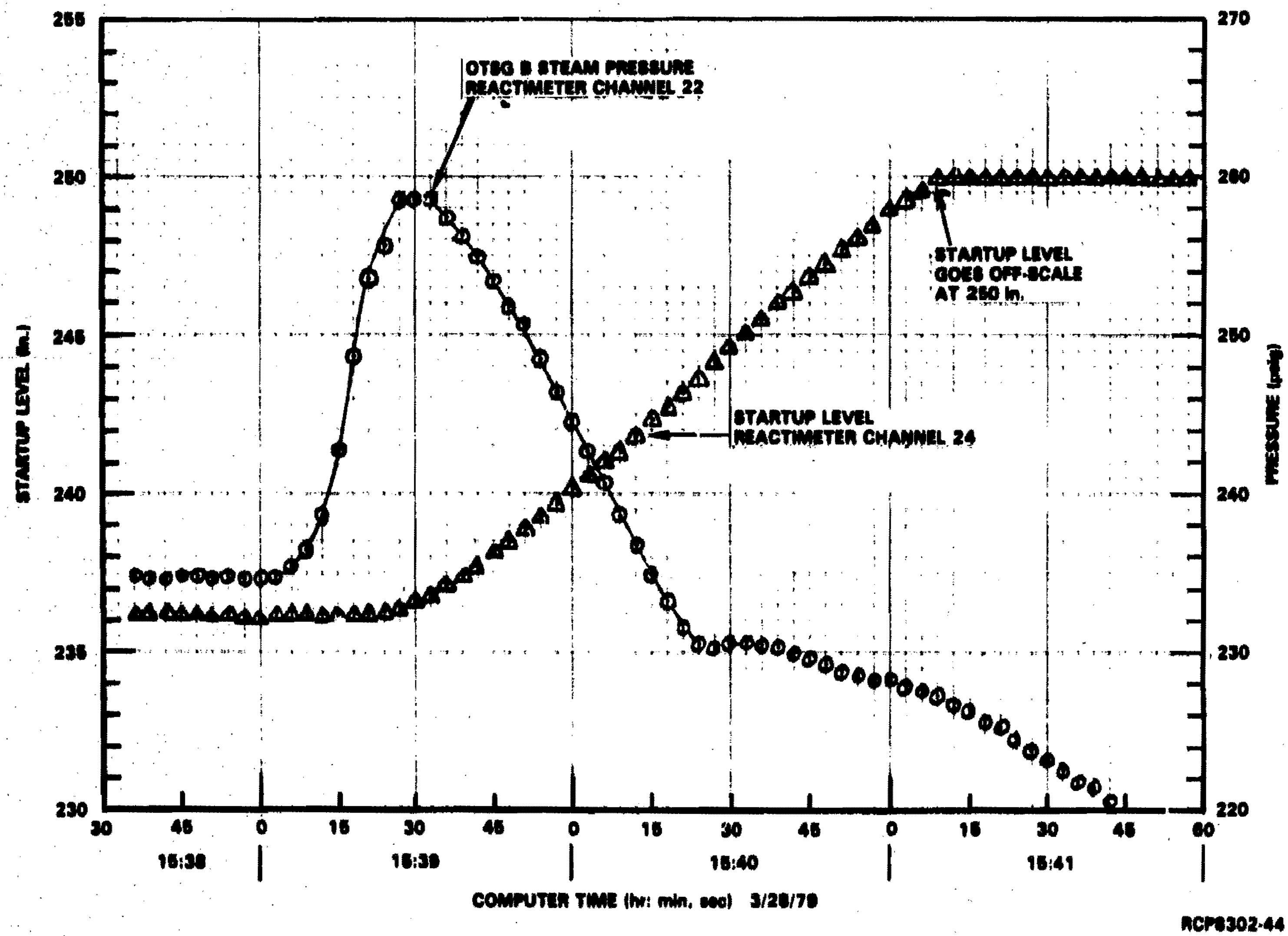

FIGURE 5. Response of OTSG B to Emergency Feed-Water Flow at 15:39.

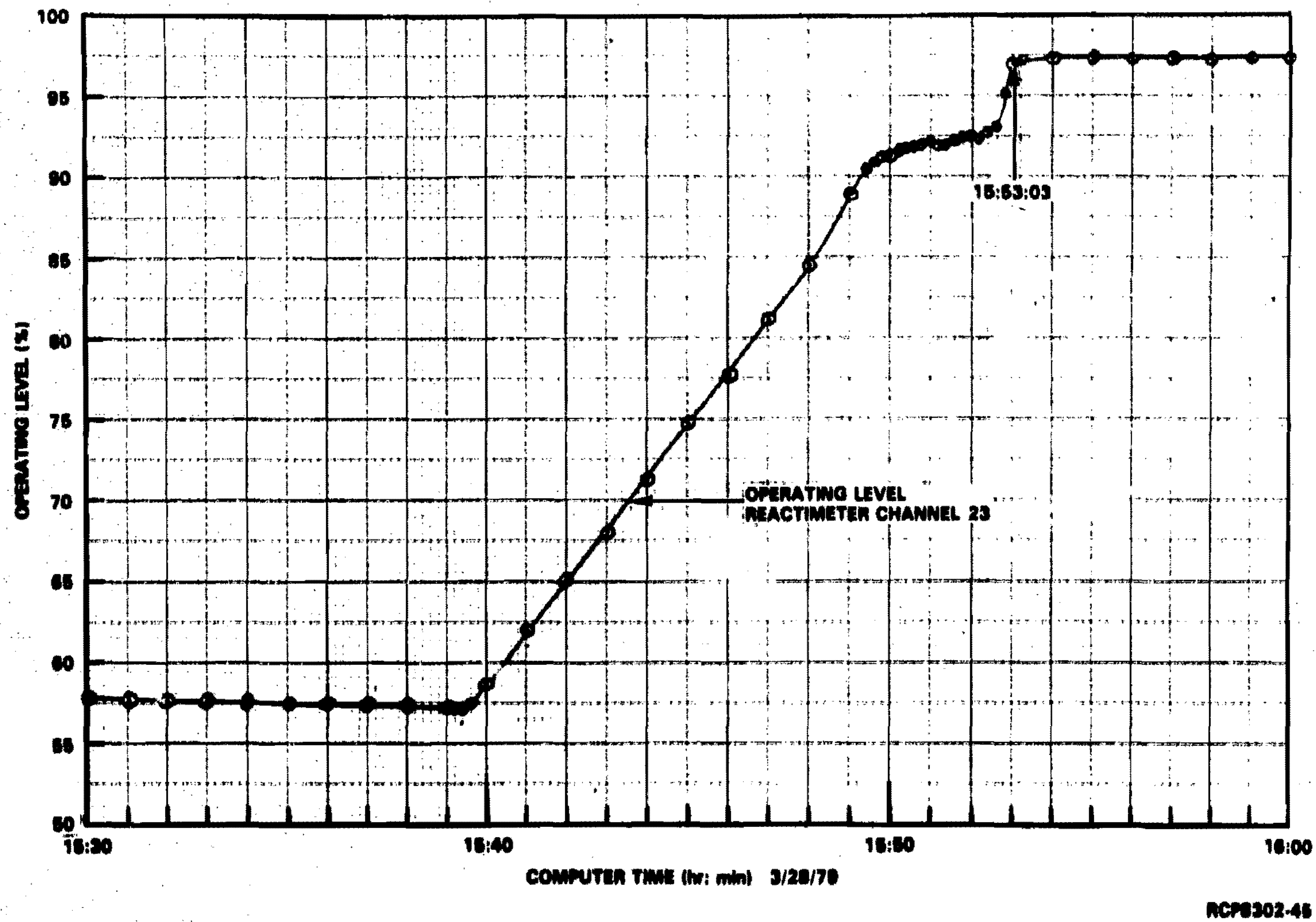

FIGURE 6. Increase in Water Level in OTSG B Due to Emergency Feed-Water Flow at 15.39.

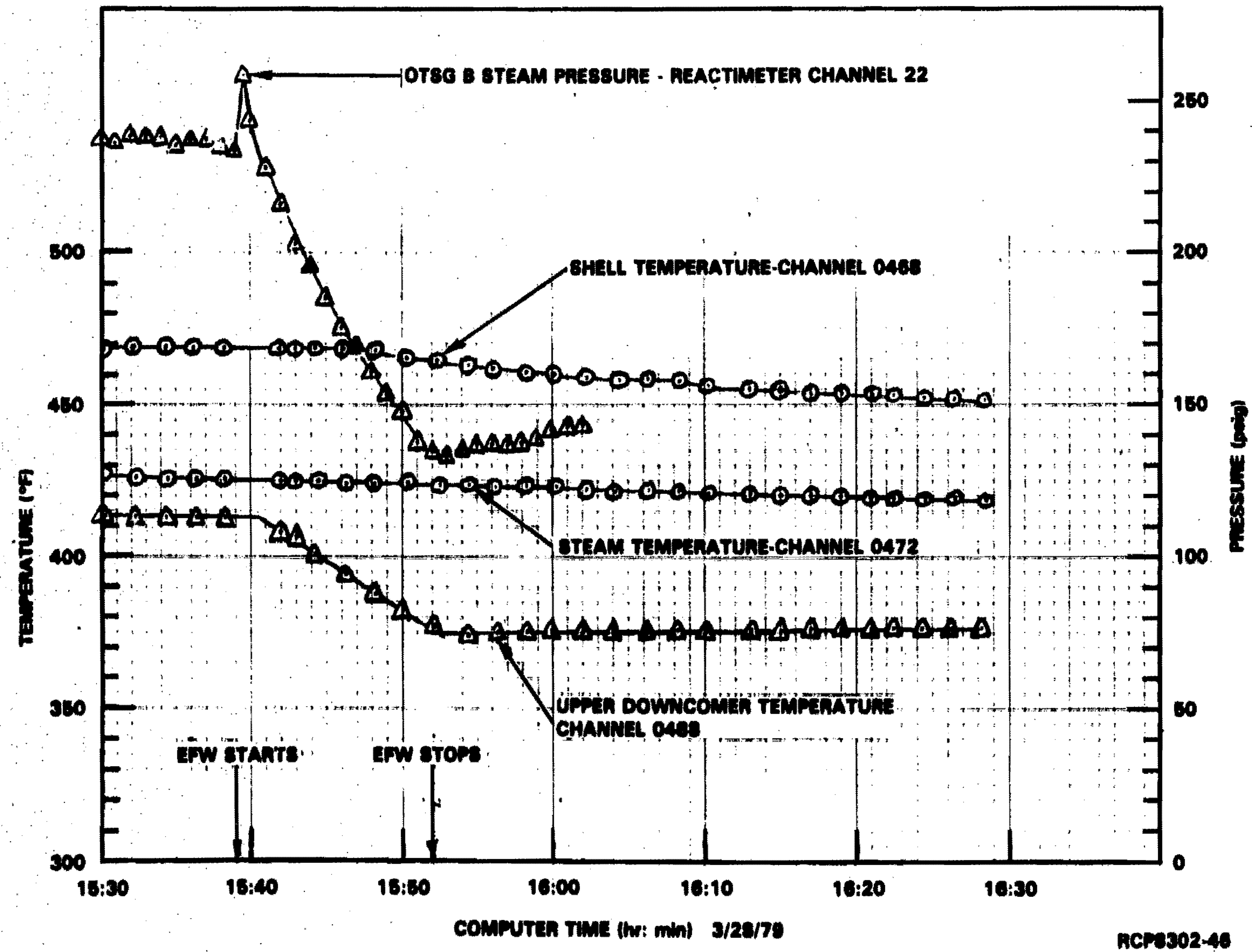

FIGURE 7. Thermal Response of OTSG B to Emergency Feed Water, 15:30 to 16:30.

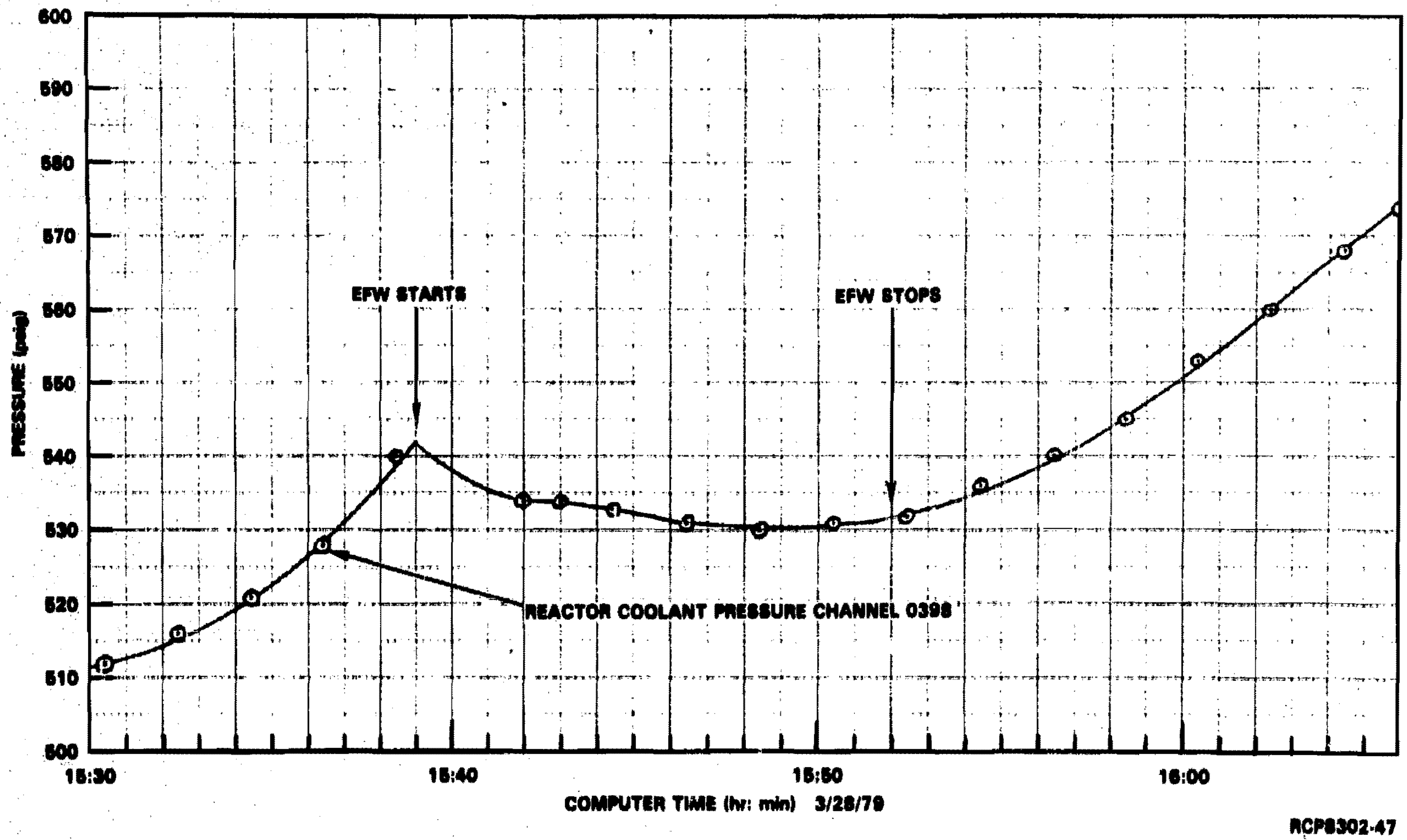

FIGURE 8. Apparent Response of RCS Pressure to Emergency Feed Water at 15:39.

A study of other system temperatures during the period of feed-water flow showed that few were affected significantly; steam pressure in OTSG A did not respond significantly.

The study of the feed-water transient at 15:39 supports the following conclusions and summary statements.

- The startup of EFW pumps is clearly indicated as a recorded event on the alarm printer. Therefore, it is expected that periods when emergency feed-water transients could occur would be identifiable from the alarm printer record.

- System response was as expected: (1) steam pressure in the affected generator first increased, then decreased, (2) liquid level increased roughly linearly with time, (3) primary steam condensation rate increased (as indicated by RCS pressure), (4) little or no response was noted in other systems that would not be expected to respond to EFW actuation in one steam generator.

- The 13:52 to 13:55 transients indicated by the reactimeter data were too fast to have been caused by an EFW transient. Further, the alarm printer did not indicate EFW operation at that time.

4.3 SYSTEM RESPONSE TO EFW OPERATION AT 07:43

Emergency feed-water pump EF-P-2A was started at 07:35:43 and the discharge pressure reached 1,573 psi 6 sec later. These timed events are recorded on the alarm printer record. An abrupt rise in steam pressure at ~07:43:24 indicates that water flow to OTSG A began at that time. Water level in the OTSG A steam generator began to increase at ~07:43:48. Both steam pressure and water level are plotted versus time on Figure 9.

Feed-water pump EF-P-2A was stopped at 08:42:51 as indicated by the alarm printer. Significant water flow, however, continued only until ~07:54; this is indicated by a peak in water level at that time. The rate of water level rise is significantly smaller than for the EFW operation at 15:39, indicating that the EFW flow rate was lower for the 07:43 EFW operation.

The EFW operation at 07:43 has the following characteristics that are similar to EFW operation at 15:39:

- Pump startup was recorded on the alarm printer.

- Feed-water flow started a number of minutes after pump startup, and was evidenced by an abrupt rise in steam pressure and an increase in water level.

- The other steam generator did not respond to the EFW flow.

Based on these similarities, the conclusion and summary statements that were made for the 15:39 EFW event also apply to the 07:43 event.

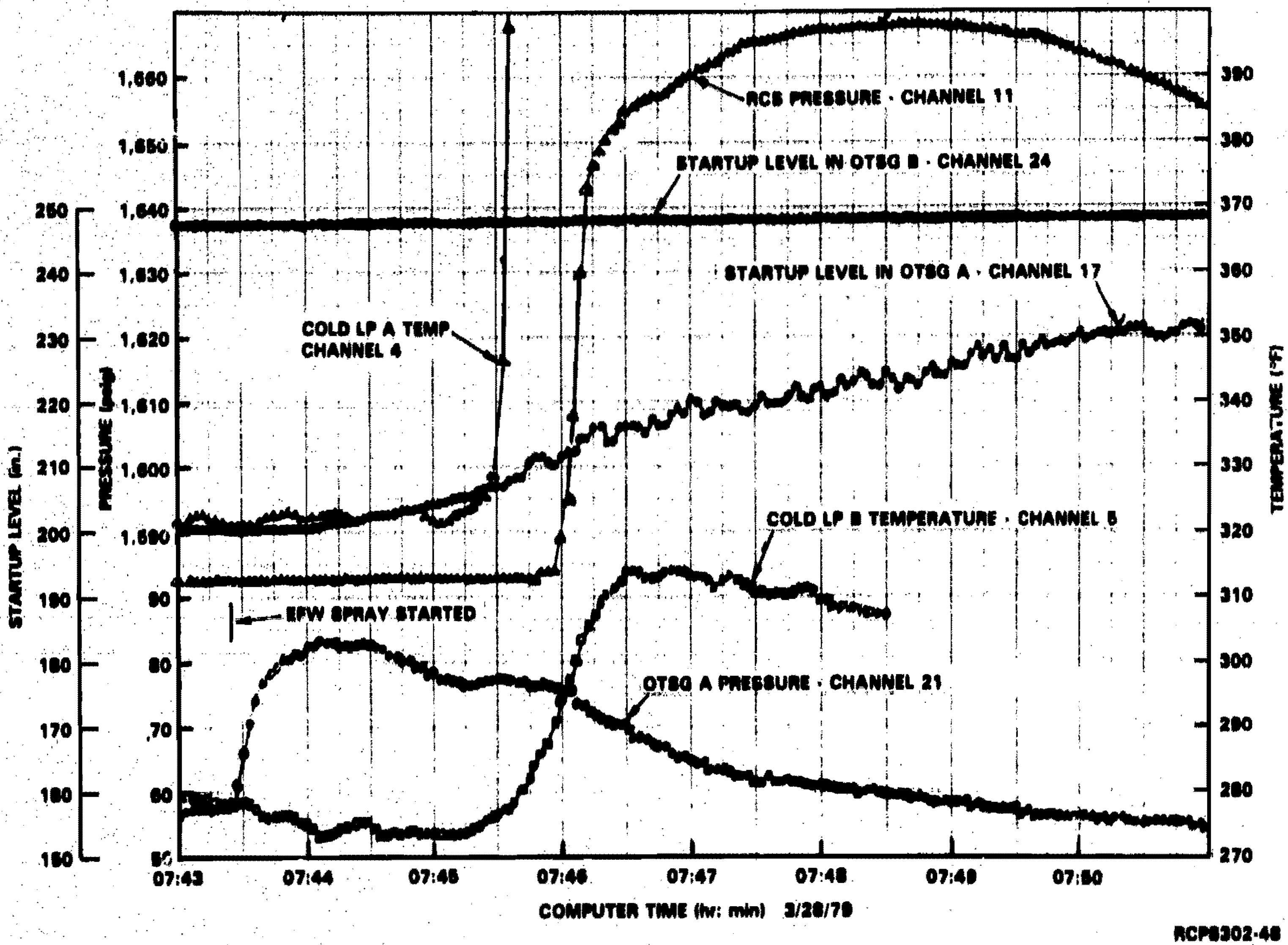

FIGURE 9. Response of System to EFW at 07:43 and Transient at 07:45

5.0 STEAM SPIKES IN THE RCS

A significant steam spike in the RCS occurred at ~06:55 as the result of the brief operation of reactor coolant pump RCP-2B. The alarm printer record shows that this pump tripped-on at 06:54:38 and reached full speed at 06:54:51. Operation of the pump caused water from the primary side of OTSG B to enter the bottom of the overheated core. Quenching of hot material and the formation of steam in the RCS caused a number of responses.

Changes in primary system parameters are illustrated on Figure 10. Reactor coolant system pressure increased from ~1,300 psi (as indicated by the wide range pressure meter) to ~2,040 psi. This increase took place over roughly a 5 min period, with most of the change occurring during the first 2 min. Pressurizer level also increased from ~300 in. to ~370 in. Thermodynamic calculations indicate that for the steam pressure changes experienced in this event, the observed increase in pressurizer level from the condensation of saturated steam is probable. Cold leg temperatures also experienced significant transients.

The indicated power range is also shown on Figure 10. This instrument, a gamma-ray sensor, experienced a sharp drop shortly after cold water was injected into the core. While a part of this drop may be attributable to water shielding, it is probable that a core rearrangement occurred when the overheated fuel and oxidized cladding were quenched with water. Discussions in Section 6.0 provide additional support for a core rearrangement at 06:55.

Response of the secondary side of the steam generators is shown on Figure 11. The data of Figure 11 show that OTSG B experienced a marked steam pressure transient and a corresponding reduction in secondary water level, whereas OTSG A exibited only minor responses. The response of OTSG B is as expected considering that the water level on the primary side was greatly reduced by the pump operation. The lowered water level caused primary steam to move into the steam generator and to contact the tubes below the secondary water level, thereby boiling the secondary water and generating large amounts of steam. In less than 1 min, OTSG B was again filled with gas (hydrogen). This prevented effective steam refluxing and caused an immediate reduction in secondary steam pressure even though the RCS pressure continued to increase. Steam generator A was not affected by the pump operation, and since it was gas filled, it was not greatly affected by the RCS pressure spike.

At ~07:45, an abrupt increase in the temperatures of the cold legs and in the RCS pressure occurred (see Fig 9). The initial RCS pressure of approximately 1,400 psi was not recorded by the narrow-range pressure indicator since it was off-scale low. The cold-leg temperature increases indicate that the cooling water flow reversed, carrying hot water from the reactor vessel to the temperature measuring location in the cold legs.

Of particular importance to the present study is the observed response of measured temperatures and pressures to the 07:45 steam spike. The following statements summarize that response.

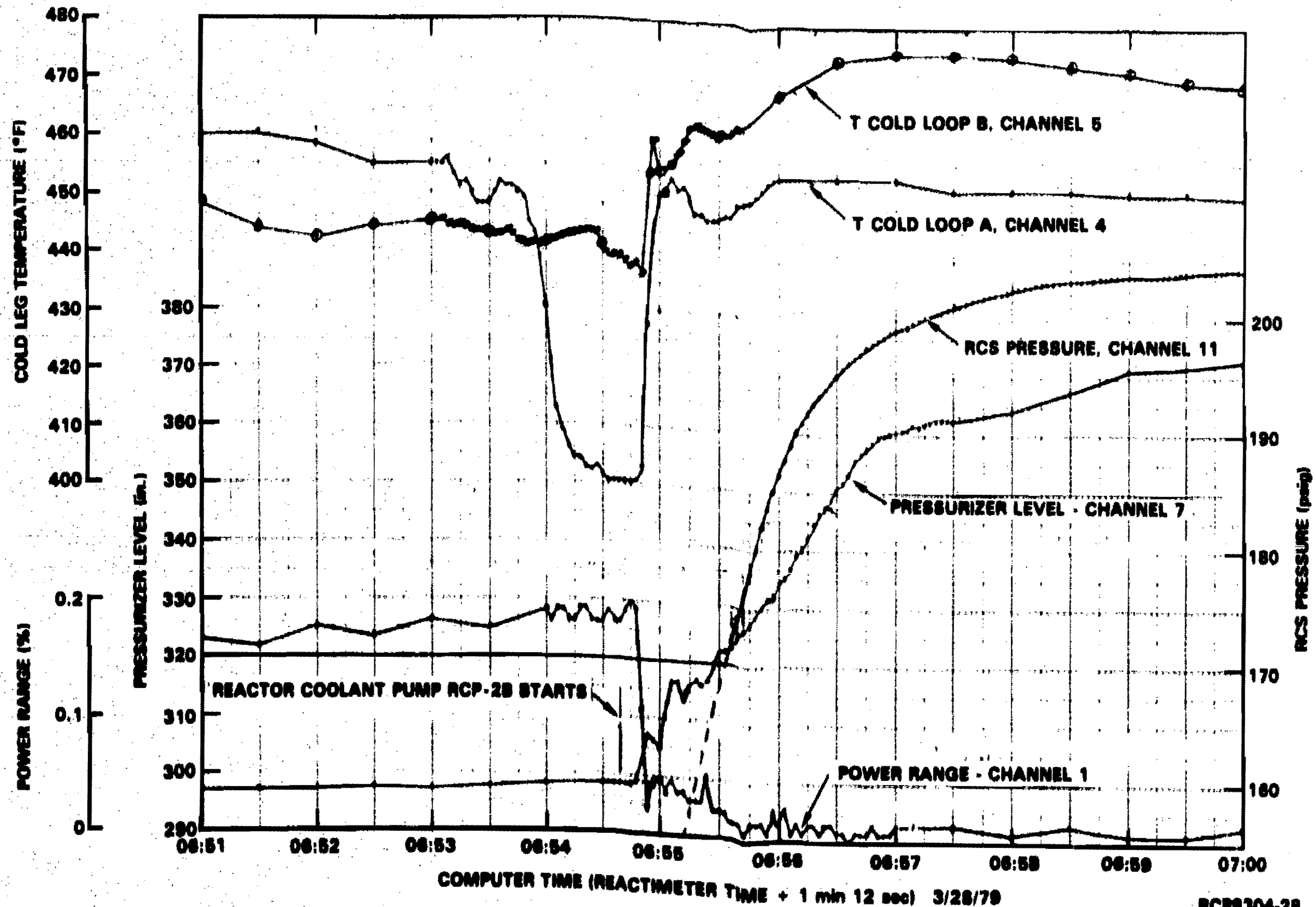

FIGURE 10. Response of Primary System to Operation of Coolant Pump RCP-28 at 06:55.

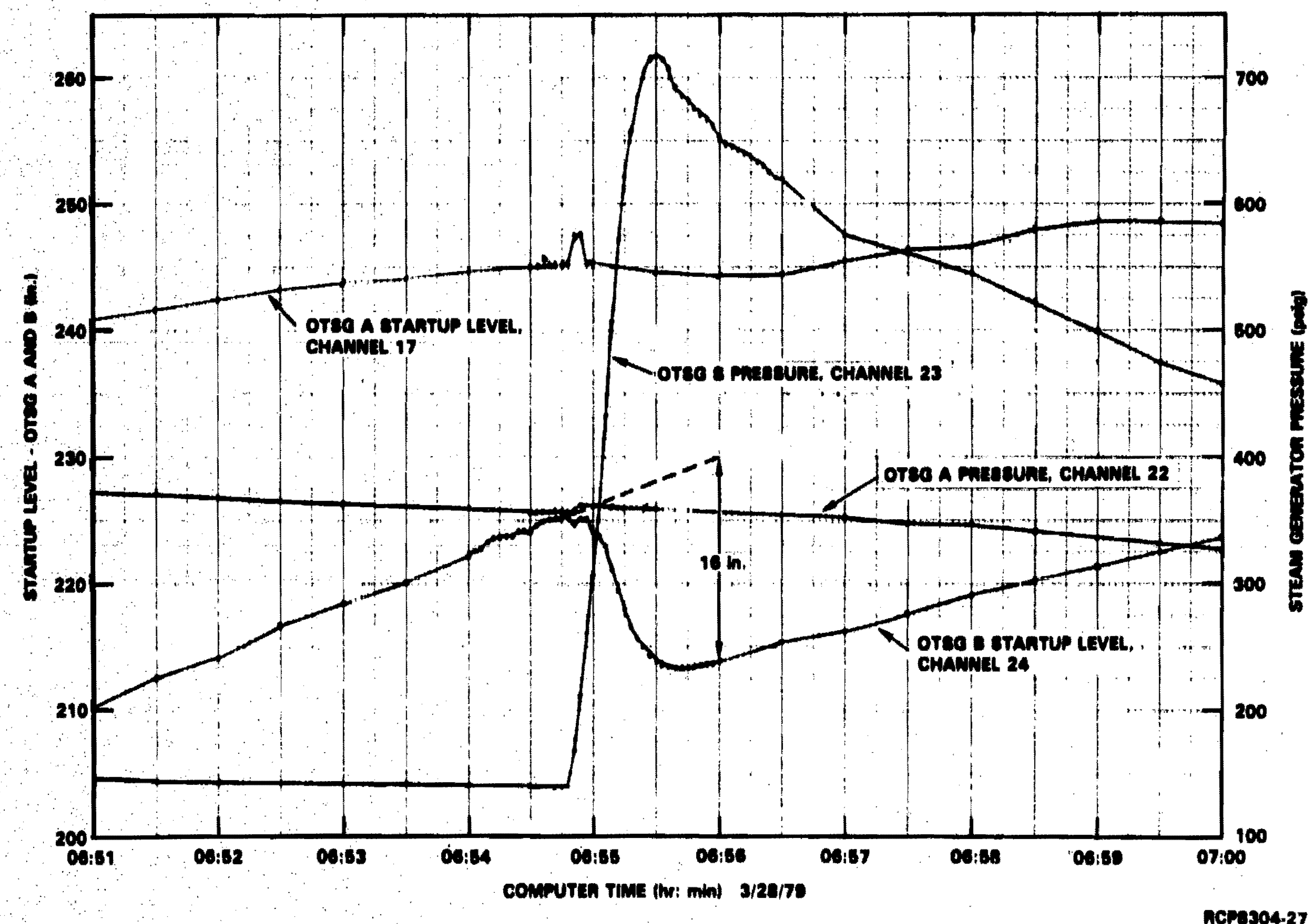

FIGURE 11. Response of Secondary Side to Steam Spike at 06:55.

1. Steam generators OTSG A and OTSB B did not respond significantly to the steam spike. This lack of response was because the generators were plugged with hydrogen gas and the reactor vessel was water filled to a level near the top of the hot leg nozzles. The opening of the PRV caused saturated steam/water to enter hot leg A which then had a minor positive influence on the OTSG A steam pressure.

2. Cold leg temperatures underwent transients that lasted for many minutes, as would be expected for the reverse flow situation.

In each of the steam transients evaluated, the duration of the steam bursts in the RCS were many minutes longer than the transient indicated at 13:52. The secondary steam pressure responses to the primary system pressure transients were small and slow. The very rapid change in OTSG B steam pressure at 6:55 was caused by the rapid removal of primary water from OTSG B and the resulting inrush of steam. The rate of pressure rise was approximately 2.5 psi/sec, which is the same as that indicated by the 13:52 reactimeter transient. The 6:55 transient is a dramatic demonstration of how rapidly a steam generator can respond to primary steam and also how effectively gas in the RCS prevents "reflux" heat transfer. It is concluded that the indicated transient at 13:52 was so fast that it could not have been caused by a steam burst in the gas-filled RCS.

6.0 OBSERVATIONS WHICH ASSIST IN ASSESSING REACTOR CORE CONFIGURATION

Much more data was studied and evaluated than is presented in this report, but the data is not pertinent to the purpose and conclusions of this report. Observations which may assist in assessing the current configuration of the reactor core follow.

6.1 PRESSURIZER AND REACTOR VESSEL WATER LEVELS

The pressurizer water-level instrumentation and level history, RCS pressure and temperature history, pressurizer spray and relief valve opening and closing, and pertinent piping configurations were studied and correlated. The conditions that allowed or caused the pressurizer to partially drain and to refill at three different times, and to remain near full during most of the first day of the accident, were determined or postulated. Figure 12 shows the pressurizer filling at two different times. It filled between 6:54 and 7:13 as a result of steam condensing in the pressurizer. The high pressure/temperature steam was generated as the hot core was quenched. The pressurizer filled again between 7:30 and 7:39 as a result of the pressurizer relief valve (PRV) being opened, and the water level in the RCS being above the level of the reactor coolant nozzles. This and other information indicate that the available data from the pressurizer level indicators are probably realistic; that after about 07:25 the reactor core was not uncovered; and that for most of the time after 07:25, the water level in the reactor vessel was near or above the reactor vessel coolant nozzles. One particular

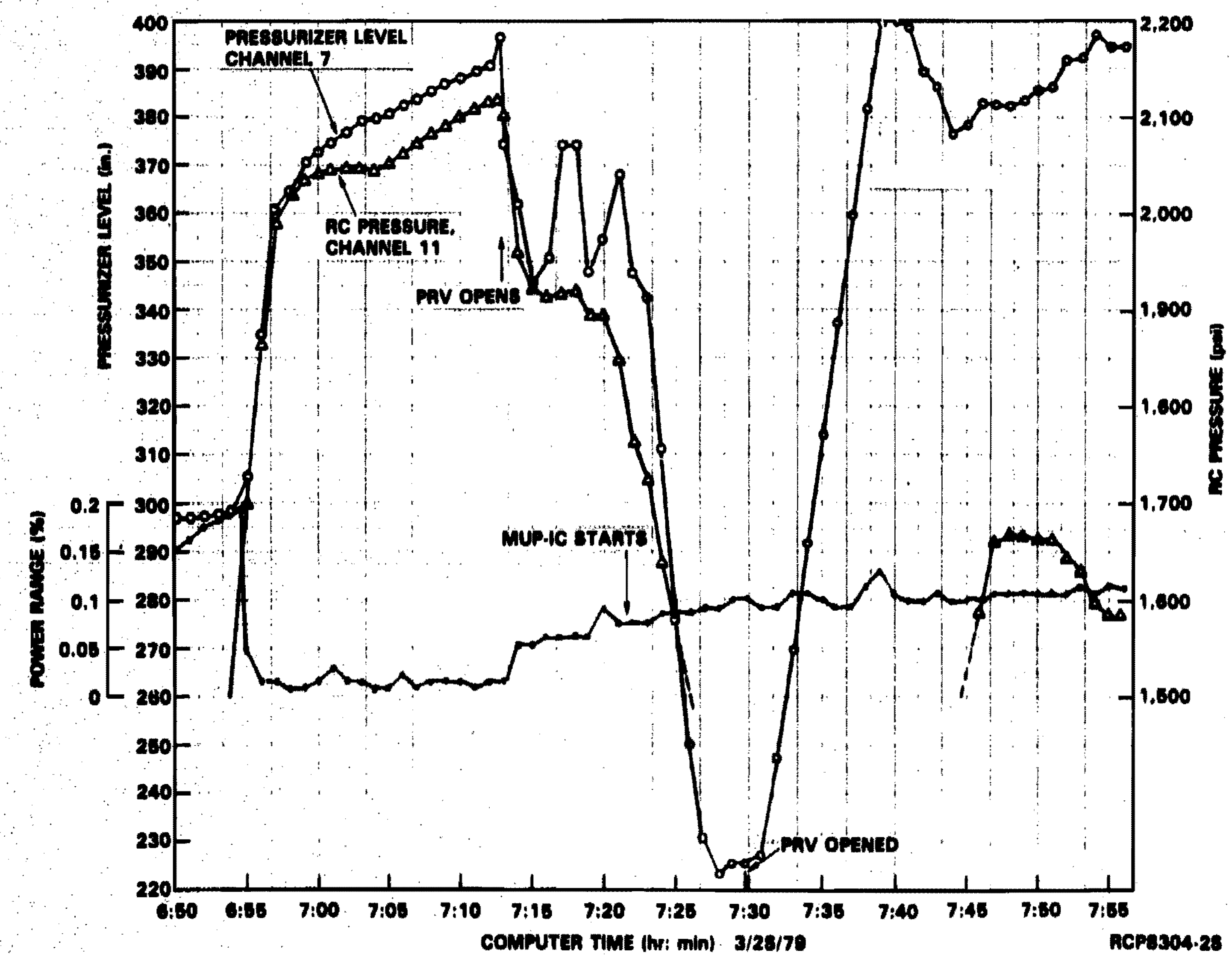

FIGURE 12. Selected Responses to the 6:54, 7:21, and 7:45 Transients.

time when a low water level in the reactor vessel has been suspected is at 07:45 (see Section 4.3). However, note from Figure 12 that the pressurizer level indicator showed that the pressurizer had just refilled at that time it appears that this would have been impossible if the water level in the reactor vessel had not been above the elevation of the reactor coolant nozzles.

6.2 REACTOR CORE HOT SPOTS

A study was made of the alarm printer record of the temperatures of the 52 instrumented fuel assemblies. There are a total of 177 assemblies in the core. The results are summarized in Figure 13. Figure 13 shows a cross section of the core and the 52 instrumented elements. A color bar at the top of each square shows whether the temperature was above or below 700°F at the end of the day. The last time during the day when the temperature changed above or below 700°F is also shown for each element. The fuel elements that cooled below 700°F during (or shortly after) four identified transients are shown by black bars. Further, the temperature measured between the hours of 08:00 and 09:30 is shown. The study shows that at least 17 of the 52 assemblies (33%) were still at temperatures hotter than 700°F in the areas of the thermocouple junctions (some newly formed) 23 hr after the start of the accident. This hot condition existed even though one reactor coolant pump was operating, coolant temperatures were ~250°F, and the RCS pressure was about 1,000 psi. Note that only the peripheral fuel elements cooled as water was added to the core at 6:55 and 7:21. This shows that coolant paths in the central section of the core were blocked at or prior to 6:55. Temperatures in many of the fuel channels, particularly those in the H through R half of the core, dropped shortly after the 7:45 transient and after the reactor coolant pump started. The "hot spots" and "cold channels" are widely scattered and continued to change all day. Therefore, the solidified melt slab (agglomerate) below the rubble in the core is probably highly irregular and not continuous. Also, even though the core was apparently water covered all day after about 7:25, the central section of the core was so badly plugged and overheated that the metal-water reaction probably continued to progress in and below the flow restriction. The steam surge which started at about 7:45 was probably caused by the rapid quenching of a steam bubble that had developed below the flow restriction. This damaged area should become evident as the reactor is defueled.

6.3 SUPERHEATING IN THE HOT LEGS

With the core covered after 7:25, as indicated in the preceeding sections, steam leaving the core would have been saturated, not superheated. However, temperatures in the hot legs and in the upper sections of the steam generators remained above the saturation temperature until late in the day. These temperatures responded to compression heating, expansion cooling, and steam cooling at times when the PRV was open and also when water was added to the RCS. However, during times when none of these changes were taking place, hot leg temperatures continued to rise. This is apparently a result of decay heat from fine core debris and condensed fission products distributed in the colandria above the reactor core, in the hot legs, and in the

LEGEND

- 07:47 •TIME WHEN TEMPERATURE DROPPED BELOW 700°F AND STAYED BELOW
- 599 TEMPERATURE (°F) BETWEEN 08:00 AND 09:30
- 13:50 †TIME WHEN TEMPERATURE ROSE ABOVE 700°F AND STAYED ABOVE FOR THE REST OF THE DAY
- OPEN THERMOCOUPLE - TEMPERATURES MAY HAVE REMAINED ABOVE 700°F ALL DAY

TIME WHEN TEMPERATURE DROPPED BELOW 700°F

- 06:55 TO 07:04 (RCP-2B STARTED)
- 07:21 TO 07:24 (MUP-1C STARTED)
- 07:47 TO 07:58 (CORE TRANSIENT)
- 19:33 AND 19:50 (RCP-1A STARTED)

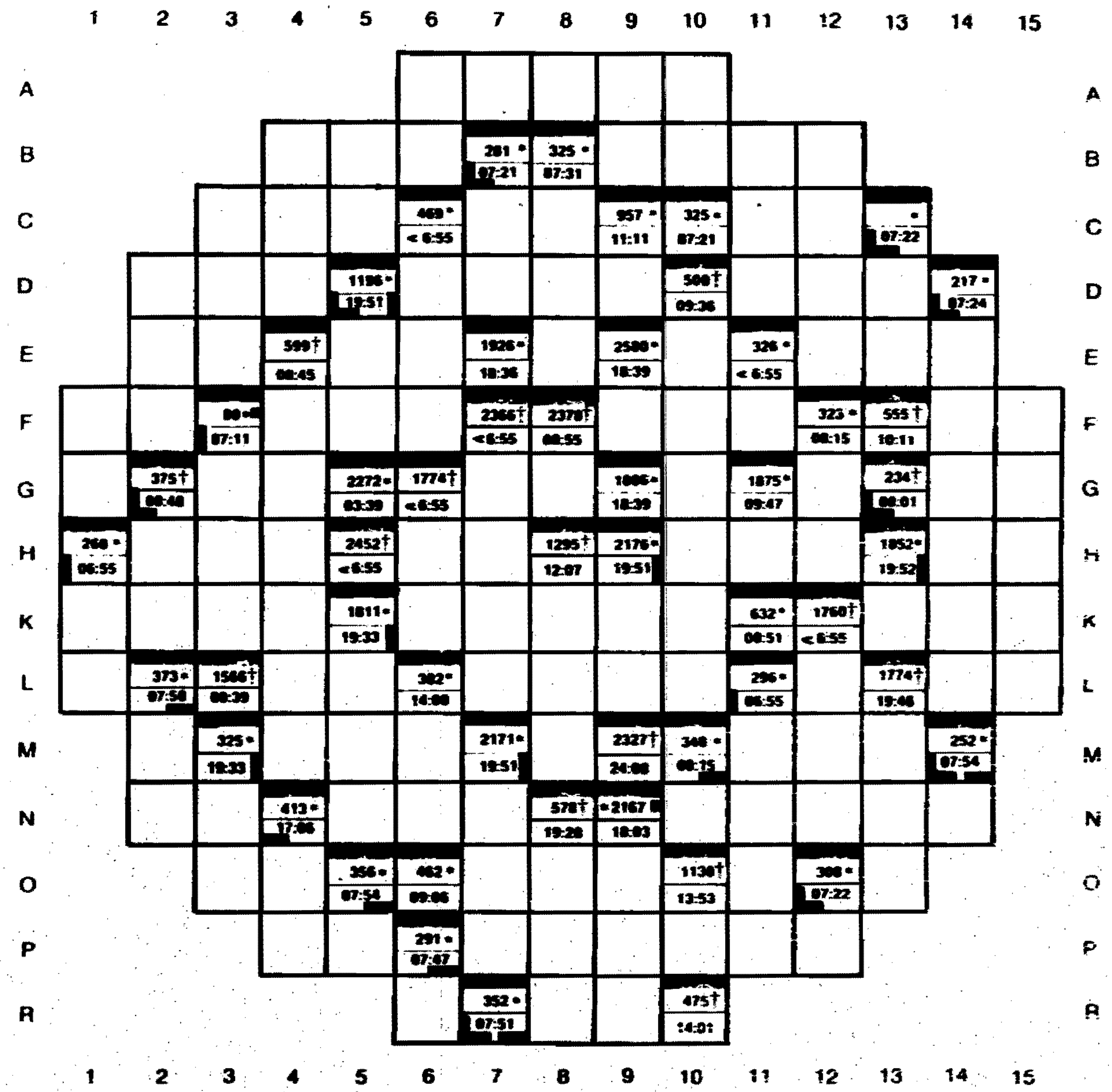

FIGURE 13. Map of Core Temperatures.

upper sections of the steam generators. Considerable quantities of fine core debris could have been ejected from the core as the water level rose rapidly in the highly overheated core at 6:54. The surrounding water would have trapped most of the particles released from the core during the 7:45 transient.