II C 1a

REACTOR PRIMARY SYSTEM BEHAVIOR FROM FOUR TO TWENTY HUNDRED

MARCH 28, 1979

Information on the behavior of a number of reactor primary system parameters has been gathered from several sources. The sources listed in Table A-1 are for those parameters found useful in the analysis of the behavior of the system during the accident. In several cases, the behavior had to be inferred from other data, such as for the opening and closing of the block valve upstream of the pilot operated relief valve (PORV), in which the position of the block valve had to be inferred from an analysis of the reactor building pressure strip chart for changes in slope and of the alarming and clearing of the alarm for the tailpipe temperatures of the PORV as shown on the alarm printer.

The time conventions used in this discussion are as follows: the time since the start of the accident is given in hours and inutes (i.e., 1 hr. 15 min.), assuming a time zero of 04:00:00/on March 28, 1979, while clock time is given on the 24 hour clock time basis and Eastern Standard Time (i.e., 05:33:22 is 5 hours, 33 minutes, and 22 seconds of a 24 hour day).

The abbreviations used in the status summaries and much of the text following are defined in Table A-2.

The plant parameters that seem to have some correlation to each other and to the total system behavior are plotted in Figures A-1 and A-2. The time scales of each of the plotted parameters have been matched to the best accuracy possible, but in no case should a time coincidence of better than about 3 minutes be expected for events or responses that actually were simultaneous.

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TABLE A-1

SOURCES OF DATA ON SYSTEM PARAMETERS

Parameter	Data Source: Reactimeter Log	Alarm Printer	Utility Typer	Hourly Computer Log	Strip Chart	Inferred from Alarm Printer	Inferred from Strip Charts
Hot and cold leg temperatures (OTSG)(T_H, T_C)	X		X	X	X		
Reactor system pressures (RCP)	X		X	X	X		
OTSG pressures and levels	x		x	x			
Pressurizer level (PZR)	X		X	X			
Pressurizer temperature (T_{pzr})			X				
Pressurizer surge line temperature (T_{surge})			X				
Source Range Monitor Counts (SRM)					X		
Pressurizer spray valve position	X						
Pressurizer vent valve operation							
Pressurizer Pilot Operated Relief Valve (PORV)						X-tail pipe temp alarms	X-building & reactor pressures X-building temperature

Parameter	Data Source: Reactimeter Log	Alarm Printer	Utility Typer	Hourly Computer Log	Strip Chart	Inferred from Alarm Printer	Inferred from Strip Charts
Pressurizer Block Valve (Block Valve)						X-tail pipe temp alarms	X-building & reactor pressure: X-building temperature
Makeup pump operation		X					
Reactor Coolant Pump Operation (RC-P1A, 2A, 1B, 2B)		X					
In-core Thermocouple temperatures (In-core Tlc)		X-plus one set of in-strument measurements					
Self-powered neutron detectors (SPND)		X					

TABLE A-2

DEFINITIONS AND ABBREVIATIONS .

RCP - reactor coolant pressure, reactor primary system pressure

RC pumps - reactor coolant pumps 1A and 2A, on OTSG A, 1B and 2B on OTSGB

MU-P - makeup pumps 1A, 1B, and 1C

PZR level - indicated level of water in the pressurizer in inches

Atmos. Dump Valve - the valve that allows the steam developed in either or both steam generators to be dumped to the atmosphere outside the reactor building

OTSG A - once-through steam generator A

OTSG B - once-through steam generator B

Surge Line Temperature - temperature indicated by a thermocouple strapped on the outer surface of the surge line between the OTSG A hot leg and the pressurizer.

PZR Temperature - temperature in °F measured in the interior of the pressurizer, just above the heaters, by a resistance thermometer called an RTD.

SRM - counts per second of the Source Range Monitor (SRM), sensing thermal neutrons from the reactor core, primarily from the peripheral bundles, and is, in this accident, mostly an indicator of water level in the downcomer in the reactor vessel. However, sudden changes in count levels may also be indicative of major changes in geometry of the core.

T_{HA} - temperature in °F of the hot leg between the reactor vessel and OTSG A, measured by an RTD about 54 inches below the tangent point of the curve at the top of the hot leg.

T_{HB} - hot leg temperature for OTSG B

T_{C1A}, T_{C2A}, T_{CB} - temperature in °F of cold legs 1A, and 2A of OTSG A, and either 1B or 2B (believed to be 2B) for OTSG B, measured a few inches below the inlet to the pertinent reactor coolant pump.

PORV - pilot operated relief valve on the pressurizer.

Block Valve - the gate valve positioned in the line between the pressurizer and the pilot operated relief valve (PORV or EMOV) that was stuck in the open position.

Engineered Safety System Actuation - a series of valve and pump actuations automatically performed when certain safety limits in the total reactor system are exceeded. It includes isolation of the reactor containment building, tripping of MU-P1B (unless the trip is bypassed), starting of MU-P1A and 1C, opening of the four "16" valves for maximum makeup flow of circa 1000 gpm total from two MU-P's, start of containment sprays, start of decay heat pumps, etc.

Steaming to Condenser or Condenser Vacuum - the normal mode of heat removal from the system is by steam production in the OTSG, steam passage through the generating turbines, and condensation in the steam condenser. The flow of steam to the turbines can be bypassed.

Pressurizer Spray Valve - the valve in the pressurizer spray line connecting the outlet side of RC-PA to the top of the pressurizer and used for "spraying down" the pressurizer in normal operation to decrease the system pressure.

Pressurizer Vent Valve - a separate venting valve located on the top of the pressurizer which can be used to reduce system pressure.

Time marks on most of the strip charts can be placed with no more than about $\pm$ 3 minutes accuracy, even though sudden changes in the recorded parameter were fitted to similar changes in other parameters to $\pm$ 1 minute or better. Since the accuracy of the chart drives is not known, and "fits" between neighboring "accurate event" time points may be several feet apart on charts having nominal speeds of 4-8 inches per hour and several inches apart on charts driven at nominally 1 inch per hour, the matching cannot be made better. In addition, the same signal from one sensing instrument recorded on two separate data acquisition systems was in one case displaced approximately 63 seconds at the start of the accident (03:59:33 and 04:00:36) and approximately two minutes at about 10 hours later (14:36:20(?) and 14:38:14(?) - both values extrapolated). This means that not only were the internal clocks of the two data acquisition systems indicating different times, but they also had different rates. These data are contained in channel 390, of the utility typer and channel MUX-2 of the reactimeter. They are plotted in Figure A-3 for times after 14:30. The signal calibrations and setting accuracies of the strip chart recording instruments on March 28, 1979 are not known and cannot be obtained at this time. The errors may be as large as 5 percent, as the wide-range chart for the reactor coolant system pressure records a pressure of 490-495 psig at 13 hrs 28 min. while channel 398 of the utility typer reports a pressure of 445 psig at that time. Also, the wide-range reactor coolant pressure chart indicates at pressure of 2200 psig at 10 hrs 13 min while the reactimeter reports a pressure of 2145 psig at that time.

Other parameters have been plotted and examined for correlation to system behavior, such as pressurizer heater trips and makeup tank levels, but no "fits" were obtained with the data presented in Figures A-1, A-2, and A-3. Thus, the data on these parameters are not reported in this section.

System Status at Successive Time Periods During the Accident

The following discussion of the sequence of events in the reactor primary system related to the damage of the core is broken into 10 time periods to allow an easier and more comprehensible presentation of important observations, events, and correlations that may enable a better understanding of the behavior of the system and the interactions occurring therein.

Period I

0 Hrs 0 Min to 1 Hr 0 Min

04:00 to 05:00, March 28, 1979

After the first few minutes of operator action and system response related to a turbine and reactor trip, the reactor primary system came to essentially a steady state condition at about 1100 psi system pressure, about 556°F coolant temperature, a relatively constant leak of mixed water and steam or steam only out the open pilot-operated relief valve (PORV), a slowly increasing build-up of voids (decreasing density of coolant) in the circulating water, and a relatively constant steam pressure in the secondary side of both steam generators (OTSG). Makeup pump 1A (MU-P1A) was operating with the flow probably throttled to a relatively low rate since the pressurizer level was high at 380 inches. Both OTSGs were filled only to about 4-5% on the operating range (this may be a minimum reading for the instrumentation, rather than a "zero" on the scale used).

INFERENCES AND COMMENTS

There is no evidence to indicate that the core was damaged at this time, even though there probably had to have been a short period of voiding in the core in the first 5 minutes when the level of the pressurizer increased quite rapidly from 158 to 400 inches (full). The OTSG-A was steaming to the condenser.

Period II

1 Hr 0 Min to 1 Hr 40 Min

05:00 to 05:40

RCP - 1100 psig ± 25
RC-P - B's off at 01:12,
A's off at 01:40
MU-P1A - on, throttled
L_{pzr} - 380 ± 10 inches
Atmos. Steam Dump Valve-
open

SRM - counts rising slowly, trace oscillating
T_H - 550°F, falling to 510°F.
T_C - 550°F, falling to 510°F.
T_{surge} - 518°F at 01:18
OTSG press. - B-980 to 160 psi
A-1000 to 780 psi
OTSG level - A-5%, B-5 to 15%

INFERENCES AND COMMENTS

During this period, the system remained relatively stable with the exception that the vibration of the reactor coolant pumps increased to the point that both "B" pumps were turned off at 1 hr 12 min to prevent damage, and both "A" pumps at 1 hr 40 min., the end of the period. The Source Range Monitor (SRM) readings became increasingly irregular as the average level increased slowly, indicating the increased amount of voids in the coolant in the downcomer. The condenser vacuum was lost at the beginning of the period, so that the atmospheric steam dump valve was automatically opened to permit heat removal from the steam generators (OTSG). Hot and cold leg temperatures were the same, and decreased about 40°F in the last 8 to 9 minutes of the period.

Period III

1 hr 40 min to 2 hr 20 min

05:40 to 06:20

RCP-1130 to 670 psi
RC-P - B - off
A - off
MU-P1A - on, throttled
L_{pzr} - 370-320 inches
Atmos. Steam Dump Valve-open
SRM - counts fell one decade in 1-2 minutes, regained in six, rose another decade in 15, leveled off.
OTSG A - pressure fell from 760 to 530 psi, level rose from 5% to 50% operating range (OR) and held

OTSG B - pressure 160 psi, level fell from 15% to 4% O.R. and held.
T_{HA} - rose from 525°F to 680°F, fell to 655°F.
T_{HB} - held at 532-536 for 16 min, fell to 513°F, rose to 570°F.
T_{CA} - fell from 510 to about 495°F.
T_{CB} - fell from 510 to 480°F.

INFERENCES AND COMMENTS

When the last reactor coolant pumps were shut off at 1 hr 40 min, the circulating mixture of water and steam separated. If reverse flow had been induced in OTSG B during the operation of RC-P1A and 2A, the coolant drained to the level of the impeller faces of the B pumps, leaving the primary side of OTSG B, at most, half full. The primary side of OTSG A had to have been nearly empty during pumping in the last few minutes, so that only the water in the cold legs would have drained back into the OTSG, leaving it possibly as much as 1/4 full. The drastic decrease in SRM counts indicated that the downcomer was full to about the top of the core immediately after the pumps were turned off. The steady rise in SRM counts over the next 20 or so minutes indicated that the level of the coolant water in the core dropped from about the top of the core to less than half full and leveled off. Various estimates give a level from 7 to 9 feet or more from the top of the core. The water boiled off in the core was condensed in the two OTSGs or vented out the pressurizer.

Since the B OTSG may have been filled to the level of the RC-P casing, the condensate in it may have been immediately returned to the core by dribbling through the horizontal section of the B cold legs. However, the level in the primary side of the A OTSG was considerably lower at the start of the period, so that much more condensate was required to fill it to the point of returning the condensate to the core. The core was then being cooled by refluxing and the loss out the pressurizer and the letdown line. As the coolant in the core was boiled off, the exposed fuel rods began to heat up, because they were cooled only by steam at very low flow rates. When the hottest part of the fuel rods reached a temperature of about 1500°F ±100°F, the cladding of the fuel rods ballooned and burst, and released the gases from the gap between the fuel pellets and the interior surface of the cladding. It is estimated in Section ___ that the hottest fuel rods in the center bundle (highest power) reached temperatures above 3500°F about 35 minutes after the pumps were turned off, and many others reached such temperatures in the minutes following that period. The hot and cold legs of the OTSGs were voided, and superheated steam was produced in the top of the core in the first few (5) minutes after the top of the core was uncovered. The period ended when the block valve for the PORV was closed and the loss of system pressure and coolant out the open pressurizer PORV was stopped. However, the loss of coolant out the letdown line continued.

Period IV

2 hr 20 min to 2 hr 54 min

06:20 to 06:54

RCP - 670 to 2130 psi
RC-P - all off
MU-P1A - on, throttled.
L_{pzr} - level at 300 inches until 2 hr. 54 min., then rose to 380 inches
Atmos. Steam Dump Valve- open
OTSG A - pressure fell steadily from 530 to 295 psi, level rose from 50% to 68% O.R.

OTSG B - pressure held at 190 ± 10 psi, then rose to 430 psi, level rose from 4% to 50% O.R.
SRM - counts slowly decreased until 2 hr. 54 min.
T_{HA} - rose from 640°F to 810°F over the period, then to 770°F.
T_{HB} - rose from 570°F to 800°F over the period.
T_{C1A} - fell from 495°F to 400°F and recovered to 430°F.
T_{C2A} - rose from 495°F to 500°F, fell to 450°F.
T_{CB} - fell from 480°F to 440°, rose to 470°F.
Block valve - closed at 2 hr 20 min.

INFERENCES AND COMMENTS

The leak out the pressurizer PORV stopped when the PORV block valve was closed at 2 hr 20 min. If the wide range reactor system pressure recording strip chart can be indexed to an accuracy of ± 3 minutes, it appears that the decrease in pressure in the reactor primary system stopped abruptly at 2 hrs 12 min and began a relatively rapid increase at least 4 minutes before the block valve was closed at 2 hr 20 min (8 ± 3 minutes for the strip chart, ± 1 minute on block valve closure). The pressure ramp shows two definite inflection points, at 2 hrs 25 min and 630 psig indicated and at 2 hrs 54 min and 1300 psig indicated. The first occurred very close in time to "jogs" in the hot and cold leg temperatures for the OTSG, and the second appears to be in time coincidence with the starting of RC-P2B at 2 hrs 54 min.

The rapidly increasing hot leg temperatures for both OTSGs can occur only if superheated steam is present in the hot legs, and they are voided of water. A particular point to be observed is that the pressurizer level indicator showed a rise in the pressurizer of 74 inches in 5 minutes. The question to be answered is "if this is a true indication, where did the water come from?" This change in level is equivalent to 237 cubic feet of water. (3.2 ft^3 volume per inch of level in the pressurizer). It is thought that the major oxidation damage to the Zircaloy cladding occurred during this period, and that parts of the fuel rods reached temperatures over 3600°F. This is discussed in detail in the section ____ on "Damage to the Core."

Period V

2 hrs 54 min to 3 hrs 12 min

06:54 to 07:12

RCP - 1300 to 2100 to 2140 psi.
RC-P - 2B on.
MU-P1A on, throttled.
L_{pzr} - 330 to 380 to 360 inches.
Atmos. Dump Valve - closed at 3 hr.
OTSG A - pressure fell steadily, level from 68 to 60%.
OTSG B - pressure from 300 to 410 to 380 psi, level from 40 to 60%.
Steaming to condenser.

SRM - count rate dropped one decade in seconds and then rose to recover most of the drop by 3 hrs 12 min.
T_{HA} - 810 to 770 to 780°F.
T_{C1A} - 400 to 480°F.
T_{C2A} - 460 to 430 to 540°F.
T_{CB} - 440° to 470° to 445°F.
Block valve - closed.
PZR Spray Valve - Open.

INFERENCES AND COMMENTS

The insurge of water from the OTSG B when the RC-P2B was turned on probably produced considerable structural damage to the very hot and embrittled fuel rods, producing fractured and shattered fuel rods in the upper portion of the core as well as a very rapid pressure rise by the large quantity of steam formed when the water hit the hot fuel rods. Observations reported by the operators indicate that there was water flow through the RC-P2B pump for only a very short time (a few minutes at most), since the vibrations and low power in the pump were again observed very shortly after it was started. If the OTSG B had been half full at the time the 2B pump was started, less than about 1000 ft^3 of water would have been pumped into the core. Since this is about 1 1/3 times the free volume of the core and downcomer over the length of the core, the core should have been covered and quenched, even if it had not been partially filled at the time the pump was started. The behavior of the temperatures of the hot legs indicate that they had not been covered, and that superheated steam was being emitted by the core all through this period. It thus seems likely that OTSG B had been not more than 1/4 full at the time the RC-P2B pump was started or that pump suction was lost very quickly. The very sharp increase in reactor coolant pressure starting at 2 hrs 54 min. was probably due to a very large burst of steam produced when the water from the OTSG B hit the very hot core.

Period VI
3 hrs 12 min to 5 hr 18 min
07:12 to 09:18

RCP - 2150 to 2000 psi, held 2000 to 1500 psi then fell to 1240 psi with 3 intermediate periods of increase.
RC-P - off
MU-P1A on until 4 hrs 21 min, then locked out 1B and 1C on at 4 hrs 27 min, 1B and 1C on HPI for 6 minutes at 3 hr 18 min and 3 hrs. 57 min, 1C on normal at 3 hrs 24 min for 12 min, on normal for 17 min at 4 hrs 3 min.
L_{PZR} - 350" to 230" in 13 min then to 400" in 20 min, remained above 390" for remainder of period.
Atmos. Dump Valve - opened at 4 hrs 30 min.
Condenser - steaming until vacuum lost at 4 hrs 30 min.
OTSG A - fell from 200 to 40 psi at 3 hrs. 42 min, decreased to 20 psi at 4 hrs 30 min, rose to 80 psi at end of period. Level ranged from 60% to 48%.
OTSG B - pressure fell slowly from 380 to 320 psi, level rose from 58% to 65% and held.
SRM - count rate dropped one decade abruptly at 3 hrs 18 min when makeup pumps on HPI, fell steadily about 1/3 decade to 3 hrs 43 min, and slowly decreased to about $2x10^3$ cps at end of period.

T_{HA} - 780°F at start of period, to 790°F at 3 hrs 18 min, fell to 700°F at 3 hrs 28 min, rose to 760°F at 3 hrs 42 min, fell to 690°F at 4 hrs, then to 700°F ± 10° for rest of period.
T_{HB} - very similar behavior but with peak temperatures to about 820°F and ending at 745°F.
T_{C2A} - from 450°F at start to 485°F in 6 min, to 320°F at 3 hrs 43 min to 510°F at 3 hrs. 48 min 450°F at 4 hrs to 190°F at end of period.
T_{C1A} - 480°F at start to 320° at 3 hrs 41 min to 440 °F at 3 hrs 45 min to 310°F at 4 hrs 4 min to 350°F at 4 hrs 13 min to 300°F at 4 hrs 30 min and lost on chart until 10 hrs 30 min.
T_{CB} - fell from 445°F at start to 220°F at end of period with several oscillations of 20-40° with sharp changes in slope.
Block Valve - open and closed several times in period.
PZR Spray Valve - open from 3 hrs 42 min to 4 hrs 6 min.

INFERENCES AND COMMENTS

At the start of the period, the reactor coolant pressure held constant at 2000 psi. When the makeup pumps were turned to high pressure injection (HPI) of about 500 gpm from each pump, the influx of water apparently chilled the downcomer region, and the pressure dropped very rapidly to 1500 psi and leveled off as MU-P1C was changed from HPI to normal flow. When HPI by

MU-P1A was stopped in another six minutes, the pressure in the system rose to about 1560 psi at 3 hrs 39 min.. When the block valve was opened at 3 hrs 42 min and the pressurizer spray valve was opened at the same time, the system pressure decreased to 1480 psi and then increased to 1710 psi very quickly. When HPI was again initiated at 3 hrs 56 min on both MU-P1A and 1C, the pressure again began a decrease to 1510 psi at 4 hrs 6 min. The pressurizer spray valve was closed at 4 hrs 6 min and the block valve opened for about six minutes between 4 hrs 12 min and 4 hrs 18 min. The block valve was opened again at 4 hrs 36 min and remained open for the rest of the period. The RC pressure decreased rapidly when the MU-P1C was started again at about 4 hrs 27 min to 1310 psi, and rose to 1390 psi at 4 hrs 54 min, even though the block valve was opened at 4 hrs 36 min. The pressure then decreased to about 1250 psi at the end of the period, when the block valve was again closed to repressurize the system.

MU-P1A was shut down and "locked out" for the remainder of the accident because the operators were having considerable difficulty in keeping it in operation. It tripped off and had to be restarted many times during the first four hours of the accident period. When it could not be restarted after the last trip at 4 hrs 21 min, the operators "locked it out" to prevent its actuation during activation of the ES system, and replaced it with MU-P1B. However, there was a period of about six minutes when no makeup coolant was flowing into the system. It should be noted that the major responses of the system seem to occur with the operation of MU-P1C, the block valve, and the pressurizer spray valve. Operation of MU-P1A or 1B seemed to have little or no effect on either system temperatures or the pressure.

The large and sudden increases in the cold leg temperatures of OTSG A were coincident with the opening of the block valve and the pressurizer spray valve. The sharp but relatively small increase in the SRM signal was also coincident with the opening of these two valves.

Period VII
5 hrs 18 min to 7 hrs 39 min
09:18 to 11:39

RCP - increased from 1240 psi to 2150 psi, cycled between about 2150 psi and 1850-1900 psi with about 2 min pressure increase, about 1 min pressure decrease.

RC-pumps - off

MU-P - both 1A and 1C operating, with various degrees of throttling

L_{PZR} - constant at 400 inches

Atmos. Dump Valve - open

OTSG A - pressure dropped slowly from 80 psi to less than 20 psi at 7 hrs, remained below 20 psi for rest of period. Level at 48% until refill started at 5 hrs 54 min, reaching 100% operating range at 7 hrs.

OTSG B - pressure dropped slowly from 320 psi to 290 psi at end of period

T surge - 310°F at 5 hrs 15 min.

T_{PZR} - 345-350°F in last half hour of period.

SRM - count rate dropped slowly from beginning to end of period, with one small "bump" occurring between 6 hrs 45 min and 7 hrs 6 min.

T_{HA} - increased from 690°F at start to 735°-740° at 6 hrs and remained at 735 ± 5°F to end of period.

T_{HB} - paralleled T_{HA} exactly but at 50°F higher temperature.

T_{CA} - temperature record appears that of 2A. Rose from 190°F at start to 220°F at 5 hrs 45 min and for rest of period.

T_{CB} - dropped from 220°F at start to 210°F at 5 hrs 30 min, then gradually fell to 185°F at end.

Block Valve - cycled open and closed to bleed off pressure to prevent opening of safety valves.

PZR Spray Valve - closed.

INFERENCES AND COMMENTS

In this period, the operators planned to collapse the "steam bubbles" in the hot legs of the OTSGs by pressurizing, so that the system could ultimately be put into the natural circulation mode of cooling. Since the system pressure was increasing to the level at which the safety valves would be opened, the block valve was manipulated to keep the pressure as high as possible without "lifting the safeties." The system would increase in pressure from about 1900 psi to 2070-2100 psi in two to two and one-half minutes (114 to 150 seconds), and decrease from about 2100 to about 1980 psi in about 70-75 seconds. This procedure was continued for more than 1-1/2 hours. The operators feared that the block valve would fail, and they would be left with no control of system pressure. The decision was then made to depressurize to less than 400 psi so that the system could receive coolant from the core flood tanks. During this period, the OTSG A was filled to 100% of the operating range (O.R.), but OTSG B was left isolated and at 60% O.R. Nothing of consequence can be noted in any of the system parameters, with the exception of a small "bump" in the SRM counts at 6 hrs 45 min to 7 hrs. The block valve was opened at 7 hrs 39 min to remain open for more than 1 1/2 hrs. Pressurizer temperatures were requested from the plant computer by the operators for the first time during the accident. The temperature of 350°F in the pressurizer indicated a steam pressure of 135 psia existed in the vapor space, and the remainder of the pressure was due to a noncondensable gas, presumably hydrogen.

Period VIII
7 hrs 39 min to 10 hrs 21 min
11:39 to 14:21

RCP - dropped from 2050 psi at the start to 1580 in 4 minutes, to 1460 at 7 hrs 51 min. to 1120 psi at 7 hrs. 57 min, then at an exponential decay to about 500 psi at 9 hrs. Held 500-490 psi to 9 hrs. 48 min (utility typer gives 440-450 psi), rose to 550 psi at 10 hrs 5 min and fell to 520 psi at end of period.

RC pumps - off

MU-P1C-on until 9 hrs 6 min, 1A on for entire period. HPI on both at 9 hr 50 min.

L_{pzr} - 395-400 inches.

Atmos. Dump Valve - closed at 9 hrs 15 min, no heat removal from system except letdown flow and when pressurizer valves open.

OTSG A - pressure near atmospheric to 10 hr 18 min to 40 psi at 10 hr 21 min, level constant at 95% O.R.

OTSG B - pressure decreased slowly from 280 psi to 250 psi except for small increase to 310 psi at 7 hrs. 54 min. Level constant at 60-65% except for short time rise to 66% at 7 hrs 54 min.

T-surge - requested twice by operators, 310°F at 8 hrs and 330°F at 8 hrs 18 min. Surge line temperature is not reported again.

T_{pzr} - pressurizer temperatures were requested several times by the operators (circa 350°F), and then were reported as "trend data" in Operators Group C Summary afterwards. Temperature held at 350°F with slight increase with time until 9 hr 30 min when an increasing rate of temperature began. At 10 hr 21 min, the pressurizer temperature was within a few degrees of or equal to saturation temperature for the system. It did not rise higher than saturation temperature for the system for the remainder of the accident.

SRM - count rate increased slowly from start of period until 9 hrs 48 min, showed a small sharp increase and decrease, then returned to the same curve as before, and remained constant for rest of the period.

T_{HA} - dropped sharply from 730°F at start of period to 700°F in 6 min at 7 hrs 51 min, then very slowly increased to 715°F at 9 hrs 51 min, dropped sharply to 660°F at 10 hrs, and dropped slowly to 650°F at end of period with one excursion to 630°F and return in 9 minutes.

T_{HB} - paralleled T_{HA} behavior except 50 to 80°F higher, ending period at 725-730°F.

T_{CA} - fell slowly from 220°F at 7 hr 39 min to 160°F at 9 hr, rose to 220°F at 10 hr and held for remainder of period.

T_{CB} - fell gradually from 185°F at 7 hrs 39 min to 150°F at 9 hr and held.

Block valve - closed at 9 hr 15 min for 6 min, closed at 9 hrs 32 min for 17 min, opened from 9 hrs 49 min through end of period.

PZR Spray Valve - opened at 8 hr closed at 9 hrs, opened at 10 hrs.

Pressurizer Vent Valve - opened at 7 hrs 54 min, closed at 9 hrs 9 min.

Engineered Safety System Actuation - at 9 hr 50 min on high building pressure, decay heat pumps started, reactor building isolated, reactor building sprays started, both makeup pumps on HPI for one minute. Reactor building spray pumps stopped at 9 hr 56 min.

Reactor Building Pressure - spiked to 28 psi at 9 hr 50 min, observable on strip chart recording reactor building pressure, and as an inverse pressure on the OTSG steam pressures (since the pressure sensors use building pressure as the reference pressure).

INFERENCES AND COMMENTS

During this period, the operators were attempting to "blow the system down" to get to a pressure low enough to allow the system to be opened to the core flood tanks. The pressure leveled off at about 450-460 psi without dropping below that for about 45 minutes. Also, the system pressure remained at 450-460 psi for almost 30 minutes even with the PORV block valve closed for the time period around 9 hrs 30 min. At about the time of the ES actuation, when the makeup pumps went onto HPI, the system pressure started rising slowly to about 500 psi at about 10 hrs 6 min and then slowly dropped down to about 460 psi at the end of the period.

The reactor building pressure pulse recorded at 9 hrs 50 min on the reactor building strip chart was thought by the operators to be a spurious signal or "electrical noise", both then and later. However, the inverse of the pressure pulse can be seen by plotting the steam pressures of the OTSGs for the time period 9 hrs 45 min-9 hrs 55 min as shown in Figure A-4, with the data taken from the reactimeter tabulation at 3 second intervals. The pressure sensors of the OTSGs use the reactor building pressure as the reference pressure. The data show that the pressure rose to a peak over a 9-second time interval, decayed to nearby the original pressure in about 100 seconds, and then dropped suddenly to below the original pressure. This was the "hydrogen burn" to be discussed later.

While the reactor core was "floating" on the core flood tanks from 8 hrs 30 min to 9 hrs 12 min, the response of the core flood tank pressure showed that only a small amount of water could have entered the primary system. For a part of the time, the pressure in core flood tanks was rising as indicated by the pertinent strip chart.

One of the more important observations of the period may be that the temperature of the pressurizer rose to the saturation temperature for the system (based on the system pressure) for the first time since they separated at about the time the primary coolant pumps were turned off at 1 hr 40 min.

The "blip" in the SRM count rate strip chart should be noted, but no cause be assigned to it, and it is not quite in time coincidence with the reactor building pressure spike at 9 hrs 50 min, though it may be within the timing coincidence error of the several strip charts and data acquisition systems.

Period IX
10 hrs 21 min to 13 hrs 15 min
14:21 to 17:15

RCP-fell from about 460 psi at the start of the period to 420-420 psi with a drop to 409 psi for one-two minutes at 10 hrs 36 min, and rose back to 420-425 psi. Slow rise starting at 11 hrs 10 min, leveled off at 650-660 psi at 12 hr 39 min forrest of period.

RC-pumps - off

MU-P 1B on for entire period, throttled. 1C on for 6 min at 10 hr 30 min, on for 10 min at 11 hr 18 min and for 3 min at 11 hr 33 min. No HPI in the period.

L_{pzr} - 380-400 inches from 10 hrs 21 min to 11 hrs 3 min, dropped very rapidly to 175 inches at 11 hrs 18 min, held 175 inches to 11 hr 33 min, rose steadily to 400 inches at 12 hrs 30 min, dropped to 390-380 psi for rest of period.

Atmos. Steam Dump Valve - closed

Condenser Vacuum - pumps started at about 13 hrs

OTSG A - pressure rose from 40 psi at start of period to 80 psi at 10 hrs 45 min with abrupt change at 10 hrs 30 min. Dropped slowly to about 50 psi at 11 hr 45 min, then rose at increasing rate to 160 psi at end of period. Level constant at 97-98% operating range.

OTSG B - pressure dropped slowly from 250 psi at start t [illegible]0 at 11 hr 30 min, then rapidly to 150 psi at 11 hr 54 min and held at 150 for rest of period. Level dropped from 60% at start to 57% at 11 hr 33 min and rose rapidly to 96% O.R. at 12 hrs, holding 96% for rest of period.

T_{pzr} - rose slowly or level, within a few degrees of saturation temperature for pressure of the system throughout the period.

SRM - count rate increased only very slightly during the entire period.

T_{HA} - dropped very rapidly from 650°F at 10 hr 21 min to 500°F at 10 hrs 32 min, rose very rapidly to 570°F at 10 hr 40 min, fell to 460°F at 11 hrs 6 min, started rapid rise at 11 hrs 15 min to 560°F at 11 hrs 23 min, rose slowly to 590°F and held to 12 hr 33 min, dropped at increasing rate to circa T_{sat} = 500°F at 13 hr 6 min and held at T_{sat} for rest of the period.

T_{HB} - rose slowly from 7[illegible]°F at start to 755°F at 12 hr 33 min, dropped very rapidly to 630°F at 12 hr 42 min, rose to 710°F at 13 hrs 3 min and to 715°F at 13 hrs 15 min.

T_{CA} - two curves observable, 1A and 2A cold legs, behavior is different. T_{C2A} preceded T_{C1A}, and reached higher temperatures. T_{C2A} reached 440°F at 11 hr 21 min, T_{C1A} reached max of 400°F at same time. Both were about 360 ± 10°F at 11 hr 36 min and both reached T_{sat} circa 480°F at 12 hrs 15 min. Both held at T_{sat} for remainder of the period.

T_{CB} - fell from 150°F at start of period to 125°F at 11 hr, held 125°F to 11 hr 45 min, rose rapidly to peak at 170°F at 12 hr, fell slowly to 145°F at end of period.

Block Valve - closed at 11 hr 9 min, opened at 12 hr 30 min, closed at 12 hr 40 min, opened at 12 hr 52 min, and open for rest of period.

PZR Vent Valve - opened at 12 hrs 45 min, closed at 12 hr 57 min.

PZR Spray Valve - closed at 11 hr 57 min.

INFERENCES AND COMMENTS

During this period of time, the reactor primary system displayed some of the symptoms of thermal-hydraulic behavior expected of a system having condensible vapor in it. The hot and cold legs of OTSG A showed a behavior indicating that there was again steam flow and condensation in the "A" steam generator, and the response of the steam generator pressure was in accordance. However, the pressurizer level dropped 230 inches between 10 hrs 54 min and 11 hrs 18 min, equivalent to a volume displacement of 736 ft^3. The system pressure showed a rise of less than 100 psi and it was delayed relative to the drop in the pressurizer level.

Though the OTSG A hot leg temperature reached the saturation temperature for the system (based on system pressure) for a short time, it rose to about 100°F superheat again for much of the remainder of the period, and again fell to saturation temperature at the end of the period. The cold leg temperatures for OTSG A reached the system saturation temperature in the middle of the period and held it for the rest of the period. The OTSG A hot and cold leg temperatures, the pressurizer temperature and the system saturation temperature were the same for the first time since the reactor coolant pumps were turned off.

Period X
13 hrs 15 min to 16 hrs
17:15 to 20:00

RCP - the pressure was constant at 600 psi until 13 hr 25 min, rose to 2350 psi at 14 hr 48 min fell to 2320 psi at 15 hr 35 min, dropped almost instantly to 1500 psi, rose rapidly back to 2120 psi, and fell to 1350 psi at 15 hr 50 min.

RC-pumps - pump 1A "burped" at 15 hr 33 min to check operation, started again at 15 hr 50 min to run for many days.

MU-P1B on for entire period, 1C started at 13 hrs 21 min, throttled at 14 hr 41 min, stopped at 14 hr 43 min, run for 7 min at 15 hr 32 min and 11 min at 15 hr 45 min.

L_{pzr} - dropped rapidly from 390 inches at 13 hr 18 min, rose slowly to 290 inches at 13 hr 54 min and rapidly to 400 inches at 14 hr 21 min.

Atmos. Steam Dump Valve - closed.

Steaming to Condenser - started at 14 hrs for OTSG A.

OTSG A - pressure dropped slowly from 160 psi at start of period to nearly zero at 15 hr, rose from circa 10 psi at 15 hr 30 min to 70 psi at 15 hr 42 min, and fell to 20 psi at 16 hr. Level-constant at 95-96% except for "dip" to 88% at 13 hr 51 min.

OTSG B - pressure constant at 150 psi to 15 hr 30 min, dropped to 40-50 psi at 16 hr.

T_{pzr} - increased slowly or held steady for entire period-no decrease. Started at saturation temperature for the system pressure, but did not increase with it as system pressure rose to 2350 psi. Reached 520°F at 16 hrs.

SRM - count rate was steady or showed only very slight increase over the entire period except for "bump" at 14 hr 30 min.

T_{HA} - rose from circa 500°F at start of period to 590°F at 14 hrs 45 min, and fell slowly to 575°F at 15 hr 33 min, dropped sharply to 420°F when RC-P1A "burped", rose again to 525°F at 15 hr 50 min and dropped to 365°F when RC-P1A started.

T_{HB} - responded as T_{HA} but 150-200°F higher.

T_{CA} - T_{C2A} started rapid drop from 490°F at 13 hr 30 min to 315°F at 13 hr 45 min to 280°F at 14 hrs 9 min, held to circa 14 hr 45 min, and started to rise to 415°F at 15 hr 33 min, dropped to 330°F, and ended period at 365°F. T_{C1A} behaved much the same way after falling slowly from 490°F at 13 hr 30 min to 425°F at 14 hrs.

T_{CB} - Held 145°F from start of period to 14 hrs, rose rapidly to 210°F at 14 hr 15 min and slowly to 230°F at 14 hr 39 min, fell to 210°F at 15 hr 33 min, and rose to 365°F at 15 hr 50 min.

Block Valve - closed at 13 hr 24 min, remained closed for rest of period.

PZR Vent and Spray Valves - closed.

INFERENCES AND COMMENTS

At the start of this period, the operators decided to repressurize and to increase makeup flow to collapse the "steam bubbles" thought to exist in the hot legs of the steam generators. The pressurizer temperature continued its slow rise, but did not follow the saturation temperature based on the system pressure. This indicated that the system was not being pressurized by a steam bubble in the pressurizer but by makeup flow and other factors. The system pressure showed a very rapid increase at 14 hr 35 min from 1400 psi to 1900 psi in less than two minutes, and the rate of increase then slowed, indicating a massive input of heat to the system vapor phase had occurred. The reactor coolant pump 1A was successfully "burped" at 15 hr 33 min and flow, motor amperage and pump vibration were found to be acceptable. The motor had to cool for 15 minutes before it could be started again.

RC-P1A was started again at 15 hrs 50 min to run continuously for several days. The hot and cold leg temperatures almost immediately merged to within about 5°F of the same value, or 365°F, though the "quenching" of the hot leg of OTSG B appeared to be delayed by one to two minutes. The system pressure dropped very rapidly to 1350 psi, rose to 1400 psi in about 8 minutes, and then fell smoothly and slowly to 1000 psi at 18 hrs (22:00). MU-P1B continued to run. The system was "stable", the core was being cooled by flowing water, and OTSG A was steaming to the condenser.

IIC1b

CORE DAMAGE AT THREE HOURS

There appears to be no evidence that the fuel rods of the reactor core had been damaged before the reactor coolant pumps RC-P1A and 2A were turned off at 1 hr. 40 min of accident time. No reactor building radiation alarms had been activated, the temperatures indicated by the incore thermocouples had not been hot enough to be recorded by the alarm printer as going off-scale, and none of the self-powered neutron detectors (SPNDs) had been shown by the alarm printer as being "bad".

Shortly after the hydrogen "burn" in the reactor containment building was accepted as a real occurrence, calculations indicated that the amount of hydrogen present in the containment at the time of the "burn," and left in the primary system as either a hydrogen gas bubble or as dissolved hydrogen in the reactor coolant, was equivalent to 35-40% of the Zircaloy present in the core having been converted to zirconium dioxide. This was the first measure of damage to the core, and applied to the amount of damage to the core at the time of the "burn."

Later, a simple set of calculations of the heat-up of the fuel rods were made (Ref.) to produce bounding estimates of core damage using simplified assumptions, constant specific heats, constant rate of boil-off, a constant heat loss fraction, and manual and graphical solutions. This estimate gave a total of 25-30% of the Zircaloy cladding (fueled length only) converted to zirconium oxide at 3 hours, and estimated the depth of damage to reach as much as 6 feet from the top in the central region of the core. In the worst case estimate, a large part of the cladding above the 6-foot level had reacted with the ZrO_2 to form a liquid euthetic phase at 3455°F. This flowed into the gap between the fuel and the cladding to react with the UO_2 fuel, partially dissolving it, and formed a liquid phase of Zr-U-O termed "liquified fuel." At most, about 10% of the fuel present in the upper half of the core was thought to have formed "liquified fuel." In the least damage case (decay heat only, no heat of oxidation of the Zircaloy added for heat-up), it was estimated that the depth of embrittlement of the Zircaloy cladding was essentially unchanged from the worst case, but the extent of formation of "liquified fuel" was confined to only a few feet of the highest power central fuel assembly. No attempt was made to continue the calculations beyond 3 hours of accident time because of a lack of information of sufficient accuracy to permit calculations to be made. As the damage estimate of 25-30% conversion was made at 3 hours, there is no significant disagreement with the estimate of 35-40% at 9.9 hours.

THE TMI BOIL CODE

A code called TMI BOIL (Ref) has been written recently to calculate more precisely the time-temperature relationship for the fuel rods in TMI-2, using relatively precise analytical expressions, no simplifying assumptions, and parametric treatment of several of the system variables. The code has been written so that the accident "scenario" can be varied over wide ranges and the calculations fitted into the scenario parametrically. Specifically, the code does not require an exact knowledge of the makeup and letdown flows, but it does require a stated rate of change (as one of te parameters) of level of coolant in the core. In addition, it

(a) calculates the steam production rate as a function of the length of the fuel rod submerged in coolant, the system pressure, the time in the scenario and the rate of coolant level change.

(b) calculates the specific heat of the fuel rod as a function of temperature.

(c) calculates analytically the heat of oxidation at each node, time, and temperature.

(d) calculates the radiative heat transfer coefficient and adds it to the conductive heat transfer coefficient.

(e) uses parametraically the conduction heat transfer coefficients, the final depth of boil-off, the rate of boil-off, the assembly power (radial peaking factor times a fixed axial power profile), and the presence of "chilling" rods (such as control and poison rods).

(f) calculates the total steam produced in each time increment, and the surplus of steam exiting the fuel subchannels for each time increment.

(g) reports the axial node in 1 inch increments, the elapsed time in minutes, the fuel (cladding) temperature in °F, the steam temperature in °F, the steam flow rate in lbs./hr., the thickness of Zircaloy metal left in the wall (not converted to oxide), and the ratio of the oxidation heat to the decay heat at each node.

(h) calculates the total number of gram moles of hydrogen produced.

(i) cuts off oxidation heat of Zircaloy-steam reaction at 3600°F, assumes molten material is formed between oxide and metal which leaves the node, and reports thereafter, at that node, the thickness of metal remaining when the node reached 3600°F (3600°F assures melting of the alpha Zircaloy whether or not the eutectic with the Zircaloy oxide is formed).

(j) assumes the time as zero at the time the top of the fuel stack is first uncovered.

The code TMI BOIL has been used (REF) to calculate the time-temperature relationship for the fuel rods using the following set of parametric values:

(1) boildown to 7, 8, or 9 feet from the top of the fuel stack.

(2) a time of boiloff of 20 minutes for most scenarios, but 30 or 33 minutes for certain scenarios.

(3) radial peaking factors in the assemblies of 1.467, 1.2, 1.0, and 0.622 (a spread reasonably presentative of the core)--power in the assemblies at each node is obtained by multiplying the radial peaking factor (rpf) by the axial power profile value at each node.

(4) conduction heat transfer coefficients over a range of representative of low steam flow rates (3 and 10)

(5) the boildown and refill scenario proposed by EPRI in NSAC-1

RESULTS

The principal results are presented in summary form in Tables B-1 and B-2, and in Figure B-1 to B-16. The effects of varying the parameters can be seen in Tables B-1 and B-2 on the time and location of bursting of the fuel rods at 1500°F (assuming that bursting occurs at 1500°F), the time and location of the first formation of the Zr-U-O liquid phase (assumed to have formed at 3600°F) and of the maximum depth of formation from the top of te fuel stack, and of the time and location of the maximum temperature reached in te fuel rod. The Figures B-1 to B-16 show the time-temperature curves for one-foot nodes on the fuel rods over a time interval of 80 minutes.

Since the time zero for the TMIBOIL calculation is the time at which the top of the fuel stack was first uncovered, the time scale can be moved any place along the clock time axis (or accident time axis) as needed to examine the effects of modifying an accident scenario.

DISCUSSION

In general overview of the TMIBOIL calculational results, and the known "facts" of the TMI-2 accident sequence, it is believed that boiloff of 7 feet produces too little damage (considering the amount of hydrogen produced and the amount of core inventory of radioactivity released), and the boiloff to 9 feet produces too much. It appears that the boiloff to 8 feet ±1/2 foot produces damage values not inconsistent with known levels of hydrogen, radioactivity release, maximum temperatures, etc.

Tables B-1 and B-2 present most of the same data in somewhat different order, so that comparisons of several parameters are made easier.

In Table B-1, the effect of changing the power in the assembly on the significant points can be seen by inter-comparing lines 1 through 4. As the power in the assembly increases, the location of the burst (defined as the first position on the rod to reach 1500°F) can be seen to rise towards the top of the fuel rod, and the time to burst decreases from 29 to 20.6 minutes. Also, the location of the first formation of liquid phase (3600°F) rises, and the time to formation decreases. It may seem surprising that the maximum depth of liquid phase formation decreases with increasing power in the assembly, but this is due to the increasing rate of steam production with increased power.

The effects of changing the maximum depth of boiloff can be seen by inter-comparison of lines 1-4 with 8-11 and 12-15. Note that for the 7 foot level of boiloff, the peak temperature on the fuel rod increases from 3042 to 2600°F with decreasing assembly power from rpf 1.467 to rpf 0.622 and only the lowest power assemblies on the periphery of the core reach temperatures high enough to form the Zr-U-O "liquified fuel" phase.

It is important to note that the ranges of time to burst and the location of the burst do not vary very much for te different levels or rates of boildown, being in the neighborhood of 4 inches of range of level across the core, and with differences of 7-10 minutes between first and last bursts for each of the boildown levels. Changes in most of the parameters do not have a large effect on time vs temperature, or on burst time and elevation. The largest effects are observed in the influence of level of boildown on the first and maximum levels of liquification and on the peak temperature reached. The calculations for the 9 foot level of boildown (from the top of the core) indicate that more than three-fourths of the core had exceeded temperatures of 5200°F (melting point of UO_2) for a depth of about 2 feet at an elapsed time of 78 minutes from start of core uncovering.

The estimate of damage present in the core at 3 hours depends on the time assumed for the first uncovering of the core. The best evidence available for determining this time is shown in Figure A-4, where the temperatures of the hot and cold legs of the two OTSGs, and the levels of coolant on the secondary side are plotted as functions of clock time.

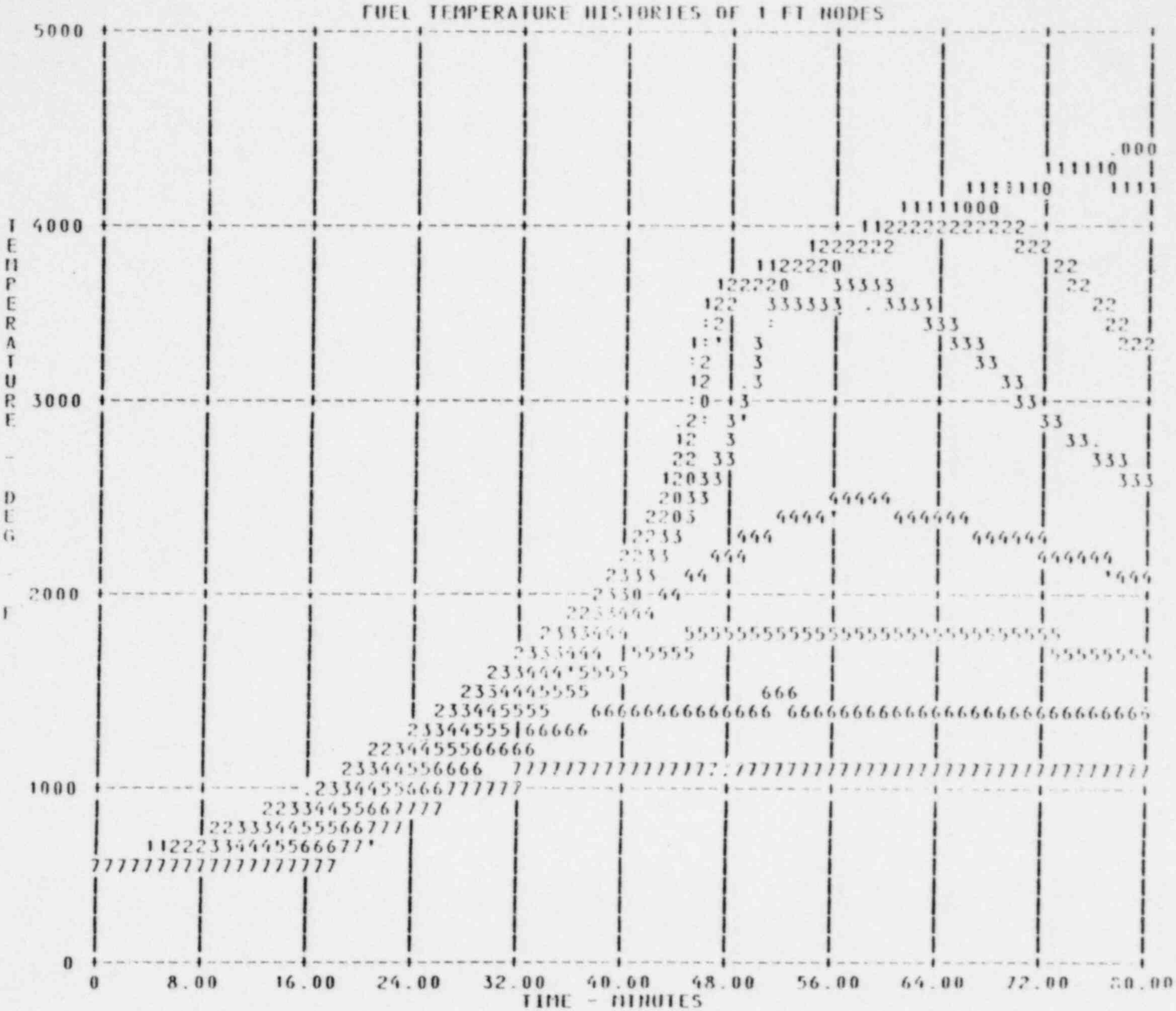
There are two possible interpretations of these data. When the prior level in OTSG B is considered (shown in Figure A-1), it can be argued that the first break in the curves for the hot leg temperatures of both steam generators at 05:42 hours (1 hr. 42 min of accident time) indicates that superheated steam was detected in both A and B steam generators at the top of the hot legs. The continued rise and subsequent decrease in temperature for OTSG B could indicate flow of superheated steam into a condenser that was saturating in heat. The reversion of OTSG A hot leg temperature to a decreasing temperature-time relationship, paralleling the previous curves, and the succeeding curves for the cold legs, could indicate that OTSG A could absorb no significant amount of heat (it was already known to have been "boiled dry") until its refilling had begun. Thus, it can be argued that the core was first uncovered at 102 minutes. It can be stated with certainty that the core had been uncovered no later than 05:52 (1 hr. 52 min. or 112 minutes of accident time), since the OTSG A hot leg temperature began at that time a rise that did not stop (other than for two short inversions) until a temperature of about 820°F was reached at 06:52 (2 hr. 52 min or 172 minutes accident time). These two times, 102 and 112 minutes of accident time, allow placement of the TMIBOIL zero time and time at which the RC-P2B pump was stated, so that bounds for the amount of damage to the core at 3 hours can be estimated. It must be assumed that at least a small amount of water was pumped by RC-P2B into the core to reverse the heatup of the fuel rods, even if for only a few minutes.

If it is then assumed that the TMIBOIL calculations for boiloff to 8 feet in 20 minutes apply (the best estimate from the amount of hydrogen and radioactivity released, the SRM data, etc), the PORV block valve was closed at 06:20 (2 hrs. 20 min accident time), and the RC-P2B was started at 06:54 (2 hrs 54 min accident time), then the amount of core damage at 07:00 (3 hrs accident time) can be founded.

With these assumptions, it can be estimated that the great majority of the fuel rods burst befor the block valve was closed at 140 minutes, first "liquified fuel" formed between 7 minutes before and 3 minutes after the block valve was closed, the maximum depth o formation of "liquified fuel" in the hot assembly occurred by the time the block valve was closed or within 10 minuted thereafter, and between 20 and 30 minutes later in the lowest power assembly, and that the maximum temperature reached in the fuel rods was circa 4410°F for a "middle power" assembly at between 20 and 30 minutes after the block valve was closed, and between 5 and 15 minutes before the RC-P2B was started. Additionally, peak temperatures of circa 4410°F were reached in more than two-thirds or more of the core by the time the RC-P2B was started.

The maximum penetration of the formation of "liquified fuel" was to about 41 inches in the lowest powered assemblies on the periphery of the core, and to 36 inches in the center of the core (the steam production rates decreased greatly as the periphery of the core was approached, and thus the cooling capability of the steam flow).

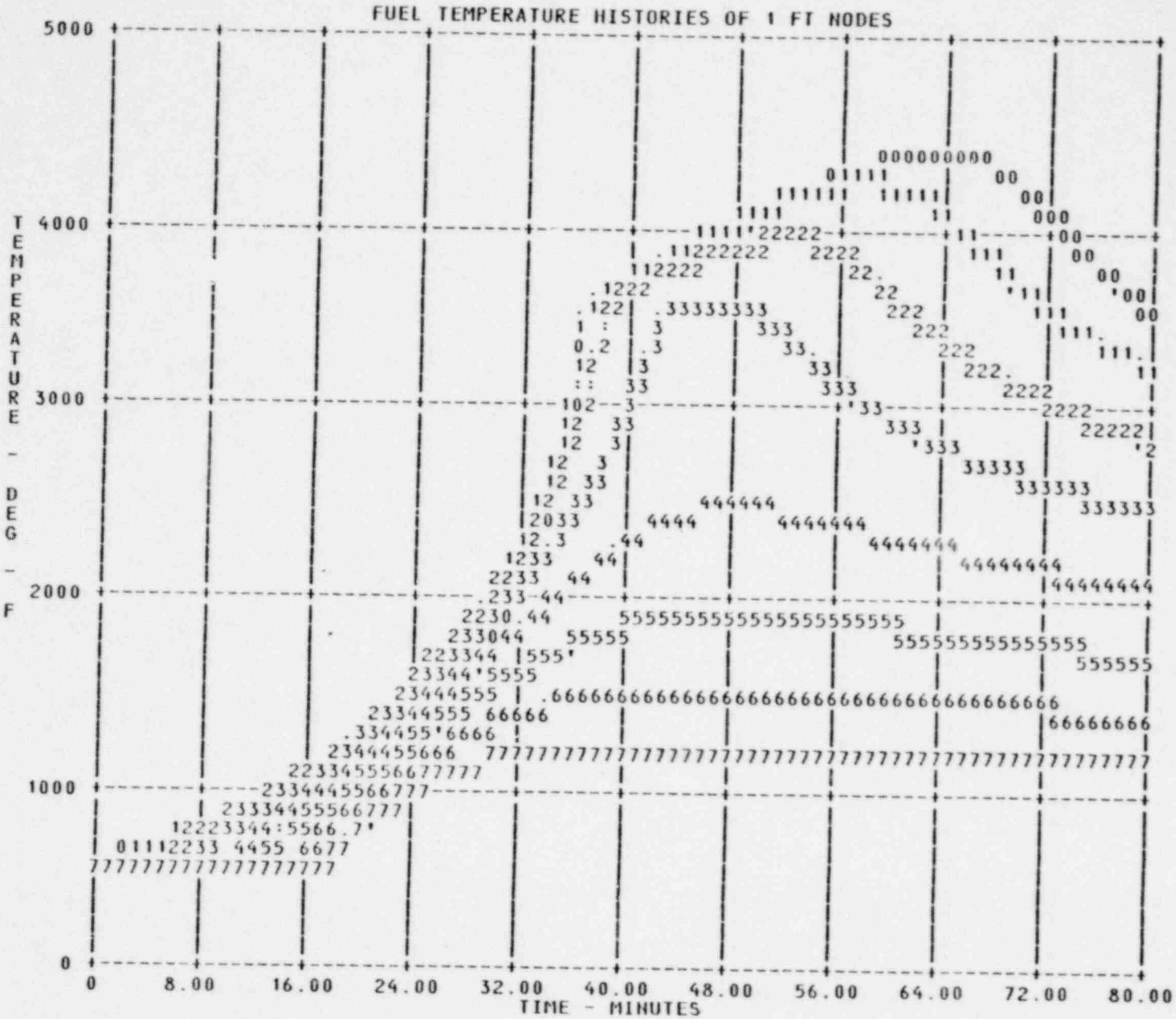
Additionally, it is estimated that the amount of Zircaloy converted to oxide at 07:00 (3 hrs accident time) is between and of the Zircaloy in the fueled part of the core, and between and of the total Zircaloy in the core, including plenum regions and end plugs. These amounts are equivalent to and lb.-moles of hydrogen, respectively. Since there is evidence that more hydrogen was produced at a later time, this is <u>not</u> to be taken as an estimate of the amount of hydrogen present in the containment and the primary system at 13:54 (9.9 hours accident time), the time of the "hydrogen burn" in the containment.



JOB 9753
10/18/79
Depth- 8 ft
Time- 20 min
h_e 3
rpf 0.622

Figure B-1

6



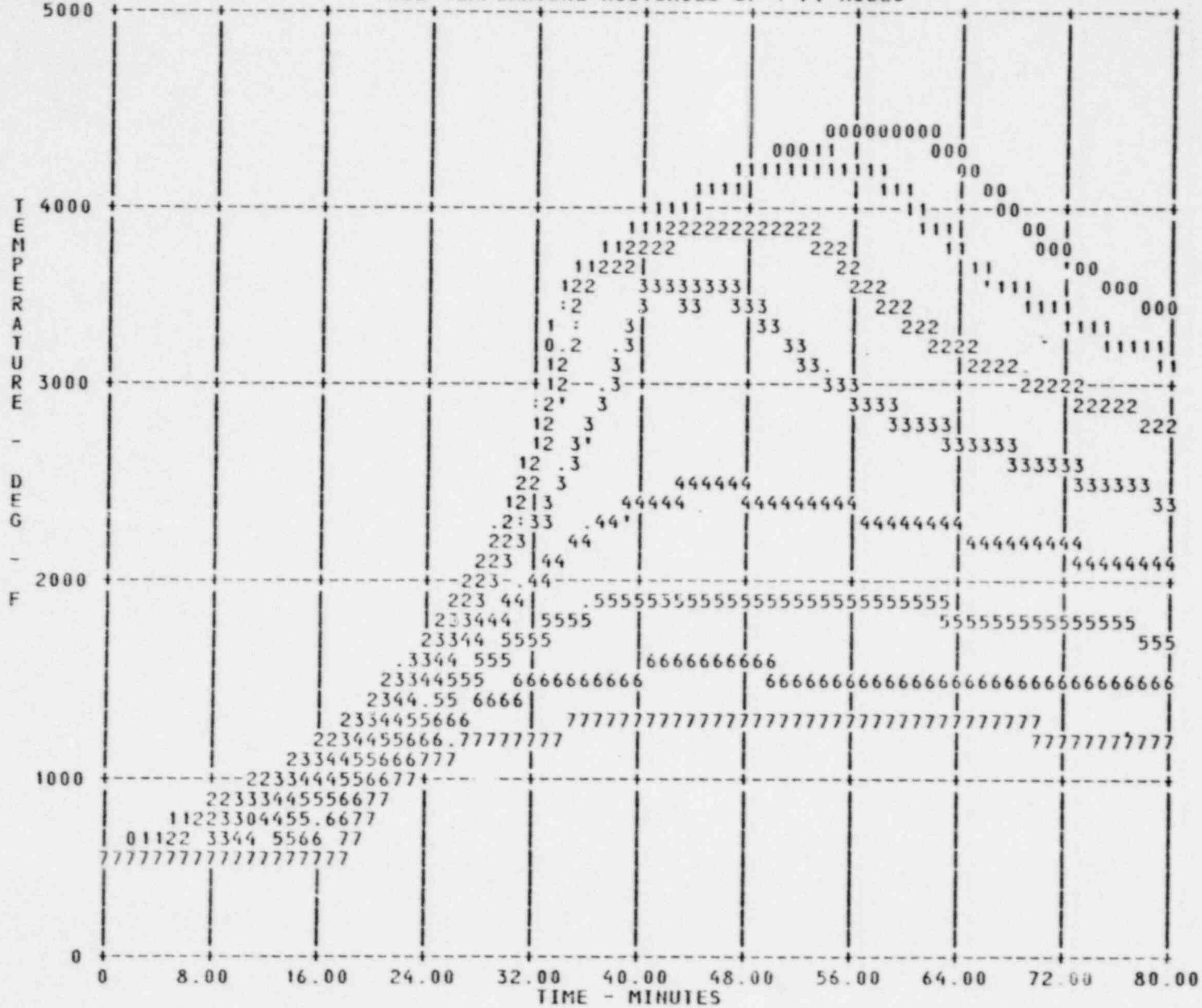
JOB 9742
10/18/79
Depth= 8 ft
Time=20 m to 8'
h_c=3
rpf=1.0

FIGURE B-2

POOR ORIGINAL

9742 8' 1.0 3

FUEL TEMPERATURE HISTORIES OF 1 FT NODES

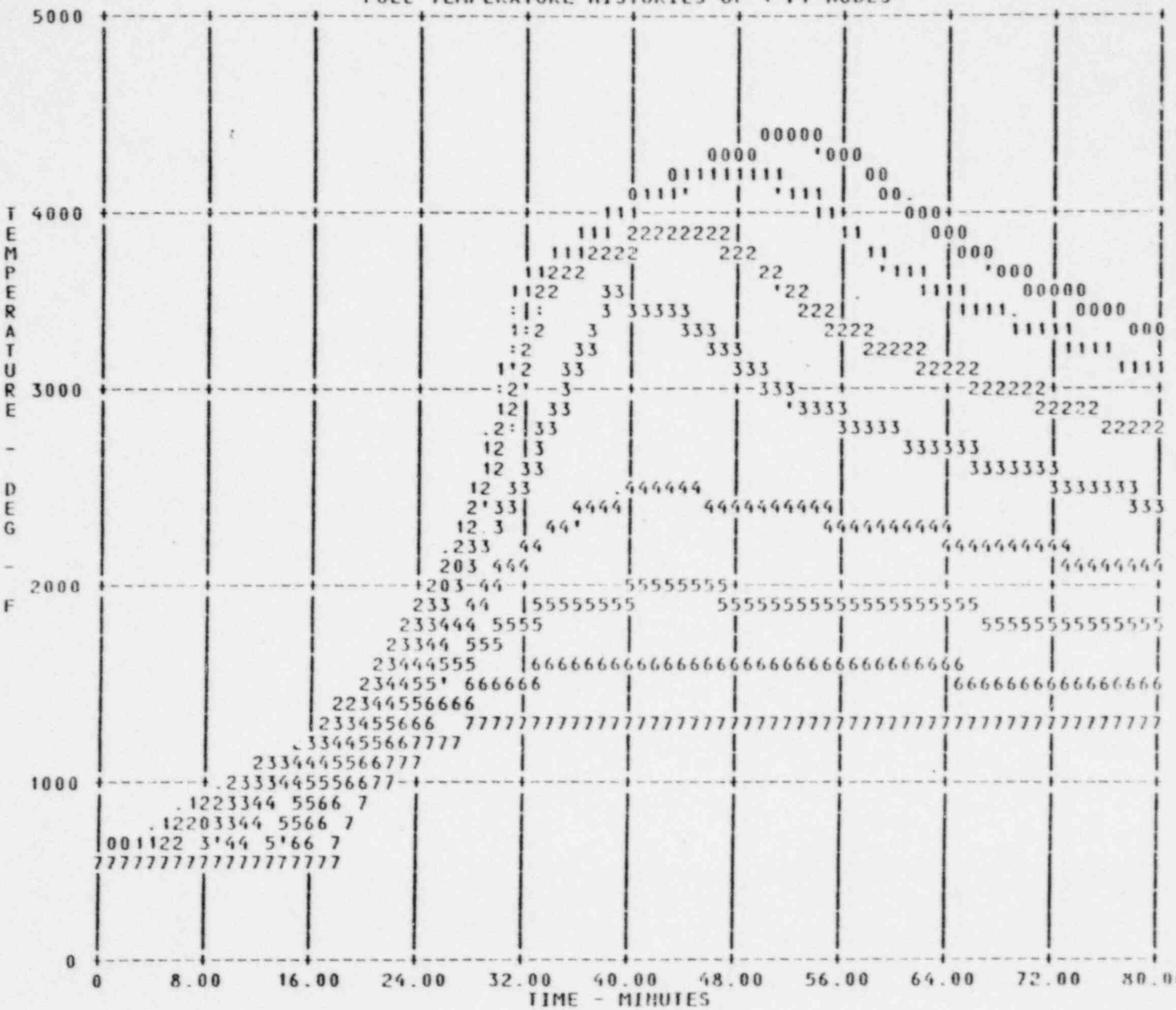


JOB 9731
10/18/79
Depth-8 ft
Time- 20 m to 8'
$h_c = 3$
rpf= 1.20

FIGURE B-3

9731 8' 1.20 3

FUEL TEMPERATURE HISTORIES OF 1 FT NODES

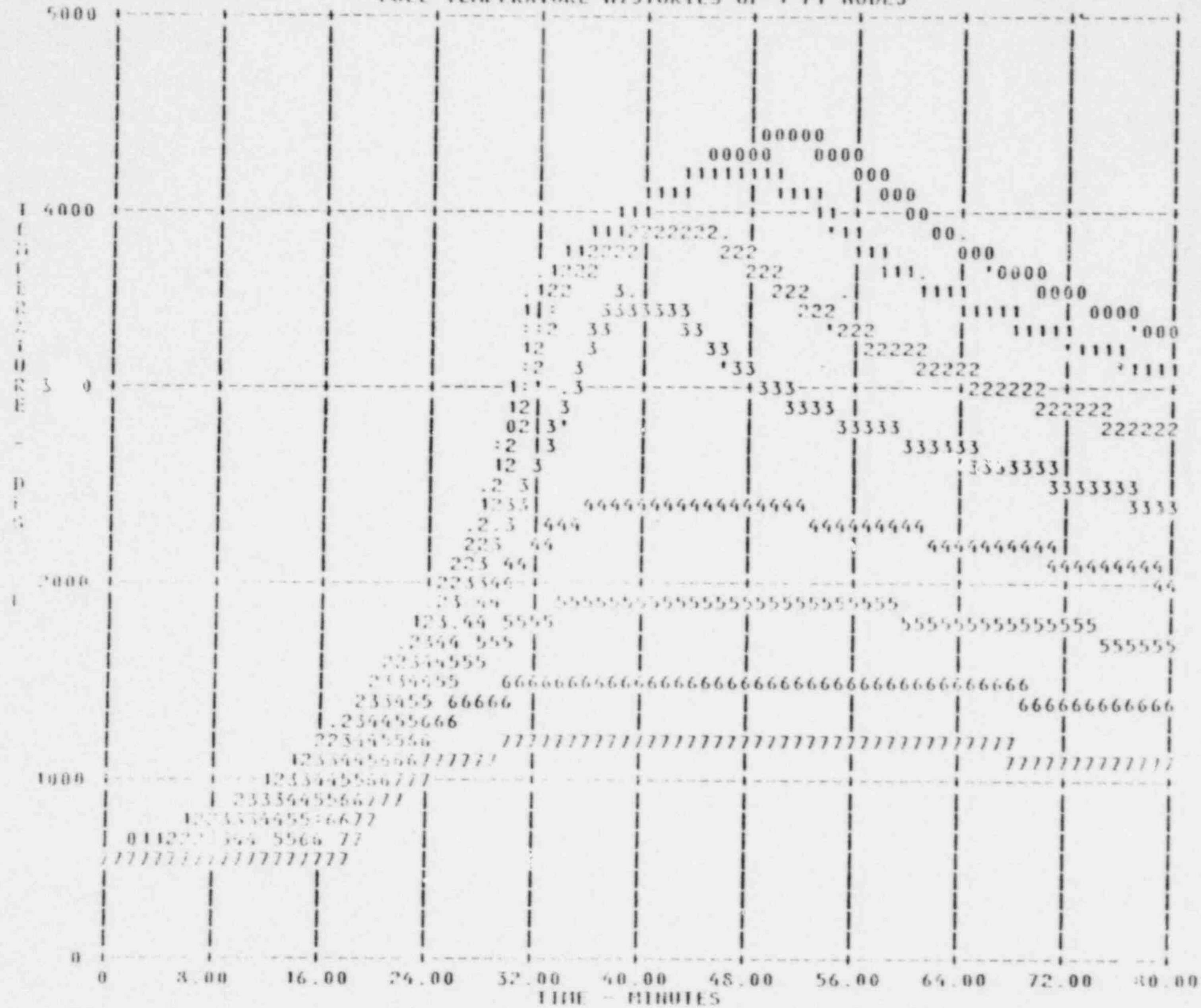


JOB 9722
10/18/79
Depth 8'
Time–20 m to 3'
h_c 3
rpf 1.167

Figure B-4

POOR ORIGINAL

FUEL TEMPERATURE HISTORIES OF 1 FT NODES



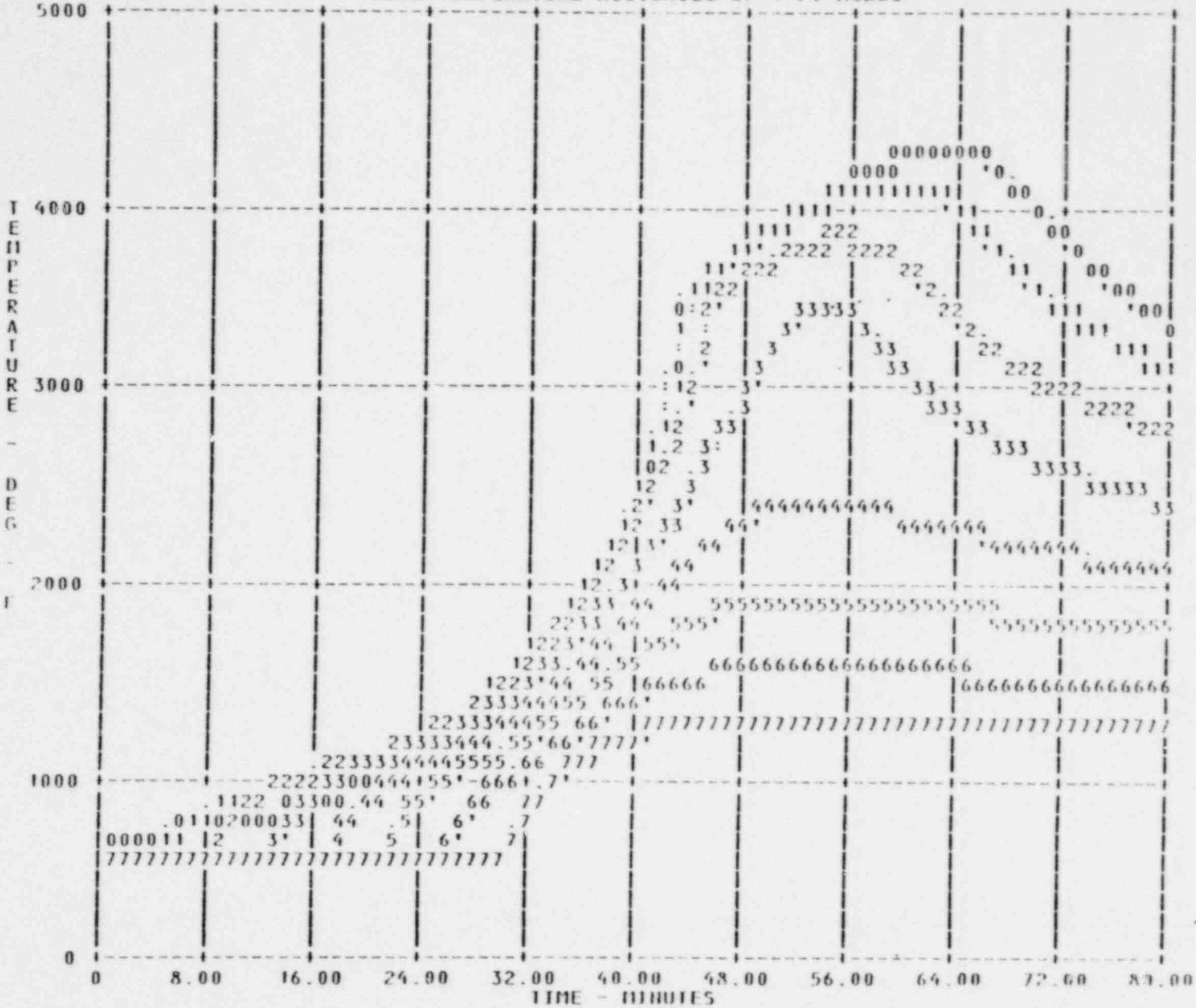
JOB 9767
10/18/79
Depth 8 ft
Time 20 m to [illegible]
h_c 10
rpf 1.467

POOR ORIGINAL

Figure B-5

8

FUEL TEMPERATURE HISTORIES OF 1 FT NODES



JOB 170 10/24/7

Without cold ro

Depth- 8 Ft

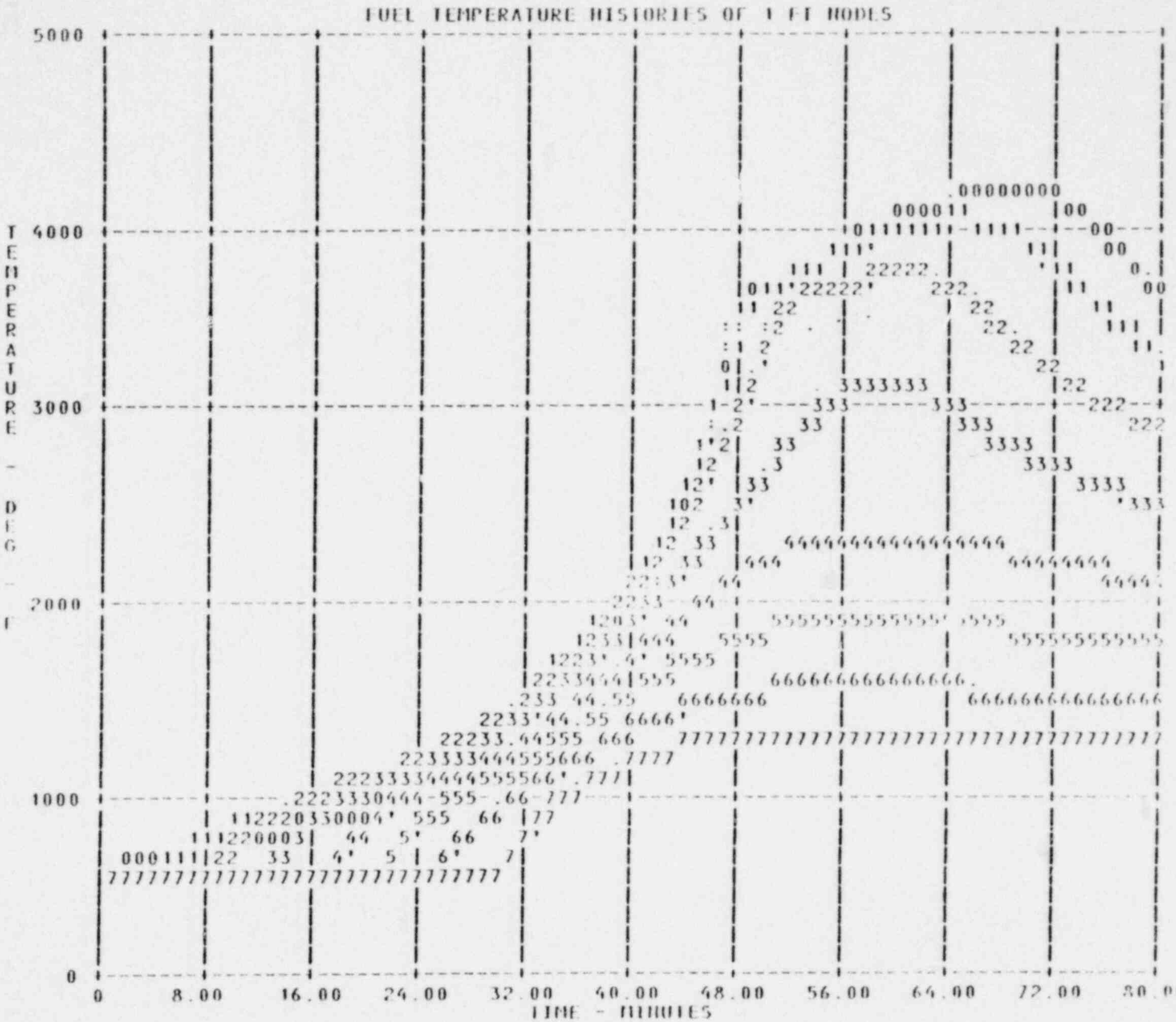
Time- 33 min to

h 3

rpf= 1.467

Figure B-6

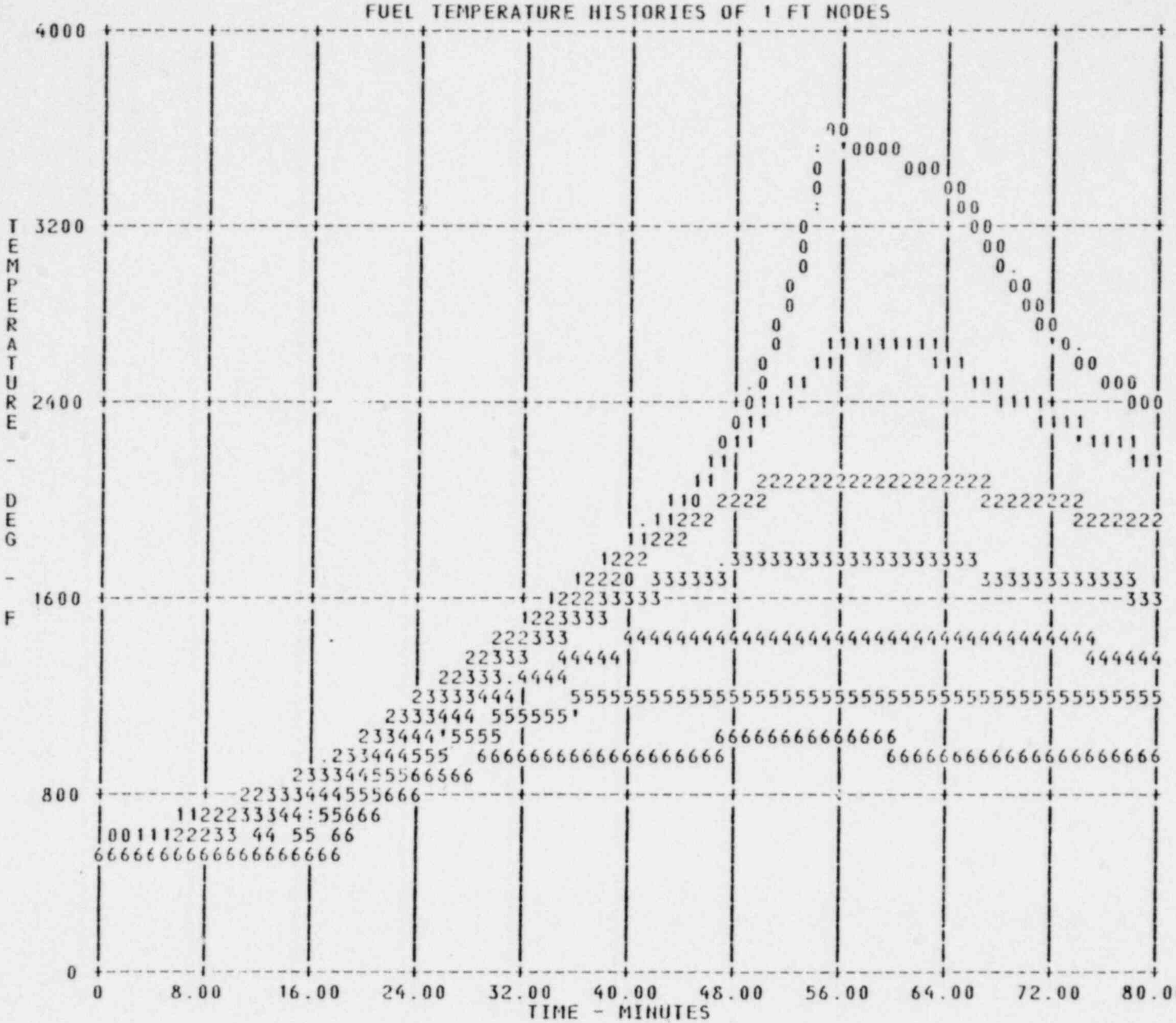
POOR ORIGINAL



JOB 7770
10/24/79
With Cold Rod
Depth- 8 ft
Time-35 min to
$h_c = 3$
rpf = 1.467

Figure B-5

POOR ORIGINAL



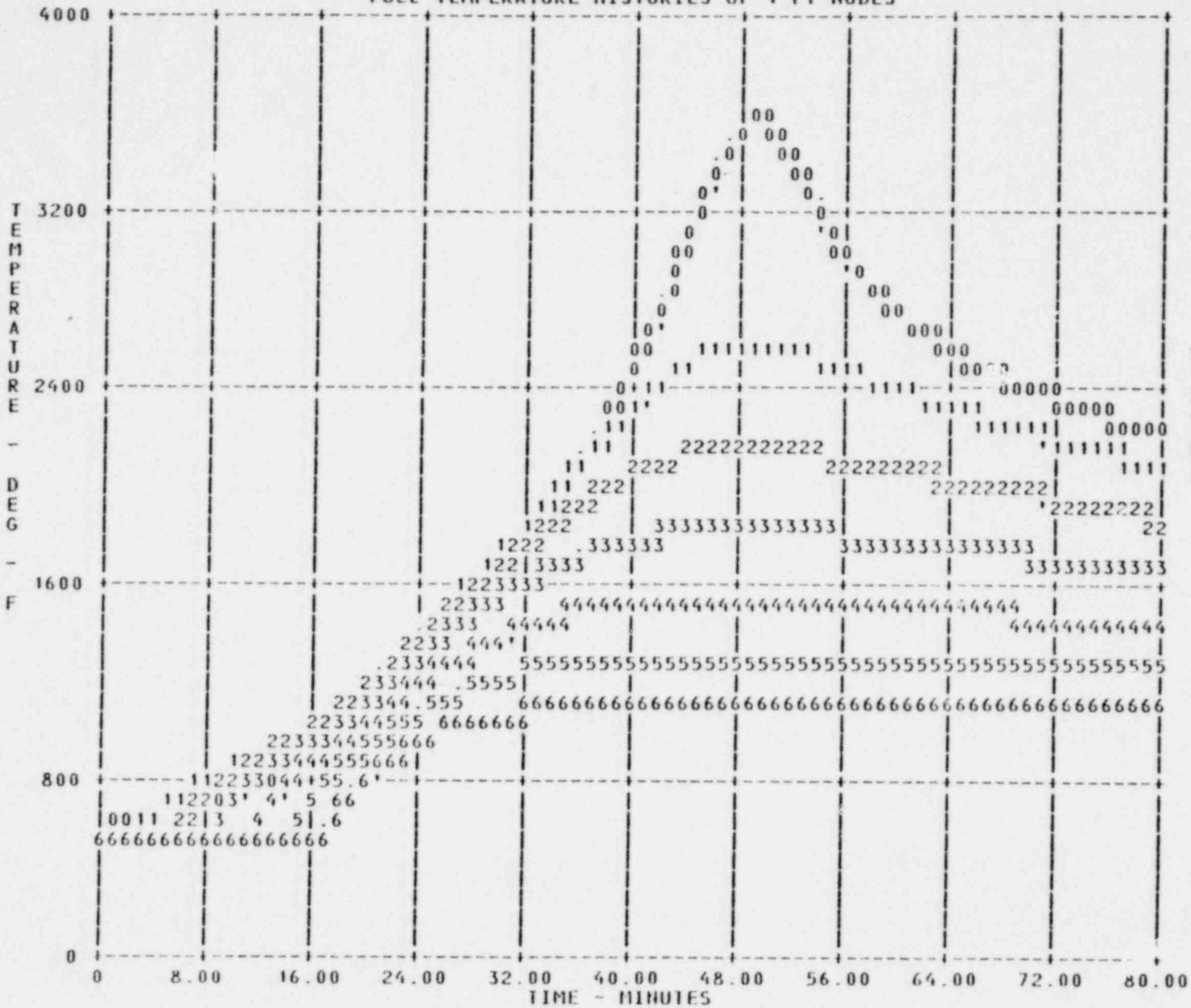
JOB 2337
10/19/79
Depth= 7 ft
Time= 20m to 7'
h_c=3
rpf=0.622

FIGURE B-8

2337 1 7' 0.622 3

PANEL 1 FRAME

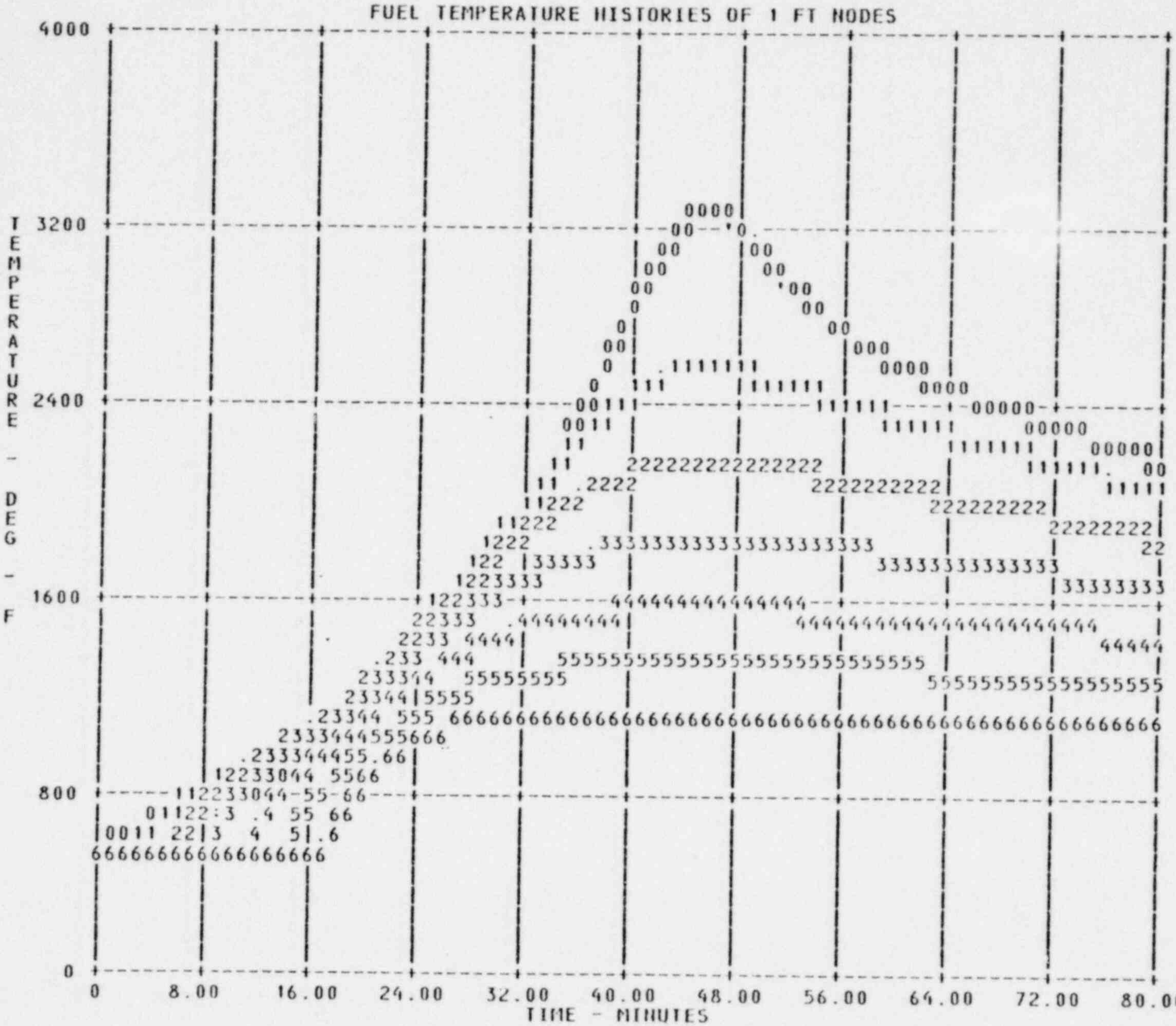
FUEL TEMPERATURE HISTORIES OF 1 FT NODES



JOB 2318
10/19/79
Depth= 7 ft
Time= 20m to 7'
h_c = 3
rpf= 1.0

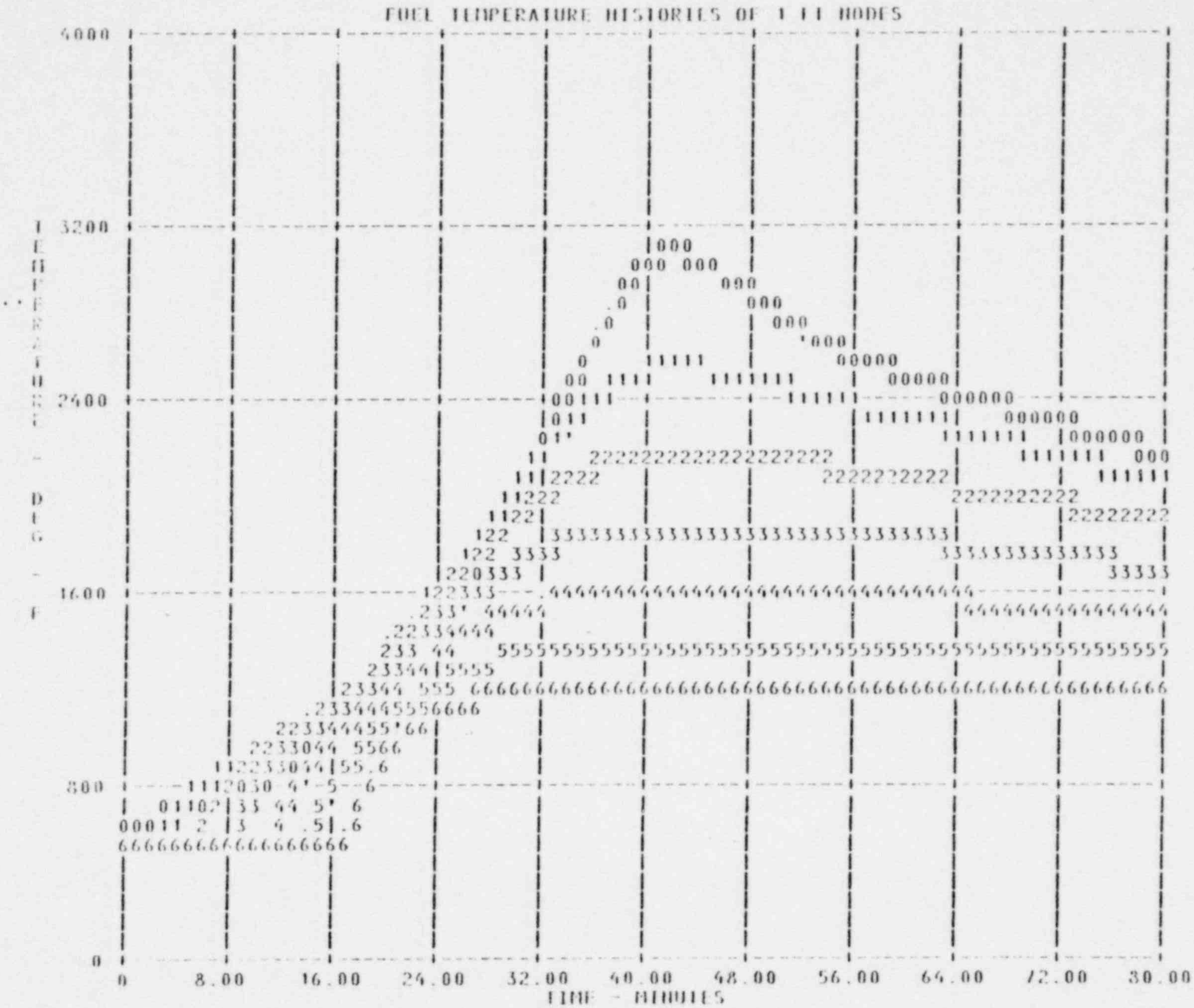
FIGURE K 9

PANEL 1 FRAME



JOB 2299
10/19/79
Depth= 7 ft
Time= 20 m to 7'
h_c = 3
rpf= 1.20

FIGURE B-16

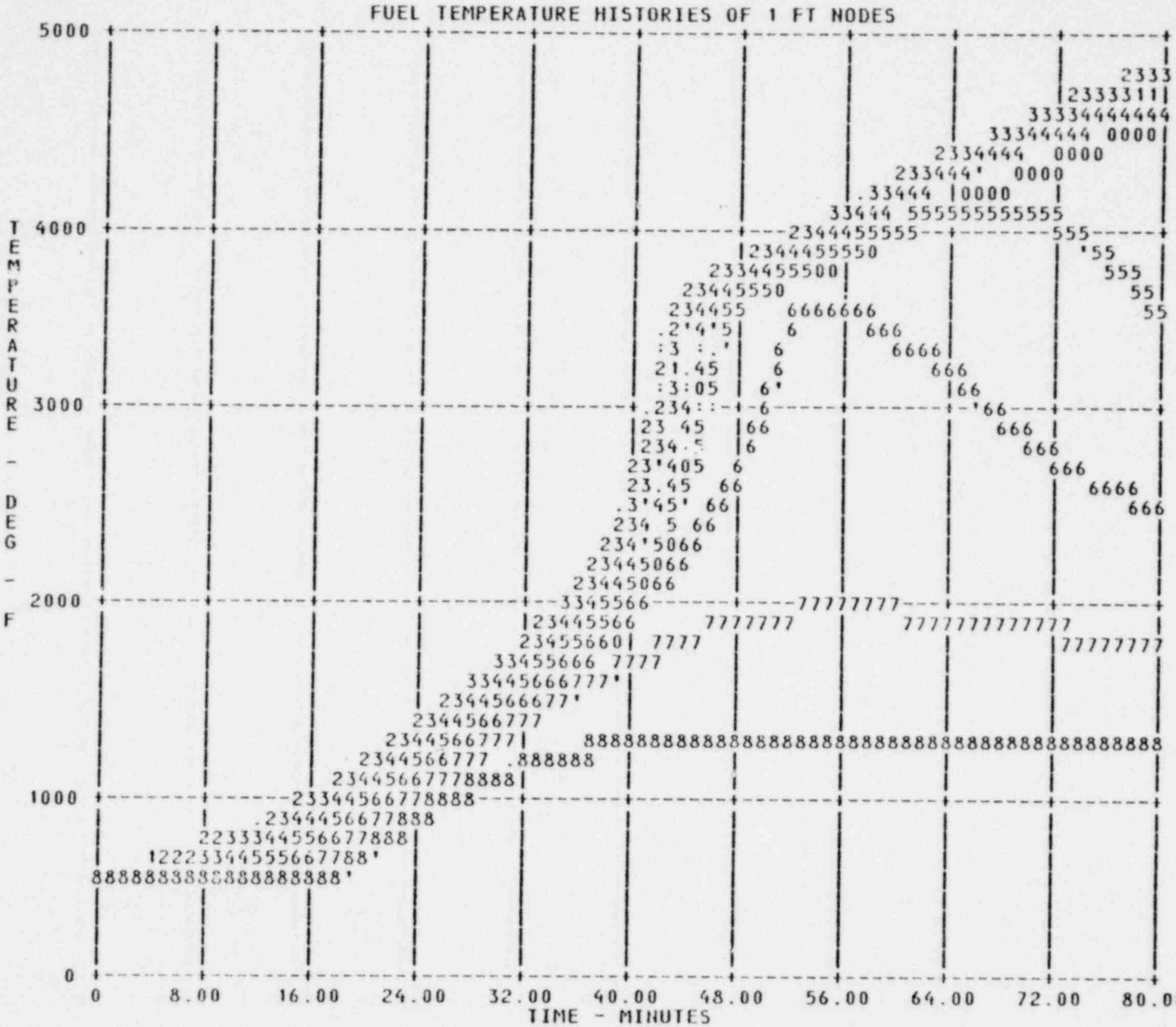


JOB 4799
10/18/79
Depth 7 ft
Time 20 m to 7
h_c 3
rpf 1.467

Figure B-11

POOR ORIGINAL

PANEL 1 FRAME

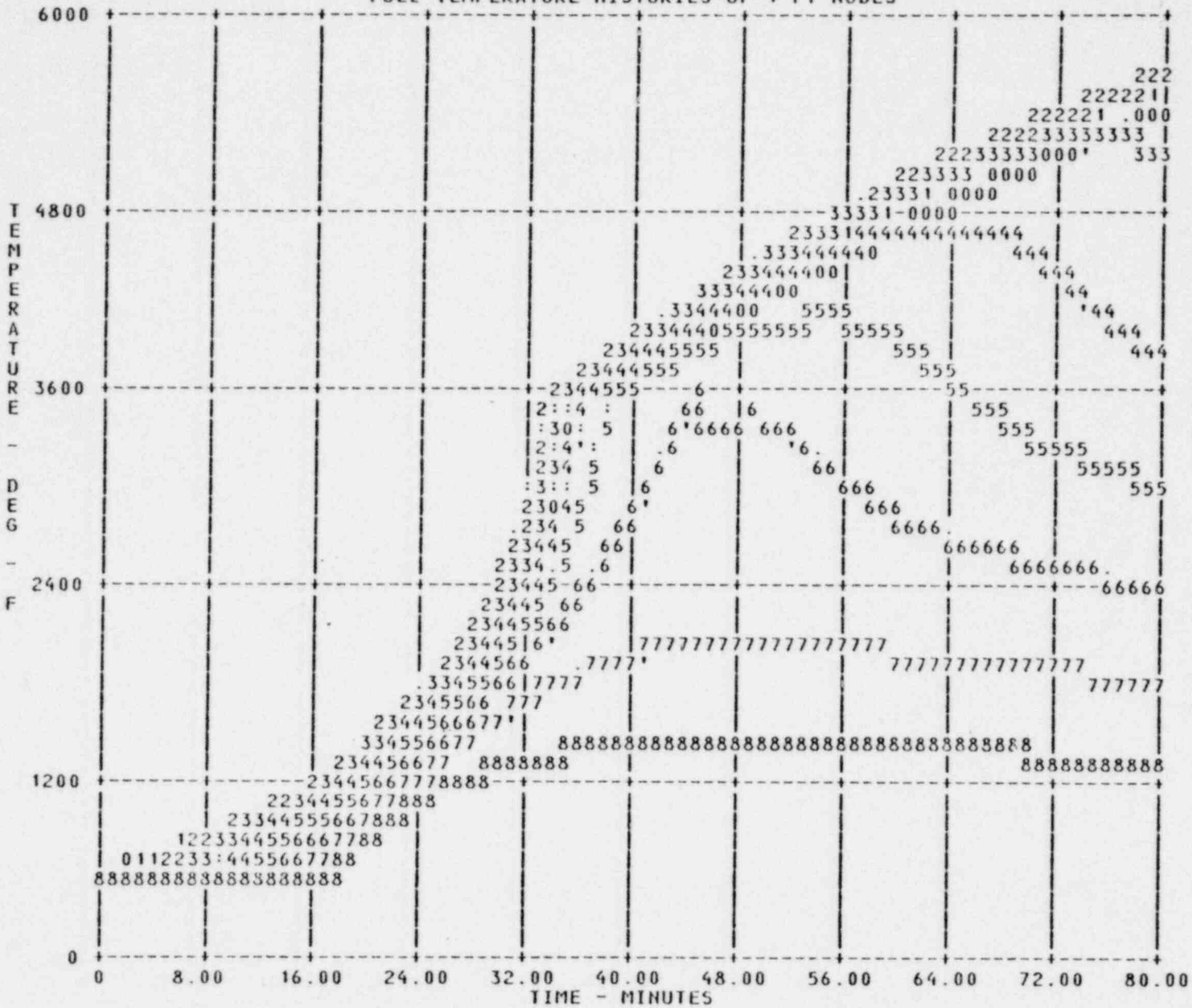


JOB 2470
10/19/79
Depth=9'
Time=20m to 9'
h_c=3
rpf=0.622

FIGURE B-12

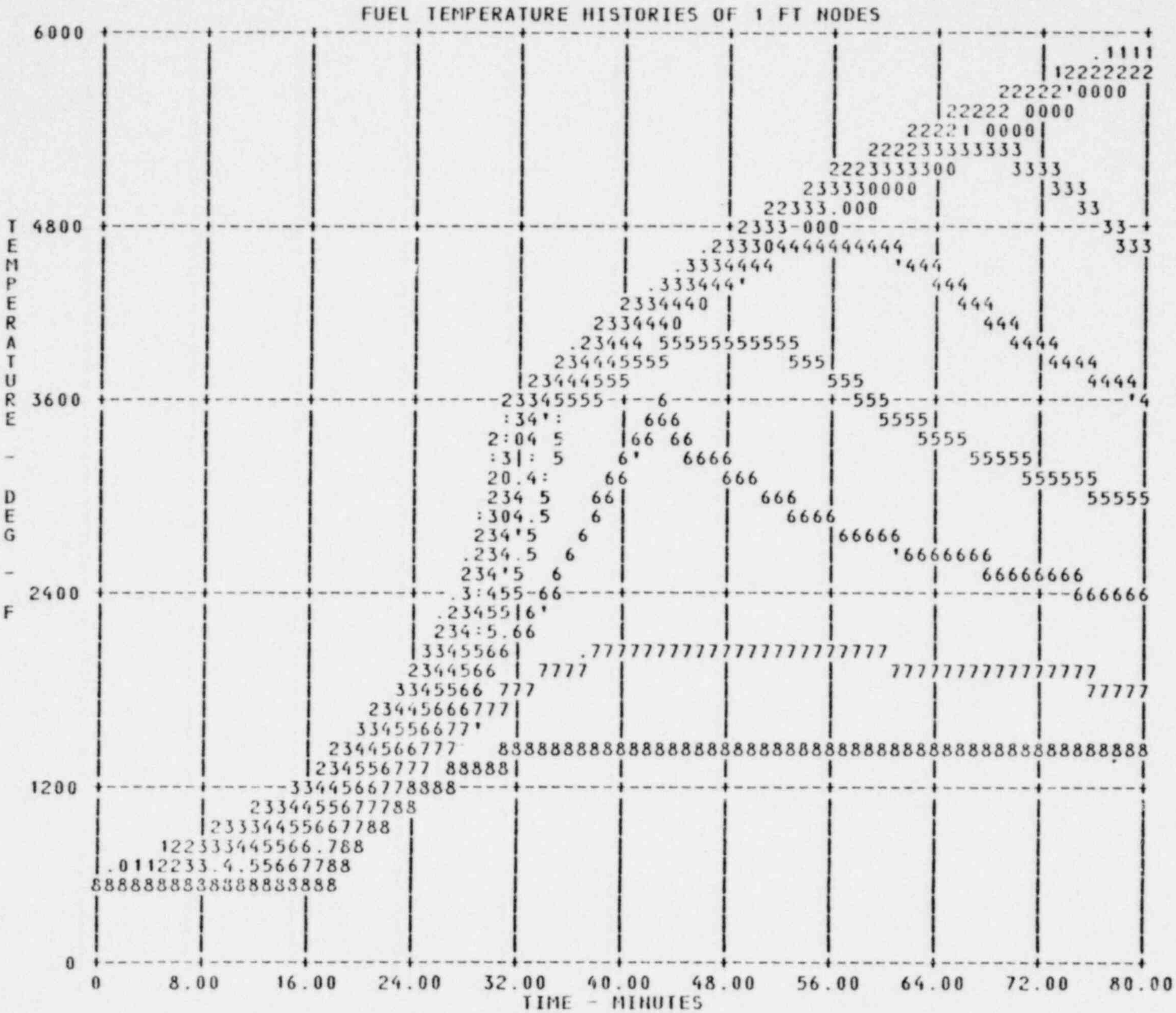
10/4/79 1 2470 1, 9' 0.622 3

FUEL TEMPERATURE HISTORIES OF 1 FT NODES



JOB 2455
10/19/79
Depth= 9 ft
Time=20m to 9'
h_c=3
rpf= 1.0

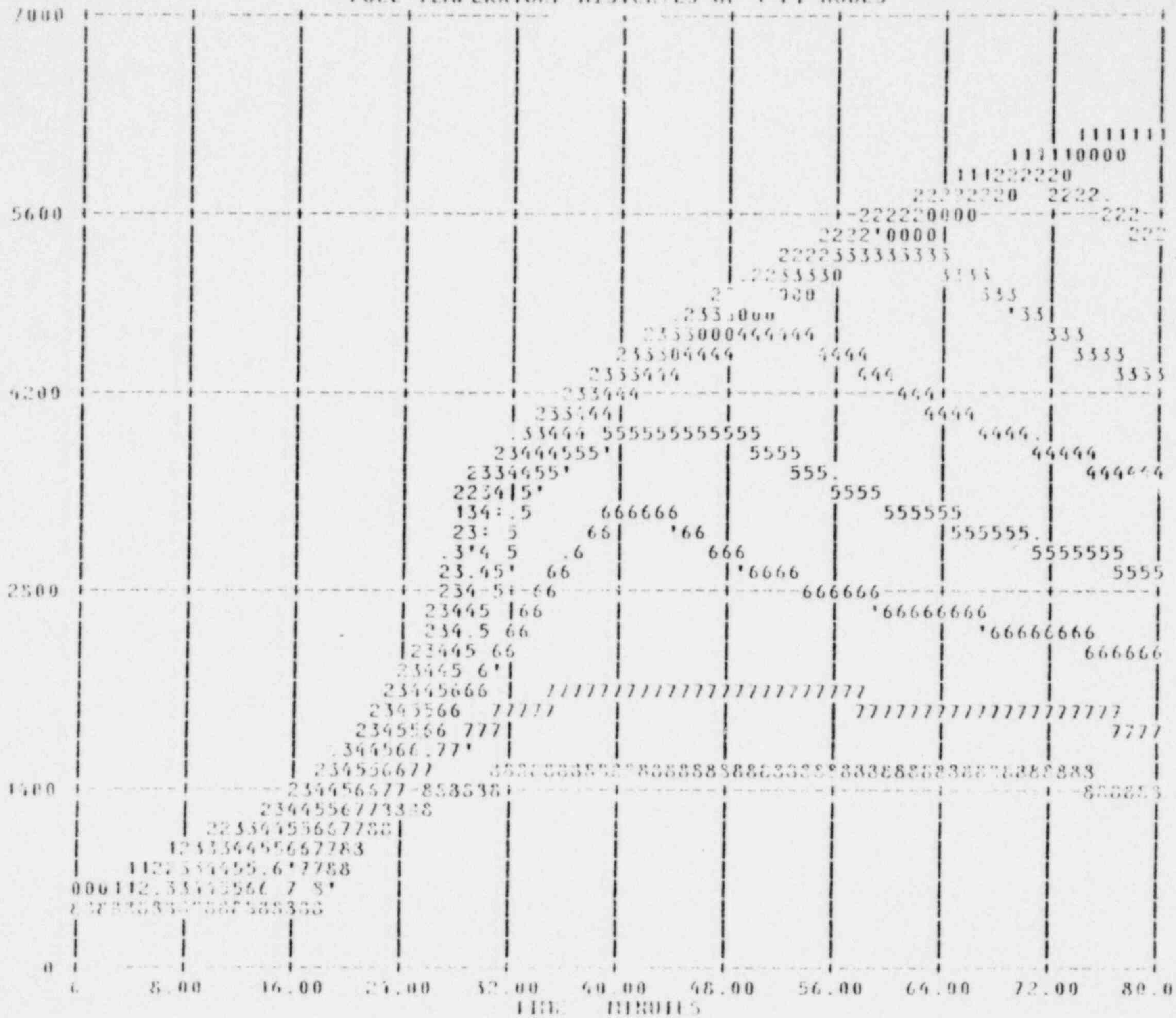
FIGURE B-13



JOB 2438
10/19/79
Depth=9 ft
Time=20m to 9'
h_c=3
rpf=1.20

FIGURE B-14

FUEL TEMPERATURE HISTORIES OF 1 FT NODES



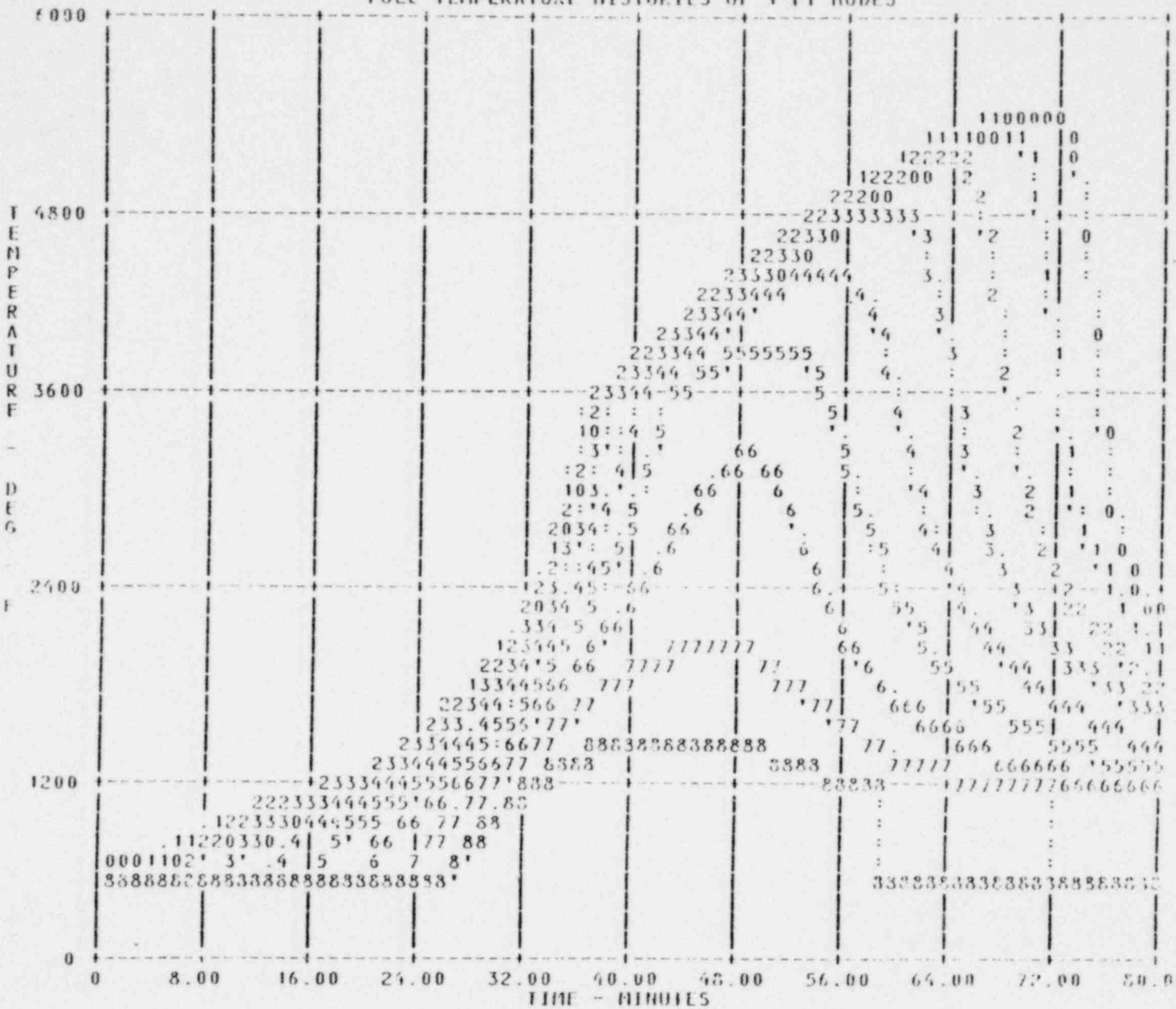
JOB 2121
10/19/79
Depth 9 ft
Time-20 m to 9'
h_c 5
rpf 1/167

POOR ORIGINAL

Figure B(5

8

FUEL TEMPERATURE HISTORIES OF 1 FT NODES



JOB 2427
10/24/7?
Depth- 9 ft
Time-30 min to ?
18 min at 9'
refill to 5'
"EPRI"scenario
h_c 3
rpf 1.467

Figure B-16

POOR ORIGINAL

TABLE 8-1

TMIBOIL CALCULATIONS ON CORE DAMAGE AT 3 HOURS

	Boiloff				1500°F Burst		Liquefaction				Peak Temperature			Comments
							First		Maximum					
	Depth Ft	Time Min	Power (Rpf)	hc hc	Depth Inches	Time Min	Depth Inches	Time Min	Depth Inches	Time Min	°F	Depth Inches	Time Min	
1	8	20	0.622	3	22	29	14	46.2	41	57	4358	1	77.5	
2	"	"	1.0	"	20	23.2	10	36.5	39	48	4410	1	62.5	
3	"	"	1.2	"	19	21.5	7	33.8	37	42	4412	1	57.5	
4	"	"	1.467	"	18	20.6	6	31.3	36	38	4370	1	52.5	
5	"	"	"	10	17	21.5	9	31.6	36	38	4362	1	50	
6	"	33	"	3	13	30	3	44	35	52	4280	1	61	without cold rod
7	"	33	"	"	16	31.9	4	48	31	56	4195	1	70	with cold rod
8	7	20	0.622	"	16	32.2	1	55.0	2	55.4	3600	2	55.4	steam flow at peak temp. 1.02 lb/hr
9	"	"	1.0	"	15	26.8	--	--	--	--	3549	1	50.0	steam flow at peak temp. 1.60 lb/hr
10	"	"	1.2	"	15	24.1	--	--	--	--	3265	1	45.0	steam flow at peak temp. 1.89 lb/hr
11	"	"	1.467	"	16	22	--	--	--	--	3042	1	40.0	steam flow at peak temp. 2.30 lb/hr
12	9	20	0.622		25	27.0	24	43.8	74	56.1	4796	29	77.5	temp. still increasing at 77.5 min.
13	"	"	1.0	"	22	21.5	18	33.6	72	45	5590	20	77.5	temp. still increasing at 77.5 min.
14	"	"	1.2	"	22	20.5	17	30.1	72	42.5	5892	15	77.5	temp still increasing at 77.5 min.
15	"	"	1.467	"	21	20	17	28.5	71	39	6194	9	78	temp still increasing at 78 min.
16	9	30	1.467	"	16	24.7	16	36.5	70	48	5444	1	70	EPRI NSAC-1

TABLE B-1
TMIBOIL CALCULATIONS ON CORE DAMAGE AT 3 HOURS

Boiloff Depth Ft	Boiloff Time Min	Power (Rpf)	hc/hc	1500°F Burst Depth Inches	1500°F Burst Time Min	Liquefaction First Depth Inches	Liquefaction First Time Min	Liquefaction Maximum Depth Inches	Liquefaction Maximum Time Min	Peak Temperature °F	Peak Temperature Depth Inches	Peak Temperature Time Min	Comments
7	20	1.467		16	22	--	--	--	--	3042	1	40.0	
8	20	1.467	"	18	20.6	6	31.3	36	38	4370	1	52.5	
9	20	1.467		21	20	17	28.5	71	39	6194	9	78	
9	30	1.467		16	24.7	16	36.5	70	48	5444	1	70	EPRI NSAC-1
7	20	1.2		15	24.1	--	--	--	--	3265	1	45.0	
8	20	1.2		19	21.5	7	33.8	37	42	4412	1	57.5	
9	20	1.2		22	20.5	17	30.1	72	42.5	5892	15	77.5	
7	20	1.0		15	26.8	--	--	--	--	3549	1	50.0	
8	20	1.0		20	23.2	10	36.5	39	48	4410	1	62.5	
9	20	1.0		22	21.5	18	33.6	72	45	5596	20	77.5	
7	20	0.622		16	32.3	1	55.0	2	55.4	3600	2	55.4	
8	20	0.622	3	22	29	14	46.2	41	57	4358	1	77.5	
9	20	0.622		25	27.0	24	43.8	74	56.1	4796	29	77.5	

#C7c

CORE DAMAGE AT FOUR HOURS

The behavior of the primary system pressure, the OTSG A cold leg temperatures, and the SRM strip chart data indicate that something of significance happened in the core at about 07:45 (3 hrs. 45 minutes accident time). An examination of the several system parameters, and the available sefl-powered neutron detector (SPND) data as shown by the alarm printer can provide a possible scenario and analysis.

Particular notice should be paid to the fact that the pressurizer spray valve had been actuated from 2 hrs 54 min to 3 hrs 15 min accident time, and from 3 hrs 42 min to 4 hrs 7 min, that the pressurizer spray line connects the OTSG A cold leg just at the outlet of RC-P2A to the top of the pressurizer (see Figure), there is a very large and very rapid increase in temperature for the 2A cold leg between 3 hrs 42 min and 3 hrs 48 min (See Figure A-1), a lesser increase of the 1A cold leg preceeding that for the 2A cold leg by about 3 min, an increase in system pressure of about 120 psi with the PORV block valve open, and only a small change in cold leg B temperature. It is to be emphasized that when the pressurizer spray valve is open, the pressure between the top of the pressurizer (i.e., the venting pressure to the outside) and the cold leg 2A of the OTSG A are equalized except for any water remaining in the spray line.

An examination of the alarm printer data for the SPNDs indicates that a large number (in 18 of 52 strings) at levels 1 and 2 (1 1/2 and 3 feet from the bottom of the fuel stack) went off-scale between 3 hrs 45 min and 3 hrs 48 min. The indication on the alarm printer of "bad" for these SPNDs indicates that their temperatures were well above 1000°F at that time (Ref.) and can only mean that such temperatures or greater were present at the 1 1/2 and 3 foot levels (from the bottom of the fuel stack) at that time.

Fourty-four of the 52 in-core thermocouples indicated temperatures above 700°F at 3 hrs 45 min when system saturation temperatures were circa 600°F, and 18 of them indicated temperatures above 1500°F in the time period between 08:00 and about 09:00 (Ref.).

All of the above observations can be explained by the following scenario, which, unfortunately, is without conclusive proof. When the pressurizer spray valve was opened the first time in the period between 2 hrs. 54 min and 3 hrs 15 min, some of te volume of water in the pressurizer spray line drained into the 2A cold leg, and probably a small amount of the volume of water contained in the pressurizer itself drained into the hot leg of OTSG A. Once the pressurizer spray line was emptied, it formed a direct gas path from the downcomer to the outside of the primary system through the top of the pressurizer, without requiring flow through the A hot leg, the surge line and the main volume of the pressurizer. Thus, the following can be postulated. With both the PORV block valve and the pressurizer spray valve open from 3 hrs 42 min to 3 hrs 57 min, the OTSG A hot leg empty of fluid, the pressurizer voided a portion of its contents into the OTSG A hot leg, onto the top of the hot core. A large burst of superheated steam and pressure was generated, the pressure of the system rose more than 100 psi (even though the PORV block valve was open), the open spray valve allowed a path for steam and hydrogen gas from the top of the core through the core barrel vent valves into the downcomer, up the 2A cold leg, up the spray line to the top of the pressurizer, and out the PORV to the reactor coolant drain tank (RCDT). If the OTSG A cold legs had not been filled to the point of refluxing into the downcomer and the core (remember that the letdown flow of 70 gpm-500 ft^3/hr-at 2000 psi comes from the OTSG 1A cold leg), then superheated steam would have been present in the outlet region of the RC pump, and could well have been drawn through the pump into the vertical section o the cold leg below the pump to condense on the cold water present in that cold leg. Thus, the cold leg temperature sensor could have been heated up by superheated steam flowing counter current through the pump to condense in the vertical section of the cold leg (see Figure   in Section   ). When the water from the voiding pressurizer hit the top of the overheated core, the core was quenched from the top to form a "rubble bed" at a lower level in the core. The normal fuel rod geometry gives a packing density of about 45% in the core, i.e., 45% fuel rod (including control and poison rods, instrumentation tubes, and power shaping rods) and 55% water space. A normal packing density for randomly sized and shaped particles approaches 65%. If the upper part of the core (nearly 1/2 as estimated in "Core Damage at Three Hours" Section   ) is compacted from 45% to 65% by shattering when the water from the voiding pressurizer hit it, additional oxidation of Zircaloy can be expected from the freshly exposed inner surfaces of fuel rod cladding. If the core is still not covered b coolant, then the increase in SRM counts (due to a change in core geometry), the increase in system pressure (a large input of energy into the vapor space of the primary system) and the greatly increased number of off-scale level 1 and 2 SPNDs can be explained by the formation of additional "liquified fuel" in the shattered and rubbled core by its densification and concentraiton of decay and oxidation heat. In many cases, the "liquified fuel" would have flowed down a "cold rod" such as an instrumentation tube, or a control rod guide tube, to form a casing around it below the average core water level. This then would have formed a steam jet through the annuli of the instrumentation tube (and the annulus of the control guide tube), to keep the upper parts of the tubes (the SPNDs, and the in-core thermocouples) cooled enough to survive until the overall system was cooled below about 2600°F. This postulated scenario can explain most of the observations at about 3 hrs 45 minutes accident time, but it does not explain the rise in pressurizer level oservable in Figure A-1 starting at 3 hrs 27 min and continuing to 3 hrs and 45 min.

If this scenario is corred, then the highest temperatures probably occurred and the greatest amount of "liquified fuel" were formed in the time period between 3 hrs 45 min and 4 hrs accident time for any time period in the 16 hour-long accident. This time period may have been that in which the closest approach to a "core melt-down" event occurred.

II C 1 g

II-C-1-g Estimate of Core Damage from Release of Radioactivity from the Core

At the time of shutdown, the reactor core contained fission products, activation products, and actinides. Some of these--notably Kr, and Xe, are gaseous and can diffuse through the fuel pellet to collect in the gap between the fuel and the cladding. To a lesser extent, the halogens (iodine and bromine) can also diffuse into the fuel/clad gap. Any perforation of the cladding can release these fission products into the reactor coolant.

If the fuel temperatures are higher than operating temperatures, but well below melting, other radioactive materials are volatilized and can diffuse out. Also, diffusion of the noble gases and halogens is enhanced, so that a larger fraction of these can be released. The release of cesium is quite variable; this could be because of compound formation. Because of this variability, it is not possible with the present state of knowledge to determine precisely the temperature at which a reasonably large fraction of the cesium would be liberated; however, the general concensus is that this would be not lower than 1300°C (2370°F).

At still higher temperatures, such that liquefaction or melting of the fuel occurs, some fraction of other fission products such as tellurium can be released. Data reported in /1/ show that the escape of tellurium depends also on many factors other than temperature. Under oxidizing conditions some ruthenium might be released before melting. In general, rather large fractions of both tellurium and ruthenium are released in melting, but under some conditions can also be released before melt. The presence of Ru and Te do not prove that melt has occurred, but the absence is a good indicator that it has not occurred. More recent experimental work /2,3/, while tending to confirm previous data, has not resolved all the questions regarding conditions--especially of temperature--under which fission products would be liberated.

Many of the fission products and most of the actinides occur as refractory oxides, and are only released in relatively small amounts even at elevated temperatures. However, if damaged fuel pellets are rewetted, some of the more refractory radioactive material can be leached out. This process is slow, and only small fractions of these materials find their way into the coolant by leaching. The longer the damaged fuel is in contact with water, the more of these materials are released.

Fission products and actinides can be divided into typical release groups, based on the ease with which they are volatilized. One such grouping (from Reference 4) is, in order of decreasing volatility:

- I Noble Gases (Kr, Xe)
- II Halogens (I, Br)
- III Alkali metals (Cs, Rb)
- IV Tellurium
- V Alkaline earths (Sr, Ba)
- VI Noble metals (Ru, Rh, Pd, Mo, Tc)
- VII Rare earths and actinides
- VIII Refractory oxides of Zr and Nb.

The fraction of gaseous and volatile fission products released depends on the temperature and the size of the fuel fragments. If the temperature is high, or if the fuel is highly fragmented, nearly complete release of the volatile materials can be assumed.

Under the conditions which have been calculated for the accident at TMI-2 /5/, nearly complete release of groups I and II can be assumed from all fuel which was severly damaged, plus some additional fraction from fuel rods whose cladding was perforated without damage to the fuel. This additional amount from perforated but otherwise undamaged rods is probably partly balanced by the amount not released from severely damaged fuel

A major fraction of group III and a much smaller fraction of group IV could have been released from the most severely damaged fuel. Small fractions--perhaps of the order of 10 percent or less--could have been released from perforated but otherwise undamaged rods, but this cannot be well estimated.

Very small fractions of the remaining groups might have been released from the very hottest fuel. The principal mechanism for release of these refractory materials is probably leaching, however. Leaching from irradiated UO_2 has not been very thoroughly studied. However, the work of Katayama /6,7/ and of Forsyth and Eklund /8/ has shown that the leaching rates are slow, being comparable to those from glass. Quantitative data--especially for the temperatures and conditions obtaining in TMI-2, are too sparse for a reliable calculation of the rate of leaching, especially when one considers that the condition of the damaged fuel is completely unknown. However, it can be said that only a small fraction of the most refractory material would be expected to have found its way into the reactor coolant. An approximate leaching calculation is presented in the Appendix.

It has been generally agreed that the principal fuel damage probably started at about three hours after turbine trip. There was probably only minor damage before 2 hours. The calculated total inventory /9/ of fission products, activation products, and actinides is given below for 3 hours after shutdown.

TABLE I

Activity in Release Groups[a]

Group	Activity
I	2.97×10^{8} Ci
II	4.47×10^{8} Ci
III	4.6×10^{7} Ci
IV	1.61×10^{8} Ci
V	3.85×10^{8} Ci
VI	6.34×10^{8} Ci
VII	2.69×10^{9} Ci
VIII	4.80×10^{8} Ci
TOTAL	5.11×10^{9} Ci[b]

Notes: (a) A few elements of low total activity, notably Fe, Cu, As, and Sb have been arbitrarily located on the basis of melting point.

(b) Total does not quite agree with calculated total activity because of rounding.

Radioactive material released to the reactor coolant could have been partially flushed to the containment through the open PORV (RC-R2). Some of the activity could have been flushed to the containment prior to the containment isolation; this could then have been pumped to the auxiliary building. However, the coolant could only have contained a minute fraction of the total activity at this time; it is highly improbable that a significant fraction was released before the reactor building sump pumps were shut down. There is a possibility--which is not substantiated /10/--that more water leaked to the auxiliary building after pump shutdown. This leakage would have terminated at 3 hrs. 56 minutes when the reactor building was isolated.

Most of the activity flushed out of the RCS probably remained in the reactor building. Some additional material would have volatilized from the makeup tank. Aside from these losses, which are not expected to be very large, estimates of the total activity released from the fuel can be made by analyzing reactor building air and water samples, the reactor coolant, and the auxiliary building tanks.

Iodine is quite volatile, so that it might be supposed that some significant fraction would be found in the air. However, the very high solubility of iodine in water and the strong tendency of atmospheric iodine to plate out on surfaces quickly reduce the amount of iodine in the air to low levels. Cesium, being less volatile, would not be expected to be present in the air in any significant quantity. On the other hand, the solubility of xenon and krypton is very low; these gases will be found almost entirely in the air.

To summarize, one can expect nearly complete release of noble gases, iodine, and cesium from damaged fuel, even if the temperature is below the melting point. Significant releases of tellurium, ruthenium, and more refractory materials will occur only if the temperature approaches the melting point. Most of the noble gases will be found in the air and most of the other fission products will be found in water.

Analyses of samples of containment air, reactor coolant water, and auxiliary building tank water are summarized in Ref. /11/. Reactor coolant analyses show between 7 and 15 percent of the calculated inventory of iodine and cesium isotopes to be in the coolant. If these measurements are corrected for dilution by water from the borated water storage tank, the fractions would be about a factor of 3 higher. Results for refractory materials show great variation. A sample taken on April 10 was analyzed by four different laboratories. The variation from laboratory to laboratory was great; indicating low confidence in the results. Analyses of krypton and xenon isotopes in the containment atmosphere also showed considerable variation. However, based on the most abundant isotopes (^{85}Kr and ^{133}Xe) there seemed to be 29-62% of the core inventory of noble gases in the containment air. Only 2-3% of the iodine and cesium was found in the auxiliary building tanks.

On August 28, 1979, a hole was drilled into the reactor building and samples of sump water were removed. Analyses of these samples showed 22-48% of the core inventory of iodine and cesium to be in the reactor building sump water /12/. In addition to iodine and cesium, very small amounts of Ru, Zr, Nb, Sb, La, and Ag were found. As expected, little ^{90}Sr was found. The amounts corresponded to at most a few millionths of the core inventory. About 0.02% of the core inventory of ^{129m}Te was found.

All of these sample analyses were corrected for decay of the radionuclides to the time of analysis. This correction process is certainly more accurate than the analyses themselves, i.e., the accuracy of the estimates does not depend on the accuracy of the decay calculation. Table II is a recapitulation of the release of volatiles.

From these results one can cautiously conclude that between 40 percent and 60 percent of the core inventory of release groups I-III was released to the coolant, that only a small fraction of group IV was released, and only minute amounts of the remaining groups. The amount of refractory isotopes released is consistent with leaching (See Appendix).

These data tend to confirm other analyses of core damage. The data on radioactivity released are too sparse and variable to be able to decide on the amount of core damage with any precision; however, the following conclusions appear to be supported:

TABLE II
Total Volatile Isotopes
Released from Core

Released To	Isotope (fraction of core inventory)			
	^{133}Xe	^{131}I	^{137}Cs	^{134}Cs
Environment	0.1 (a)	-- (b)	--	--
RB Atmosphere	0.46 (c)	--	--	--
RB Water	--	0.22 (d)	0.48 (d)	0.34 (d)
RC Water	--	0.14	0.12	0.08
Aux. Bldg. Tanks	--	0.03	0.03	0.02
TOTALS	0.46	0.39	0.63	0.44

(a) Estimated, Ref. 811.
(b) Dashes indicate low values (generally less than 1%)
(c) Best estimate from data of Ref. 11.
(d) Average of observations

(a) About 50 percent of the reactor core was damaged sufficiently to release the most volatile fission products.

(b) the low fractions of tellurium, ruthenium, and strontium indicate that no significant quantity of fuel reached the melting point of UO_2 (5200°F).

(c) the amount of refractory isotopes in the reactor coolant is consistent with leaching.

References:

1. D.M. Haaland, "Release of Radioactivity from the Core," Chapter 7 of Core-Meltdown Experimental Review, SAND 74-0382, Sandia Laboratories, Albuquerque, NM, August 1975.

2. H. Albrecht, V. Matschoss, H. Wild; Experimental Investigations of LWR-Core Material Release at Temperature Ranging from 1500-2800°C, 7th Water Reactor Safety Information Meeting, Nov. 5-9, 1979.

3. A.P. Malinauskas, R.A. Lorenz, J.A. Collins, "Fission Product Release from Defected LWR Fuel Rods," 7th Water Reactor Safety Information Meeting, November 5-9, 1979.

4. U.S. NRC, Reactor Safety Study, WASH-1400 (NUREG-75/014), Appendix VII, Washington, D.C., October 1975.

5. M.L. Picklesimer, "Bounding Estimates of Damage to Zircaloy Fuel Rod Cladding in the TMI-2 Core at Three Hours after the Start of the Accident, on March 28, 1979," Memorandum for file, U.S. NRC, Washington, D.C., June 1979.

6. Y.B. Katayama and J.E. Mendel, "Leaching of Irradiated LWR Fuel Pellets in Deionized Water, Sea Brine, and Typical Ground Water," ANS Trans, 27:447, Nov.-Dec. 1977.

7. Y.B. Katayama, Leaching of Irradiated LWR Fuel Pellets in Deionized and Typical Ground Water, BNWL-2057, Battelle Northwest Laboratories, Richland, Washington, July 1976.

8. E.B. Eklund and R. Forsyth, Laekning av bestraalad UO_2--braensle, (Leaching of irradiated UO_2--fuel), TPM-BL, AB Atomenergie, Studsvik Sweden, February 1978 (in Swedish).

9. Letter from D.E. Bennett, Sandia Laboratories, to J. Murphy, U.S. NRC (PAS), dated April 4, 1979, Re: Sandia-ORIGEN core calculations.

10. U.S. NRC, Investigation into the March 28, 1979 Three Mile Island Accident by the Office of Inspection and Enforcement, NUREG-0600, Washington, D.C., August 1979.

11. Letter from Harold R. Denton (NRR) to Vincent L. Johnson, September 28, 1979.

12. Letter from J.T. Collins (TMI-2 support) to J.Y. Lee (TMI-2 support), "Preliminary Analysis Results of TMI-2 Containment Building Water Samples," August 31, 1979.

APPENDIX

CALCULATION OF LEACHING FROM REACTOR FUEL

Analysis of a sample of reactor coolant taken on March 29, showed only a minute fraction of a percent of the core inventory of refractory elements (Sr, Ru, Ba) in the reactor coolant. A sample taken on April 10 showed about 1% of the core inventory of Sr, Ba, La and Mo. The fractions were quite variable -- both from element to element and from laboratory to laboratory -- but a figure of 1% represents a reasonable average.

A fit to the data of Katayama /1/ gives, for early time,

$$w = 3.2 \times 10^{-4} t^{-.9} A \quad \text{(average of } ^{90}Sr \text{ and } ^{137}Cs)$$

w = leaching rate, g/day

t = time in days

A = surface area, cm^2

Then the total amount leached is

$$W = \int_0^T w \, dt = .0032 \, t^{0.1} A$$

where W is total leached in grams.

The total weight of fuel is 9.31×10^7 g./2/

If it is assumed that 1/3 of the fuel is damaged, the mass of damaged fuel is 3.1×10^7 g.

If the damaged fuel is in the form of spheres of uniform size, the volume of each is

$$v = \frac{4 \pi r^3}{3}$$

and the surface area is

$$a = 4 \pi r^2.$$

The number of spheres is then

$$n = \frac{V}{v} = \frac{M}{\rho v}$$

where M is the mass of fractured fuel, and ρ is fuel density. The total surface area is then

$$A = na = \frac{3M}{\rho r}$$

The apparent surface area is

$$A_{app} = \frac{(0.01)\ (3.1 \times 10^7)}{(.0032\ t^{0.1})}$$

For April 10 (t=14)

$$A_{app} = 7.4 \times 10^7\ cm^2.$$

The equivalent radius sphere is

$$r = \frac{3M}{A\rho} = \frac{(3)\ (3.1 \times 10^7)}{(7.4 \times 10^7)\ (10.9)} = 0.12\ cm$$

This appears to be rather small for an average size -- a diameter of about .090"; however, the precision of the concentration data is so poor that a factor of two larger would also be completely reasonable. Also, if the fraction of fuel badly enough damaged to be leached is larger, the average radius would be larger.

Experimental data /3/ indicate that particle sizes under similar (LOCA) conditions might be of the order of 0.2 cm.

Although the calculations cannot be made with any precision, it appears that the presence of refractory elements in the reactor coolant can be explained by leaching alone.

References for Appendix:

/1/ Y.B. Katayama and J.E. Mandel; "Leaching of Irradiating LWR Fuel Pellets in Deionized Water, Sea Brine, and Typical Ground Water," ANS Trans., 27:447 (Nov.-Dec. 1977)

/2/ Final Safety Analysi Report, Three Mile Island Nuclear Station--Unit 2, Metropolitan Edison Co., Jersey Central Power and Light Co., Pennsylvania Electric Co.

/3/ INEL paper at WRSIM (details to be provided later).

IIC3

II-C-3 Hydrogen Production, Removal, and Hazard

1. Hydrogen Production

During dryout and uncovering of the core, decay heat causes the temperature of the fuel rods to increase. The water remaining in the reactor vessel boils off, and the steam flowing past the fuel rods gives some cooling, but not enough to prevent the temperature rise. As the temperature of the zircaloy cladding increases, the rate of oxidation by the steam increases. The oxidation reaction releases heat, causing the cladding temperature to go even higher. This phenomenon, and the method of calculating the extent of oxidation is described in Reference 1 and in Section ___ of this report.

Hydrogen production by two mechanisms has been postulated: by the reaction of steam with zirconium

$$2H_2O + Zr \rightarrow Zr\,O_2 + 2H_2,$$

and by the radiolytic decomposition of water vapor

$$2H_2O \rightleftarrows 2H_2 + O_2.$$

The second mechanism is known to be inoperative (see section 5.1). As soon as an excess of hydrogen exists from the first mechanism, the reverse reaction (recombination) proceeds faster than decomposition, so that no net hydrogen or oxygen is produced by radiolytic decomposition.

A number of estimates of the amount of hydrogen produced by the metal-water reaction have been made. For example, Picklesimer /1/ made an early estimate of 240-260 kg of hydrogen in the first 3 hours. Cole /2/ estimated 350 kg in the same time frame. A later estimate by Cole /3/ based on more realistic calculations, indicated that 450 kg at 6.5 hours was more likely produced. This includes less than 10 kg from oxidation of stainless steel. The President's Commission Technical Staff estimated from 434 kg to 620 kg/4/.

2. Hydrogen Accounting

The calculation of Cole (3) also includes the partioning of hydrogen between the RCS and containment. This partitioning is important in accounting for the removal of hydrogen. Because the estimate of Cole lies within the bounds of Ref. 4, it will be used as a starting point for the analysis.

Cole estimated that at 6.5 hours, 250 kg was in the RCS and 200 kg was in the containmnt. In later depressurization, between 7.5 and 14 hours, about 100 kg additional is believed to have been added to the containment. At the time of hydrogen burn, there might have been 150 kg in the RCS and 300 kg in the containment. The calculated amount burned, based on the peak overpressure, was 267 kg /4/. Ref. 3 estimated that there was 330-360 kg at the time of burn. Measurement of the hydrogen concentration on March 31 indicated about 80 kg at that time; so that the amount consumed from Ref. 3 would be 250-280 kg. The lower estimate of Ref. 4 would have given about 350 kg in containment, and hence about 100 kg in the RCS. The maximum production of Ref. 4, which is considered less likely, would give an RCS content of 270 kg.

These estimates subsume, that little hydrogen was produced during later depressurization. This is believed likely; even if some of the core was uncovered again, the rods exposed would already have been at least partially oxidized, and further oxidation would have been slow.

The estimated "most likely" amount remaining in the RCS, 100 kg, includes the amount in solution (about 26 kg at 1000 psi and 280°F) as well as that in a bubble (about 74 kg). At a pressure of 1000 psia and 280°F (typical of conditions during the several days following the accident) this would be 645 cubic feet. If we add about 1.6 lb-moles of fission gases and 3.2 lb-moles of helium to this, the total of all noncondensible gases in the bubble is 684 cubic feet at 1000 psia and 280°F (29,000 cubic feet at STP).

The largest amount considered for the RCS, 270 Kg, would give 244 Kg in the bubble, for a total volume of 2166 cubic feet at 1000 psia and 280°F (92,000 cubic feet at STP).

Bubble size calculations extrapolated back to 16 hours (see section 4) give a volume of 1470 cubic feet at 1000 psia. If the "most likely" hydrogen estimate is correct, this bubble volume would be about 44% hydrogen--the remainder could be any other gas, mostly steam. For example, a 786-cubic- foot bubble of steam in a "hot spot" within the damaged core would be possible. The maximum estimate of 270 Kg is impossible if the bubble size calculations are correct. This lends credence to the belief that the smaller quantity is more reasonable.

Based on the "most likely" quantities, the hydrogen accounting is then as follows:

Produced	--	450 kg
Released to containment	--	350 kg
Burned	--	270 kg
Remaining in containment	--	80 kg
Remaining in RCS at 16 hours	--	100 kg
In solution at 16 hours	--	26 kg
In bubble at 16 hours	--	74 kg

3. Calculation Of Bubble Size

The bubble size was calculated during the course of cooldown and bubble removal by Met Ed and B&W. The same physical principle -- the compliance of a liquid containing a gas bubble -- was invoked by each. After the accident, Sandia carried out an independent investigation /3/ at the request of the TMI/SIG. The results, as given in Figures 1 and 2, show that the bubble was about 1470 cubic feet at 2000, March 28 and was completely gone at 1800, April 1.

Although each organization has used the same basic principle, the equations appe r different, because different simplifying assumptions have been used.

The Met Ed formula [/5/] is the simplest. It neglects the compressibility and thermal expansion of water and the solubility of hydrogen. These simplifications lead to a consistent 300 cubic foot overprediction of the bubble size at 875 psi.

The B&W formula [/6/] includes these effects, but neglects changes in vapor mass in the pressurizer and the effect of the hemispherical lower head of the pressurizer, and does not consider the partial pressure of water vapor. The net result is generally about 5 percent underprediction of bubble size.

The Sandia formula/3/ includes all of these terms, but neglects the effect of leakage during bubble size experiments, the compliance of the steel vessel and change of density because of temperature change during experiments. The effect of the last two terms is known to be small. The leakage effect has not yet been evaluated but is also expected to be small.

Each bubble experiment was performed by subjecting the RCS to a known change in pressure and deducing the associated change in volume. From this the compliance of the liquid-gas system was calculated, and hence the size of the bubble. It should be noted that the size of the bubble changes because of two causes: because of compression of the gas, and because more of the gas goes into solution at the higher pressure. The latter effect is one of those that was neglected by Met Ed.

Even if an accurate formula is used which includes all the physical effects, the inherent inaccuracy of the measuring system would make an accurate prediction nearly impossible. One needs to measure small changes in volume corresponding to small changes in pressure in a very large system, using instruments which are not of laboratory quality.

3.1 Error Analysis

An error analysis of the Sandia formula has been carried out. The errors in bubble size are dependent on the conditions of the experiment and on the size of the bubble. Conditions for most of the bubble size experiments were approximately as follows:

RCS Pressure - 1000 psi

RCS Temperature - 280°F

Pressurizer Level - 250 in.

Makeup Tank Level - 45 in.

Makeup Tank Temp. - 81°F

At a bubble size of 500 cubic feet, the errors to be expected for errors in each of the measured quantities are:

RCS Pressure - 12.2 ft^3/psi error

RCS Temperature - 1.58 ft^3/degree F error

Pressurizer Level - 97.3 ft^3/inch error

Makeup Tank Level - 181.4 ft^3/inch error

Solubility - 4.43 ft^3/percent error.

Errors in each of the measured quantities could be as great as 2 percent of full range[7]. However, data are normally more accurate than this, and 2 percent of each reading is considered more likely. An error of 10 percent in solubility is considered reasonable. Then the possible total errors are:

error due to RCS pressure error = 244 ft^3

error due to RCS temperature error = 9 ft^3

error due to pressurizer level = 486 ft^3

error due to makeup tank level = 163 ft^3

error due to solubility = 44 ft^3

All errors would probably not occur simultaneously, and would not normally all have the same sign. Note, however, that the largest error - that due to pressurizer level error - is nearly as large as the bubble and several of the errors are large fractions of the bubble size. This clearly explains the great variability in bubble size estimates.

4. Bubble Removal

Except for changes in dissolved hydrogen due to changes in RCS pressure and temperatures, degassing at a constant rate of letdown would give a constant rate of bubble shrinkage. Figure 1 shows the results of bubble calculation with the Sandia formula, along with a least squares fit for removal rate. Also shown on this Figure is a removal rate calculated by B&W and a one-standard-deviation error band about the Sandia fit. Figure 2 shows the same data, except that the ordinate is total hydrogen in the RCS--that in the bubble and that dissolved in the coolant. The time of removal can be taken to be the intercept with the horizontal axis in Figure 1. The data show that the bubble had disappeared by 1800 on April 1, ±3 hours.

The removal of hydrogen was accomplished both by letdown and by pressurizer venting. It is not possible to estimate accurately the amount removed by each. However, from the fact that the hydrogen in the containment atmosphere increased by only a modest amount during venting, it can be assumed that venting was not the principal removal mechanism.

The removal rate by letdown is

$$\frac{dn_H}{dt} = \frac{1}{M_H}\frac{dm_H}{dt} = \frac{1}{M_w}\frac{dm_w}{dt}(N_{AR} - N_{AM}),$$

where $\frac{dn}{dt}$ is the molar letdown rate, moles/minute,

M is molecular weight, and

$\frac{dm}{dt}$ is the mass letdown rate

N_A is mole fraction of hydrogen in solution, and the subscripts H and w refer to hydrogen and water and R and M refer to RCS and makup tank conditions. The mole fraction in solution is, by Henry's law,

$$N_A = \frac{P}{K}*$$

where P_A is the partial pressure of hydrogen in the gas and K is the Henry's law constant. For RCS conditions $K = 9.3 \times 10^5$, and for makeup tank conditions K_z 11×10^5 /8/. These values are for 300°F and 75°F, the nearest tabulated points to 280°F and 80°F. The partial pressure of water vapor is taken to be equal to the saturation pressure at the indicated temperature. This is not strictly accurate but is within a few percent. The partial pressures of hydrogen are 933 psia* and 39.6 psia at total pressures of 1000 psia and 40 psia for the RCS and makeup tank. With these values, letdown removes 9.64×10^{-4} moles of hydrogen per mole water.

The letdown rate as given in post-accident notes was about 30 gpm, except for times when the letdown cooler was plugged. An average rate was probably about 25 gpm. This is a mole rate of 10.64 lb-moles of water of hydrogen per min, referred to RCS conditions, for the 94 hours of bubble removal. This would have removed 52.6 Kg.

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*Note that Dalton's law must hold for a bubble of mixed gases. If a separate bubble contains pure steam, Dalton's law cannot be applied to the total.

Leakage is estimated to be 5-6 gpm. It is assumed that all leakage is to reactor building conditions, where the partial pressure of hydrogen is so low as to be negligible in comparison with RCS conditions. The molar removal rate is then 0.001 moles hydrogen per mole water, and 5 gpm (again referred to RCS conditions) will remove 10.5 Kg in 94 hours.

The amount remaining (74-52.6-10.5), or 10.9 kg, could have been removed by pressurizer venting. This would cause only a 0.2% increase in containment hydrogen content. This explains why a marked increase in hydrogen content due to venting was not observed. Leakage should have caused an additional 0.2% increase in containment hydrogen content.

The amounts removed using the "most likely" original amount are:

Letdown	52.6 Kg	(71%)
Leakage	10.5 Kg	(14%)
Venting	10.9 Kg	(15%)
Totals	74 Kg	(100%)

It should be noted that no exotic or improbable mechanisms need to be invoked to explain the bubble disappearance.

5. Bubble "Hazard"

5.1 Oxygen Content

Assurance had been given as early as March 29 by a B&W scientist that no oxygen problem existed. This information was given to   and did not, apparently reach the NRC officials informing the public until much later.

On March 30, and 31, T.E. Murley contacted a number of specialists in response to a request by Roger Mattson. The responses are summarized in Reference 10. The early information given to Mattson was based on experiences from a BWR and from the Advanced Test Reactor (ATR) and hence was not applicable to a PWR, and certainly did not apply to the situation at TMI-2, in which the coolant had a large amount of hydrogen in solution. Other scientists questioned were unable to give definitive answers promptly.

Notes taken at the time at the NRC Emergency Center, including those by Mattson, do not indicate that anyone disagreed with the possibility of a hydrogen-oxygen explosion. Among those queried on the effects and probability of explosion was B&W. The only note found to indicate mild disagreement is the record of a conversation with B&W to the effect that "[B&W] feels that H_2 recombination is taking place under gamma flux." There are notes indicating that other experts basically agreed with the estimates of oxygen production. On April 1, the word from B&W was that "[B&W official] <u>thinks</u> not flammable."

The opinion was almost universal that the bubble would be explosive, either very soon or in a matter of some days.

On April 1, other data began to be received that contradicted the belief that the bubble contained oxygen. In the meantime, however, other scientists had been asked about the possibility of explosion, and still others were delivering opinions on the damaging effects of explosions. Highly vocal comments were received from a number of supposed experts suggesting, for example, that hydrogen would combine with zirconium to form zirconium hydride. It was difficult to sort out the facts in the confusing melange of differing opinions.

In view of the disagreement by the experts, the following summary was prepared on April 1:

Flammability Limit	5% O_2 in pure H_2
O_2 Production Rate	1% per day
Current O_2 concentration	5%
Detonation Limit	12% O_2 in pure H_2

Emergency Center notes for April 1 show that information that there was no oxygen was increasingly being received. On April 2 virtually all incoming information was that there was no oxygen.

A wide cross-section of experts was involved; NRC staff, National Laboratories, NRC contractors, Department of Energy Laboratories, the academic community, and reactor manufacturers. At some time on April 1, the weight of opinion was that oxygen was probably not present. Even then, however, explosion and structural experts -- who had not yet been advised of the latest findings -- continued to give opinions on the hazard of explosions.

5.2 Explosive Hazard in Reactor Vessel

A number of computations of the effect on the reactor vessel of a hydrogen detonation, given that an explosive mixture existed (which was physically impossible). These calculations, of which/9/ is typical, generally showed that major damage to the reactor vessel was unlikely, although some showed that the strength of the upper head might be marginal. Generally, specialists in explosive damage would be unable to predict the effects on the basis of such calculations without experiments.

Of equal interest is whether fragments of the reactor vessel could have been propelled with sufficient velocity to breach the containment. Specialists are generally agreed that this is so improbable that it can be virtually ruled out, especially since any explosive fracture would be highly unlikely.

Since there was no possibility of an explosive mixture being formed, the whole question is academic, and it can be concluded that there was explosive hazard.

Considering the lack of unanimity on March 31, the decision to consider the bubble potentially explosive was correct. In the face of contradictory opinions, it is proper to give consideration to the worst case.

5.3 Explosive Hazard in Containment

A more realistic hazard was the possibility of sudden depressurization, with release of the hydrogen from the RCS to the containment. This was unlikely, but possible. If the entire inventory of hydrogen had been added to the containment, an explosive mixture might have been formed.

Analysis of the containment atmosphere on March 31 showed 1.7% H_2, 15.7 O_2, and 82.6% n_2 for one sample, and 1.7% H_2, 16.5 O_2, and 81.8% N_2 for another. At a temperature of 80°F, and pressure of 14.3 psia, the latter would be 86.1 lb-moles H_2, 835.9 lb-moles O_2, and 4144 lb-moles N_2. The addition of all the hydrogen in the RCS--100 kg or 110 lb-moles--would raise the hydrogen concentration to 3.8%. This is still below the flammable limit. However, if the entire bubble were hydrogen, there would be an addition of 185 lb-moles. This would give a hydrogen concentration of 5.2% which could be flammable. However, it should be noted that the burning of about 290 lb-moles on March 28 did not damage the containment. Therefore, the burning of 270 lb-moles or less on March 31 likewise would not damage the containment.

6. Summary

The most likely estimate for hydrogen production is 450 kg, equivalent to oxidation of approximately 50% of the cladding. It is possible that the amount produced could have been as great as 520 kg. A total gas volume of 1470 cubic feet was probably present in the RCS at 2000, March 28. The fraction of hydrogen in this bubble or bubbles could have been 44-100%. The hydrogen was removed from the bubble by letdown, leakage, and venting; no unusual mechanisms need to be hypothesized to account for bubble removal.

The variability in estimates of bubble size came from different methods of computation being used by different organizations, and from the inherent inaccuracy in the method of measurement. The bubble disappeared about 1800 on April 1.

There was no oxygen in the bubble, and therefore no possibility of explosion. The incorrect perception of an explosion hazard stemmed from contradiction among supposed experts. This perception was known or should have been known to be false by some time on April 1.

A flammable mixture in containment, due to release of all the hydrogen, would have been possible, but very unlikely. Even if it had occurred, the containment would not have been damaged.

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2. R.K. Cole, Generation of Hydrogen During the First Three Hours of the Three Mile Island Accident, NUREG/CR-0913, SAND 79-1357, Sandia Laboratories, July 1979.

3. Letter from R.K. Cole, Sandia Laboratories, to M. Picklesimer, NRC/TMI/SIG, dated October 15, 1979, with attachments.

4. Technical Staff Analysis Report -- Summary, p. 4-3, President's Commission on the Accident at Three Mile Island, October 1979.

5. Metropolitan Edison, "Bubble Size Calculations," various dates.

6. Letter from James H. Taylor, Manager of Licensing, B&W, to John Bickel, ACRS, July 20, 1979; Subject: "TMI-2 Incident, Gas Bubble Volume Estimate."

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10. Memorandum for Files, T.E. Murley, "Record of Actions, Three Mile Island Accident, March 28-April 6," U.S. NRC (RES), June 1979.

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QUALIFICATION OF INSTRUMENTATION AND PLANT DATA

1. Environmental Qualification of Instrumentation

Instrumentation within the reactor building, if part of the RPS or SFAS, ~~are~~ is required to meet the following environmental conditions:[1, 2]

Normal Conditions	Post Accident Conditions
40-120°F, atmospheric pressure, 40%-70% relative humidity, and 25 mR/hr.	286°F, 51.3 psig, 100% humidity, and total integrated radiation exposure 2 x 10^4 roentgens (24-hour operability).

Cables are generally qualified to higher environmental requirements. There is no reason to believe that the TMI environment should have damaged the RPS and SFAS systems in the first day of the accident.

Other instrumentation has been classified according to whether it is required for safety or not required for safety. The former category includes instrumentation required for safe shutdown and accident monitoring instrumentation. Some of the instrumentation in each subclass is also contained in the RPS and SFAS, and as such, is qualified to the environmental extreme shown above. Accident monitoring instrumentation (Table I) is also designed to operate in the post-accident environment. However, instrumentation required for safe shutdown (Table II) ~~was~~ is not required to be qualified to these conditions, unless it also forms part of the RPS or SFAS.

The most vital data are received from accident monitoring instrumentation. However, it is clear that systems and controls designed for safe shutdown are also vital for post-accident management. In addition, there is a clear need for instrumentation to enable the plant to be maintained in a stable, safe condition after shutdown.

The least severe qualifications are required of instrumentation "not required for safety." This category includes automatic reactor coolant pressure control, pressurizer temperature measurement, automatic pressurizer level control, the integrated control system, and the control rod drive control system.

It should be noted that pressurizer heaters in the automatic mode are included under "systems not required for safety." However, in the event of failure of the automatic reactor coolant pressure control system, pressurizer heaters in the "manual" mode are not included in any category.

2. Limits of Operability

The ranges of capability of instrumentation systems are shown in Table III. The ranges of indication available to the operators are shown in Table IV.

Small excursions past the limits of operability should not damage instrumentation systems. However, excursion past the indicating limits means that the information will not be available to the operators.

3. Acceptability of Plant Data

The acceptability of data depends on a number of factors, some of which are subjective and difficult to quantify. Sensors, signal conditioning equipment, and data display and recording devices are all subject to inherent error. In addition to the error in equipment which is nominally in good working order, there is a problem of reliability; that is, some items are prone to malfunction. There is also a perceived reliability, which could vary from the actual reliability. Finally, there is the question of utility; vitally needed data might be accepted even if the accuracy and reliability could not be guaranteed.

3.1 Accuracy (Some of this section may be changed when calibration data are received from TMI)

Required accuracies of broad classes of data are specified in the FSAR.[1] It can be expected that instrumentation in good repair will always fall within the accuracy limits shown in Table I.

Periodic calibrations are performed to ensure that instrumentation accuracy falls within the limits set by the FSAR. The frequency of calibration for the reactor protection system, engineered safety features actuation system, and monitoring instrumentation is covered in the Technical Specifications.[3] Surveillance Procedures provide further limits on frequency of calibration and allowable tolerance; these are at least as restrictive as the FSAR and Tech Specs. For example, steam generator pressure is required by the FSAR to be within a tolerance of ±2% of full scale (± 24 psi). However, the Surveillance Procedure[4] requires actual calibration to be within a tolerance of ± 13.4 psi. The reason for the more restrictive requirements of the Surveillance Procedures is that instrumentation can "drift" between calibrations. By requiring greater accuracy at the time of calibration, it is hoped that the FSAR tolerance will not be exceeded at any time. In addition, each item of instrumentation equipment (sensor, signal conditioner, indicator, or recorder) is covered by an "MTX Data Sheet." The Data Sheets give manufacturer's tolerances, which can be more restrictive yet.

Errors found at the most recent calibration are given in Table VI. It will be seen (at least immediately after calibration) that the requirements of the FSAR were met.

Two sources of error are not covered either by specifications or calibrations; reading error and chart timing error.

Reading error of charts or meters is governed by the width of the recording or indicating band and by the fineness of graduations. A chart on which the recording is spread out over a wide band can obviously be read with greater accuracy than one on which the reading is tightly crowded into a narrow space. Likewise, finely graduated charts or meters can be read with higher accuracy than coarsely graduated charts. However, coarse graduations can often be more easily read "at a glance."

As a rule of thumb, it is estimated that reasonable reading accuracy to 1/2 the finest graduation is possible; however, on a few very finely graduated charts accuracy only to the finest whole graduation is considered reasonable. Table VI shows achievable reading accuracy for a number of stripcharts. It will be seen that each channel should be readable to an accuracy at least as good as the specified instrument accuracy.

Chart timing error should be easy to assess. It ought to be possible to read to 0.1 in. accuracy; at the most common chart speeds (2 in/hr and 1 in/hr) the reading error would not exceed 3-6 minutes. However, the following improper practices have been found at TMI-2:

1. Time-of-day not accurately or clearly marked.
2. Charts translated without new markings.
3. Chart speed does not match speed written on chart.
4. Insufficient fiducial time markings.
5. Chart speed obviously changed during recording.

Because of these improper practices, the only way that timing can be read with any confidence is to locate two known events and measure the distance between them. Even this gives no assurance that the chart has not been tampered with between events. Time can be established on a few charts with an accuracy of ± 3 min. However, as a general rule, ± 12 min. or even greater must be considered representative.

These remarks do not apply to reactimeter data. There is no possibility of an amplitude error other than the instrument channel error, and time can be matched to within a few seconds. Therefore, in attempting to match reactimeter data to strip-chart data, disagreements are resolved in favor of the reactimeter.

3.2 Reliability of Data

Data channels which had given trouble in the past would probably be perceived as less reliable than those which had operated without difficulty. From a sample of 45 incidents reported in the "Instrument Out of Service Log," 42% were alarms, 33% were radiation monitors, 13% were temperature channels, 4% were pressure channels, and the remainder were equally divided among level, flow, and electrical channels. It is probable that alarms and radiation monitors would be perceived as less reliable than other data. There seem to have been slightly more problems with temperature channels than with some others; this is not likely to be significant, given the small size of the data sample.

Past operation of the data channels cannot give much information on the actual (as opposed to perceived) reliability. Conditions during the accident -- temperature, humidity, radiation, etc. -- were much more challenging than at any time in the past history of the plant. For example, the peak temperature measured on the incore thermocouples (2580°F) is near to the liquidus temperature of the Inconel sheaths (2600°F). Melting of junctions and rewelding of false junctions is a distinct possibility. As another example, voiding of the pressurizer reference leg due to evolution of dissolved hydrogen is possible. Degradation of insulation due to high temperature, humidity and radiation in the reactor building could have caused false readings. Whether environmental extremes caused misperformance of instrumentation can only be a matter of conjecture. Even if a channel is found to be inoperative in a postmortem examination, it will not usually be possible to determine at what time the failure occurred.

Perceived reliability is probably lower for out-of-range channels. The plant computer uses the same symbol for data out of range as for bad data. It is generally not possible to determine whether out-of-range data are correctly indicated without access to other information.

High reliability can be ascribed to data which is confirmed from an independent source. Redundant reactimeter and stripchart data generally tend to be confirmatory, although the low accuracy and poor legibility of some of the stripcharts makes comparison difficult. Particularly, RCS pressure and temperature data appear to be well confirmed. Conversely, PORV block valve opening and closing times cannot be unequivocally confirmed.

Estimates of data reliability are given in Table VII.

The most vitally important information would have been core water inventory. Lacking this information, the operators depended on an inappropriate substitute, pressurizer level. Dependence on pressurizer level actually caused incorrect actions to be taken. Similarly, the lack of emergency feedwater flow indication caused the operators to seize on a set of substitutes - discharge pressure, "eleven-valve" opening and steam generator level. The set of substitutes did eventually lead to the correct conclusion, but only after a considerable delay.

Perhaps nearly as important as the lack of some needed data was a confusing excess of unneeded information. As an example, one of the factors in the alarm printer falling behind was the great number of alarms relative to the feedwater heaters. These were not germane to the situation, and suppression would have helped clear the computer for more useful tasks.

Utility of data was compromised not only by the absence of some useful data, but also by less than optimum display. There was little consideration given to grouping of the most useful data for accident management, and some display devices are difficult to read.

3.4 Utility of Data for Historical Reconstruction

For a reconstruction of the accident sequence, additional data would have been useful. This is especially true of understanding the motivation for actions taken. A voice recording of operator discussions would probably have been helpful for this task.

The improper practices with regard to stripchart marking have hindered reconstruction. Training in the importance of correct marking and stricter administrative control should ensure better marking practices. Also, some consideration should be given to historical reconstruction when selecting the channels to be recorded.

Accident reconstruction would also have been aided by more complete data recording on tape. The reactimeter data were very helpful, but would have been even better if the entire range of each channel had been recorded and if the data channels had been specifically selected for accident analysis. Postmortem analysis would be easier - and better - if a similar recording device were dedicated to analysis of accidents and other off-normal occurrences.

4. Experience at TMI-2

Conditions of high humidity and radiation have continued at TMI-2 ever since the accident, although the humidity is probably now well below 100%. There has also been considerable flooding by water which is far from pure. The possibility of cables being under water - or worse, in a conducting solution - for a matter of months was not considered in design. The total integrated radiation qualification of many systems may have been exceeded. However, no failures have been ascribed to this condition alone. It is clear, however, that the requirement that systems be operable for 24 hours in an accident environment is far too lenient.

4.1 Instrumentation and Control Failures

The most notable failure was, of course, the pilot operated relief valve, RC-R2. It should be noted that this valve is at the bottom of the list of importance ("not related to safety") and hence was qualified to the most lenient criteria. The same is true of pressurizer heaters, which had a history of tripping repeatedly.[5]

Pressurizer level indicators also failed. These are considered to be "accident monitoring instrumentation," and as such are designed for the post-accident environment. The first such failure occurred at 2114, March 29, 1979.[6] This was more than 24 hours after accident initiation, and hence does not - at least technically - demonstrate a lack of compliance with the environmental qualifications.

Some incore thermocouples appear to have been damaged in the accident. These were considered "not related to safety" and hence would not be expected to necessarily survive environments more severe than normal operation. No matter what category these instruments had been placed in, the ferociously severe core environment probably would have damaged them. Temperatures of 3000°F and higher would be most challenging for any instruments in the present state of the art. The same is true of self-powered neutron detectors. However, consideration should be given to always installing thermocouples as a matter of course, and ~~so~~ to protecting leads from high temperatures to the maximum extent possible.

5. Conclusions

The RPS and SFAS systems, and to some extent, accident monitoring systems, are environmentally qualified for post-accident environments. Systems required for safe shutdown are not so qualified. Pressurizer heaters in the "manual" mode do not appear to come under any instrumentation category. No category is established for instrumentation required to maintain stable conditions after shutdown. Existing qualifications call for 24-hour operation in the accident environment.

Accuracy of instrumentation appears, from pre-accident calibration, to be adequate. Poor control room practices have resulted in difficulties in chart reading.

Reliability of alarms and radiation monitors might have been perceived to be lower than other data channels. Considerable confidence can be placed on most RCS parameters, within the specified accuracy, and subject to the remarks on chart timing error. PORV block valve opening and closing times cannot be reliably determined.

Utility of data for operation was compromised both by the lack~~s~~ of some vital data, and by a confusing superfluity of low priority data. Little thought appears to have been given to the utility of data for historical reconstruction of the accident. Inappropriate substitutes were used for unavailable data.

Failures of instrumentation can be ascribed to too lenient environmental qualification, to flooding for which instrumentation was not qualified, and to qualification for too short a time.

6. Recommendations

Many of the recommendations made in the following sections are covered in Revision 2 to Reg Guide 1.97[6] and Draft ANS 4.5.[7] Therefore, if these guides are adopted, the recommendations marked with an asterisk (*) will be superfluous.

6.1 Specification of Environmental Qualifications

Operation of cables and some sensors after flooding should be considered. The time for post-accident operation should be lengthened.*

6.2 Categories of Systems

Accident monitoring and safe shutdown systems should be qualified to full accident conditions.* In addition, a category for "systems required to maintain the plant in a stable condition" should be established, with qualification to full accident conditions.* Careful review of instrument and control systems should be carried out to make sure that items like pressurizer heaters do not get left out or are placed in improper categories.

6.3 Accuracy and Reliability of Data

Administrative review of instrument repair records is necessary so that unreliable systems will be upgraded. Stricter control on stripchart marking should be instituted:

6.4 Utility of Data

Data presented to the operators should be reviewed to make sure that important data are continuously available.* Consideration should be given to layout so that important data can be readily assimilated without being diluted by less important displays.*

Recording devices dedicated to historical reconstruction of accidents or off-normal incidents ~~institutes~~ should be installed. Control room voice recorders, magnetic tape or disk recording of important parameters, and dedicated stripcharts are examples.

References

1. Final Safety Analysis Report, Three Mile Island Nuclear Station - Unit 2, Metropolitan Edison Co., Jersey Central Power and Light Co., and Pennsylvania Electric Co., Docket No. 50-320.

2. IEEE Standard No. 279-1968.

3. U.S. NRC, Three Mile Island Nuclear Station Unit 2 Technical Specifications, NUREG-0432, 1978.

4. "Three Mile Island Nuclear Station Unit 2 Surveillance Procedure 2302-R28", Metropolitan Edison Co., July 1978.

5. TMI-2 Alarm Printer, March 28, 1979.

6. U.S. NRC, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," Reg Guide 1.97, Rev. 2. (Proposed).

7. American National Standard "Functional Requirements for Post Accident Monitoring Capability for the Control Room Operator of a Nuclear Power Generating Station," ANS-4.5 (Draft), September 1979.

TABLE I

ACCIDENT MONITORING INSTRUMENTATION

End-Point Parameters	Recognize Accident Condition	Functioning of Mitigating Equipment	Follow Course of Accident/ Transient	Required for		
				LOCA		
				Large	Small	Transient
ESF busses energized		X		X	X	X
Pressurizer Level	X		X			X
SG Press.	X		X			X
R.C.Press. (wide range)	X	X	X	X	X	X
R.C.System Flow	X					X
Containment Press.	X	X	X	X	X	X
Emer. Feed Press.		X				X
Containment Isolation		X		X	X	X
Area Rad. Monitor & Grab Sampling	X		X	X	X	
R.C.Temp. hot/cold	X		X		X	X
D.H.Cooler Outlet Temperature			X	X	X	X
D.H.Pump Suction Temp.			X	X	X	X
HPI Flow		X	X	X	X	X
LPI Flow		X	X	X	X	X
BWST Switch-over Valves		X	X	X	X	

TABLE I - continued

End-Point Parameters	Recognize Accident Condition	Functioning of Mitigating Equipment	Follow Course of Accident/ Transient	Required For		
				LOCA		
				Large	Small	Transient
Feed Latch (valve indication)	X	X				X
H_2 Content (grab sample)			X	X	X	
SG Level (Startup & Operate Range			X		X	X
Reactor Bldg. Spray Pump Flow		X		X	X	X
Pressurizer Electromatic Relief		X				X

TABLE II

SYSTEMS REQUIRED FOR SAFE SHUTDOWN

Control Rod Drive Control System

Makeup Pump Control

Letdown Line Isolation Valve Control

BWST Suction Valve Control

EFW Control

Pressurizer Spray Valve Control

Electromatic Relief Valve Control

Decay Heat Removal System Controls

Nuclear Services Closed Cooling Water System

Nuclear Services River Water System

Supporting Systems (Electrical, Air, etc.)

TABLE III

SYSTEM RANGES

Item	System Desig.	Ind. Type	Range
Reactor Coolant Temperature	RC-5A-TE3 (Cold Leg)	Indicator	50-650 F
	RC-5A-TE-4 (Cold Leg)	Indicator	50-650 F
	RC-5B-TE3 (Cold Leg)	Indicator	50-650 F
	RC-5A-TE4 (Cold Leg)	Indicator	50-650 F
	RC-15A-TE1 (Hot Leg)	Recorder	0-800 F
	RC-15A-TE2 (Cold Leg)	Recorder	0-800 F
	RC-15A-TE3 (Cold Leg)	Recorder	0-800 F
	RC-15B-TE1 (Hot Leg)	Recorder	0-800 F
	RC-15B-TE2 (Cold Leg)	Recorder	0-600 F
	RC-15B-TE3 (Cold Leg)	Recorder	0-800 F
Reactor Coolant Pressure (SFAS Input)	RC-3A-PT3	Recorder	0-2500 psig
	RC-3A-PT4	Indicator	0-2500 psig
	RC-3B-PT3	Indicator	0-2500 psig
Pressurizer Level	RC-1-LT1	Indicator	0-400 in.
	RC-1-LT2	Recorder	0-400 in.
	RC-1-LT3	Recorder	0-400 in.
Pressurizer Temperature	RC-2-TE1	Indicator	0-700 F
	RC-2-TE2	Indicator	0-700 F
OTSG 'A' Level	SP-1A-LT1	Indicator	0-600 in.
	SP-1A-LT2	Recorder	96-388 in.
	SP-1A-LT3	Recorder	96-388 in.
	SP-1A-LT4	Indicator	0-250 in.
	SP-1A-LT5	Indicator	0-250 in.
OTSG 'B' Level	SP-1B-LT1	Indicator	0-600 in.
	SP-1B-LT2	Recorder	96-388 in.
	SP-1B-LT3	Recorder	96-388 in.
	SP-1B-LT4	Indicator	0-250 in.
	SP-1B-LT5	Indicator	0-250 in.

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TABLE IV

INFORMATION READOUTS AVAILABLE TO THE OPERATOR FOR MONITORING CONDITIONS IN THE UNIT

	Measured Parameters	Total No. Reqd. Ch.	No. of Ch. Available (1)	No. of Sensors in a Channel	Types of Readouts	No. of Readouts (2)	Indicator Range	Indicator Accuracy (3)	Indicator Location	PURPOSE OR/USAGE
1.	Source Range Neutron Level	1	2	2	B, F	3	10^{-1} to 10^{+6} cps	± 3	A, B, D	A
2.	Source Range Start-up Rate	1	2	2	A, F	3	-1 to 10 dpm	± 3	A, B, D	A
3.	Intermediate Range Neutron Level	1	2	2	B, F	3	10^{-11} to 10^{-3} amp	± 3	A, B, D	A, B
4.	Intermediate Range Start-up Rate	1	2	2	A, F	3	-1 to 10 dpm	± 3	A, B, D	A
5.	Power Range Neutron Level	3(2) ****	4 4	4	A, F	3	0 to 125% FP	± 2	A, B, D	A(B)
6.	Power Range Neutron Level Imbalance	3(2) ****	4	4	A, F	3	-62.5 to 62.5% FP	± 2	A, B, D	A(B)
7.	R.C. Loop Outlet Temp.	2(1/Loop)	6(3/Loop)	3/Loop	A, E, F	4/Loop	520F-620F	± 2	B, C, D	B
8.	R.C. Unit Outlet Temp.	*	*	*	E	1	520F-620F	± 2	B	
9.	R. C. Loop Inlet Temp. (Narrow Range)	2(1/Loop)	4(2/Loop)	4/Loop	A, E, F	4/Loop	520F-620F	± 2	B, D	B
10.	R. C. Loop Inlet Temp. (Wide Range)	2(1/Loop)	4(2/Loop)	2/Loop	A, F	2/Loop	50F-650F	± 2	B, D	B
11.	R. C. Unit T_c	*	*	*	A	1	520F-620F	± 2	B	
12.	R. C. Loop Avg. Temp.	*	*	*	A	1/Loop	520F-620F	± 2	B	
13.	R. C. Unit Avg. Temp.	*	*	*	E	1	520F-620F	± 2	B	
14.	R. C. Loop Temp. Diff.	*	*	*	A	1/Loop	0-70F	± 2	B	
15.	R. C. Unit ΔT_c	*	*	*	A	1	-10F to +10F	± 2	B	
16.	R. C. Loop Pressure (Wide) (Narrow)	1	1	1	A, E A, E, F	2	0-2500 psig 1700-2500 psig	± 2	A, B, C A, B, D	B
17.	Pressurizer Level	1	3	3	A, E, F	3	0-400" H_2O	± 2	A, B, C, D	B
18.	Pressurizer Temp.	1	2	2	A, F	2	0-700 F	± 2	B, D	B
19.	R. C. Loop Flow	2(1/Loop)	2(1/Loop)	2/Loop	A, F	2/Loop	0-90x10^6 lb/hr.	± 3	A, B, D	B

TABLE IV - continued

	Measured Parameters	Total No. Reqd. Ch.	No. of Ch. Available (1)	No. of Sensors in a Channel	Types of Readouts	No. of Readouts (2)	Indicator Range (3)	Indicator Accuracy	Indicator Location	PURPOSE OR USAGE
20.	R. C. Total Flow	*	*	*	E	1	0-180X10^{6}lb/hr.	$\pm$ 3	A, B, E	
21.	Steam Gen. Full Range Level	2 (1/Loop)	2(1/Loop)	1/Loop	A, F	2/Loop	0-600" H_2O	$\pm$ 2	B, D	B
22.	Steam Gen. Start-Up Range Level	2 (1/Loop)	2(1/Loop)	2 /Loop	A, F	3/Loop	0-250" H_2O	$\pm$ 2	A, B, C, D	B
23.	Steam Gen. Operate Range Level	2 (1/Loop)	2(1/Loop)	2 /Loop	E, F	2/Loop	0-100%	$\pm$ 2	B, D	B
24.	Emergency F.W. Status	2 (1/Loop)	1/Loop	1/Loop	C, F	2/Loop	-	-	B, D	
25.	Emergency F.W. Press.	2 (1/Pump)	2(1/Pump)	1/Pump	A	1/Pump	0-100%	$\pm$ 2	B, E	B
26.	Containment Pressure (RPS) (SFAS)	2 2	4 3	1	A,E,F	3	0-100 psig (-5 psig to -10 psig)	$\pm$ 1	B	
27.	Containment Isolation Status	4	4	1/valve	C	1/valve	-	-	B	
28.	Containment Temp.	*	*	20	A	1	0-300 F	$\pm$ 2	B	
29.	Steam Gen. Outlet Press.	2(1/Loop)	4(2/Loop)	2/Loop	A, E, F	3/Loop	0-1200 psig	$\pm$ 2	A, B, C, D	B
30.	Steam Temperature	2(1/Loop)	4(2/Loop)	2/Loop	A, F	2/Loop	100F-650F	$\pm$ 2	B, D	B
31.	Start-up F.W. Flow	2 (1/Loop)	2(1/Loop)	1/Loop	A, E, F	3/Loop	0-1.5X10^{6}lb/hr.	$\pm$ 2	B, D	A
32.	Main F.W. Flow	2 (1/Loop)	4(2/Loop)	2/Loop	A, E, F	3/Loop	0-6.5X10^{6}lb/hr.	$\pm$ 2	B, D	B
33.	Feed Water Temperature	2 (1/Loop)	4(2/Loop)	2/Loop	A, F	2/Loop	0-500F	$\pm$ 2	B, D	B
34.	Nuclear Services River Water Pump Discharge Pressure	1/Pump	4(1/Pump)	1/Pump	A, F	3(1)/Pump	0-100 psig	$\pm$ 1	B, D, E	B
35.	N.S. River Water Pump-Motor Amps.	1/pump	4(1/pump)	3/Channel	A,F	12 (Total)	0-100 Amps	$\pm$ 1	B, D, E	B
36.	N.S. River Water Hdr. Temperature	1/Hdr	2(1/Hdr)	1/Hdr	A, F	3(1)/Hdr	20-220F	$\pm$ 1	B	B
37.	N.S. Cooler Outlet Temperature	1/Cooler	2(1/Cooler)	1/Cooler	A, E, F	3(1)/Cooler	20-220F	$\pm$ 1	B, D, E	

POOR ORIGINAL

TABLE IV - continued

	Measured Parameters	Total No. Reqd. Ch.	No. of Ch. Available (1)	No. of Sensors In	Types of Readouts	No. of Readouts (2)	Indicator Range	Indicator Accuracy (3)	Indicator Location	PURPOSE OR/USAGE
38.	Decay Heat Closed System Service Cooler River Water Outlet Temp.	1/Cooler	2(1/Cooler)	1/Cooler	A, E, F	3(1)/Cooler	20-220F	± 1	B, D, E	B
39.	Nuclear Services River Water Pump Disch. Hdr. Pressure	1/Hdr	2(1/Hdr)	2/Hdr	A, F	3(1)/Hdr.	0-100 psig	± 1	B, D, E	B
40.	Decay Heat Service Cooler Cooling Water Inlet Temperature	2 (1/Cooler)	1/Cooler	1/Cooler	A, F	3(1)/Cooler	20-220F	± 1	B, D, E	
41.	Decay Heat Service Cooler Cooling Water Outlet Temperature	1/cooler	1/cooler	1/cooler	A	1/cooler	20-220F	± 1	B	B
42.	Decay Heat Closed Cooling System Disch. Pressure	2 (1/Pump)	4(2/Pump)	1/Pump	A	2(1)/Pump	0-100 psig	± 1	B, E	B
43.	Decay Heat Closed Cooling Surge Tank Level	1/Tank	2(1/Tank)	1/Tank	A	2(1)	0-5' 6"	± 1	B, E	B
44.	Nuclear Services Closed Cooling Pump Suction Hdr. Pressure	1	1	1	A, F	2	0-30 psig	± 1	B, D	B
45.	Nuclear Services Closed Cooling Pump Disch. Hdr. Pressure	1	1	1	A, F	2	0-100 psig	± 1	B, D	B
46.	Nuclear Services Closed Cooling Service Coolers Inlet Temperature	1	1	1	A, F	2	20-150 F	± 1	B, D	B
47.	Nuclear Services Closed Cooling Service Coolers Outlet Temperature	1/Cooler	2(1/Cooler)	F	2(1)/Cooler		20-150F	± 1	D, E	B
48.	Nuclear Services Closed Surge Tank Level	1	1	1	A	2 (1)	0-8' 0"	± 1	B, E	B
49.	Core Flooding Tank Level	2/Tank	4(2/Tank)	1/Channel	A, F	2/Channel	0-14' 0"	± 2	B, D	
50.	Core Flooding Tank Pressure	4(2/Tank)	4(2/Tank)	1/Channel	A	1/Channel	0-800 psig	± 2	B	A

POOR ORIGINAL

TABLE IV - continued

Measured Parameters	Total No. Reqd. Ch.	No. of Ch. Available (1)	No. of Sensors in	Types of Readouts	No. of Readouts (2)	Indicator Range	Indicator Accuracy (3)	Indicator Location	PURPOSE OR/USAGE
51. Make-Up Pump Suction Hdr. Pressure	1	1	1	A	1	0-100 psig	± 1	B	B
52. High Pressure Injection Flow	1/Loop	4(1/Loop)	1/Loop	A	1/Loop	0-600 gpm	± 2	B	B
53. Decay Heat Removal Reactor Outlet Temp.	1/Loop	2(1/Loop)	1/Loop	A	1/Loop	0-350F	± 2	B	B
54. Decay Heat Removal Pump Discharge Pressure	1/Pump	1/Pump	1/Pump	A	1/Pump	0-600 psig	± 2	E	B
55. Decay Heat Removal Flow	1/Loop	1/Loop	1/Loop	A	1/Loop	0-5000 gpm	± 3	B	B
56. Borated Water Storage Tank Temperature	1	1	1	A	1	0-200F	± 2	B	
57. Borated Water Storage Tank Level	2	2	1	A, F	3	0-56' 0"	± 2	B, D	
58. Sodium Hydroxide Storage Tank Level	1	1	1	A	2 (1)	0-50' 0"	± 2	B, E	
59. Sodium Hydroxide Storage Tank Temp.	1	1	1	A	1	0-200F	± 2	B	
60. Decay Heat Removal System Cooler Outlet Temperature	1/Cooler	2(1/Cooler)	1/Cooler	A	1/Cooler	0-300F	± 2	B	B
61. Spent Fuel Cooling Pump Discharge Pressure	1/Pump	2(1/Pump)	1/Pump	A	2 (1)/Pump	0-160 psig	± 2	B, E	B
62. Spent Fuel Water Cooler Outlet Temperature	1/Cooler	1/Cooler	1/Cooler	A, F	3 (1)/Cooler	0-250F	± 2	B, D, E	B
63. Spent Fuel Pool Temp.	2	1	1 /Channel	A, F	2(1)	0-200F	± 2	B, D, E	
64. Spent Fuel Surge Tank Level	1	1	1	A	1	0-40"	± 2	B	B
65. Borated Water Pump Discharge Pressure	1	1	1	A	2 (1)	0-160 psig	± 2	B, E	B
66. Spent Fuel Cooling Flow to Demineralizer	1	1	1	A	2 (1)	0-250 gpm	± 2	B, E	B
67. Area Gamma Monitors	20 (20 monitors)	20(1/Monitor)	1/Monitor	B, E	**	0.1 to 10^4 mr/hr.	± 2 of set	A, B	B

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TABLE IV - continued

	Measured Parameters	Total No. Reqd. Ch.	No. of Ch. Available(1)	No. of Sensors In	Types of Readouts	No. of Readouts(2)	Indicator Range	Indicator Accuracy(3)	Indicator Location	PURPOSE OR/USAGE
68.	Reactor Building Dome Monitor	1	1	1	B, E	3	10^3 to 10^8 mr/hr.	± 2 of set point	A, B	B
69.	Atmosphere Monitors (Particulate, Iodine and Gas)	12 ***	12(1/Monitor)	1/Monitor	B, E	**	10^1 to 10^6 counts per minute	± 2 of set point	A, B	B
70.	Gas Monitor	4	4(1/Monitor)	1/Monitor	B, E	**	10^1 to 10^6 counts per minute	± 2 of set point	A, B	B
71.	Liquid Monitor	10	1/Monitor	1/Monitor	B, E	**	10^1 to 10^6 counts per minute	± 2 of set point	A, B	B
72.	Failed Fuel Detector (Gamma and Liquid)	1	1	1	B, E	3	10^1 to 10^6 counts per minute	± 2 of set point	A, B	B

Legends: **Type of Readout**
A Linear Scale Indicator
B Log Scale Indicator
C-Indicator Light
D Digital Indicator
E Recorder
F-Plant Computer Output

Indicator Locations
A-System Cabinets
B-Control Room
C-Local Auxiliary Panels
D-Plant Computer Printout
E-Local

PURPOSE OR/USAGE
Blank- Information only
A- Total number of channels required for unit start-up according to Tech. Specs.
B- Total number of channels considered to be essential for safe, normal operation

NOTES: (1) Number of transmitters which are fed by the sensors providing the signal to the instrument string.
(2) Number in parenthesis indicates the number of local indicators with no electrical channel.
(3) Accuracy as a percent of full scal
*Two or more signals combined to produce indicated parameter.
**Multiple readouts (more than 3)
***Includes 2 portable monitors
**** Assumes one channel in bypass

TABLE V

ACCURACY REQUIRED BY FSAR

Parameter	Range	Accuracy, % of Range	Accuracy, in Units
RC Outlet Temp. NR*	520 - 620°F	± 2	± 2°
RC Inlet Temp. NR*	520 - 620°F	± 2	± 2°
RC Inlet Temp. WR*	50 - 650°F	± 2	± 12°
Loop ΔT	0 - 70°F	± 2	± 1.4°
Loop Press. WR	0 - 2500 psig	± 2	± 50 psi
Loop Press. NR	1700 - 2500 psig	± 2	± 16 psi
Pressurizer Level	0 - 400 in.	± 2	± 8 in.
Loop Flow	0 - 90×10^6 LB/HR	± 3	± 2.7×10^6 LB/HR
Startup Range	0 - 250 in.	± 2	± 5 in.
Operate Range	0 - 100%	± 2	± 2%
RB Press.	0 - 100 psig	± 1	± 1 psi
RB Temp.	0 - 300°F	± 2	± 6°F
St. Gen. Press.	0 - 1200 psig	± 2	± 24 psi
Steam Temp.	100 - 650°F	± 2	± 11°F
HPI Flow	0 - 600 gpm	± 2	± 12 gpm
BWST Level	0 - 56 ft.	± 2	± 1.12 ft.

* RPS temperature loops must be accurate to ± 1%.

TABLE VI

ESTIMATED RECORDER READING ACCURACY

Parameter	Range	Est'd Rdg. Accuracy	Req'd Instument Accuracy
RCS Temp.	0 - 800°F	5°	16°
Steam Gen. Temp.	0 - 800°F	5°	16°
RCS Unit Tave.	520 - 620°F	1°	2°
RCS Unit Outlet Temp.	520 - 620°F	1°	2°
RCS Press. (WR)	0 - 2500 psig	25 psi	50 psi
RCS Press. (NR)	1700 - 2500 psig	5 psi	16 psi
React. Bldg Press. (NR)	-5 - + 10 psig	0.2 psi (a)	0.15 psi
Reactor Bldg. Press (WR)	0 - 100 psig	1 psi (a)	1 psi
React. Bldg. Temp.	0 - 200°F	1°	6°
Steam Press.	600 - 1200 psig	5 psi	24 psi
Pressurizer Level	0 - 400 in.	2.5 in.	8 in.
Steam Gen. Level	0 - 100%	1%	2%
Makeup Tank Level	0 - 100%	1%	2%
SRM and IRM	8 decades	0.1 decade (b)	(c)
Rad. Monitors	5 decades	0.1 decade (b)	(d)
RCS Flow	0 - 110×10^6 lb/hr	1×10^6 lb/hr	5.4×10^6 lb/hr

Notes:

(a) Chart alternates between wide and narrow range. Reading of each trace is difficult when not in its own range.

(b) Log scale - accuracy varies. This is an estimated average.

(c) 3% of full range.

(d) 2% of setpoint; varies with instrument.

TABLE VII

ESTIMATED DATA RELIABILITY

Data	~~Est'd~~ Data Primary Source	Confirmatory Sources	Reliability Ranking
RCS Pressure	Reactimeter	Stripcharts, utility printer	Good
RCS Temp.	Reactimeter	Stripcharts, utility printer	Good
Press. Level	Reactimeter	Stripcharts, utility printer	Good[a]
Press. Temp.	Utility Printer	---	Good[b]
OTSG Level	Reactimeter	Stripcharts	Good
OTSG Press.	Reactimeter	Stripcharts	Fair
EFW Flow	OTSG Level Change	None	Very poor
MU Flow	Operator recollection	BWST Level	Poor
PORV Block Valve Opening	Operator recollection	Tailpipe temp., RB Press. and Temp.	Poor
BWST Level	Logs	None	Fair[b]
Core Temps.	Incore T/C's (alarm printer)	One set of manually read voltages	~~Fair~~ Poor[c]
Pump Start and Stop	Alarm printer	Operator recollection	Good

(a) Subject to possibility of reference leg voiding.

(b) Only available at a few discrete times.

(c) Only available when passing to or from alarm status; range too narrow; possible false junctions.

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	---	Plant status prior to accident: Unit 1 is shut down for refueling. Unit 2 is operating at between 97-98% of full power. The Integrated Control System (ICS) was in automatic. Pressurizer heater and spray controls were in manual. Feedwater pumps FW-P1A and FW-P1B, condensate pumps CO-P1A and CO-P1B and condensate booster pumps CO-P2A and CO-P2B were in operation. Make-up pump MU-P1B was in service. Operators were attempting to transfer spent resins from a condensate polisher to the resin regeneration tank. In this operation air at 100 psig and demineralized water at approximately 160 psig are used. Plant parameters as printed by the hourly log typer at 0300: RCS: Pressure- Loop A - 2165 psig Loop B - 2148 psig Flow - 137 million Lb/hr Temperatures- Loop A TH - 606DegF TC - 556-558DegF Loop B TH - 606DegF TC - 557DegF Pressurizer level 229 inches Make-up Tank at 77 inches Make-up Flow 70 Gpm Steam Generators- Pressure: A - 908 psig B - 905 psig Temperature: A - 595DegF B - 594DegF Levels: A - 257 inches B - 264 inches Percent full power - 97.928				1,2,3,10

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Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	-1 sec. (0400:36)	Condensate pump CO-P1A tripped.	Annunciator (Panel 17) Status lights (Panel 5) Alarm printer (operating without delay at this time)	Check valve in air line to condensate polisher was found to be frozen in the open position. This could have admitted water to the control air system. Condensate booster pumps CO-P2A and CO-P2B found tripped after turbine trip.	It has been postulated that the cause of the trip was closure of the polisher outlet/inlet valves because of water in the control air system. The polisher outlet and inlet valves were found to be closed after the turbine trip, but tests of similar valves have not substantiated this hypothesis.	1,2,3,4
	0 sec. (0400:37)	Feedwater pumps FW-P1A and FW-P1B tripped.	Annunciator (Panels 15 and 17) Pump discharge meter (panel 5) Alarm printer (delay-4 sec.)		Could have tripped on low suction pressure or trip of condensate booster pumps.	1,2,3,4
	0 sec.	Turbine trip.	Annunciators (panels 5 and 17) Meters (panel 5) Status lights (panel 5) Alarm printer (delayed)		Normal following trip of feedwater pumps.	1,2,3,4, 5,6
	0 sec.	Emergency feedwater pumps EF-P-2A, EF-P-2B and EF-P-1 came on.	Status lights (panel 4) Alarm printer (delayed)	Block valves EF-V12A and EF-V12B were closed.	Startup of emergency feedwater is automatic on loss of main feedwater pumps.	1,2,3,4
	+1 sec.	Turbine throttle and governor valves closed.	Meters (panel 5) Alarm printer (delayed)		One throttle valve did not show closed.	1,4,11

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Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	3 sec.	RCS pressure reaches the set point of the pilot operated relief valve (PORV) RC-R2. PORV opens. (Set point = 2255 psig)	Status light (panel 4)		Pressure in reactor coolant drain tank (RCDT) begins to increase.	1,2,3,6
	8 sec.	Reactor trips on high pressure. (Set point = 2355 psig)	Annunciator (panel 8) Status light and meter (panel 14) Neutron flux meter (panel 4)	Reactimeter indicates peak pressure of 2346 psig. Wide range strip chart shows a peak of 2435 psig.	Reactimeter sampling rate may be too coarse to catch peak. The code safety valves may have lifted momentarily, if the higher indicated pressure is correct.	1,2,3,4, 5,6
	8 sec.	Pressurizer heater banks 1-5 tripped.	Status light (panel 4)		Pressurizer was evidentally switched from manual to automatic control.	1,2,3,4
	9 sec.	Main steam pressure peaks at 1070 psig.	Meter (panel 4) Stripchart (panel 17)			1,2,3,6
	9 sec.	Confirmed all rods inserted.	Status lights (panel 4) Alarm printer (delayed)			1,3,4
	13 sec.	Let down secured. Operator attempts to start make-up pump MU-P1A	Annunciator (panel 8) Status light (panel 3) Alarm printer (delayed) Letdown flow meter (panel 3)	Pump failed to start.	The switch for the make-up pump must be held in the start position for 2.5 sec. Observation of status light would have shown that pump did not start. The purpose of these actions is to minimize pressurizer transient.	1,2,3,4
	13 sec.	RCS pressure reaches setpoint for PORV closure (setpoint = 2205 psig).	Status light (panel 4)	Valve did not close.	Light "off" indicates solenoid de-energized. There is no actual position indicator.	1,2,3,6

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	13 sec.	Condensate hotwell low level alarm. (21.72 inches).	Meter (panel 5) Alarm printer (delayed)			2,3,4
	14 sec.	Pressurizer heater groups 1-5 returned.	Status light (panel 4) Alarm printer (delayed)		Automatically energized on decreasing pressure. Setpoints = 2105 psig for 1-3 and 2120 psig for 4-5, with pressure decreasing.	1,2,3,4
	14 sec.	Emergency feedwater pumps reach full discharge pressure.	Meters (panel 4) Alarm printer (delayed)		Emergency feedwater valves EF-V11A and EF-V11B will not open until OTSGs reach 30 inches.	1,2,3,4
	15 sec.	Pressurizer spray valve closed.	Status light (panel 4)			2,6
	15 sec.	Pressurizer peaks at 255 inches.	Meter (panel 5) Stripchart (panel 4)		RCS parameters are normal.	1,3,6
	28 sec	OTSG "A" reaches 30 in.	Meter (panel 4) Annunciator (panel 17)		Emergency feedwater valves EF-V11A and EF-V11B should begin to open. These valves apparently opened more slowly than usual; however, no flow was possible because the block valves were closed.	3,6
	28 sec.	Condensate hotwell level returned to normal.	Meter (panel 5) Alarm printer (delayed)			3,4
	30 sec.	High temperature alarms on outlet temperatures for PORV (239.2DegF) and one code safety valve.	Stripchart (panel 10) Alarm printer (delayed)		Alarms were not considered abnormal, because the PORV had previously opened.	1,2,3,4
	30 sec.	RCS pressure reaches low pressure trip setpoint (1940 psig)		Reactimeter data.		1,2,4,6

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	40 sec.	Both OTSGs alarm low.	Annunciator (panel 17) Meter (panel 4) Alarm printer (delayed)			1,4
	41 sec.	Start make-up pump MU-P1A. Open valve MU-V16B to increase make-up flow.	Annunciator (panel 8) Status light (panel 3) Alarm printer (delayed)		Pump was started by a second operator, who saw that the first at-attempt was unsuccessful. Pumps A and B are now both operating.	1,2,3,4
	41 sec.	Open valve DH-5A	status lights (panel 8)		Allows makeup to be drawn from BWST	1
	48 sec.	Pressurizer level reaches minimum 158.5 inches and starts to increase.	Met[illegible] (panel 5) Str[illegible]t (pa[illegible] 4)		Minimum level is not as low as usual for this transient.	1,2,3,6
	1 min.	Code safety valve (RC-R1A) outlet temperature alarms high (204.5Deg).	Stripchart (panel 10) Alarm printer (delayed 1 min.)		This does not necessarily indicate that the code safety valves lifted; opening of PORV would also cause increase in code safety valve outlet temperature.	1,2,3,4
	1 min., 13 sec.	Condensate high level alarm.	Meter (panel 5) Alarm printer (delayed 1 min.)	Hotwell level reject valve was later found to be inoperative. Instrument air line to level controller was broken.	Condensate hotwell level control and other secondary side problems were constantly occurring, distracting operators' attention from the accident.	2,3,4
	1 min., 18 sec.	OTSGs reach minimum level on start-up range instrumentation (A: 11 inches; B: 15 inches).	Steam pressure: Meter (panel 4) OTSG level: Meter (panel 4)		Indicates dryout. No feedwater was being admitted. Dry-out indicated by low steam pressure, low level, increasing RCS temperature. Operator verified EF-V11A and B opening.	1,2,3,6 Ref. 1 and 3 times are in error.

Date	Time after initiation	Event	Information available to operator	Post-accident calculations and data	Remarks	References
3/28/79	1 min., 26 sec.	Reactor coolant drain tank (RCDT) temperature reaches 85.5DegF	Meter (panel 8A) Alarm printer (delayed 1 min.)		RCDT temperature was gradually increasing as RCS coolant was released from the EMOV.	1,2,3,4 (Reference 1 time is in error)
	1 min., 30 sec.	RCS pressure reaches 1727 psig.	Meter (panel 4) Stripchart (panel 4) Alarm printer (delayed 1 min.)		RCS pressure was decreasing, pressurizer level was increasing, RCS temperature was increasing. Pressure normally trends in the same direction as level and temperature following feedwater transient and reactor trip.	1,4,6 (Reference 1 time is in error)
	2 min., 2 sec.	ESF actuation. Make-up pump MU-P1B trips. Make-up pump MU-P1C starts at 2min. 4sec. DH removal pumps 1A and 1B start.	ESF: Annunciator (panel 13) Status lights (panels 3 and 13) MU pumps: Annunciator (panel 8) Status lights (panel 3) DH pumps: Status lights (panels 3 and 13) Meters (panels 3 and 8) Alarm printer (delayed 2 min.)		Actuation on low RCS pressure (setpoint = 1600 psig.) Make-up pumps A and 1C operating with valves MU-V16 wide open.	1,2,3,4,5
	3 min., 13 sec.	RCDT relief valve opens (120 psig).	Pressure: Meter (panel 8A)	Reactimeter--not available to operators.		2,3
	3 min., 13 sec.	ESF emergency injection bypassed by operator.	Operator action Alarm printer (delayed 3 min.)		Bypass leaves all equipment operating, but generator now has control.	1,2,3,4
	3 min., 26 sec.	RCDT high temperature alarm (127.2DegF)	Meter (panel 8A) Alarm printer (delayed 3 min.)	Reactimeter shows oscillations, possibly caused by RCDT safety valve lifting momentarily.	Further indication of open PORV.	3,4

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	4 min. (approx.)	Operator throttles make-up valves (MU-V16) to reduce injection flow.	Flowmeter (panel 8)		Purpose of throttling is (a) to reduce rate of rise of pressurizer level (b) to prevent pump damage as RCS pressure drops.	1,2,3
	4 min., 38 sec.	Operator stops make-up pump MU-P1C.	Operator action Annunciator (panel 8) Status light (panel 3) Pressurizer level: Meter (panal 5) Stripchart (panel 4)		Make-up valves MU-V16C and MU-V16D were fully closed. Operator throttles valves MU-V16A and M-V16B in an attempt to control rising pressurizer level.	1,2,3,4
	4 min., 52 sec.	Operator starts intermediate closed cooling pump IC-P-1A.	Annunciator (panel 8) Status lights (panels 8 and 13) Meters (panel 8) Alarm printer (delayed 5 min.)		erator is preparing to put a second letdown cooler in operation, so that letdown flow can be increased. He is attempting to recover control over the still increasing pressurizer level.	1,2,3,4
	4 min., 58 sec.	Letdown flow alarms high (Greater than 160 gpm)	Meter (panel 3)		Alarm printer is now so severely delayed that it is of little value to operators; alarm printer will not be listed as information available to operator from now on.	1,2,3,4 (Reference 1 time is in error)
	5 min., 0 sec.	Pressurizer level hits peak of 377 inches, momentarily decreases to 373 inches at 5min. 18 sec., and then begins to increase again.	Meter (panel 5) Stripchart (panel 4)			1,2,3,6
	5 min., 15 sec.	Start condensate pump CO-P1A.	Annunciator (panel 17) Status light (panel 5) Meter (panel 5)			2,3,4

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	5 min., 17 sec.	Attempt to start condensate booster pump CO-P2B (trips at 5min. 20 sec.)	Annunciator (panel 17) Status light (panel 5) Meter (panel 5)		Cause of trip is apparently low suction pressure. Auxiliary operator has reportedly realigned polisher correctly for restart.	1,2,3,4
	5 min., 50 sec.	RCS pressure reaches minimum (1350 psig), then begins to increase. Temperature reaches saturation.	Meters and stripcharts (panel 4)		Reaching saturation temperature means that steam voids can form in system; steam is being formed rapidly enough to reverse pressure decline.	1,3,7
	5 min., 51 sec.	Pressurizer level goes offscale high (greater than 400 inches).	Meter (panel 5) Stripchart (panel 4)			1,2,3,6
	6 min.	RCDT pressure begins erratic, rapid rise.	Meter (panel 8A)	Reactimeter data (not available to operators).	Indicative of two-phase flow through PORV.	1,2,3,6
	6 min., 24 sec.	Condensate booster pump CO-P2B trips after another attempted start.	Annunciator (panel 17) Meter (panel 5) Status light (panel 5)			1,2,3,4
	6 min., 54 sec.	Letdown cooler high temperature alarm (139DegF)	Stripchart (panel 10)			3,4
	6 min., 58 sec.	Letdown flow throttled to 71 gpm.	Meter (panel 3)		Purpose of throttling is to reduce cooler outlet temperature and halt decline of RCS pressure.	1,2,3,4
	7 min., 29 sec.	Reactor building sump pump WDL-P1A starts.			Pump outlet was believed aligned to the miscellaneous waste holdup tank; however, the latter tank's level does not show the appropriate changes. Pump was apparently aligned to the aux. building sump tank, which had a blown rupture disk.	1,2,3,4

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	8 min.	Operator finds emergency feedwater block valves EF-V12A and EF-V12B shut, and opens them.	Status lights (panel 4)		Clues to blocked feedwater flow: low OTSG level, low steam pressure, high emergency feedwater discharge pressure. Clues to initiation of flow: drop in discharge pressure, noise from loose parts monitor, increase in steam pressure.	1,2,3,6
	8+min.	RCS temperatures begin to decrease.	Stripcharts (panel 4 and panel 10) Meter (panel 4)		Resumption of heat transfer through steam generators.	3,6
	8 min., 58 sec.	Condensate pump CO-P1A trips again.	Annunciator (panel 17) Meter (panel 5) Status light (panel 5)		Another recurrence of secondary side problems, apparently unrelated to accident, but very troublesome to operators.	1,2,3,4
	9 min., 7 sec.	Intermediate range NIs drop below scale, source range NIs energized.				1,4
	9 min., 13 sec.	Condensate booster pump low suction pressure alarm (14.7 psig).			See remarks above.	2,3,4
	9 min., 30 sec.	Turbine bypass valves placed in manual.	Operator action		ICS was not responding adequately to increased steam steam pressure.	1
	10 min., 0 sec.	RCP high vibration alarm.	Annunciators (panel 8 and panel 10) Meter (panel 10)		Indication of voids in system. Apparently not recognized.	3
	10 min., 15 sec.	Pressurizer level comes back on scale and drops rapidly.	Meter (panel 5) Stripchart (panel 4)			2,3,6
	10 min., 19 sec.	Reactor building sump pump WDL-P2B starts.				1,2,3,4

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	1(min., 24 sec. to 11 min. 43 sec.	Make-up pump MU-P1A was stopped, started, stopped, and restarted.	Annunciator (panel 8) Status light (panel 3) Meter (panel 3)		After final restart, pump 1A runs throttled for 3 hours 23 minutes.	1,2,3,4
	10 min., 48 sec.	Reactor building sump high level alarm (4.65 feet).		Alarm printer.	PORV discharge goes into RCDT, then out RCDT relief valve.	1,2,3,4
	11 min., 24 sec.	Intermediate cooling water temperature from RCDT cooling is offscale (225DegF).		Alarm printer.		1,4
	13 min.	The operators are attempting to establish a 30-inch level in the OTSGs.	Meter (panel 4) Stripchart (panel 4) Stripchart (panel 5)		Operators may have throttled valves EF-V11A and B.	1
	13 min., 13 sec.	Operators stopped decay heat removal pumps DH-P1A and B.	Operator action Status lights (panel 13 and panel 3) Meters (panel 8 and panel 13)			1,2,3,4
	13 min., 27 sec.	Cond nsate booster pump suction header low pressure alarm clears.		Alarm printer.		2,3,4
	14 min., 50 sec.	Reactor coolant pump alarms begin to be received on pumps 2A and 1B.	Annunciators (panel 8 and panel 10) Meter (panel 10)	Alarm printer.	Many rapidly alternating "norm/high" or "norm/low" alarms on pump speed, seal leak tank level, backstop oil flow, etc. These were probably caused by high vibration levels of the RCPs and all associated equipment, but might not be perceived as such. The great number of RCP related alarms was a major factor in the alarm printer getting so far behind time.	1,2,4

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	15 min., 27 sec.	RCDT rupture diaphragm bursts. RCDT pressure drops suddenly; reactor building pressure increases 1 psi.	Annunciator (panel 8A) Meter (panel 8A)	Reactimeter (not available to operators).	The drain tank is now completely open to the reactor building atmosphere, and will overflow.	1,2,3,6
	15 min., 43 sec.	Condensate booster pump low discharge pressure alarm (307 psig).		Alarm printer.		2,3,4
	16 min., 04 sec.	Restarted condensate pump CO-P1A.	Operator action Annunciator (panel 17) Meter (panel 5) Status light (panel 5)			2,3,4
	16 min., 30 sec.	RCS becomes subcooled.	RCS Pressure: Meter and stripchart (panel 4) RCS Temperature: Meter and stripchart (panel 4) Stripchart (panel 10)	Calculations based on reactimeter temperature data and stripchart pressure data.	RCS has been near saturation or saturated for 10 min. For 30 min. to 1 hr. hereafter, the RCS remains slightly subcooled or barely saturated.	1,6
	16 min., 12 sec.	Condensate booster pump suction low pressure alarm.		Alarm printer.		2,3,4
	19 min., 23 sec.	Reactor building air exhaust duct shows increased radiation level.	Meter (panel 12) Stripchart (panel 12)		Probably due to dislodged "crud". Possibly slight cracking of fuel clodding.	1,2,3
	22 min., 17 sec.	Source range NI was higher than expected: the operator manually tripped the reactor.	Meter (panel 4)	Post-accident analysis indicates increase was due to voids in coolant in downcomer.	Operator was not aware of reason for increase.	1,2,4
	22 min., 44 sec.	OTSG "A" low level alarm cleared.	Annunciator (panel 17)		OTSG "B" clears 4 minutes later.	1,4

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	24 min., 58 sec.	Operator requested PORV and code safety valve outlet temperatures. PORV outlet was 285.4DegF, code safety outlets were 263.9DegF and 275.1DegF.	Utility printer	High temperatures, coupled with blown RCDT rupture disk and increasing reactor building pressure, give sufficient indication of PORV being open. Note, however, that RCDT parameters are displayed behind the control panels. RB pressure can be read on panel 3.	Operator believed high temperatures were due to (a) slow cooldown from the opening at 3 sec., and (b) known leakage.	3,8
	25 min.	Letdown cooler high radiation alarms.		Analyses indicate that little significant fuel damage could have occurred at this time, although there is some possibility of clod rupture on high-powered fuel rods.	This is probably due to a "crud burst," which would not be unexpected in a transient event.	3
	25 min., 44 sec.	Emergency feedwater low discharge pressure alarm received.			Operator has apparently shut steam driven emergency feedwater pump to slow the rate of rise of water in the steam generators.	1,2,4
	26m 26s to 27m 51s	The operators request the computer print RCS temperatures, PORV outlet temperature, pressurizer level.	Utility printer			1,2,8
	26 min.	Stopped steam driven emergency feedwater pump.				
	28 min.	Operator closes valves supplying emergency feedwater to OTSG "B".	Operator action		Intend to slow rate of rise in level.	1
	30 min., 21 sec.	Diesels manually shutdown.			Auxiliary operator has been sent to diesels to shut them down locally. Diesels cannot now be started from control room.	1,4

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	31 min.	Operator requests "Sequence of Events Review" from computer.	Utility printer			1,8
	32 min., 36 sec.	Incore thermocouple R-10 alarms offscale high (T greater than 700DegF)		Alarm printer not received by operator for nearly 1/2 hour.		1,2,3,4
	36 min., 08 sec.	Emergency feedwater pump EF-P2B stopped by operator.	Meters and status lights (panel 4)		Further actions to stop rate of rise of OTSG levels.	1,2,3,4
	38 min., 10 sec.	Reactor building sump pumps turned off.		Approximately 8300 gallons have been pumped to auxiliary building.		1,2,3,4
	40 min.	Operator checked RCDT pressure and temperature.	Meters (panel 8A)			3
	40 min.	Increased source range count rate.	Meter and strip-chart (panel 4)	RCS conditions again at or approaching saturation.	Possible voiding in core region.	2
	44 min.	Operator requests printout of pressurizer level indicator differential pressures.	Utility printer		Operator attempts to determine if level indication is correct. Conclusion: all instruments agree.	1,8
	45 min.	Letdown cooler count rate increased slowly over an order of magnitude. Peak 2 x 10^4 cpm.	Meter and strip-chart (panel 12)		Increase and recovery are more indicative of crud burst than of fuel failure.	2,3,8
	50 min.	OTSG "A" level trending downward; OTSG "B" level trending upward.	Meters and strip-charts (panel 4) Stripchart (O.R.) Channel 5			1,6
	52 min.	Operator requests computer print condenser pressure and emergency feedwater pump #1 discharge pressure.	Utility typer		EF-P1 has previously been shut down.	1,8
	59 min.	Condensate High Temperature Alarm.				

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	59 min.	Polisher inlet and outlet valves were manually opened.	Report from personnel at polisher.		Still could not establish hotwell level control--broken air line to reject valve.	1
	1 hour	Plant status: All RCP's running, make-up pump MU-P1A operating. Feeding OTSG "A" directly. Feeding OTSG "B" via cross-connect. Hourly log typer has the following data for 0500: Reactor Coolant Flow: 103×10^6 Lb/hr Loop A: TH 550DegF TC 546-547DegF Pressure: 1061 psig Loop B: TH 550DegF TC 547DegF Pressure: 1041 psig Steam Pressure "A" 1003 psig "B" 1011 psig Steam Temperature "A" 579DegF "B" 580DegF Make-up Flow: 102 gpm The PORV is open (unknown to the operators). The RCS is near saturation, probably having extensive voids in the core. Coolant pump operation has become severely degraded, with reduced flow and high vibration. Difficulties with the condensate system have plagued the operators the past hour; the condition of the RCS appeared outwardly stable, in that pressure and temperature were not changing rapidly.		Calculated Decay Power = 32.8 MW	Note that steam temperature is actually higher than hot leg temperature; this shows that cooling at this time is being provided by make-up water, which is being blown out through the PORV.	10,12

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	1 hour, 1 min.	The operator stopped circulation water pumps CW-P1B, C, D and E. (Opens atmospheric dump).	Operator action Meters and Status lights (panel 17)		This is done to permit use of the power operated emergency main steam dump valves (MS-V3A and MS-V3B) to control main steam pressure. The continuing deterioration of the condensate system has made an atmospheric dump necessary.	
	1 hour, 2 min.	Alarms from 1 hour 2 minutes to 1 hour 13 minutes are on the utility printer.	Utility printer	Alarms are delayed about 1-1/2 hours.	The action causing this change is taken at 2 hours 39 minutes after initiation.	4,8
	1 hour, 11 min.	Reactor building air cooling coil B emergency discharge alarm.	Flowmeter (panel 25)		The fact that this alarm clears in 30 sec. indicates that it may be spurious.	1,3,4
	1 hour, 12 min.	Operator requests alarm status of of reactor coolant pumps and motors.	Utility printer	RCP flow is down 35% from normal.	All pumps show oil lift pump discharge pressure alarms; 1A, 2A, and 1B show full speed alarms; all pumps show backstop oil flow alarms; 2A shows seal leak tank level alarms. These alarms may not be intrinsically valid, but were probably caused by severe vibration conditions.	1,8
	1 hour, 13 min.	Operator stops reactor coolant pump RC-P2B because of increasing vibration and decreasing flow and amperage. Pump RC-P1B is stopped a few seconds later.	Flow: meter and stripchart (panel 4) Vibration: Annunciators (panel 8 & 10) Amperage: Meter (panel 4) Status light (panel 4)	Coolant has now been clearly saturated again since about 1 hour after initiation.	Loop B pumps were tripped in order to maintain pressure on pressurizer spray line from loop A. Steam pressure on B side began to drop, indicating stagnation.	1,2,3,6

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	1 hour, 13 min.	Alarms are lost for 1 hour, 24 minutes.			Ref. 1 has the time for this event at 1 hr 2 min, with a duration of 2 hrs 49 min. These times are obviously wrong.	2,3,4,8
	1 hour, 15 min.	Boron concentration measured at 700 ppm. Source range high ([redacted] cps) and increasing. 1.E4	B conc: sample results Source range: Meter and stripchart (panel 4)	Low boron concentration may have been due to dilution of liquid in sample line by condensed steam. High count rate may have been caused by voids in the downcomer.	Operators fear restart, do not realize that voids are forming in coolant. Checked reactor trip procedures.	1,2,7
	1 hour, 20 min.	An increase in the letdown line radiation monitor was observed.	Stripchart (panel 12)		Increased steadily for the next 45 minutes, then stays offscale high.	1,9
	1 hour, 20 min.	Operator gets printout of: Pressurizer surge line temp. (514DegF) PORV outlet temp. (283DegF) Code safety outlet temps. (211, 219DegF) Pressurizer spray line temp. (497DegF) Condensate pump outlet press (164 psi).	Utility printer		The operators now have adequate information to deduce that the PORV is open--(a) No reduction in outlet temperature, and (b) PORV outlet 70Deg hotter than code safety outlets.	1,2,3,7
	1 hour, 27 min.	OTSG "B" was isolated.	Operator action	Assumption of leak in steam generator cannot be supported by later information.	Operators assume that low steam pressure and high reactor building pressure are caused by steam leak.	1,3,5
	1 hour, 30 min.	Intermediate range and source range neutron instrumentation both increase.	Stripchart (panel 4)		Ref. 2 postulates that increased voiding makes the downcomer annulus more transparent.	1,2,3,7
	1 hour, 30 min.	Boron concentration down to 400-500 ppm in RCS; activity 4 μci/ml (factor of 10 increase).	RCS sample analysis.	Analysis indicates gross fuel failure improbable at this time.	Increased activity could be the result of a crud bust; count level recovers.	1

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	1 hour, 30 min.	Secondary side steam flow from OTSG A increased.	Pressure: Meter (panel 4), stripchart (panel 17) Level: Meters (panel 4), stripcharts (panels 4 & 5)	Inferred from pressure and level changes.	Indicates increased heat transfer in OTSG "A". Temperature of reactor coolant in A loop has been trending upward slightly; apparently temperature difference across OTSG is now sufficient for OTSG to remove a significant amount of heat from RCS. Loop A cold leg temperatures next decrease, because of increased heat loss.	2,6
	1 hour, 34 min.	OTSG "A" boils dry again.	Meters (panel 4) Stripcharts (panels 4 & 5)			1,2,3,6
	1 hour, 37 min.	Intermediate range neutron instrumentation drops off-scale; source range decreases suddenly by a factor of 30.	Meters and stripcharts (panel 4)	Analysis (Ref. 2) indicates that separation of liquid and vapor probably occured.	Flow in A loop may have now deteriorated to the point that vapor is no longer being circulated. Loop A flow is 30 000 Lb/hr. Normal flow is 60 000 Lb/hr. Flow is now dropping rapidly.	
	1 hour, 37 min.	Operator increases flow to OTSG "A" in an effort to reestablish level.	Operator action Meters (panel 4) (EF-V11A)		Will raise level to 50% on operating range, in an effort to establish natural circulation.	2,3

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	1 hour, 41 min.	Operator stopped both Loop A coolant pumps, because of high vibration, decreasing and erratic flow.	Vibration: Annunciator (panel 8) Annunciation and Meter (panel 10) Flow: Meter (panel 4) Pump operation: Status lights and meters (panel 4)		The pump has been operating without adequate suction head. Further operation could cause severe damage.	1,2,3,6,7
	1 hour, 42 min.	Source range count rate increased two decades. Intermediate range comes on scale and increases one decade. Operators commence emergency boration.	Meters and stripcharts (panel 4)	No data available to substantiate either of conflicting hypotheses However, the fact that recorded source range and intermediate range (NI-3 and NI-7) follow each other closely lends credence to Reference 2.	Increase has been postulated (Ref. 2) to be due to lower downcomer water level as th core dries out. This conclusion is disputed by Reference 1--believes instrument error.	1,2,3,7
	1 hour, 43 min.	Hot and cold leg temperatures begin to diverge. The cold leg temperature drops and hot leg temperature rises.	Meter and stripchart (panel 4) Stripchart (panel 10)		After temperatures exceed the narrow range indications, average temperature is the average of the narrow range limits, rather than the average of hot and cold leg temperatures.	1,2,3,6,7
	1 hour, 52 min.	Trying to achieve 50% level on OTSG "A".	Operator action Meters (panel 4) Stripcharts (panel 4 & 5)			1
	1 hour, 54 min.	Operator requests computer print Sequence of Events Review.	Utility printer		Review disproves contention of Reference 3 that make-up pump 1C has been started.	1,8

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	2 hours	Plant Status: Make-up pump MU-P1A is operating, no reactor coolant pumps are operating, OTSG-A is being dumped to the atmosphere because of a general breakdown of the condensate system. The hourly log typer shows the following data for 0600: Reactor coolant temperatures: Loop A TH = 558oF TC (offscale) Loop B: TH = 528oF TC (offscale) Pressures: Loop A = 735 psig Loop B = 715 psig Make-up flow 99 gpm Steam Pressures: A = 685 psig B = 190 psig Steam Temperatures: A = 536oF B = 532oF OTSG Levels: A = 154 in. B = 79 in, (OTSG B is isolated) PORV is still open. Boron sample 400.		Calculated decay power = 25.7 MW		10,12
	2 hours, 5 min.	OTSG "A" reaches 50% level. Throttled back EF-V11A.	Meters (panel 4) Stripcharts (panels 4 & 5)			1,2,6,7
	2 hours, 11 min.	Loop A hot leg temperature offscale high.	Annunciator (panel 8) Stripcharts (panels 4 & 10) Meter (panel 4)		TAVE will not be correctly shown.	3,6
	2 hours, 15 min.	Reactor building air sample particulate radiation monitor goes offscale high.	Annunciator, meter, and stripchart (panel 12)		Possibility of gross fuel damage at this time.	2,3,7
	2 hours, 18 min.	Operator requests computer printout of PORV and safety valve outlet temperatures. PORV-228.7DegF, safety valves 189.5DegF, 194.2DegF.	Utility typer			1,2,3,8

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	2 hours, 18 min.	Operator closes PORV block valve RC-V2.	Status light (panel 4)			1,2,3
	2 hours, 18 min.	Reactor building temperature and pressure immediately decrease.	Stripchart (panel 3)			1,2,3,7
	2 hours, 18 min.	RCS pressure begins to increase.	Meter and strip-chart (panel 4)		Because of block valve closure. No physical evidence of an increase in makeup flow.	1,2,3,7
	2 hours, 24 min.	Reactor building air sample gas channel monitor increased and went offscale.	Annunciator, meter, and strip-chart (panel 12)			3,7
	2 hours, 28 min.	Loop B hot leg temperature goes offscale high.	Annunciator (panel 8) Stripcharts (panels 4 & 10) Meter (panel 4)		There is now clear evidence of super heating in the hot legs. This could have shown the operators that the core was beginning to become uncovered.	3,6
	2 hours, 35 min.	Operator begins feeding OTSG "B" to 50% level.	Meters (panel 4) Stripcharts (panels 4 & 5)			1,2,3,6,7
	2 hours, 38 min.	Letdown cooler A radiation monitor went offscale high. Numerous area radiation alarms received.	Annunciators, meters, and stripcharts (panel 12)	Calculations suggest possiblity of fuel damage at this time.	Letdown sampling secured due to high radiation.	1,3,7
	2 hours, 40 min.	Emergency boration started.			Increasing levels of neutron instrumentation lead operators to fear restart.	1,3
	2 hours, 44 min.	Incore instrumentation panel monitor goes offscale high.	Meter and strip-chart (panel 12)			2,3

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	2 hours, 45 min.	See Remarks.			Reference 3 has make-up pump 1C tripped at this time. Ref. 3 had start at 1 hr 54 min. This contention is disputed by Reference 1 (which erroneously states that Ref. 3 has no start time).	
	2 hours, 45 min.	Numerous radiation alarms begin.	Panel 12		Radiation alarms are now indicative of extensive fuel damage.	3
	2 hours, 46 min.	Unsuccessful attempt to start Reactor Coolant Pump RC-P1A.	Status lights and meters (panel 4)			2,3
	2 hours, 47 min.	Alarms back on alarm printer. Alarm printer brought up to date.	Operator action		Alarms for the period from 1 hr 13 min to 2 hrs 47 min were irretrievably lost.	1,2,3,8
	2 hours, 48 min.	90 percent of core T/C's offscale high. Self-powered neutron detectors indicate readings.	Alarm printer		Readings on SPND's caused by high temperatures. Flood of readings swamps alarm printer, causing it to lose time again. Core T/C reading not available to operators.	2,4
	2 hours, 52 min.	Unsuccessful attempt to start RC-P2A.	Status light and meters (panel 4)		Does not appear on alarm printer.	1,2,3
	2 hours, 53 min.	Control of hotwell level regained.	Alarm printer (delayed several minutes) Meter (panel 5)		Broken air line to reject valve was repaired.	1,2,3,4
	2 hours, 53 min.	Unsuccessful attempt to start RC-P1B.	Status lights and meters (panel 4)			1,2,3

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	2 hours, 54 min.	Start RC-P2B.	Status lights and meters (panel 4)		Had to jumper start interlocks to start pump. Flow was shown momentarily and then dropped to near zero. The pump ran with high vibration.	1,2,3,4,6
	2 hours, 54 min.	Pressurizer heater groups 1-5 tripped.	Status lights (panel 4)			1,2,3,4
	2 hours, 55 min.	Pressurizer spray valve opens.			Operation of pressurizer spray is impossible without reactor coolant pump operation.	6
	2 hours, 55 min.	7 in core T/C's on scale.	Alarm printer (delayed 1/2 hour)	Slug of water from RC-P2B apparently gave some cooling.	These readings were back on scale indicating that they had just returned from an offscale condition.	1,3,4
	2 hours, 55 min.	RCS pressure suddenly increased to 2140 psig.	Annunciator (panel 8) Meter and stripchart (panel 4)		Slug of water from cold leg gave rise to rapid boiling.	1,3,4
	2 hours, 55 min.	HPI reset by increased pressure.	Annunciator (panel 13) Status lights (panels 3 & 13)		Set point 1845 psig.	2,3,4
	2 hours, 55 min.	Source range and intermediate range neutron instrumentation dropped sharply.	Meters and stripcharts (panel 12)		Slug of water filled downcomer, giving better shielding.	1,3,4
	2 hours, 56 min.	Start circulating water pump CW-P1B, and CW-P1E. (Close atmospheric dump, resume steaming to condenser).	Meters and status lights (panel 17)		This allows control of main steam pressure by turbine bypass valves, and use of condenser.	1,2,3,4
	2 hours, 56 min.	Site Emergency declared.			Reason: radiation alarms.	1,2,3

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	2 hours. 59 min.	Reactor Building Purge Unit area monitor and Fuel Handling Building area monitors increased. Fuel Handling Building Air Supply Fans turned off.	Annunciators, meters, strip-charts (panel 12)			
	3 hours	Plant Status: Make-up pump MV-P1A is operating, the PORV block valve is closed, reactor coolant pump RC-P2B is operating but with very little flow. Both steam generators were being fed, with dump to condenser from steam generator "A" only. The hourly log typer gives the following information for 3 hrs 1 min: RCS Pressures: Loop A = 2055 psig Loop B = 2051 psig RCS Temperatures: Loop A: TH offscale high TC offscale low Loop B: TH offscale high TC offscale low Pressurizer level: 375 inches. MU Flow 125 gpm Steam Pressures: A = 308 psig B = 416 psig Steam Temperatures: A = 495DegF B = 520DegF		Calculated decay power = 22.3 MW		10,12
	3 hours +	Pressurizer level offscale high.	Meter (panel 5) Stripchart (panel 4)			1,6,7
	3 hours, 2 min.	RCS Loop B hot leg temperature reaches 800DegF.	Stripchart (panel 4)		This is offscale on multipoint recorder.	1,7
	3 hrs 3m	Hotwell low level alarm.				2,3,4

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	3 hours, 4 min.	Shut turbine bypass valves from steam generator B. Shut emergency feedwater valves to steam generator B.	Status lights (panels 4 & 5)		Condenser vacuum pump exhaust radiation monitor had increased. A leak from primary side was suspected. This completely isolates steam generator B.	1,2,3
	3 hours, 5 min.	Source range and intermediate range detectors increasing.	Meters and strip-charts (panel 4)		Indicates dropping water level.	1,3,7
	3 hours, 7 min.	Condensate storage tank low level alarm.		Alarm printer delayed 50 min.		2,3,4
	3 hours, 10 min.	Emergency feedwater pump (EF-P2A) was stopped.	Status lights and meters (panel 4) SG level: panels (panels 4 and 5)		Steam generator level above 50% on startup range.	1,2,3,4
	3 hours, 11 min.	Condenser hotwell low level alarm cleared.	Meter (panel 5)			2,3,4
	3 hours, 12 min.	Opened PORV block valve.	Status light (panel 4) Operator action	Inferred from pressure and temperatures. Time cannot be specified accurately.	Attempt to control RCS pressure. Outlet high temperature alarm, pressure spike in RCDT, drop in RCS pressure, increase in reactor building pressure.	1,2,3,4,7
	3 hours, 13 min.	Pressurizer spray valve closer.				6
	3 hours, 13 min.	Stopped reactor coolant pump RC-P2B.	Status lights, meters, and stripcharts (panel 4)		Zero flow, low current, high vibration.	1,2,3,4
	3 hours, 14 min.	Intermediate closed cooling pump area radiation monitor increased.	Annunciator, meter, strip-chart (panel 12)			2,3,7

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	3 hours, 20 min.	ESF manually initiated. Make-up pump MU-P1C starts. Loop A hot leg temperature drops.	Annunciator (panel 13) Status lights (panels 3 & 13)	Rapid quenching probably caused major fuel damage.	Reason for actuation was low RCS pressure. HPI gives water in core.	1,2,3,4
	3 hours, 20 min.	Source range and intermediate range detectors drop suddenly.	Meters and stripcharts (panel 4)		Indicates reflooding.	1,2,3,7
	3 hours, 21 min.	Many radiation alarms received. The control building (except the control room) was evacuated.	Annunciators, meters, stripcharts (panel 12)		Indication of major core damage.	1,2,3,7
	3 hours, 24 min.	General Emergency declared.				1,2,3
	3 hours, 26 min.	Pressurizer high level alarm clears.	Meter (panel 5) Stripchart (panel 4)			1,2,4
	3 hours, 27 min.	ESF actuation reset.	Operator action			1,4
	3 hours, 30 min.	BWST low level alarm at 53'.				1,4
	3 hours, 30 min.	Shut PORV block valve.	Status light (panel 4)		Time of closing is very uncertain. May have been closed at 3 hours 17 minutes. Definitely closed before 3 hours 34 minutes.	1,2,3,4
	3 hours, 33 min.	Pressurizer high level alarm level increasing rapidly.	Meter (panel 5) Stripchart (panel 4)			1,2,4
	3 hours, 35 min.	Auxiliary building basement flooded. High radiation readings in many areas of auxiliary building.	Meters and stripcharts (panel 12) Annunciators (panel 12)			1,7
	3 hours, 35 min.	Start emergency feedwater pump EF-P2A.	Status lights and meters (panel 4)		OTSG A level had been falling.	1,2,3,4
	3 hours, 37 min.	Make-up pump MU-P1C stopped.	Status lights and meters (panel 3) Annunciator (panel 8)		Apparently stopped to slow rate of rise in pressurizer level.	1,2,3,4

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	3 hours, 41 min.	Open PORV block valve. Outlet temperature alarms.	Status lights (panel 4)		Time is in doubt. Could have been opened earlier.	1,2,3,4
	3 hours 45 min.	Pressurizer spray valve opens				6
	3 hours, 46 min.	Sudden jump in source range detectors.	Meters and strip-charts (panel 4)		May have been due to sudden steam flashing or change in core geometry.	1,2,7
	3 hours, 54 min.	Reduced feed to OTSG "A".	Operator action			1,6
	3 hours, 56 min.	ES actuates on high reactor building pressure. Reactor building isolated.	Annunciator (panel 13)		This is the first time the reactor building has been isolated.	1,2,3,4
	3 hours, 56 min.	Make-up pump MU-P1C starts.	Annunciator (panel 8) Stripcharts and meters (panel 3)			1,2,3,4
	3 hours, 56 min.	Close PORV block valve.	Status light (panel 4)		Inferred from reactor building pressure.	7
	3 hours, 56 min.	Intermediate closed cooling pumps 1A and 1B tripped.	Annunciators, status lights, meters (panel 8)		By building isolation.	1,2,3,4
	4 hours	Plant status: make up pumps MU-PIA and B operating. No reactor coolant pumps operating. The hourly log typer gives the following information: RC Pressures: Loop A=1460psig Loop B=1453psig RCS Temperatures: Offscale Pressurizer level 381 inches Steam Pressures A=30psig B=358psig (isolated) Steam temperatures A=468DegF B=499DegF		Calculated decay heat=20.3 MW	ES actuation and RB isolation have just occurred	10,12
	4 hours	ESF and reactor building isolation defeated.	Annunciator (panel 13)			1,2,4

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
	4 hours	Started intermediate closed cooling pumps 1A and 1B.	Annunciators, status lights, meters (panel 8)		Necessary for letdown cooling.	1,2,3,4
	4 hours to 5 hours, 30 min.	Incore thermocouples being manually read.			This permits reading beyond range of 700DegF. Range from readings was 80-2580DegF.	1,2,3
	4 hours, 8 min.	Start reactor coolant pump RC-P1A.	Status lights, meters, strip-charts (panel 4)		Purpose of start was to observe current and flow. Started satisfactorily, but running current was low, and flow was zero.	1,2,3,4
	4 hours, 9 min.	Stop reactor coolant pump RC-P1A.				1,2,3,4
	4 hours, 15 min.	Open PORV.	Status light (panel 4)		Inferred from reactor building pressure.	7
	4 hours, 17 min.	Stop make-up pumps MU-P1A and 1C.	Status lights and meters (panel 3) Annunciator (panel 8)		No make-up pumps now running.	1,2,3,4
	4 hours, 18 min.	Attempt to restart MU-P1A.			Switch apparently then put in "pull-to-lock" position. MU-P1A will not now start on ESF actuation.	1,2,4,8
	4 hours, 19 min.	ESF actuates on high building pressure. Decay heat pump DH-P1A starts. Intermediate cooling pump 1A trips. MU-P1A and C do not start.	Annunciator (panel 13)		One channel actuated, one channel defeated. 2/3 logic satisfied. Immediately bypassed.	1,2,3,4
	4 hours, 19 min.	Cleared ESF actuation.	Annunciator (panel 13)			1,2,3,4
	4 hours, 19 min.	Restart intermediate cooling pump 1A.	Annunciators, status lights, meters (panel 8)			1,2,3,4

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	4 hours, 20 min.	Close PORV.	Status light (panel 4)		Inferred from reactor building pressure.	7
	4 hours, 22 min.	Pressurizer spray valve closes				6
	4 hours, 22 min.	Operator starts make-up pump MU-P1B.	Status lights and meters (panel 3) Annunciators (panel 8)			1,2,3,4
	4 hours, 24 min.	Pressurizer heater groups 1-5 return to service.	Annunciator (panel 8) Status lights (panel 4)		All heaters now in service.	1,2,3,4
	4 hours, 26 min.	Letdown cooler high temperature alarm.			Probably a late alarm when ESF was cleared.	1,2,4
	4 hours, 27 min.	Start make-up pump MU-P1C.	Status lights and meters (panel 3) Annunciator (panel 8)			1,2,3,4
	4 hours, 31 min.	Pressurizer heater group 10 trips.	Status lights (panel 4) Annunciator (panel 8)			1,2,3,4
	4 hours, 31 min.	Stop condenser vacuum pumps. Broke condenser vacuum.	Status light (panel 17) Annunciator and Stripchart (panel 17)		Condenser vacuum had been seriously degraded previously. Auxiliary boiler out of service.	1,2,3,4
	4 hours, 31 min.	Opened main steam dump valve (MS-V3A)	Meter (panel 5)			2,3
	4 hours, 35 min.	Incore thermocouple readings printed out.	Utility printer		Range from 310DegF to offscale.	1,8
	4 hours, 36 min.	Letdown high temperature alarm clears.				1,4

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Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	4 hours, 36 min.	Open PORV block valve.	Status light (panel 4)		Inferred from reactor building pressure.	7
	4 hours, 42 min.	Emergency feedwater pump EF-P2A stopped.	Status lights and meters (panel 4)	Steam generator could only be operating in reflux mode. Heat removal capability is low.	Steam generator level had risen and would now remain up.	1,2,4
	4 hours, 44 min.	Letdown cooler A radiation monitor went offscale low.	Stripchart and meter (panel 12)		Apparently failed.	2,3,7
	4 hours, 46 min.	Pressurize heater groups 4-5 trip.	Annunciator (panel 8) Status lights (panel 4)		Did not come on again for rest of 3/28.	1,3,4
	4 hours, 47 min.	Incore temperature readings again printed out.	Utility printer		Range from 378DegF to offscale.	1,8
	4 hours, 59 min.	Intermediate cooling pump area radiation monitors and reactor building emergency cooling monitors increase.	Stripchart (panel 12)			3,7
	5 hours	Plant status: No reactor coolant pumps running, make-up pump MU-P1B and 1C running, steaming through atmospheric dump valve, only reflux circulation, many radiation monitors offscale (containment dome monitor up to 6000 R/hr), RCS pressure 1266-1296 psig, steam pressure (A) = 43 psig, temperature 454DegF, hot leg superheated, PORV block valve open.		Calculations indicate that a large quantity quantity of hydrogen was now in the RCS. Calculated decay power = 18.9 [illegible].		
	5 hours, 15 min.	RCS pressures (1203, 1164, 1126 psig! and pressurizer surge line temperature (303DegF) printed out.	Utility printer			1,8
	5 hours, 15 min.	Decision made to repressurize system.	Operator action	Bubble contained hydrogen. There was no possibility of collapsing the bubble.	Believed hot legs contained steam bubble. Hoped repressurizing would collapse bubble.	1

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	5 hours, 17 min.	The alarm printer returned to service.	Alarm printer		Alarms had previously been on utility printer. Alarms are 1 hr 26 min behind time.	3,4
	5 hours, 18 min.	Closed PORV block valve.	Status light (panel 4)			1,2,3
	5 hours, 19 min.	Decay heat pump DH-P1A stopped.	Status light (panel 3)		Switch placed in "pull-to-lock".	1,4
	5 hours, 24 min.	ES actuates on building pressure. Pumps MU-P1A and DH-P1A do not come on. ES immediately defeated. Intermediate cooling pump 1A trips and is immediately restarted.	Annunciator (panel 13) RB Press: strip-chart (panel 3)			1,2,4
	5 hours, 29 min.	Diesels are placed in "MAINT EXERCISE" position.	Operator action		Diesels can now be started from the control room, but will not automatically start.	1
	5 hours, 31 min.	Pressurizer heater group 3 trips.	Annunciator (panel 8) Status lights (panel 4)		Remains out of service.	1,3,4
	5 hours, 35 min.	PORV outlet temperature alarms clear.			Evidence of closure at 5 hrs 18 min.	1,4
	5 hours, 35 min.	Condensate storage tank low level alarm.				4
	5 hours, 43 min.	System is repressurized. Pressure maintained by cycling PORV block valve.	Meter and strip-chart (panel 4)		Intention is to hold pressure at about 2050 psig. Intermittent outlet temperature alarms and pressure fluctuations show block valve cycling.	1,2,3,4,7

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	5 hours, 49 min.	Control room intake radiation monitors (gas, particulate, iodine) all increased.	Stripcharts (panel 12)		Nonessential personnel cleared from control room. Emergency control station moved to unit 1.	2,3,7
	5 hours, 59 min.	Auxiliary building exhaust fans stopped because of high radiation.	Stripcharts (panel 12)			2,3
	6 hours	Plant status: RCS pressure being maintained between 2050-2200 psig by cycling PORV block valve. No reactor coolant pumps running, make-up pumps MU-P1B and 1C running, hot legs superheated. Atmospheric dump from OTSG "A".		Calculated decay power = 17.8 MW.		
	6 hours, 14 min.	Raise OTSG level to 97%, using condensate pumps for feeding.	Meters and stripcharts (panels 4 and 5)			1,2
	6 hours, 14 min.	Pressurizer heater groups 1 and 2 trip, but are immediately returned.	Annunciator (panel 8) Status lights (panel 4)			2,3,4
	6 hours, 14 min.	Auxiliary building fans restarted.	Status lights (panel 25)			3,4
	6 hours, 17 min.	Control room personnel don respirators.				1,3
	6 hours, 18 min.	Operator gets "Sequence of Events Review" from computer.	Utility typer			1,8
	6 hours, 23 min.	Temperature on reactor building air cooling coils B emergency discharge goes offscale, then returns.			Indicative of severe temperature transient in reactor building.	1,4
	6 hours, 39 min.	Start fuel handling building air exhaust fans.	Status light (panel 25)			3
	6hrs 54m to 6hrs 56m	Steam generator A downcomer and shell temperatures printed by computer.	Utility printer			1,8

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	7 hours	Plant status: No reactor coolant pumps operating, makeup pumps 1B and 1C operating. Maintaining RCS pressure between 2000 psig and 2100 psig by cycling EMOV block valve.		Calculated decay heat=16.9 MW		10,12
	7 hours	Confirmed OTSG "A" not contaminated.	Measurement of steam plume.			1
	7 hours, 9 min.	Started emergency feedwater pump EF-P2A to raise steam generator level higher.	SG level: Meters and stripcharts (panels 4 and 5)			1,2,4
	7 hours, 30 min.	Steam generator A filled.	Meters and stripcharts (panels 4 and 5)			1,2,3
	7 hours, 30 min.	Note: Natural circulation cannot be achieved by repressurizing. Reactor coolant pumps have proven to be inoperable. At this time it is planned to depressurize via the PORV, with the hope of getting the pressure low enough to inject core flood tank water.				
	7 hours, 30 min.	Open PORV block valve and pressurize spray valve.	Status lights (panel 4)		On orders of Station Manager.	1,2,3,6
	7 hours, 42 min.	Defeated ESF actuation.	Operator action Status light (panel 3)		ES would have been actuated 1 min. later if it had not been defeated.	1,2,3,4
	7 hours, 44 min.	Pressurizer heater groups 1 and 2 trip but immediately return.	Status lights (panel 4)			1,3,4
	7 hours, 44 min.	Auxiliary building air exhaust fans stopped.	Status lights (panel 25)			3
	7 hours, 50 min.	Pressurizer heater groups 1 and 2 trip.	Status lights (panel 4)		May be depressurizing via pressurizer vent now.	1,3,4

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	7 hours, 53 min.	Operator gets "Sequence of Events Review".	Utility printer			1,8
	7hrs 54m to 8hrs 19m	Print out RCS and pressurizer pressures and temperatures.	Utility printer			1,8
	7 hours 58 min.	Pressurizer spray valve opens				6
	8 hours	Plant status: No reactor coolant pumps operating. Makeup pumps MU-P1B and 1C operating. PCRV block valve open. Pressurizer vent valve probably open. Depressurizing RCS Hourly log typer gives the following data: RCS Pressure: Loop A=1035 psig Loop B=1038 psig RCS temperatures offscale. Pressurizer level 395 inches Steam Generators: Pressures: A=7 psig B=3[illegible] psig Temperatures: A=422DegF B=458DegF Levels: A=374 inches B=228 inches Steaming through atmospheric dump valve from OTSGA.		Calculated decay heat=16.2 MW		10,12
	8 hours, 1 min.	Letdown cooler high temperature alarm.				1,4
	8 hours, 12 min.	Core flood tank high level alarm. (13.32 feet).	Annunciator and meter (panel 8)		Possibility of check valve leakage.	1,3,4
	8 hours, 31 min.	Start decay heat cooling pumps DH-P1A and 1B.	Status lights (panels 3 & 13)		Hoped to be able to go on decay heat removal system.	1,2,3,4
	8 hours, 40 min.	Operator requests incore thermocouple readings.	Utility printer		Most are offscale.	1,8
	8 hours, 40 min.	RCS pressure is down to 600 psig.	Meter and strip-chart (panel 4)		Indicates RCS now floating on core flood tanks. Level of CR tanks decreased very little.	1,2,3,7

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	8 hours, 43 min.	BWST level down to 32 feet.	Meter (panel 8)			1
	8 hours, 55 min.	Core flood tank high level alarm clears (13.13 ft.)	Annunciator and meter (panel 8)		Very little water injected.	1,3,4
	8 hours, 58 min.	Printout of RCS pressure and pressurizer temperature. Pressure: 483-526 psig, temperature 350DegF.	Utility printer			1,8
	9 hours	Plant status: RCS has been depressurized. Makeup pumps MU-P1A and 1C operating. EMOV block valve open, vent valve may be open. Steaming through atmospheric dump valve RCS Pressure: Loop A=473 psig Loop B=480 psig RCS temperatures offscale. Pressurizer level 399 inches. Steam pressures: A=13 psig B=296 psig Steam temperatures: A=413DegF B=449DegF		calculated decay heat =15.6 MW		
	9 hours, 4 min.	Stopped make-up pump MU-P1C.	Annunciator (panel 8) Status lights and meter (panel 3)			1,2,3,4
	9 hours 7 min.	Pressurizer spray valve closes				6
	9 hours, 8 min.	Stopped taking make-up from BWST.			Concerned that BWST would run out.	1
	9 hours, 15 min.	Closed atmospheric dump valve.	Meter (panel 5)		There is now no heat sink for the steam generators. Refs. 2 and 3 have this event at 8 hrs 30 min.	1,2,3

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	9 hours, 15 min.	Shut PORV block valve.	Status light (panel 4)		Cannot get RCS pressure low enough to go on decay heat removal system. The line at closure is not well fixed.	1,2,3,5
	9 hours, 17 min.	PORV outlet high temperature alarm clears.			Indicates that PORV block valve was definitely closed by this time.	1,3,4
	9 hours, 20 min.	Letdown cooler high temperature alarm clears.				1,4
	9 hours, 21 min.	PORV outlet high temperature alarm. (EMOV block valve open-time not accurately known)			Block valve must have been reopened prior to this time.	1,4
	9 hours, 32 min.	PORV outlet high temperature alarm clears. (Block valve closed-time not accurately known)			Valve must have been closed again.	1,4
	9 hours, 40 min.	Start intermediate closed cooling pump IC-P1B.	Annunciators, status light, meters (panel 8)		This clears letdown alarm.	1,4
	9 hours, 49 min.	PORV outlet high temperature alarm. (Block valve reopened)			Valve was opened again.	1,4
	9 hours, 50 min.	Pressure and temperature in containment show sudden spike.	(RB Press.: Stripchart (panel 3) RB Temp.: Stripchart (panel 25) Audible "thump"	Hydrogen combustion in containment.	Pressure spike was believed to be "electrical noise". Max. Pressure 28 psig. Not ascribed to detonation at the time.	1,2,3,6,7

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	9 hours, 50 min.	ESF actuation on high/high building pressure (setpoint = 28 psig). Decay heat pumps DH-P1A and 1B start. Int. cooling pumps IC-P1A and 1B trip. Reactor building isolates. Reactor building sprays start. Make-up pump MU-P1C starts. Reactor building isolated.	ES: Annunciator and status lights (panel 13) Status light (panel 3) MU-P1C: Annunciator (panel 8), meters and status lights (panel 3) Spray: Status lights (panels 13 and 15) Meters panel 3)		Later "Sequence of Events" review showed only that 4 psi had been received on 4 channels. Log entry has "4 psi" apparently based on this print-out. Reactor coolant pump air temperatures alarmed high (cleared in 1 minute).	1,2,3,4,5
	9 hours, 51 min.	Stopped make-up pump MU-P1C.	Annunciator (panel 8) Status lights and meter (panel 3)			1,2,3,4
	9 hours, 51 min.	480 v motor control centers 2-32A and 42A trip.				1,4
	9 hours, 55 min.	Pressurizer heater group 8 trips.	Status light (panel 4)			1,2,3,4
	9 hours, 56 min.	Reactor building spray pumps stopped.	Meters (panel 3) Status lights (panels 13 & 15) Annunciator (panel 8)	Sprays operated for 5 min 40 sec.	Observed that pressure and temperature had been brought down.	1,2,3,4
	9 hours, 57 min.	Stopped decay heat pumps, DH-P1A and 1B.	Status lights and meters (panel 3) Status lights (panel 13)		Cannot get pressure down far enough for decay heat system. Core Flood tanks remain floating, with intermittent changes of level.	1,2,3,4
	10 hours	Plant status: RCS pressure 512-522 psig, temperatures offscale, no RCPs running, make-up pump MU-P1B running, pressurizer shows 400 inches, no secondary heat sink.		Calculated decay power = 15.1 MW.		

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	10 hours	Opened PORV block valve.	Status light (panel 4)		Outlet temperature alarms high 1 minute later.	2,3,4
	10 hours, 3 min.	Operator gets "Sequence of Events" covering last ESF actuation.	Utility printer			1,8
	10 hours 5 min.	Pressurizer spray valve opens				6
	10 hours, 6 min.	Pressurizer heater groups 1 and 2 returned to service, but trip again less than 2 minutes later.	Status light (panel 4)			1,3,4
	10 hours, 28 min.	RCS loop A outlet temperature comes back on scale. Goes to minimum of 548DegF and stays on scale for 10 minutes.	Stripchart and meter (panel 4)		Ref. 1 postulates pressurizer dumped to 100 p. Operators may have believed they now had control of pressurizer level.	1,2,3,7
	10 hours, 32 min.	Make-up pump MU-P1C started. RCS pressure had dropped to 440 psig.	Annunciator (panel 8) Meter and status light (panel 3) RC Press: Meter and Strip-chart (panel 4)			1,2,3,4
	10 hours, 33 min.	Pressurizer heater groups 1 and 2 return to service.	Status light (panel 4)			1,3,4
	10 hours, 35 min.	RCS pressure drops to 409 psig, then starts to rise again.	Meter and strip-chart (panel 4)			2,7
	10 hours, 36 min.	Make-up pump MU-P1C stopped.	Annunciator (panel 8) Meter and status light (panel 3)			1,3,4
	10 hours, 38 min.	RCS loop A outlet temperature goes offscale again, then comes back on and continues to drop.	Stripchart and meter (panel 4)			3,6,7
	10 hours, 39 min.	Pressurizer heater groups 1 and 2 tripped again.	Status light (panel 4)			3,4

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	10 hours, 40 min.	Auxiliary building sumps now full.	Observation			1
	10 hours, 44 min.	Auxiliary building fans came on.	Stripchart (panel 25)		Ran for 30 minutes.	3,7
	11 hours	Plant status: Makeup pump MU-PIB operating. EMOV block valve open. RCS Pressure: Loop A=415 psig Loop B=421 psig RCS Temperatures TH-A=525DegF All others offscale Pressurizer level 378 inches. Steam pressures: A=63 psig B=266 psig Steam temperatures A=404 psig B=431 psig OTSG levels: A=371 inches B=224 inches		Calculated decay heat=14.6 MW		10,12
	11 hours, 6 min.	Pressurizer decreased to 180 inches in next 18 minutes. RCS loop A temperature increases.	Annunciator (panel 8) Stripchart (panel 4) Meter (panel 5)			1,3,6,7
	11 hours, 10 min.	Respirators removed in control room.	Operator action			1,3,5
	11 hours, 10 min.	Shut PORV block valve.	Status light (panel 4)			1,2,3
	11 hours, 18 min.	Start make-up pump MU-PIC pressurizer low level alarm.	Annunciator (panel 8) Meter and status light (panel 3) PZR level: Annunciator (panel 8) Stripchart (panel 4)			1,2,3,4
	11 hours, 27 min.	Computer printout of PORV and pressurizer safety valve outlet temperatures.	Utility printer		EMOV outlet 191De[illegible] safety valves 171[illegible]egF and 175DegF.	1,8

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	11 hours, 28 min.	Stopped make-up pump MU-P1C.	Annunciator (panel 8) Meter and status light (panel 3) Pressurizer (panel 4)		Pressurizer level increasing.	1,2,3,4
	11 hours, 29 min.	Pressurizer heater groups 1 and 2 tripped again.	Status light (panel 4)			1,3,4
	11 hours, 33 min.	Start make-up pump MU-P1C.	Annunciator (panel 8) Meter and status light (panel 3)			1,2,3,4
	11 hours, 34 min.	Start emergency feedwater pump EF-P2B.	SG level: Stripcharts and meters (panels 4 and 5) EF-P2B: Status lights and meters (panel 4)		To raise level in OTSG "B" to 97% to 99% range.	1,2,3,4
	11 hours, 36 min.	Stop make-up pump MU-P1C.	Annunciator (panel 8) Meter and status light (panel 3)		Pressurizer level continues to climb.	1,2,3,4
	11 hours, 44 min.	Pressurizer low level alarm clears at 206 inches.	Annunciator (panel 8)			1,4
	11 hours, 52 min.	Stopped emergency feedwater pump EF-P2B.	SG level: Strip-charts and meter (panels 4 and 5)		Steam generator B at 97%.	1,2,4
	11 hours, 54 min.	Pressurizer high level alarm at 260 in.	Annunciator (panel 8)			1,4

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	12 hours	Plant status: No reactor coolant pumps running. Make-up pump MU-P1B running. RCS pressure 560 psig and rising. Pressurizer level 294 inches and rising. RCS Temperatures: Loop A TH = 590DegF TC 340DegF and rising Loop B TH 620DegF TC 180DegF OTSG "B" isolated and full. OTSG "A" without heat sink, pressure 44 psig and falling, and nearly full.		Calculated decay heat = 14 MW.	There is no indication of natural circulation. Very little of the decay heat is being removed, except by make-up water and by occasional opening of PORV block valve. Gradual heatup of RCS is causing temperature and pressure to rise. Attempting to control pressure by juggling make-up and PORV block valve.	
	12 hours 6 min.	Pressurizer spray valve closes				6
	12 hours, 11 min.	Computer printout of selected incore thermocouples.	Utility printer		Almost all offscale.	1,8
	12 hours, 14 min.	Auxiliary building exhaust fans restarted.	Stripchart (panel 25)			3
	12 hours, 22 min.	Pressurizer level goes offscale.	Stripchart (panel 4) Meter (panel 5)			1,6
	12 hours, 30 min.	Open PORV block valve.	Status light (panel 4)		Attempting to depressurize further. PORV outlet alarms 5 minutes later	1,2,3,4
	12 hours, 40 min.	Close PORV block valve.	Status light (panel 4)		Closing time in doubt; Ref. 2 has 12 hrs 46 min.	1,2
	12 hours, 48 min.	Pressurizer level back on scale.	Stripchart (panel 4) Meter (panel 5)			3,6
	12 hours, 52 min.	Open PORV block valve. (?)	Status light (panel 4)	RCDT temperatures suggest the block valve remains closed the rest of 3/28.	This is extremely doubtful. Safety valve alarms clear a few minutes later, which is inconsistent with opening.	2

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	13 hours	Plant status: RCS at low pressure without secondary heat sink. Make-up pump MU-P1B operating. RCS Pressure: A=613 psig B=613 psig RCS Temperatures: THA=522 DegF All others offscale. Pressurizer level 379 inches Steam pressures: A=95 psig B=172 psig Status of PORV block valve not positively known; believed to be closed.		Calculated decay heat=13.8 MW		10,12
	13 hours, 2 min.	Start condenser vacuum pumps VA-P1A and 1C.	Status light (panel 17)	Condenser vacuum will be restored in a few minutes.	The auxiliary boiler has finally been returned to service and is now supplying turbine gland seal steam (this is a necessary prerequisite to using the condenser). Pump 1A trips, but is restarted 10 minutes later.	1,2,3,4
	13 hours, 20 min.	Reactor building pressure starts to go negative. Pressurizer level starts to drop. RCS pressure 637 psig and falling. Pumping 425 gpm with two HPI pumps. It is now the the intention to repressurize, collapse bubbles (hopefully) and begin steaming from OTSG "A".		High points were actually hydrogen filled. Collapse of loop bubbles was still impossible.	It is operators belief that main condenser will soon be available.	1,5,7
	13 hours 23 min.	Start makeup pump MU-P1C	Annunciator (panel 8) Meter and status lights (panel 3)			1,2,3,4
	13 hours, 25 min.	PORV outlet temperature alarm clears. (Block valve closed-time not known)			Indicates valve is definitely closed.	1,4
	13 hours, 26 min.	Pressurizer heater groups 1 and 2 trip.	Status light (panel 4)			1,2,3,4

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	13 hours, 38 min.	RCS pressure bottoms out at 611 psig and begins increasing.	Utility printer Meter and strip-chart (panel 4)			1,5,8
	13 hours, 45 min.	OTSG "A" now steaming.	Meter (panel 4) (Steam pressure)		Some difficulty earlier encountered with outlet valve--now cleared.	1,5,7
	13 hours, 52 min.	OTSG "A" high level alarm clears at 81.3%.	Annunciator (panel 17) Meter (panel 4)		Indicates some heat transfer--steaming down.	1,4
	13 hours, 59 min.	OTSG "A" high level alarm again.	Annunciator (panel 17) Meter (panel 4)		Indicates now feeding OTSG.	1,4
	14 hours	Plant status: RCS at low pressure; pressure increasing. EMOV block valve closed. Makeup pump MW-P1B operating. RCS Pressure: Loop A=857 psig Loop B=868 psig RCS Temperatures: THA=549 DegF All others offscale. Pressurizer level 312 inches		Calculated decay heat=13.5MW		10,12
	14 hours, 20 min.	RCS pressure 1200 psig, BWST 23'.	Meter and strip-chart (panel 4)			1,5
	14 hours, 25 min.	Pressurizer heater groups 1 and 2 returned to service.	Status light (panel 4)			1,3,4
	14 hours, 39 min.	Closing valve MU-V16B. MU flow now 120 gpm. RCS pressure 2080 psig.	MU-V16B: Status light (panel 3) Flow: Meter (panel 8) RCS P.: Meter and stripchart (panel 4)		RCS is now fully re-pressurized. Valve is throttled to reduce flow.	1,5,6
	14 hours, 41 min.	Cutting back on valve MU-V16C. MU flow 105 gpm.	MU-V16C: Status light (panel 3) Flow: Meter (panel 8)			1,5

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	14 hours, 43 min.	Stop make-up pump MU-P1C. MU-V16C closed. RCS pressure 2275 psig.	MU-P1C: Annunciator (panel 8) Meter and status light (panel 3) RCS P.: Meter and stripchart (panel 4)			1,4,5
	14 hours, 47 min.	Holding at 2300 psig. Operators have now decided to "bump" a reactor coolant pump.	Meter and strip-chart (panel 4)		BWST level 22 ft. 80 gpm letdown flow. 32 gpm seal injection, 20 gpm make-up flow.	1,5
	14 hours, 48 min.	Alarm printer fails. Not available until 15 hrs. 10 min.				1,4
	14 hours, 59 min.	Many radiation monitors come back on scale.	Stripcharts (panel 12)			2,3
	15 hours	Plant status: PORV block valve closed. MU-P1B operating. RCS pressures: A=2285 psig B=2304 psig Attempting to collapse bubbles.		calculated decay heat=13.2 MW		10,12
	15 hours, 10 min.	Alarm printer back in service, but almost illegible.				1,4
	15 hours, 11 min.	Computer prints out reactor coolant pump and make-up pump status on request.	Utility printer		Only MU-P1A now operating.	1,8
	15 hours, 15 min.	Start DC reactor coolant pump oil lift pumps.	Auxiliary operator action		AC pumps not operable, due to power loss at motor control centers. Had to send personnel to auxiliary building to start. RCP's will not start without oil lift pump running.	1
	15 hours, 16 min.	Full condenser vacuum reestablished.	Stripchart (panel 17)			1,5

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	15 hours, 22 min.	Start condensate booster pump CO-P2B.	Annunciator (panel 17) Meter and status light (panel 5)		To complete filling of OTSG "B".	1,4
	15 hours 32 min.	Start makeup pump MU-P1C	Annunciator (panel 18) Meter and status lights (panel 3)		1,2	
	15 hours, 32 min.	Stop condensate booster pump CO-P2B.				1,4
	15 hours, 33 min.	Start reactor coolant pump RC-P1A. Ran for 10 sec., then stopped.	Status light (panel 4) Meters: amperage, flow (panel 4)		Starting amperage normal, flow OK. RCS pressure and temperature immediately drop, then start to rise again. ESF actuates, but was bypassed.	1,2,3,4,5
	15 hours, 39 min.	Stop make-up pump MU-P1C.	Annunciator (panel 8) Meter and status light (panel 3)			1,3,4
	15hrs 49m	Start make-up pump MU-P1C.				1,3,4
	15 hours, 50 min.	Start reactor coolant pump RC-P1A.	Status light (panel 4) Meters and strip-chart (panel 4)	Adequate core cooling now has been established.	Satisfactory operation.	1,2,3,4
	15 hours, 55 min.	Stop decay heat pumps DH-P1A and 1B.	Status lights and meters (panel 3)			1,4
	15 hours, 56 min.	Stop make-up pump MU-P1C.	Annunciator (panel 8) Status light and meter (panel 3)			1,3,4

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/28/79	16 hours	Plant status: Reactor coolant pump RC-P1A is operating, make-up pump MU-P1B is running, the plant is now being well cooled with a heat sink to the condenser. Reactor coolant flow 28 million lb/hr. Pressurizer level 400 inches. RCS pressure 1310-1330 psig. Loop A: TH 520 Deg F T_C = Deg F Loop B: T_H Deg F T_C = 282 Deg F Steam pressure (OTSG "A") = 76 psig (OTSG "B") = 99 psig Steam generator levels: A = 414 inches B = 393 inches. WDL-T8B to Unit 1. This tank had been filled before the accident, and was now being emptied to accept water from the auxiliary building sump.		A bubble of noncondensible gas had collected in the upper head of the reactor pressure vessel. Calculated decay power = 12.9 MW.	Steam generator B is isolated.	6,7
	18 hours, 18 min.	Bubble reestablished in pressurizer.	Meter (panel 5) Stripchart (panel 4)		Went back offscale at 18 hrs. 30 min.	1,5,6
	17 hours, 25 min.	Valve DH-V187 from the decay heat pumps to the RCS was opened.			Indicates intention to depressurize.	5
	17 hours, 29 min.	Commenced transfer of material from Auxiliary Building Neutralizer Tank	Operator action			1,3
	18 hours, 34 min.	Letdown flow is lost.	Flowmeter (panel 3)	Probably due to plugging with boric acid.		3

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/29/79	(All times after 0000 on Mar. 29 are given as time of day) 0000.	RCS pressure 1026 psig RCS temp. 240DegF Pressurizer level 362 inches. Steam pressure (A) 25 psig.				
	0020	Stopped transfer fom WDL-T8B to Unit 1.				1,5
	0051	High pressure drop observed across letdown prefilters.	Alarm printer			5,4
	0055	Secured auxiliary building and fuel handling building ventilation.				1,5
	0210	Restarted auxiliary building and fuel handling building ventilation.				1,5
	0211	The control room gas and particulate radiation monitors showed high levels. Control room personnel donned masks.	Stripcharts (panel 12)			5
	0300	Pressurizer level and RCS pressure dropping slowly. Loop A: TC = 238DegF Flow = 28 million lb/hr Pressurizer temperature 549DegF RCS pressure 1028 psig Pressurizer level 400 inches OTSG "A" at 95%.				5,6
	0315	Control room radiation monitors dropped and respirators were removed.	Stripcharts (panel 12)			5
	0400	Plant status: RCP-1A running. Loop A: TC = 234DegF Loop B: TC = 233DegF Flow 28 million lb/hr RCS Pressure = 998 psig Pressurizer Temperature 547DegF Pressurizer Level 394 inches.				5,6

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/29/79	0435	Vented make-up tank MU-T1 to vent header by opening MU-V13.				1,5
	0443	Seal water high temperature alarm on RCP 2A. Requested seal water temperatures on all RCPs.	Alarm printer Utility printer		High temperatures on RC-P1B, 2A, 2B (all nonoperating).	4,5,8
	0504	RCP seal water temperature alarms cleared.	Alarm printer			4
	0510	RCS pressure 969 psig TC "B" = 284DegF Pressurizer temperature 543DegF Pressurizer level 352.5 inches.				5
	0615	RCS pressure 945 psig TC "B" 284DegF Pressurizer temperature 540DegF Pressurizer level 341 inches BWST 20.5 feet.				5
	0630	Sprayed down pressurizer. Level rose from 345 inches to 367 inches, pressure dropped 50 psi.	Level: Meter (panel 5) Stripchart (panel 4) Pressure: Meter (panel 4) Stripchart (panel 4)			5
	0631	Letdown flow (25 gpm) reestablished after raising intermediate cooling temperature.	Meter (panel 3)			5
	0710	RCS pressure 899 psig TC "B" = 283DegF Pressurizer level 352 inches.				5
	0715	Pumped auxiliary building sump tank to auxiliary building neutralizer tank.				1,5

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/29/79	0716	Letdown flow shifted to RCBHT "B".			When makeup tank was vented, aux. bldg. radiation increased.	1,5
	0845	Commenced transfer from WDL-T8A to Unit 1.		Calculated decay heat at 1000 = 10.3 MW.	Preaccident water being transferred to make room in tank.	1,5
	1215	Plastic sheet put down on aux. bldg. floor to reduce rate of release.				1
	1240	Shut off turbine building, control bldg., control and service bldg. sump pumps.			High level in industrial waste treatment system. Overflowing and draining to settling pond. Leakage to river.	1
	1315	Started industrial waste treatment system.			Discharges to river.	1
	1410	Shut down industrial waste treatment system.		Xenon measurement was false.	Because of apparent Xenon release.	1
	1458	Shifted letdown from RCBHT "B" to "C".				5
	1600	Pumped auxiliary building sump tank to WDL-T8A.			Will later pump sump to sump tank.	1,5,9
	1610	Restarted industrial waste processing system.				1
	1815	Stopped industrial waste treatment system.				1
	1900	Washed down auxiliary building floor under the plastic.			After pumping sump to sump tank.	5
	1920	Letdown flow was 20 gpm.	Meter (panel 3)			9
	1945	Lined up M.U.T. degassing system through Unit 1 sample system to Unit 2 vent header.				9

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/29/79	2020	Started degassing M.U.T. via sample system.			Secured 10 min. later.	1,5
	2035	Opened MU-V13 for 5 sec.			Vents makeup tank to vent gas system.	9
	2036	Isolated nitrogen to waste gas header.			To keep pressure down.	9
	2040	Significant increase in fuel handling building exhaust gas monitor.	Stripchart (panel 12)		From 300 mr/hr to 1 r/hr	9
	2045	MU-T1 vented to waste gas header.			Cautiously (to keep in waste gas header down). Reduce M.U.T. pressure to 55 psi.	1,5
	2105	Secured venting MU-T1.				9
	2114	LT2 pressurizer level indicator failed.	Alarm printer		Returned to service at 2230.	4,5,9
	2200	Decided no leak in OTSG "B".			Pressure steady at 25 psig. Level steady at 380 inches.	5
	2330	Vented MU-T1 to waste gas vent header.			Cycling MU-V13 at 2 sec. periods.	5,9
	2400	Now believed steam bubble in reactor vessel. TC "A" 325DegF RCS pressure 1105 psig Pressurizer level 325 inches.				5
3/30/79	0058	MU-T1 level decreasing, pressure increasing.				5,9
	0130	Shut turbine bypass value for 5 minutes.			Temperature increased 8DegF.	5,9
	0150	Vented MU-T1 to waste gas decay tank WDG-T1B.			Secured at 0215.	1,5,9

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/30/79	0155	Secured transfer from aux. bldg. sump tank to neutralizer tank WDG-T8A.				1
	0215	Shut off all sump pumps from turbine building and control building area.				1,5,9
	0315	Pumped control building area sump to turbine building sump.			Using temporary pump.	1,5,9
	0330	Vented MU-T1 to waste gas decay header.			Secured at 0350 Tank pressure: A = 50 psig, B = 80 psig.	1,5,9
	0346	Cycling MU-V376.			To try to reestablish letdown.	5,9
	0430	Started industrial waste discharge filter system, discharging to river from mechanical draft cooling tower blowdown line.			Sump level 76%.	1,5,9
	0435	Liquid pressure relief valve MU-R1 on MU-T1 opened, venting MU-T1 to reactor coolant bleed holdup tanks. MU-T1 level dropped to zero. Shut MU-V12. Seal flow dropped. Pressure in RCBHT's went offscale. Realigned make-up to BWST.			Increase in gas discharge, coincident with venting.	1,5,6,9
	0530	Flow to RCP seals adjusted to 7.2 gpm each, using needle values.				5,9
	0710	Venting MU-T1 to vent header via MU-V13.				1,5,9
	0750	Started waste transfer pump WDL-P5A pumping from RCBHT to MU-T1.		Calculated bubble at 0730--893 cu.ft. (CPU).	Unsuccessful because of high pressure in MU-T1 (80-84 psi) stopped at 0753.	1,5,9
	0753	Added 371 gals demineralized water to MU-T1 and boric acid from CA-T1.			So as not to draw from BWST. Finished at 0800.	5,9

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/30/79	0805	Secure seal water injection to non-operating pumps RC-P2A, 1B, 2B.			Log says 1A; obvious error.	5,9
	0815	Open MU-V12, shut DH-V5A, switch MU-P1A to MU-T1. Commenced adding water and boric acid to MU-T1.			Finished at 0820. Added 300 gal. shift suction from BWST to MU-T1.	5,9
	0855	Sent personnel to start hydrogen recombiner.				5
	0900	Venting MU-T1.				5,9
	0908	Shut DH-V5B.				5,9
	0940	Shut off OTSG "A".			To heat RCS to 280DegF for 7 minutes.	5,9
	1045	Closed MU-V17. Commenced bleeding letdown to RCBHT "A". Began reducing pressurizer level to 100".				
	1120	RCS status: TCA 280DegF Pressure 1043 psig. Pressurizer level 390 inches. Pressurizer temperature 560DegF.				9
	1220	Started transfer of miscellaneous waste tank to Unit 1.		Calculated bubble at 1240 (GPU) = 308 cu.ft.		1,5
	1405	Attempted to open WDG-V30B to vent WDG-T1B into reactor building.			Unsuccessful. Finally opened at 1442.	1,5,9
	1410	Switched letdown from MU-T1 to RCBT "A".			Switched back to MU-T1 at 1420.	9
	1442	Venting waste gas decay tank "B" WDG-T1B to reactor building.			Stopped at 1450.	1,9
	1502	Added 462 gal. from RCBHT "A" to MU-T1.				5
	1530	Fuel handling exhaust unit ARM and aux. bldg. access corridor ARM climb from 240 mr/hr and 70 mr/hr at 1145 to 700 and 160 mr/hr.	Stripcharts (panel 12)		Decline slowly to 100 and 35 mr/hr on 4/1/79.	9

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/30/79	1600	RCS status: TCA = 280DegF, pressure 1049 psig, pressurizer level 215 inches, pressurizer temperature 557DegF, BWST level 15.5 ft.				9
	1634	Turned off all pressurizer heaters to calculate rate of RCS pressure drop.	Status lights (panel 4) RCS pressure: Meter and stripchart (panel 4)			5,9
	1650	Letdown temperature high. Opened valve MU-V376 to cool down.	Stripchart (panel 10)		Cleared at 1655.	9
	1704	Started RCP-2A oil pump. K3 relay failed and pump tripped.			Ground fault.	5,9
	1719	Added 200 gal from RCBTA to MU-T1.				5
	1730	Lining up to pump from Unit 1 spent fuel tank to Unit 2 surge tank and then to Unit 2 BWST.				9
	1810	Found and replaced blown fuse on RC-P2A control circuit.				
	1850	Starting to refill BWST at 4000 gal/hr.		Calc. bubble at 1907 (GPU) = 1806 cu.ft.	CR log says being filled from Halliburton truck. BWST level 15.5 ft.	5,9
	1920	Switched letdown to RCBHT "C".			RCBT "A" filled.	5
	1945	Isolated letdown from RCBHT's.				5,9
	2036	Added 300 gal from RCBHT "A" to MU-T1.				9
	2053	Shut off feedwater to OTSG "A".			Steaming down.	9
	2132	Venting pressurizer to RCDT.				
	2200	Oil pumps on RCP-2A tested satisfactorily.			Until 2217.	5,9
	2229	Transferring misc. waste tank to Unit 1.				5

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/30/79	2240	Restored feed to OTSG "A". TCA 285DegF, pressure 1029 psig, pressurizer level 215 inches, BWST 16.5'.				9
	2310	Starting to vent pressurizer again.			Completed 0140, 3/31.	5,9
	2330	"Gas bubble" noted for first time in C.R. log.			Volume given as 400 cubic ft.	5
	2347	Added 300 gal. from RCBHT "A" to MU-T1. Pressure in MU-T1 at 43.5 psig.				5
3/31/79	0145	Venting RCS.				5,9
	0205	Reactor building equipment hatch contact reading 60 r/hr. WDG-T1A and 1B contact readings 40 r/hr.				5,9
	0315	Secured venting. Waiting for hydrogen recombiner to be placed in operation.				5,9
	0325	Shift supervisor, shift foreman, and CROs reviewed Emergency Procedure for loss of RC-P1A.				5,9
	0400	RCS status: TCA 282DegF, pressure 1060 psig, pressurizer level 215 inches, pressurizer temperature 550DegF, BWST level 18 ft.		Calculated decay power = 7.43 MW.		9
	0423	Auxiliary operators instructed not to enter auxiliary building without a "teletector".				9
	0546	Pressure in MU-T1 is 32 psig.				5
	0518-0638	Taking hydrogen samples from reactor building.				5,9
	0548	Turbine bypass valves from OTSG "A" closed from 47% open to 44% open to heat up RCS.				9

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/31/79	0735	Reduced pressure in RCS to 1025 psig using pressurizer spray. Level after spraying was 233 inches.				5,9
	0753	Commenced venting pressurizer while heating and spraying simultaneously.			Secured at 0803.	9
	0828	Venting pressurizer (same as 0753).			Secured at 0846.	5,9
	0907	Venting pressurizer.			Secured at 0917.	5,9
	0930	Sump and tank levels: MWHT, 7 ft; Aux. Bldg. sump 3.2 ft/ aux. bldg. sump tank 3.4 ft; waste gas vent heater 20 psig. Auxiliary sump tank lined up to MWHT.				5,9
	0935	Venting pressurizer.			Secured at 0957.	5,9
	0950	Drained spent fuel surge tank to Unit 2 BWST.		Calc. bubble (GPU) at 1032 = 860 cu.ft.		5,9
	1312	Venting pressurizer.			Secured at 1350.	
	1344	Transferring water from Unit 1 spent fuel fuel pool to Unit 2 spent fuel surge tank with two sump pumps. Pumping (intermittently) to Unit 2 with SF-P1A.				5,9
	1425	Venting pressurizer.			Secured at 1500.	5,9
	1511	Halted transfer from Unit 1 spent fuel to Unit 2 BWST, until spent fuel refilled. BWST level 26.5 ft.				5,9
	1537	Cracked pressurizer vent valve.			Closed at 1619.	5,9
	1542	Secured turbine bldg. ventilation.				5,9
	1656	Venting pressurizer.			Secured at 1737.	5,9
	1741	Pressure in MU-T1 vent to zero. Closed MU-V13.				5,9
	1815	Cracked pressurizer vent valve.			Closed 1850.	5,9

Date	Time [illegible]	Event	Information available to operators	Post-accident calculations and data	Remarks	References
3/31/79	1858	Opened MU-V13. MU-T1 pressure equalized with waste gas vent header. Discharge level increased.				9
	1950	Cracked pressurizer vent valve.			Closed at 2034.	5,9
	2110	Pressurizer venting: 2110-2139 2221-2352				5,9
	2124	Transferring water from SF surge tank to BWST.		Calc. bubble (GPU) at 2245 = 894 cu.ft. B+W = 487 cu.ft.		5,9
4/1/79	0016	Pressurizer vented at same frequency throughout the day.				5,9
	0029	Opened bypass valve on OTSG "A" slightly to compensate for higher RCS temperature.				9
	0750	Stopped transfer of water to BWST BWST level 40.5 ft.		Calc. bubble (GPU) at 0731 = 564 cu.ft.		5
	0930	Transferring MWHT WDL-T2 to Unit 1 Reactor building hydrogen concentration 2% throughout day.				5
	1500	Reduced RCS pressure to 1000 psig.				9
	2020	WDG-T1A and 1B is 86 psig.				9
4/2/79	1000	Lost auxiliary boiler for 2 minutes.				9
	1347	Hydrogen recombiner in service.		Calc. bubble at 1315 (GPU) 174 cu.ft..		9
4/3/79	0906	Reduced steaming on OTSG "A".		Calculated decay power = 5.4 MW.		9
	0950	Slowly raised OTSG level to 97%.				9
	1830	DC ground faults, RCP alarms.				9

Date	Time after initiation	Event	Information available to operators	Post-accident calculations and data	Remarks	References
4/3/79	2400	Plant status: TCA 281DegF T_{HA} 281DegF T_{CB} 281DegF T_{HB} 278DegF Pressure 1050 psig BWST level 54 feet Reactor building pressure 1.3 psig Reactor building temp. 87.5DegF.		Bubble gone.		

II-D Alternative Accident Sequences

In this section, a number of accident sequences are discussed which are somewhat different from the actual accident progression. These "alternative sequences" have been established and evaluated to address particular questions which have arisen from the Special Inquiry Group's (and other group's) investigation of the accident.

Section II-D-1 deals with methods for amelioration of the accident. More specifically, Section II-D-1.1 is related to methods by which the damage to the fuel could have been ameliorated, and the reasons for the lack of success of the "anticipated" procedures. Section II-D-1.2 provides a comparable description related to the amelioration of the releases of radioactive material.

Section II-D-2 describes analyses which were performed to address specific questions concerning the effect of certain operator actions (or inactions) or equipment failures. This section thus addresses a number of "what if" questions such as "what if the operators had not reduced the high pressure injection system flow," or "what if the PORV block valve had not been closed when it was."

II-D-1 Amelioration of the Accident

II-D-1.1 Amelioration of Fuel Damage

As is discussed in Section II-C, the integrity of the TMI-2 core was threatened primarily during the first 16 hours of the accident. The majority of the

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damage was done in the time period from about 2 to 4 hours into the accident; the remainder of the 16 hours was then spent in attempts to recover from this damage. The subject of the time of final refilling of the core region continues as a point of controversy. Because of this controversy, the matter of the effectiveness of the various core cooling methods attempted between 4 and 16 hours also remains unresolved. However, there appear to be some methods not utilized which would have been more likely to succeed. These include:

PORV Block Valve Closure

The PORV discharge line temperature was obtained from the plant computer a number of times between the beginning of the accident and when the block valve was closed at 2.3 hours. The first such time was at 25 minutes. Had the block valve been closed at this time, the loss of coolant would have been ended before significant amounts of water were lost. During similar incidents involving stuck-open PORVs (TMI-2 on March 29, 1978, Davis-Besse on September 14, 1977, Oconee 3 on June 13, 1975), block valve closure occurred during roughly comparable time periods. Since no fuel damage resulted from these similar events, it can reasonably be expected that no fuel damage would have resulted at TMI-2 had the block valve been closed at 25 minutes.

Computer analysis has been performed for the Special Inquiry Group to further evaluate the effect of block valve closure at 25 minutes. This analysis, discussed in greater detail in Section II-D-3, also indicates that valve closure at 25 minutes would have stopped the accident before serious damage to the fuel began (Ref. 1,.2).

At the times of subsequent requests for discharge line temperatures, additional coolant had been lost from the reactor coolant system. After some time period, this loss of coolant would have become sufficiently great that closure of the block valve would not, in itself, have reversed the deteriorating situation. In these cases, the use of the high pressure injection system in one of the modes discussed below would also have been necessary.

Use of the High Pressure Injection System

Effective use of the high pressure injection (HPI) system would have provided (and actually eventually did provide) the means to cool down the RCS and the core. Several modes of HPI system use, by itself or in conjection with other systems, were possible. These included:

-- Continuous high pressure injection system flow at high flow rates.

The high pressure injection (HPI) system was automatically actuated a number of times during the first 16 hours of the accident. Because of the continued reliance on pressurizer level instrumentation, the flow rate from the system was substantially reduced by the operators following each actuation.

Operation of the HPI system at its full flow rates would have repressurized the reactor coolant system (RCS) and refilled it with coolant. With the RCS again filled with water (with some pockets of noncondensible gas after some time into the accident), a flow path from the HPI system, into the RCS, and out the PORV and safety valves would have been established. In this mode of HPI system use, heat removal from the RCS is achieved by the heatup of water as it passes through the RCS. Upon depletion of the normal source of water from the HPI system (the borated water storage tank), a flow path from the reactor building emergency sump and using the water lost out the PORV and safety valves could have been established.

This method of cooling was not attempted on March 28, apparently because of the reluctance of the crew in the control room to force open the safety valves (Ref. 3).

-- Use of the HPI system in conjunction with reactor coolant pump operation.

Between the times of 100 minutes and 16 hours, all reactor coolant pumps (RCPs) were off (with two brief exceptions), so that forced flow cooling of the core was not occurring. Restart of one of these pumps would have reestablished forced flow cooling to the core, with heat removal being achieved through the OTSGs. However, attempts to restart an RCP in this time period met with limited success, apparently because of low water inventories and pressures in the RCS. Use of the HPI system in support of the restart of an RCP would have provided the needed additional coolant and pressure. Thus the combination of HPI system use and restart of an RCP should have been successful in cooling down the reactor core.

Between the 4 and 16 hour period, RCS pressure was increased to high pressures twice -- once at about 5 to 6 hours and maintained for over 2 hours, and again at about 14 to 15 hours. During the latter repressurization, a reactor coolant pump was started, providing the long-term, stable method of cooling. There is no evidence that attempts to start a reactor coolant pump were made during the earlier repressurization.

The "anticipated" procedure to be followed during a loss-of-coolant due to a small break in the RCS would be to allow the automatically-actuated high pressure injection system to operate. For a "break" such as a stuck-open PORV, the HPI would restore RCS pressure and coolant inventory and maintain core cooling. When RCS pressure and pressurizer level are restored to specific levels (as defined in the emergency procedures), HPI system flow is supposed to be decreased by valve manipulation.

This anticipated procedure apparently did not work for a number of reasons. These include:

-- Operator failure to recognize that a loss-of-coolant accident was occurring,

-- Pressurizer level indication was misleading,

-- Operator lack of understanding on how to recover from such an accident, once recognized.

Table II-D-1

Description of Alternative Accident Sequences and Results

Accident Sequence	Parameter Analyzed	Computer Co_ Used RELAP (Ref. 1)	TRAC (Ref. 2)	MARCH (Ref
Base Case --Reactor coolant pumps tripped at 73-100 minutes --Emergency feedwater delivered at 8 minutes --PORV block valve closed at 2.3 hours --High pressure injection system in "degraded" mode (throttled back from full flow)		X	X	X
Alternative Sequence 1 (Section II-D-2.1) --High pressure injection system allowed to operate at full flow rates.	Effect of operator decision to substantially throttle back HPI flow.			X
Alternative Sequence 2 (Section II-D-2.2) --High pressure injection system allowed to operate at full flow rates, and --Emergency feedwater delivery to OTSGs at 1 hour	Capability of HPI system to cool core without heat removal from OTSGs		X	X
Alternative Sequence 3 (Section II-D-2.3) --Emergency feedwater delivery to OTSGs at about 40 seconds	Effect of closure of EFW block valves until 8 minutes.		X	X
Alternative Sequence 4 (Section II-D-2.4) --Emergency feedwater delivery to OTSGs at about 1 hour	Effect of a more prolonged closure of the EFW block valves		X	X

/CORRAL (. 3)	Results
	Core continuously cooled-no fuel damage.
	Core continuously cooled-no fuel damage
	Little significant change from base case

Accident Sequence	Parameter Analyzed	Computer Cod RELAP (Ref. 1)	 TRAC (Ref. 2)	 MA (
Alternative Sequence 5 (Section II-D-2.5) --Closure of the PORV block valve at 25 minutes	Effect of operator error in not closing the block valve after the first check of PORV discharge line temperature	X		
Alternative Sequence 6 (Section II-D-2.6) --Closure of the PORV block valve at 3.3 hours	Effect of a more prolonged operator error before closure of the block valve			
Alternative Sequence 7 (Section II-D-2.7) --One reactor coolant pump per loop shutdown at 73 minutes.	Effect of method of shutting down RCPs, i.e., both B loop pumps first, then A loop pumps 28 minutes later.	X		
Alternative Sequence 8 (Section II-D-2.8) --All reactor coolant pumps shut down at time of reactor trip.	Effect of cooling provided by forced flow from the RCPs	X	X	
Alternative Sequence 9 (Section II-D-9) --No reactor coolant pump restart at 16 hours	Effect of timing of the pump restart			
Alternative Sequence 10 (Section II-D-2.10) --Loss of offsite AC power at 1/2 to 5 hours	Effect of operator decision to negate emergency AC power actuation system			
Alternative Sequence 11 (Section II-D-2.11) --Loss of offsite AC power during March 30 to April 1	Effect of loss of forced flow from the one operating reactor coolant pump			

e Used RCH/CORRAL Ref. 3)	Results
X	Core continuously cooled-no fuel damage
X	Substantial fuel melting
	Core continuously cooled-no fuel damage
X	
X	Core likely would be cooled down at slower rate.
X	Operator action to restore diesels required within about 15 minutes to prevent substantial fuel melting.
	Options available to prevent further core damage.

Accident Sequence	Parameter Analyzed	Computer Code Used: RELAP (Ref. 1)	TRAC (Ref. 2)	MARCH/CORRAL (Ref. 3)	
Alternative Sequence 12 (Section II-D-2.12) --Recriticality	Recriticality resulting from fuel and control rod damage.				Rec min
Alternative Sequence 13 (Section II-D-2.13) --Effect of containment design	Containment design pressure (compared to ice condenser design)				Ice cou dam

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condenser containment
d have been severely
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Table II-D-1

Description of Alternative Accident Sequences and Results

Accident Sequence	Parameter Analyzed	Computer Code Used: RELAP (Ref. 1)	Computer Code Used: TRAC (Ref. 2)	Computer Code Used: MARCH/CORRAL (Ref. 3)	Results
Base Case --Reactor coolant pumps tripped at 73-100 minutes --Emergency feedwater delivered at 8 minutes --PORV block valve closed at 2.3 hours --High pressure injection system in "degraded" mode (throttled back from full flow)		X	X	X	
Alternative Sequence 1 (Section II-D-2.1) --High pressure injection system allowed to operate at full flow rates.	Effect of operator decision to substantially throttle back HPI flow.			X	Core continuously cooled-no fuel damage.
Alternative Sequence 2 (Section II-D-2.2) --High pressure injection system allowed to operate at full flow rates, and --Emergency feedwater delivery to OTSGs at 1 hour	Capability of HPI system to cool core without heat removal from OTSGs		X	X	Core continuously cooled-no fuel damage
Alternative Sequence 3 (Section II-D-2.3) --Emergency feedwater delivery to OTSGs at about 40 seconds	Effect of closure of EFW block valves until 8 minutes.		X	X	Little significant change from base case
Alternative Sequence 4 (Section II-D-2.4) --Emergency feedwater delivery to OTSGs at about 1 hour	Effect of a more prolonged closure of the EFW block valves		X	X	

Accident Sequence	Parameter Analyzed	Computer Code Used: RELAP (Ref. 1)	Computer Code Used: TRAC (Ref. 2)	Computer Code Used: MARCH/CORRAL (Ref. 3)	Results
Alternative Sequence 5 (Section II-D-2.5) --Closure of the PORV block valve at 25 minutes	Effect of operator error in not closing the block valve after the first check of PORV discharge line temperature	X		X	Core continuously cooled-no fuel damage
Alternative Sequence 6 (Section II-D-2.6) --Closure of the PORV block valve at 3.3 hours	Effect of a more prolonged operator error before closure of the block valve			X	Substantial fuel melting
Alternative Sequence 7 (Section II-D-2.7) --One reactor coolant pump per loop shutdown at 73 minutes.	Effect of method of shutting down RCPs, i.e., both B loop pumps first, then A loop pumps 28 minutes later.	X			Core continuously cooled-no fuel damage
Alternative Sequence 8 (Section II-D-2.8) --All reactor coolant pumps shut down at time of reactor trip.	Effect of cooling provided by forced flow from the RCPs	X	X	X	
Alternative Sequence 9 (Section II-D-9) --No reactor coolant pump restart at 16 hours	Effect of timing of the pump restart			X	Core likely would be cooled down at slower rate.
Alternative Sequence 10 (Section II-D-2.10) --Loss of offsite AC power at 1/2 to 5 hours	Effect of operator decision to negate emergency AC power actuation system			X	Operator action to restore diesels required within about 15 minutes to prevent substantial fuel melting.
Alternative Sequence 11 (Section II-D-2.11) --Loss of offsite AC power during March 30 to April 1	Effect of loss of forced flow from the one operating reactor coolant pump				Options available to prevent further core damage.

Accident Sequence	Parameter Analyzed	Computer Code Used: RELAP (Ref. 1)	Computer Code Used: TRAC (Ref. 2)	Computer Code Used: MARCH/CORRAL (Ref. 3)	Results
Alternative Sequence 12 (Section II-D-2.12) --Recriticality	Recriticality resulting from fuel and control rod damage.				Recriticality potential minimal.
Alternative Sequence 13 (Section II-D-2.13) --Effect of containment design	Containment design pressure (compared to ice condenser design)				Ice condenser containment could have been severely damaged.

References - II-D-1.1

1. TMI Sensitivity Calculations (DRAFT), EG&G Idaho, Inc., November 15, 1979.

2. Analysis of the Three Mile Island Accident and Alternative Sequences (DRAFT), Battelle Columbus Laboratories, November 16, 1979.

3. (IE operator interviews)

II-D-2 Analysis of Alternative Accident Sequences

II-D-2.0 Introduction and Summary

The analysis of a set of alternative accident sequences has been undertaken as part of the Special Inquiry Group's work. The purposes of this analysis were to:

-- specifically assess the importance of various equipment failures and/or human actions (or inactions);

-- provide additional information on the physical phenomena occurring during the accident; and

-- aid in the assessment of how close this accident came to being a "core meltdown" accident.

To assist in the evaluation of certain alternative accident sequences, computer analyses were performed at the following locations:

-- RELAP code calculations at the Idaho National Engineering Laboratory;

-- TRAC code calculations at the Los Alamos Scientific Laboratory; and

-- MARCH code calculations at Battelle Columbus Laboratories.

The results of these calculations are discussed in the following sections. Detailed results may be found in References 1, 2, and 3.

The method by which the majority of the alternative sequences were determined was through the combination of critical parameters in an "event tree" logic. Such a tree displays the progression of the early portion of the accident (i.e., the first few hours) in terms of system operation and human actions. Variations in these parameters are displayed as "branches" in the event tree; thus, any variation becomes a different "branch," or alternative accident sequence in the overall event tree.

The progression of the early portion of the TMI accident is shown in Figure II-D-1. Four parameters, all of which were related to human actions, were identified as critical to this progression. The four parameters chosen were:

-- timing and method of tripping the reactor coolant pumps;

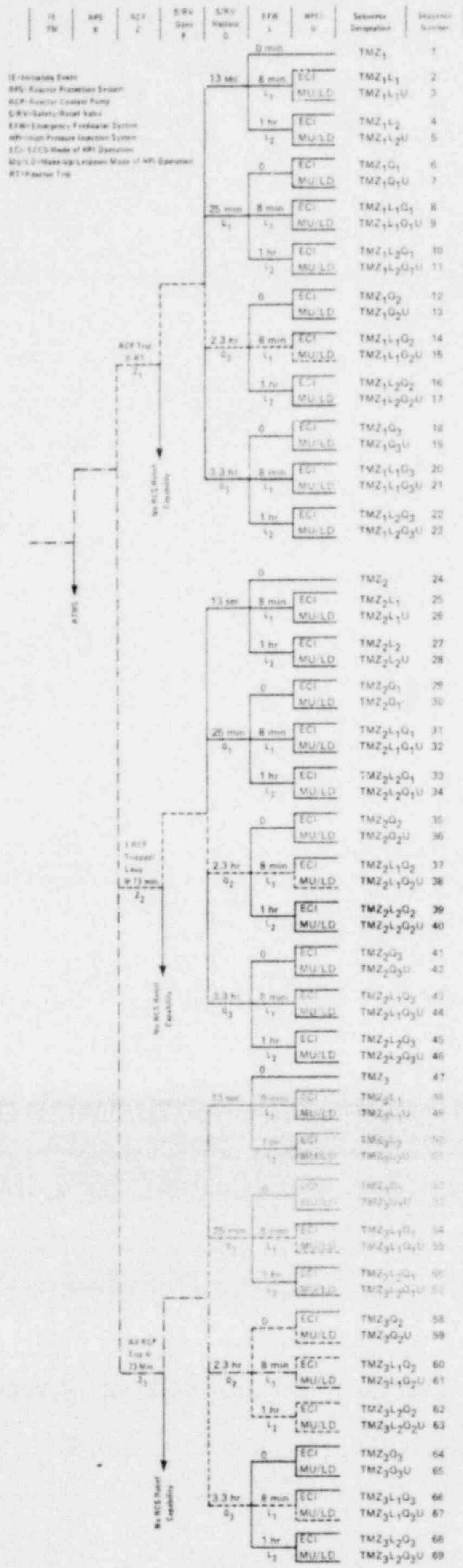
-- timing of closure of the PORV or its block valve;

-- timing of initial delivery of emergency feedwater system flow to the OTSGs; and

-- flow rate delivered from the high pressure injection system.

For each of these parameters a number of alternative values were chosen. With respect to the timing and method of tripping the reactor coolant pumps, three variations were chosen: (1) pump trip concurrent with reactor trip;

FIGURE II.D-1. Event Tree for Parameters Critical to Early Accident Progression



(2) one pump tripped per loop at 73 minutes; and (3) B-loop pump trip at 73 minutes and A loop pump trip at 100 minutes. Case 1 relates to the tripping of the reactor coolant pumps very early in transient, at the time of reactor trip. Case 2 relates to the possibility of prolonged pump operation from selective tripping of one pump in each loop. Case 3 is the "base case" of the actual timing of pump tripping.

Four variations in the time of closure of the PORV or the PORV block valve were defined. Times of closure were: (1) 13 seconds; (2) 25 minutes; (3) 2.3 hours; and (4) 3.3 hours. Case 1 relates to the normal timing of PORV closure following an interruption in flow from the main feedwater system and subsequent reactor trip. Case 2 relates to the timing of the first operator request for printout of the PORV discharge line temperature from the utility printer. Case 3 is the "base case" of the actual time of PORV block valve closure. Case 4 adds an additional 1 hour delay in closure of the PORV block valve beyond that actually experienced.

Three variations in the timing of initial emergency feedwater (EFW) flow into the OTSGs were analyzed. Times of delivery were: (1) 40 seconds; (2) 8 minutes; and (3) 1 hour. Case 1 relates to the normal time of EFW flow initiation into the OTSGs, had the discharge line block valves not been closed. Case 2 is the "base case" time of EFW delivery to the OTSGs. Case 3 relates to a delay in opening of the block valves at 1 hour rather than 8 minutes.

Two variations in the flow rates from the high pressure injection (HPI) pumps were examined. These cases were: (1) full HPI flow rate after actuation;

and (2) degraded HPI flow rates. Case 1 relates to the functioning of the HPI system without interference to throttle back flow rates. Case 2 involves the "base case" flow rates as actually resulted because of operator actions to reduce this flow.

Figure II-D-1 thus displays possible variations in the progression of events following the TMI-2 accident initiating event (i.e., the interruption of main feedwater flow), resulting from the parametric variations discussed above. From all of the possible accident sequences shown in Figure II-D-1, a set of nine sequences were chosen to examine the effects of variations in the four specific parameters. These nine cases are discussed in more detail below.

Base Case Accident Sequence

Accident sequence 61 in Figure II-D-1 is the sequence of events which actually occurred during the early portion of the TMI-2 accident. This sequence was recreated using the various computer codes to provide a basis to which the alternative accident sequences could be compared and to assist in the overall understanding of the accident. The RELAP, TRAC, and MARCH codes were all used to recreate the base case accident sequence, with RELAP and TRAC being used to analyze the time period of 0 to 2 1/2 hours and MARCH the time period of 0 to 16 hours. Detailed discussions of these analyses may be found in References 1, 2 and 3, respectively.

Alternative Accident Sequence 1

Accident sequence 60 in Figure II-D-1 was analyzed as alternative accident sequence 1. As may be seen from this figure, all parameters remain the same except that the high pressure injection system is allowed to operate at full flow rates rather than in the degraded mode resulting from operator actions to throttle back flow. This analysis thus shows the effect of the operator decision to throttle back the HPI flow. Detailed results of the analysis of this sequence may be found in Section II-D-2.1.

Alternative Sequence Analysis 2

Accident sequence 62 in Figure II-D-1 was analyzed as alternative accident sequence 2. As may be seen from the figure, two parameters are varied so that the high pressure injection system operates at full flow rates (rather than throttled back) and the delivery of emergency feedwater is delayed until about one hour into the accident. This sequence thus addresses the capability of the HPI system to provide adequate core cooling without heat removal through OTSGs. Detailed results of the analysis of this sequence may be found in Section II-D-2.2.

Alternative Accident Sequence 3

Accident sequence 59 in Figure II-D-1 was analyzed as alternative accident sequence 3. As may be seen from the figure, only one parameter is varied, this being the time of delivery of emergency feedwater flow. In this sequence, EFW is assumed to be delivered beginning at about 40 seconds, as if the EFW discharge line block valves had not been closed. This analysis thus shows the effect of these block valves being closed until 8 minutes into the accident. Detailed results of the analysis of this sequence may be found in Section II-D-2.3.

-- Failure by the plant personnel to recognize the significance of the PORV discharge line temperature readings was an additional highly significant contributor to the severity of fuel damage;

-- Had the PORV block valve not been closed at the time it was, substantial fuel melting could have occurred within an hour;

-- Either early reactor coolant pump trip or a single pump trip in each loop at 73 minutes could have prevented high fuel temperatures and minimized fuel damage.

-- The delay in delivery of emergency feedwater to the steam generators until 8 minutes had no appreciable affect on the accident progression.

II-D-2 Results of Alternative Accident Sequence Analysis

II-D-2.1 Alternative Accident Sequence 1: High Pressure Injection System Allowed to Operate at Full Flow Rates

At approximately two minutes into the accident, the high pressure injection (HPI) system was automatically actuated on a low reactor coolant system (RCS) pressure signal, resulting in the flow of 1000 gallons of water per minute into the RCS. Within 2 to 3 minutes, the operators had reduced the flow from the HPI system substantially, to the degree that the amount of water lost out the stuck-open PORV was greater than that supplied by the HPI system. Throughout the first 16 hours of the accident, the HPI system was automatically actuated a number of times; each time the high flow rates from the system were subsequently reduced by the crew.

In this alternative accident sequence, the high pressure injection system is assumed to have operated at full capacity from the initial actuation. Other parameters such as the delay of 8 minutes in delivery of the emergency feedwater are assumed to remain the same.

The results of the RELAP (Ref. 1) and MARCH (Ref. 3) calculations both indicate that the use of the HPI system at full capacity would have prevented the overheating of the fuel and the resulting release of radioactive material. These analyses show that the reactor coolant system would have remained essentially full and cool throughout the incident.

On the basis of the analysis performed for this alternative accident sequence, it is evident that the operating crew's decision to reduce the flow from the high pressure injection system was a major contributor to the severity of this accident.

II-D-2.2 Alternative Accident Sequence 2: High Pressure Injection System Allowed to Operate at Full Flow Rates, and Emergency Feedwater Delivered at 1 Hour

In this alternative sequence, the effect of HPI flow analyzed in alternative accident 1 is compounded with the effect of a prolonged human error in opening of the emergency feedwater system discharge line block valves. In the actual accident, these block valves were opened at approximately 8 minutes. In this alternative sequence, opening of these valves was delayed until 1 hour. This sequence in effect analyzes the capability of the HPI system to cool the core in the absence of heat removal through the OTSGs.

The TRAC (Ref. 2) and MARCH (Ref. 3) analyses of this alternative accident sequence are in general agreement; both indicate that fuel temperatures remain significantly lower than those achieved during the actual accident. Figure II-D-2 shows this difference in temperature based on the TRAC calculations.

The analysis of this alternative accident sequence thus indicates that adequate core cooling could have been achieved by the use of the HPI system at full capacity, even in the absence of heat removal through the steam generators (i.e., without the use of the emergency feedwater system).

II-D-2.3 Alternative Accident Sequence 3: Emergency Feedwater Delivered at 40 Seconds

In alternative accident sequence 3, it has been assumed that the emergency feedwater system discharge line block valves were not closed, so that EFW could have been delivered at about 40 seconds into the accident. The comparison of the results of this sequence to those of the base case thus shows the effect of the 8 minute delay in the initial delivery of EFW to the OTSGs.

Analysis of this alternative accident sequence has been performed using the TRAC (Ref. 2) and MARCH (Ref. 3) codes. The results of these analyses indicate that while some differences in the early progression of the accident result from this variation in delivery time, the progression beyond about 80 minutes is essentially the same (See Figure II-D-3). As such, the delay of 8 minutes in initial delivery of emergency feedwater does not appear to have significantly affected the overall course of this accident. However, since the lack of

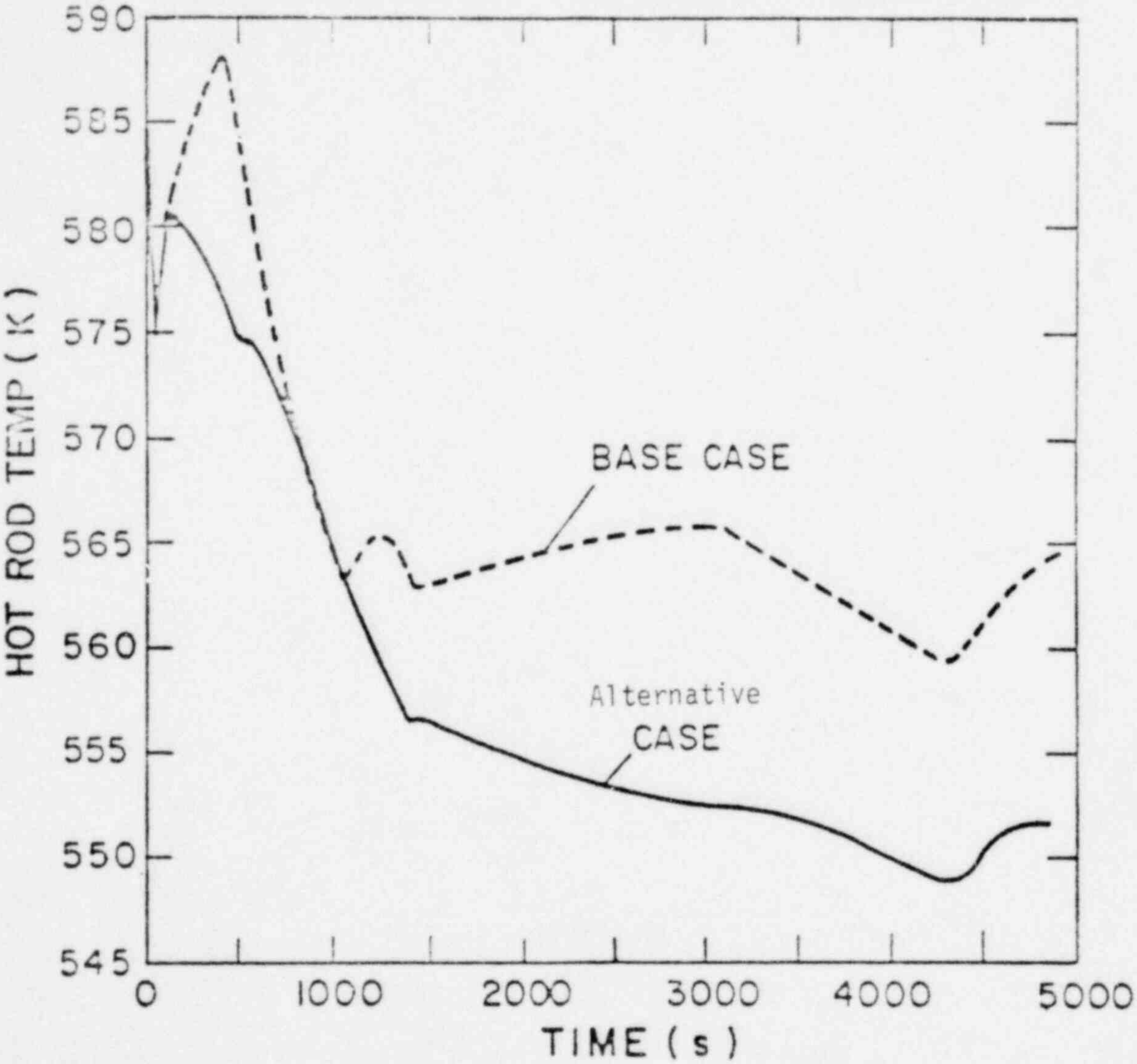


Figure II-D-2
Comparison of base case to alternative sequence 2
(EFW delay until 1 hr; full HPI flow)
(Ref. 2)

heat removal through the OTSGs apparently had some influence on the initial pressurizer turnaround and increase, the lack of EFW for 8 minutes did to some degree influence the decisions of the operating staff. In this sense, closure of the EFW block valves did contribute to the accident progression.

II-D-2.4 Alternative Accident Sequence 4: Emergency Feedwater Delivered at 1 Hour, High Pressure Injection System in "Degraded" Mode

In alternative accident sequence 4, it was assumed that the closure of the EFW discharge line block valves was not detected until about 1 hour into the accident, rather than 8 minutes. This sequence thus indicates the effect of a more prolonged operator error in failing to discover the block valve closure.

[Results to be filled in later]

II-D-2.5 Alternative Accident Sequence 5: PORV Block Valve Closure at 25 Minutes

In this alternative sequence, it was assumed that closure of the PORV block valve occurred at approximately 25 minutes. At this time in the accident the staff in the control room first requested the PORV discharge line temperature. This sequence was compared to the base case in order to assess the effect of failure to close the PORV block valve at this early time.

The analysis of this sequence was performed using the RELAP (Ref. 1) and MARCH (Ref. 3) codes. The results of these analyses indicate that the temperature in the core does not become sufficiently high that damage to the

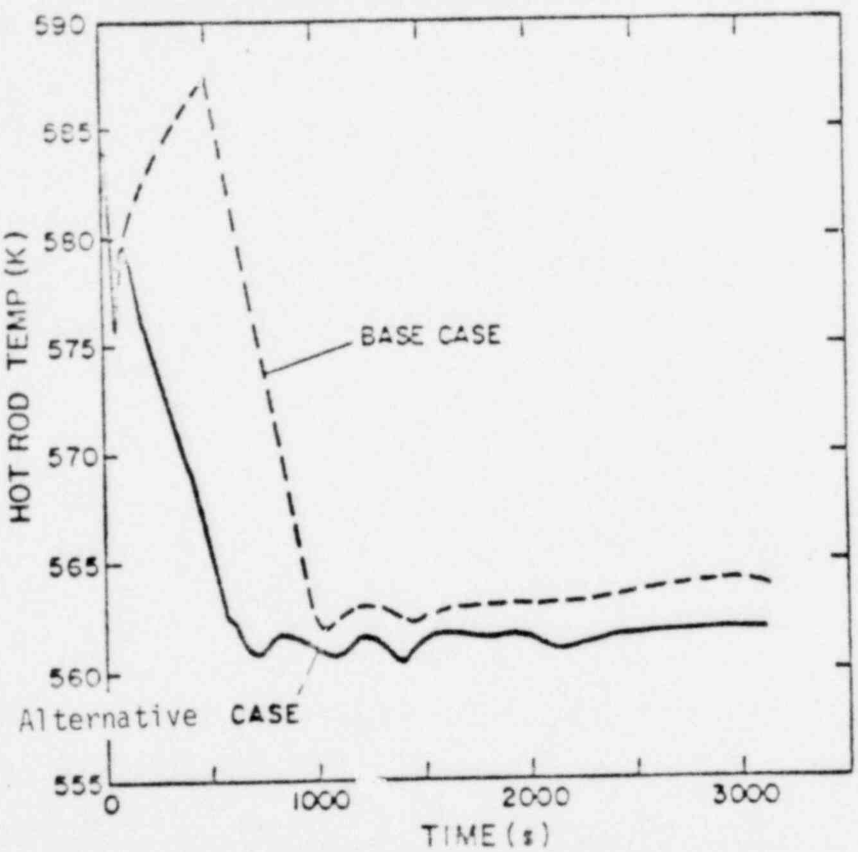


Figure II-D-3
Comparison of base case to alternative sequence 3
(EFW on at 40 seconds)
(Ref. 2)

fuel would have been expected. With the flow rates from the high pressure injection system as they were in the accident, recovery to "normal" conditions in the reactor coolant system would have taken roughly 1 1/2 hours. Thus, had the PORV block valve been closed at 25 minutes, it appears that the event would have had no significant consequences to the plant.

II-D-2.6 Alternative Accident Sequence 6: PORV Block Valve Closure at 3.3 Hours

In this alternative accident sequence, the time of closure of the PORV block valve is assumed to be delayed by an additional one hour, so that closure occurs at about 3.3 hours into the accident. The subsequent course of the accident in the time period between 2.3 and 3.3 hours has been evaluated using the MARCH code (Ref. 3), so that the importance of the timing of the operator action to close the block valve can be better understood.

The MARCH code analysis indicates the accident progression after 2.3 hours is particularly affected by the assumed makeup flow; it is also dependent on emergency feedwater flow, and the availability of the core flood tanks (CFT). Makeup flows throughout the accident are subject to considerable uncertainty; for this reason, a "best estimate" flow rate has been considered as well as several variations in flow rates. In the actual accident, emergency feedwater flow to the operable steam generator (OTSG A) was stopped at about the time of block valve closure at 2.3 hours. In the time subsequent to this, steam generator heat transfer was decreased, resulting in higher RCS pressures. Also, the availability of the core flood tanks has been questioned because of operator actions prior to 2.3 hours. It appears that the CFT isolation valves were closed early in the accident, so that the possibility exists that the tanks would not have operated if RCS pressure decreased below their setpoints.

The "best estimate" MARCH calculation of the alternative sequence indicates that, because of the lack of emergency feedwater after about 2.3 hours, RCS pressures do not decrease after this time (with the block remaining open), but rather begin a slow increase. Since the setpoint for the core flood tanks is not reached, water from these tanks is not considered to be available.

Based on the MARCH results shown in Figure II-D-5, it appears likely that the failure to close the PORV block valve until 3.3 hours would have resulted in a substantial fraction of the fuel achieving temperatures where fuel-clad eutectic formation, i.e., fuel "liquification," would occur. Thus, it appears that the TMI-2 accident could have been within an hour of becoming what is called in general terms a "core meltdown" accident.

The progression of an alternative accident sequence such as this has been analyzed beyond the time when "core meltdown" is predicted to begin, using the MARCH code. This progression is discussed in Section II-C-2.

As noted above, this conclusion is sensitive to the assumed makeup flow rate in this time period. In the "base case" for this alternative sequence, a flow rate of 90 gallons per minute is used (based on information from Ref. 8), and results in about 1/3 of the fuel "liquifying." If a flow rate of about 115 gallons per minute is used, only a small fraction of the fuel is predicted to reach eutectic-formation temperatures. In this particular case,

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less fuel liquefaction is predicted than for the actual TMI-2 accident. If a makeup flow of about 65 gallons per minute is used, MARCH predicts that a large fraction (approximately____%) of the fuel "liquifies."

II-D-2.7 Alternative Accident Sequence 7: One Reactor Coolant Pump per Loop Tripped at 73 Minutes

In this alternative sequence the method of tripping the reactor coolant pumps has been varied from that in the actual sequence. During the accident, both

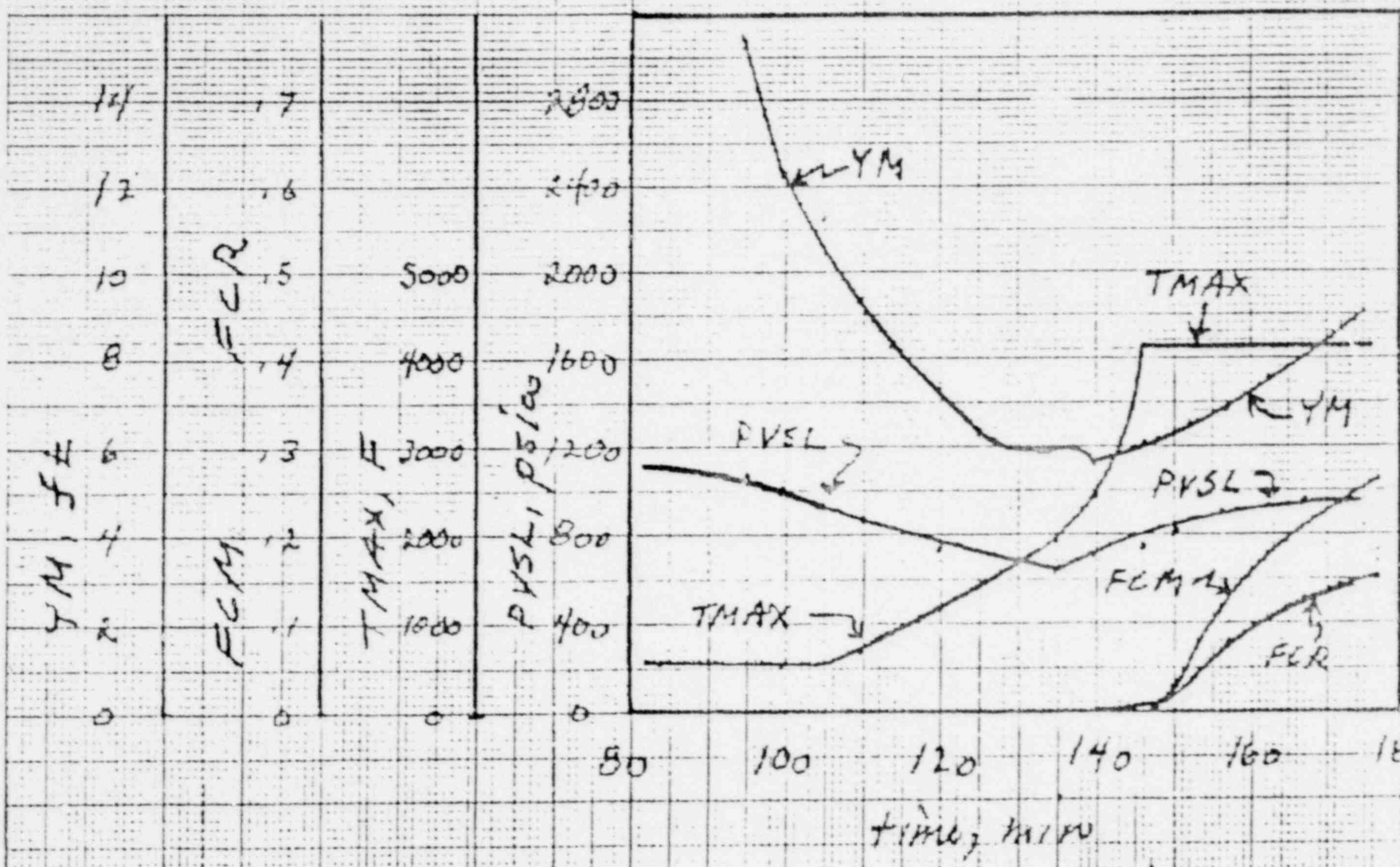
DRAFT

Fig. II-D-5

Block valve closure at 3.3 hrs.

(Ref.3)

FCM = Fraction core melted
FCR = Fraction clad reacted
PVSL = vessel pressure(psia)
TMAX = max core temperature (°F)
YM = vessel water level(ft)



16b

pumps in the B-loop were tripped at about 73 minutes, with the A-loop pumps tripped 28 minutes later. In tripping both B-loop pumps first, the water subsequently available to the A-loop pumps may have been reduced. In this alternative sequence, one pump per loop is assumed to be tripped at 73 minutes, potentially increasing the water subsequently available to the running pumps. This then may result in prolonged cooling of the core and delayed core uncovery.

Analysis of this sequence has been performed using the RELAP computer code (Ref. 1). The results of this analysis indicate that the fluid density at the suction of the reactor coolant pump remains higher in this alternative sequence than in the actual accident. As may be seen in Figure II-D-6, the fluid density at the A-loop pump suction is calculated to be about 5 lbm/ft^3 at the time of trip of these pumps in the actual accident (about 100 minutes). In contrast, this density is not achieved in the alternative sequence until roughly 135 minutes.

Also obtained in the RELAP calculations is the core inlet mass flow rate, shown in Figure II-D-7. This figure indicates that, in the alternative sequence case, the inlet flow rates decrease at a slower rate than in the actual accident and remains almost constant after the trip of the first two reactor coolant pumps.

The PORV block valve was closed at about 138 minutes into the accident, causing a marked increase in RCS pressure. The calculated pump suction fluid densities and core inlet flow rates discussed above suggest that relatively good flow could have been sustained until the time of block valve closure, had the alternative method of pump trip been used. Since reactor coolant

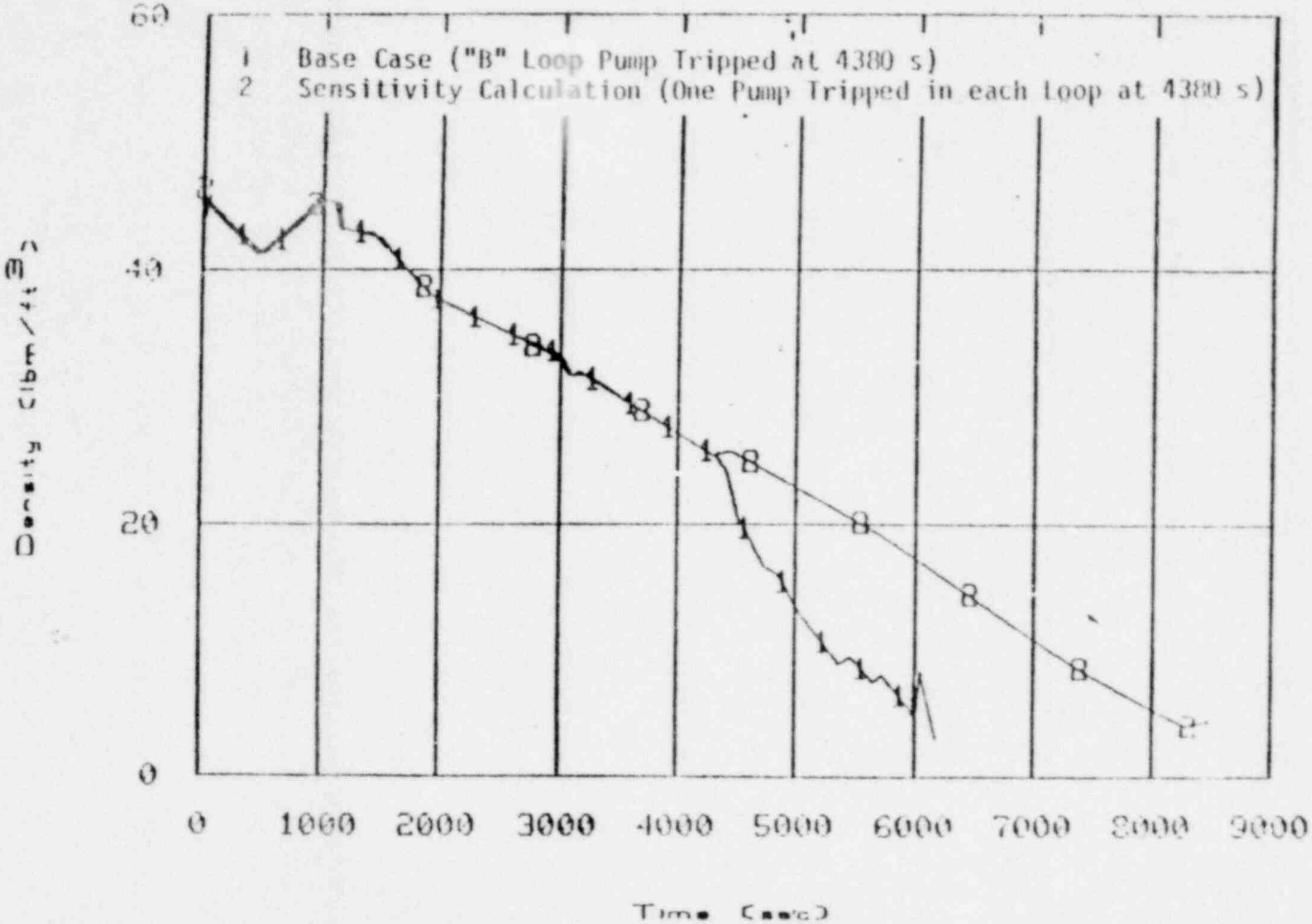


Fig. II-D-6
Effect of pump trips on fluid density in the A-1 loop pump.
(Ref. 1)

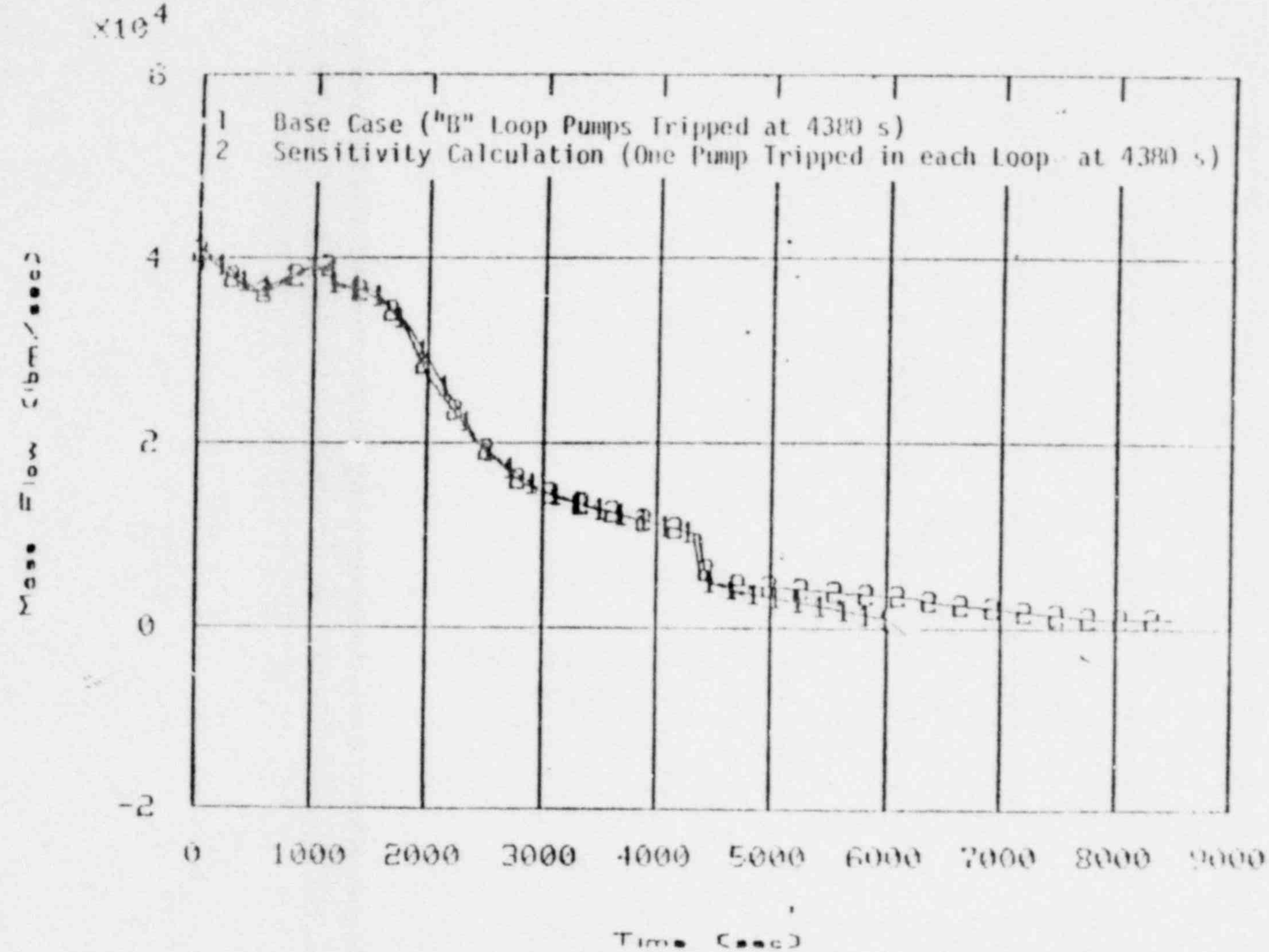


Fig. II-D-7
Effect of pump trips on core inlet mass flow rate.
(Ref. 1)

pump flow of this magnitude would have prevented high fuel temperatures, it appears that fuel damage might not have occurred had one pump been tripped in each loop rather than both pumps in one loop.

II-D-2.8 Alternative Accident Sequence 8: All Reactor Coolant Pumps Tripped Concurrent With Reactor Trip

In this alternative sequence, the reactor coolant pumps are assumed to have been tripped at the time of reactor trip; i.e., about 8 seconds into the accident. In effect, this assumption removes the contribution of these pumps to the accident progression.

The contribution of these pumps has two aspects. The forced flow of water provided by the pumps was a positive factor in keeping core temperatures relatively low. However, this same flow was forcing liquid water into the pressurizer and out the PORV, increasing the mass loss out of the reactor coolant system. This analysis thus indicates the relative significance of these competing effects.

[Results to be filled in later]

II-D-2.9 Alternative Accident Sequence 9: No Reactor Coolant Pump Restart at 16 Hours

In this sequence it has been assumed that it was not possible to restart a reactor coolant pump at 16 hours. In the actual accident one reactor coolant pump was started at that time and forced cooling of the core reestablished.

This analysis thus assesses the state of core cooling at 16 hours; i.e., whether actions to repressurize the RCS by increasing high pressure injection flow had begun to effectively cool the core, or that core conditions were continuing to deteriorate. Consideration of this alternative sequence was undertaken as part of the MARCH recreation of the first 16 hours of the accident, with additional insights being obtained from other evaluations of this time period by the Special Inquiry Group staff.

Neither the MARCH analysis or the work within the Special Inquiry Group provide conclusive answers to the question of concern. The trends in hot leg temperatures appear to indicate that some cooldown of the RCS was occurring as a result of the repressurization of the RCS beginning at about 14 hours and before the restart of the reactor coolant pump. However, this apparent cooling in the hot leg temperatures is not necessarily correlatable to decreasing fuel temperatures. Information from in-core thermocouples and self-powered neutron detectors indicate that a substantial region of the core remained very hot in this time period, with quenching of some regions occurring as the reactor coolant pump was restarted. However, no clear trend in quenching of regions is apparent before start of the pump. For this reason, one cannot conclude definitively that the core was (or was not) cooling down in this time period. As such, the criticality of reactor coolant restart at 16 hours cannot be determined conclusively.

II-D-2.10 Alternative Accident Sequence 10: Loss of Offsite Power at 1/2 to 5 1/2 Hours

In this alternative sequence, a less-directly related, less likely event has been postulated. Between about 4:30 and 9:30 on the morning of March 28, the

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emergency onsite AC power system (diesel-generators) had been disabled by the operating crew in such a way that, had offsite power been lost, all AC power would have been temporarily lost. Such a loss of offsite power was unlikely during this time period; however, the resulting loss of all AC power would have seriously affected an already bad situation.

MARCH analysis has been performed to assess the time required to result in a significant fraction of the fuel reaching eutectic-formation temperatures (Ref. 3). This analysis indicates that for a total loss of AC power beginning at about 2 hours, some fuel would reach such temperatures in about 24 minutes. The majority of the fuel is predicted to reach these temperatures within about 54 minutes after the loss of AC power.

The onset of high fuel temperatures such as discussed above can of course be prevented by the restoration of an AC power source. When questioned about the time required to restore a diesel-generator to operation, operators from TMI-2 estimated this to require about five minutes (Ref. 4). It therefore, appears likely that a loss of all AC power during the early portion of the accident could have been compensated for by prompt operator action before fuel eutectic formation occurred.

II-D-2.11 Alternative Accident Sequence 11: Loss of Offsite Power During March 30 to April 1

In this alternative sequence, it has been assumed that a loss of offsite power occurred during the time period of March 30 through April 1. In this period, core cooling was being maintained by the operation of one reactor coolant pump. A loss of offsite power during this time would have shut down this pump and other equipment such as the pressurizer heaters and the PORV block valve.

The loss of offsite power postulated here would have had varying degrees of impact on the systems potentially available for core cooling. Table II-D-2 shows the possible system options and the associated impacts of a loss of offsite power. From this table it becomes apparent that the most reasonable option would be the use of the high pressure injection system. Natural circulation cooling may have been a viable option; however, loss of RCS pressure control, and the presence of some hydrogen in the RCS may have prevented this option. Further, the lack of forced flow in parts of the damaged core region may have resulted in localized temperature excursions following the loss of offsite power. The use of the low pressure injection system would not have been possible because of the inability to depressurize the RCS using the PORV and its block valve.

Restoration of offsite power would of course have increased the number of options available to the operating crew. Restart of a reactor coolant pump would have been possible, as well as the use of the low pressure injection not previously possible.

Analysis using the MARCH code indicates that had a <u>total</u> loss of core cooling occurred on March 31, at least 20 hours would have had to elapse before fuel temperatures would have reached those needed for eutectic formation (Ref. 3). With this amount of time available for restoration of offsite power or the

TABLE II-D-2

POSSIBLE SYSTEMS OPTIONS TO MITIGATE A POSTULATED LOSS OF OFFSITE POWER ON MARCH 30-APRIL 1

System	Effect of Loss of Offsite Power
(1) High pressure injection system	NONE
(2) Natural Circulation	Loss of RCS pressure control may prevent natural circulation
(3) Low pressure injection system with depressurization caused by PORV block valve opening	No power to PORV block valve

actuation of the HPI system, it appears likely that core cooling could have been restored without further core damage. For this reason it appears that the loss of offsite power on March 30 to April 1 would not have been a serious problem.

II-D-2.12 Recriticality

In this alternative accident sequence, the potential for recriticality after the accident has been assessed. Since the high fuel temperatures experienced early in the accident may have distorted the core geometry and damaged control rods, evaluation of possible core reactivity changes was considered necessary.

A number of analyses of criticality potential were performed after the accident by the NRC staff and by B&W (Ref. 5, 6). These analyses considered degrees of fuel damage ranging from essentially no geometric distortion to a substantially collapsed core. In general, no credit was given for control rod or burnable poison material, so that dissolved boron was the only presumed poison in the core. The results of these calculations indicated that subcriticality could be maintained with boron concentrations of 1500 parts per million (ppm) for an essentially undistrubed core to about 3500 ppm for a fully damaged core.

For a core configuration suggested in Section as now existing in the TMI-2 vessel, the criticality calculations indicated that boron concentrations in the range of 1500 to 2200 ppm would be required to maintain subcriticality (Ref. 6). Since no credit is given for control rod and burnable poison material, it is likely that these estimates are somewhat conservative; i.e., a more realistic requirement for boron concentration would be somewhat less.

Reactor coolant samples taken on April 7 indicated that the coolant was being maintained at approximately 2200 ppm (Ref. 6); suggesting that the potential for recriticality was not a serious concern. Subsequent to the analysis, the boron concentration was increased to over 3000 ppm to provide an even greater margin of subcriticality.

The possibility of an inadvertent dilution of the RCS of course could have caused the return of the core to criticality and caused additional problems in the recovery process. However, since such a dilution would have to go undetected for some time to result in recriticality, it seems reasonable that operator detection and correction would be likely prior to the return to criticality.

II-D-2.13 Effect of Containment Design

Consideration has been given in this section to the effect of containment design on the course of the accident. Specifically, it has been postulated that the containment design was different from what actually exists at TMI-2; this then indicates the relative vulnerability of different containment designs to this type of accident.

The principal threat to the TMI-2 containment occurred at about 1:50 p.m. on March 28, when a hydrogen deflagration resulted in a 28 psig pressure spike. At all other times in the accident, containment pressure was at 5 psig or less. Since the design pressure of this containment building is 60 psig, little actual threat to the building existed at any time in the accident.

Table II-D-3 shows typical design characteristics for the spectrum of containment buildings used in large commercial reactors in this country. Examination of the characteristics for the large, free volume buildings indicates that these are comparable to the TMI-2 containment; as such, it is likely that these containments also would not have been seriously threatened by the hydrogen deflagration experienced at TMI-2.

The data in Table II-D-3 suggest that the pressure suppresion type of containment building is more susceptible to damage from a hydrogen deflagration of the magnitude experienced at TMI-2. For this reason each type of pressure suppression containment will be discussed individually below.

Analysis of the capability of the ice condenser containment design to withstand pressure loadings due to hydrogen burning has been performed at Battelle Columbus Laboratories (BCL), using the MARCH code (Ref. 3). In support of this analysis, BCL has evaluated data on the 28 psig pressure transient and concluded that it corresponds to the rapid burning of about 550 pounds of hydrogen. If a comparable amount of hydrogen were rapidly burned in an ice condenser containment, containment failure would be likely. Thus it appears that an ice condenser containment design would not have retained its integrity had it experienced the type of hydrogen deflagration that was experienced in the TMI-2 containment.

It is evident from Table II-D-3 that the BWR Mark I containment is of relatively high design pressure but of very small free volume. Because of these characteristics it would appear that this design could also be vulnerable to hydrogen burning. Analysis of this possible vulnerability was performed by Battelle

TABLE II-D-3
TYPICAL CONTAINMENT DESIGN PARAMETERS

Containment Type	Example Plant	Free Volume (ft^3)	Design pressure (psig)
Large Free Volume			
-- Prestressed Concrete	TMI-2	2×10^6	60
-- Free standing steal	St. Lucie	2.5×10^6	44
-- Sub atmospheric	Surry	1.8×10^6	45
-- Spherical Steel	Perkins	3.3×10^6	47
Pressure Suppression			
-- Ice Condenser	Seqoyah	1.2×10^6	12
-- BWR Mark I	Peach Bottom	2.8×10^5	56
-- BWR Mark II	Zimmer	3.9×10^5	55
-- BWR Mark III	Grand Gulf	1.7×10^6	15

Columbus Laboratories for the Reactor Safety Study (Ref. 7). This analysis indicated that, because of the combination of high design pressure and strength of the steel Mark I containment, there is the possibility that this containment could withstand the burning of large amounts of hydrogen. The amount of hydrogen that can be burned would, of course, be limited by the amount of oxygen available, even if the atmosphere were not inerted.

An additional factor in the consideration of the capabilities of the Mark I containment is that most are "inerted" with nitrogen. This reduces the amount of oxygen potentially available for recombination with hydrogen, so that the likelihood of burning (and thus the containment vulnerability) is substantially reduced.

The BWR Mark II containment design is characterized by a somewhat larger free volume and a comparable design pressure, compared to the Mark I design. It is constructed of prestressed concrete, rather than the steel of the Mark I. Because of the lack of an inerted atmosphere, the Mark II would be somewhat more vulnerable to hydrogen burning. Since no specific analysis is available on this containment design, it cannot be concluded that a hydrogen deflagration of the magnitude of that in TMI-2 would (or would not) have caused containment failure.

The BWR March III containment is the largest of the BWR containments, being roughly comparable in free volume, design pressure, and construction to the ice condenser design. This comparability in design characteristics suggests that the Mark III containment would respond in a manner similar to that predicted for the ice condenser; that is, it appears likely that a Mark III

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containment would not have survived a hydrogen deflagration of the magnitude experienced at TMI-2.

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7. Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, WASH-1400, U.S. Nuclear Regulatory Commission, October 1975.

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Section I-C Preexisting Deficiencies and Their Contribution to the Accident

At the time of the TMI-2 accident, certain conditions were present in the plant which, in retrospect, have been suggested to be deficiencies which contributed to the course of the accident. These conditions included both design and operational features of the plant, such as:

-- inadequate steam generator and pressurizer sizing;

-- lack of remotely-operable vents at the RCS high points;

-- lack of anticipatory reactor trip;

-- lack of ECCS bypass prevention;

-- inadequate radwaste handling capabilities; and

-- inadequate radiation monitoring instrumentation.

In Sections I-C-1, I-C-2, and I-C-3 below, each of these possible deficiencies is discussed in greater detail. Section I-C-1 deals with possible plant systems deficiencies, Section I-C-2 with possible command and control deficiencies, and Section I-C-3 with possible instrumentation deficiencies. At the end of the discussion of each possible deficiency, findings and conclusions, and when believed necessary, recommendations are presented. A summary of all findings and conclusions, and recommendations may be found at the end of this section

I-C-1 Possible Deficiencies in the Plant Systems Design

I-C-1.1 Responsiveness of the B&W Nuclear Steam Supply System

Since the time of the TMI-2 accident, the vulnerability of the B&W Nuclear Steam Supply System relative to other pressurized water reactor designs has been the subject of considerable discussion because of the differences in the operational responsiveness of the design. A number of features of the B&W design have been suggested as being contributors to this apparently greater vulnerability. These features are discussed individually in the sections below; the final section then integrates the individual evaluations into overall conclusions on the designs vulnerability.

-- Pressurizer Design

It has been suggested that the volume of the TMI-2 pressurizer is relatively small compared to other pressurized water reactors of comparable power. The concern in this instance is that a smaller volume would result in greater changes in level in the TMI-2 pressurizer, for any particular transient in the reactor. Since the rapid rise in pressurizer level early in the accident apparently contributed to the confusion experienced by the operating crew, the possibility of a relatively small pressurizer volume being a design deficiency contributing to this accident has been considered.

An examination has been made of the volume of the pressurizers of a number of nuclear plants, the details of which may be seen in Table I-C-1. This examination indicates that the pressurizer volume in TMI-2 is comparable to that in other plants. In fact, when this volume is normalized with respect to plant power, the TMI-2 pressurizer is slightly larger than some other plants. Thus, it does not appear that relative pressurizer size is, by itself, a significant concern.

A number of operational events have occurred in B&W plants involving loss of pressurizer level indication in both the high and low directions. These events, which are shown in Table I-C-2, may be construed to imply that the pressurizer volume is insufficient to accomodate certain transient events. However, consideration of the causes of these operational events suggests that the pressurizer volume is not directly the problem. Rather, it appears that the plant sensitivity to secondary side transients, i.e., to the amount of heat removal through the steam generators and to the rapidity of the changes in heat removal capability during transient events is the basic problem. This sensitivity is discussed in more detail below.

Another concern which has arisen with respect to the B&W pressurizer design is that the design includes a "loop seal" in the pressurizer surge line. This feature, which may be seen in Figure I-C-1, is installed as a protective device for the pressurizer heaters. In the event of a relatively slow decrease in the reactor coolant system pressure, the loop seal feature maintains water over these heaters and reduces the likelihood of overheating.

TABLE I-C-1
PRESSURIZER SIZING

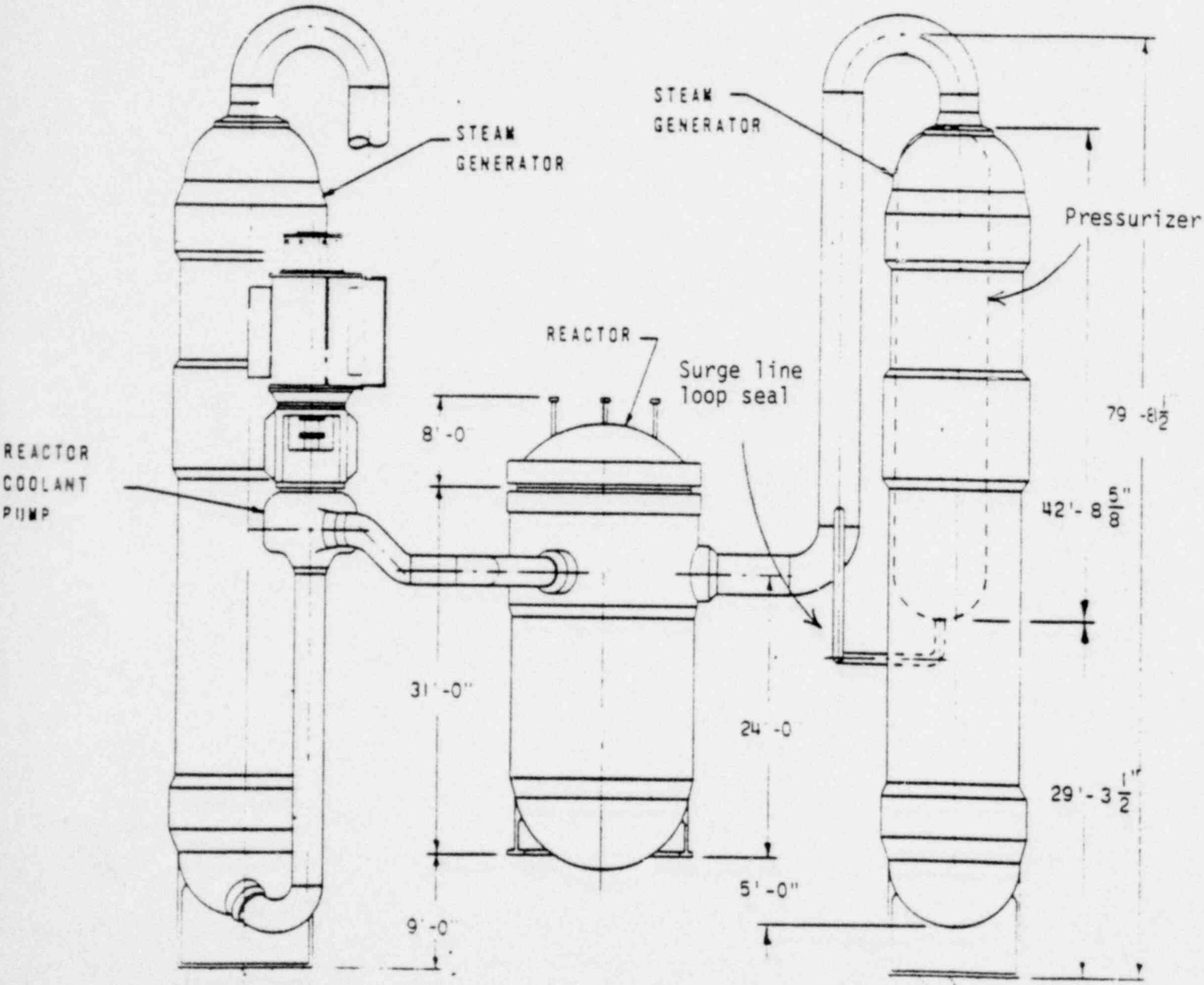
Plant	Vendor	Thermal Power (MWth)	Pressurizer Volume (ft^3)	Ratio: Pressurizer Volume (ft^3) / Thermal Power (MW)
Three Mile Island (Ref. 1)	B&W	2772	1500	0.54
Oconee (Ref. 2)	B&W	2568	1500	0.58
San Onofre 2&3 (Ref. 3)	CE	3410	1500	0.44
St. Lucie (Ref. 4)	CE	2560	1500	0.58
Surry (Ref. 5)	W	2441	1300	0.53
Sequoyah (Ref. 6)	W	3411	1800	0.53

TABLE I-C-2
INSTANCES OF LOSS OF PRESSURIZER
LEVEL IN B&W PLANTS
(Ref. 7, 8)

Plant	Date	Direction of Level Losses High	Direction of Level Losses Low	Cause
Davis-Besse	11/29/77		X	Loss of AC power
Davis-Besse	9/14/77	X		Loss of feedwater stuck-open PORV
Rancho Seco	3/20/78		X	Electrical malfunction-ICS
TMI-2	3/29/78	X		Bus failure - stuck open PORV
TMI-2	4/23/78		X	MS safety valves fail open
TMI-2	11/07/78		X	LO FW (?)
TMI-2	3/28/79	X		LO FW - stuck-open PORV

Figure I-C-1

Pressurizer Surge Line
Loop Seal Arrangement
(Ref. 1)



It has been suggested that this loop seal contributed to the artificially high pressurizer levels indicated to the operating crew. In this way,
the loop seal is suggested to have contributed to the throttling of the high pressure injection system and the resulting core uncovery and fuel damage.

In the initial 1 to 2 hours of the accident, a number of effects were influencing the pressurizer level, causing it to increase off-scale in the high direction and remain there. Among these influences were the stuck-open PORV, the high initial flow rates from the HPI pumps, the increase in coolant volume due to heating, the loop seal arrangement, and perhaps flashing in the pressurizer reference leg. At later times these effects may have been additionally compounded by the effect of hydrogen in the reference leg water.

Because of the possible presence of all the effects noted above, the particular influence of the surge line arrangement is not easily discernable. Analysis performed by Westinghouse on a small break loss-of-coolant accident in the pressurizer steam spac (which includes a stuck-open PORV) indicates that Westinghouse-designed plants would experience a similar increase in pressurizer level resulting from a stuck-open PORV (Ref. 9). Since Westinghouse plants have a vertical surge line (i.e., no loop seal), it would appear that the loop seal arrangement is not the dominant influence in causing increasing pressurizer level. For this reason, it appears that the pressurizer surge line loop seal design did not by itself have a significant impact on the accident progression.

-- OTSG Secondary Side Coolant Inventory

The design of the B&W pressurized water reactor includes steam generators which are considerably smaller (in terms of secondary side water inventory) than Westinghouse or Combustion Engineering PWRs, as may be seen in Table I-C-3. In the event of an interruption of feedwater to the steam generators, as occurred at TMI-2, the smaller size in the B&W design results in a more rapid boiling off of the secondary coolant inventory. This then results in a more rapid loss of heat removal from the primary coolant, causing its temperature and pressure to increase more quickly.

The fast response of the B&W steam generators is a favorable feature in the context of plant operational responsiveness and reliability of electrical generation. However, in the context of reactor safety, the fast response to abnormal transients requires rapid attention and intervention by the operating crew to prevent or minimize the effects on the reactor coolant system. Because of this need for rapid operator action, the B&W design is considered to be fundamentally more susceptible to human errors than other pressurized water reactor designs.

-- Use of the PORV

In the B&W pressurized water reactor design the PORV is used routinely during transient events. When such a transient event begins in the plant which causes the reactor coolant system pressure to increase, the PORV opens in an attempt to compensate for the increase. Because of this design feature, the PORV is used an average of about five times a

TABLE I-C-3

STEAM GENERATOR SECONDARY SIDE
COOLANT INVENTORIES FOR VARIOUS
PRESSURIZED WATER REACTORS

Plant	Designer	Power	Total Steam Generator Secondary Side Coolant Inventory
TMI-2 (Ref. 1)	B&W	2772 MW	110,000 lb
Oconee (Ref. 2)	B&W	2568	110,000 lb
Calvert Cliffs (Ref. 10)	CE	2560	304,000 lb
St. Lucie (Ref. 4)	W	2560	320,000 lb
Surry (Ref. 5)	W	2441	261,000
Sequoyah (Ref. 6)	W	3411	376,000

year in each B&W plant (see Table I-C-4). In contrast, the Combustion Engineering and Westinghouse pressurized water reactor designs do not routinely use the pressurizer PORV. Data assembled by these vendors, and shown in Table I-C-4, indicate that the frequency of use of the PORV is significantly less (a factor of ten or more) than in the B&W design.

Relief valves such as th pressurizer PORV have a history of failing to reclose after opening with a frequency of roughly one failure in 100 openings. Since this alve should be generally applicable to PORVs in the reactors of all three vendors, the higher frequency of use in the B&W plants should correspond to a higher incidence rate of stuck-open valves in these plants. Actual reactor operating experience confirms this predicted higher frequency of stuck open PORVs in B&W plants.

In summary, because of the routine use of the pressurizer PORV in B&W plants, the likelihood of experiencing a stuck-open valve is significantly higher in B&W plants than in other pressurized water reactors. Since a stuck-open pressurizer PORV is essentially equivalent to a small loss-of-coolant accident, it can be concluded that the likelihood of experiencing a small loss-of-coolant accident is significantly greater in B&W plants than in Combustion Engineering and Westinghouse plants.

— Lack of Anticipatory Reactor Trip

Another design feature of the B&W pressurized water reactor is the lack of an "anticipatory" reactor trip system. This feature is included to allow the plant to continue operation during less severe transients by not requiring the shutdown of the reactor. In this way the frequency and time of reactor shutdown is decreased, thereby enhancing the operational reliability of the plant.

The detailed evaluation of this design feature is described in Section I-C-2.1. Of concern here is the particular affect of this feature on the overall responsiveness of the B&W plant. In this respect, the influence of the lack of an "anticipatory" reactor trip is to decrease the time available to the operating crew to cope with the event. The delay of reactor trip causes the input of a significant amount of energy into the reactor coolant system above that which would have been input had the reactor been tripped immediately. In transient events where the normal cooling path is interrupted (e.g., when the main feedwater pumps and turbine are tripped), this additional energy input can substantially change the steam generator boil-dry time and affect the reactor coolant system pressure and temperature. The overall effect of the delay in reactor trip is thus a decrease in the time in which the operating crew has to perform necessary actions. Since human errors become increasingly likely as the time to perform actions decreases, the lack of an anticipatory trip in the B&W plants may be translated into an increased susceptibility of the B&W plants to human errors.

In an overall sense, the responsiveness of plants designed by B&W to transient events causes these plants to be more vulnerable than other pressurized water reactor designs. The method of reactor coolant system pressure control during transient events places reliance on a relief valve with a known propensity for failing in an open (i.e., unsafe) position; this design feature thus results in a significantly greater likelihood of experiencing a small loss-of-coolant accident. Other features of the plant design require the operating crew to make more hurried judgments in response to the initiating event and reduce the time available to make corrective measures. These features thus make the B&W design less "forgiving" to errors by the operating crew. The combination of these two aspects of the B&W plant operational characteristics and responsiveness, i.e., the increased propensity for experiencing a loss-of-coolant accident and for subsequent human errors, make this responsiveness a clearly significant contributor to the accident at Three Mile Island.

Findings and Conclusions

Because of particular design features and operational characteristics of B&W plants, these plants have:

-- a significantly higher likelihood of experiencing a small loss-of-coolant accident than other pressurized water reactors, and

-- a greater susceptibility to human errors during abnormal situations than other pressurized water reactors.

Recommendations

-- Modifications to the B&W design and operational characteristics are needed to reduce the frequency of demand on the pressurizer relief valve and safety valves. Plant modifications made to all B&W plants after March 28, 1979 (e.g., PORV setpoint increase, hardwired reactor trip on turbine trip) provide at least a temporary solution to this concern. These modifications are not, however, necessarily the optimum achievable; alternative methods of providing equivalent or greater degrees of protection may be forthcoming and should merit due consideration.

-- Human interactions with vital systems during transient events and accidents in B&W plants (as well as all other LWRs) should be kept to an absolute minimum to reduce the likelihood of human errors of omission or commission. Proper "human factors" consideration in system design and operation, operating and emergency procedures, test and maintenance procedures, and training is needed in this regard. Additional automation of vital systems should be a matter of foremost attention in these considerations. Because of the greater sensitivity of B&W plants to these interactions, these plants merit particular attention in this regard.

I-C-1.2 Radiological Design Adequacy of Plant Systems

During the course of the accident and the post-accident recovery, significant problems arose relating to high radiation fields in the auxiliary building. These problems influenced decisions being made at the time concerning access to and work done in the auxiliary building and the method by which the RCS would be cooled down. In this light, the contamination in the auxiliary buildings suggests possible plant deficiencies. In this section, possible deficiencies in core cooling and other safety-related equipment are considered.

Additional related deficiencies in the plant radiological waste systems are discussed separately in Section ___.

The emergency core cooling systems in PWRs are designed to initially draw coolant from an uncontaminated water supply such as the Borated Water Storage Tank at TMI-2. Upon depletion of this tank, supply lines are switched to use water in the reactor building sump. This water is then drawn into the decay heat removal pumps and pumped either back into the RCS (if RCS pressure is sufficiently low) or to the suction of the high pressure injection pumps, with subsequent flow back into the RCS. The containment spray system uses a similar method of supplying water, drawing first from the BWST, and subsequently from the reactor building sump.

Early on the first day of the accident the water collecting in the reactor building began to be contaminated with radioactive material being released from the damaged fuel. It soon became evident that this contamination could cause significant radiological problems in the auxiliary building if previously uncontaminated equipment (e.g., the decay heat removal pumps) and areas would become contaminated. For this reason a method of core cooldown was chosen which would minimize the likelihood of requiring use of sump water. Thus, in effect, the "expected" method of core cooldown following a loss of coolant accident was to be used only in the event that all other options failed.

The design basis radiological hazard for the emergency core cooling equipment and areas is established in Chapters 12 and 15 of the final safety analysis report (Ref. 1). Chapter 12 provides information on the radiation protection for onsite personnel, while Chapter 15 provides offsite dose calculations based on leakage from ECC equipment during post-accident recirculation. For the concern here, Chapter 12 is the relevant section.

Chapter 12 established the design basis upon which shielding was provided for certain components of the emergency core cooling system. That vital equipment which is part of the makeup and purification system or the decay heat removal system was shielded to compensate for the assumed radioactivity levels in the reactor coolant resulting from normal operation of the plant. Other vital equipment which is not normally used during plant operation (e.g., in the containment spray system) was not required to have even this amount of shielding (Ref. 15).

The concept of shielding of vital equipment containing contaminated water following an accident appears to be area not given consideration in the licensing of TMI-2. This deficiency created a situation during this accident where options which should have been available were in fact highly undesirable because of the possibility of contamination and the lack of adequate shielding. Thus, in the sense that the range of options available for accident recovery was reduced because of the inadequacy of radiation protection, this deficiency was a contributor to the accident.

Findings and Conclusions

Consideration of requirements on radiation protection measures for accident-level radioactive contamination of vital engineered safety features equipment does not appear to have been adequate.

Recommendations

The capability for post-accident radiation shielding and leakage control for vital equipment using potentially radioactive containment sump water should be improved in all LWRs. Accessability to surrounding areas and equipment by plant personnel during accident mitigation and recovery should be a primary consideration in this regard.

I-C-1.3 Design of the PORV

It has been suggested that the design of the PORV was deficient in two respects. The first suggested deficiency relates to the capability of the PORV to pass mixtures of steam and liquid water. The second concern relates to the possibility that the discharge piping arrangement from the PORV may have been the cause of the valve remaining open when it was supposed to close. These concerns are discussed separately below.

-- Capability of the PORV to pass two-phase flow.

It has been suggested that the pressurizer PORV was designed only to pass steam flow, and not a two-phase mixture of liquid water and steam. The possibility thus arose that, upon the complete filling of the pressurizer, the two-phase flow through the valve may have caused sufficient damage to prevent any capability for further operation.

Consideration of this concern has two aspects. First, since the PORV had already failed in the open mode prior to the pressurizer completely filling, the possibility of damage incurred at this later time is somewhat academic.

The second, more significant, aspect relates to the actual design qualification of the PORV. Investigation of this possibility was pursued with B&W. It has been determined that, as part of B&W's analysis of the ATWS issue, the capability for water discharge through the relief and safety valves was evaluated. The conclusion of this evaluation was that while these valves were not qualified for water discharge through them, this discharge would not lead to "unacceptable damage" (Ref.). It therefore appears that this concern was not of great significance.

Findings and Conclusions

The capability of the PORV to discharge two-phase or water flow appears to be sufficient to prevent serious damage to the valve.

-- Affect of PORV discharge line piping arrangement on reseating capability

It has been suggested that the piping arrangement for the discharge from the PORV pilot valve may have been such that backpressure forces in that line prevented closure of the PORV. Consideration of this possibility has been pursued; results indicate that [to be filled in later].

I-C-1.4 Inhibitions to Natural Circulation

Throughout the first day of the accident attempts were made by the operating crew to induce natural circulation cooling in the reactor coolant system.

During the time interval when the reactor coolant pumps were not providing forced circulation cooling (i.e. from about 5:40 a.m. to about 7:50 p.m.), it was judged that this mode of cooling the core was highly desirable; however, attempts to induce it were apparently unsuccessful until about 4 to 5 p.m., when some natural circulation appeared to have been achieved. In this section results are presented of an evaluation of the contribution of the plant design to the prevention of the achievement of natural circulation under abnormal circumstances.

Before entering into a discussion of the capability for natural circulation cooling in B&W plants during abnormal circumstances, some discussion of this capability during normal circumstances is useful. Since the time of the TMI-2 accident, the capabilities of B&W plants in such situations has been question. Based on operating experience where natural circulation cooling was achieved and on specific natural circulation cooling tests in B&W plants (Ref. 18), it appears that the capability for such a cooling mode under normal circumstances is satisfactory.

In the TMI-2 accident, the capability for natural circulation cooling was initially lost within a number of minutes after the turbine/reactor trip. The initial depressurization of the reactor coolant system (RCS) caused the flashing of RCS water into steam. When the last reactor coolant pumps were tripped at 5:40 a.m., the steam in the RCS collected at the various high points of the system: the upper head of the reactor vessel, and the upper sections of the hot legs (the "candy canes"). The presence of steam regions in the hot legs, in concert with the large coolant mass loss out the PORV, prevented natural circulation cooling at this time and for some time period afterward.

Beginning very soon after the reactor coolant pump trip at 5:40 a.m. the core began to be uncovered as a result of the continued coolant mass loss out the PORV. For at least the next hour, the core was uncovered and fuel temperatures rose into the 2000°F to 4000°F range, causing the generation of hydrogen from the metal-water reaction. As this gas was being produced it too was rising into the high points of the reactor coolant system. Thus, from approximately 6 to 7 a.m. to about 5 to 6 p.m. the inhibition to natural circulation already resulting from steam was being compounded by the presence of noncondensible hydrogen. Because of the combined presence of these two substances, attempts in this time period to induce natural circulation by repressurization or to reinstitute forced flow by starting a reactor coolant pump met without success.

The eventual restoration of (apparently) some natural circulation and the restarting of a reactor coolant pump some time later appear to be attributable to the escape of some of the steam/hydrogen mixture from the loop A hot leg. The apparent reason for this escape is the depressurization of the RCS beginning at about 11:40 a.m. This decrease in pressure allowed the steam/gas mixture to expand to the point that it could flow into the pressurizer through the surge line and then out into the reactor building. The reduction in the amount of blockage in the loop A hot leg then apparently allowed sufficient flow to move through the hot leg to provide some natural circulation cooling. This reduction also appears to have made the reactor coolant pump restart at 7:50 p.m. possible.

It becomes apparent from the above discussion that the RCS hot legs were a primary source of the blockage which prevented natural circulation. Since these are high points in the system, this is not unexpected; similar behavior would be expected in the U-tube region of the hot legs in Westinghouse and Combustion Engineering pressurized water reactors. However, since the hot leg high points in these PWRs are within the steam generators, where feedwater can be used to condense and "unblock" steam pockets, the problem of steam blockage is not as serious a concern as in B&W plants.

The presence of hydrogen or other noncondensible gases in the steam pockets in the hot legs makes more difficult the restoration of natural circulation cooling. Once such materal has been introduced, the natural circulation cooling capability of any PWR would be compromised. However, since the volume in which gas may be trapped is greater in B&W plants (i.e., the B&W hot leg design has a larger volume than the U-tube arrangement in other PWRs), the restoration of natural circulation cooling would likely be more difficult in B&W plants.

It therefore appears that the B&W PWR design is somewhat more vulnerable to loss of natural circulation cooling capability during abnormal circumstances. This relatively greater vulnerability is due to the design of the hot legs which makes steam or steam and noncondensible gas more difficult to remove once trapped. Because of this, the concept of remote venting capability to be discussed below should be of greater interest for B&W plants.

Findings and Conclusions

Under normal circumstances, the capability for natural circulation cooling in B&W plants appears to be adequate.

Under abnormal circumstances, the ability to restore natural circulation cooling (once lost) appears to be somewhat more difficult in B&W plants than in other pressurized water reactors.

I-C-1.5 Lack of Remote Vent Capability at the Reactor Coolant System High Points

During the first 5 days of the accident, two significant concerns arose because of the trapping of steam and noncondensible gases (hydrogen, xenon, krypton) in the various high points of the reactor coolant system. As is discussed in Section I-C-1.4 above, the presence of these substances in the RCS hot legs inhibited attempts to restore natural circulation cooling and impaired the accident recovery during the first day. During the subsequent four days, the presence of a hydrogen "bubble" in the upper head of the reactor vessel was a major concern.

It was known at that time that manual vent valves were installed at both the tops of the hot legs and the top of the reactor vessel. However, because of the radiation environment in the reactor building, it was not possible to go to the valves and open them.

In the sense that the accident recovery process was hampered by the lack of remotely-operable vents at high points of the RCS, the B&W plant design (as well as other reactor designs) may be considered to be deficient. The addition of remotely-operable valves, or the modification of presently installed manual vents, appears to be desirable change.

It should be noted that the addition of remotely-operable valves would not be without some negative safety implications. Such valves provide additional possible paths for losses of coolant from the RCS, because of the inadvertent opening of a vent valve due to equipment failure or human error, or the intentional, malicious opening by a person. Thus the addition of these vents increases to some extent the likelihood of a loss-of-coolant from the RCS. Care should be taken in the design of such a vent system to minimize the possible effects of equipment failures and human interactions.

Findings and Conclusions

The lack of a remotely-operable vent at the reactor coolant system high points was a design deficiency which significantly impeded the recovery from the TMI-2 accident.

Recommendations

The capability to remotely vent the high points in light water reactors should be provided. Since certain failures in such vents could lead to a loss of coolant from the RCS, due consideration of this possibility should be one aspect of the design requirements. Measures to reduce the likelihood of unintentional (or malicious) use of these valves also merits consideration.

I-C-1.6 Core Barrel Vent Valves

The B&W design for a pressurized water reactor includes "core barrel vent valves." These valves are installed in the upper region of the reactor vessel, and under certain conditions, permit flow from the region above the core into the downcomer region.

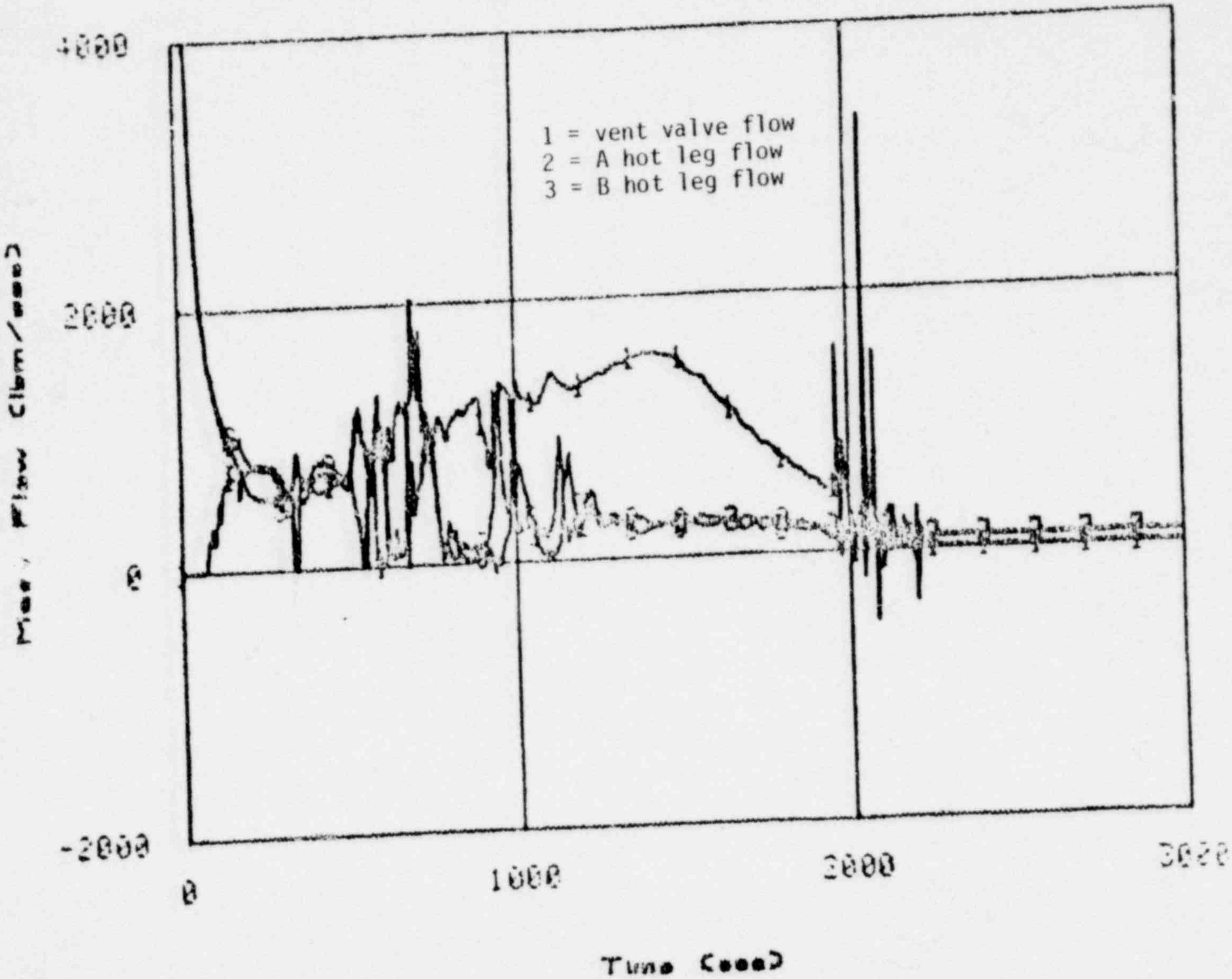
These vent valves were installed in B&W reactors in order to mitigate potential problems from the phenomena of "steam binding" during a large loss-of-coolant accident. "Steam binding" is an effect postulated to occur in such accidents in which high steam pressure above the core impedes the refilling of the core region with coolant. The vent valves are designed to relieve this pressure and thus assist in the refilling of the core region. During certain parts of the TMI-2 accident conditions were correct for the vent valves to have opened. In this circumstance, water which would have otherwise traveled into the steam generators and been cooled is returned into the downcomer and subsequently to the inlet of the core. Thus the heat removal capability of the steam generators may have been compromised.

Information on the flow through the core barrel vent valves has been obtained as part of the RELAP code calculations discussed in Section II-D-2. Figure I-C-2 shows the flow through these valves as a function of time, calculated for the alternative sequence case of reactor coolant pump trip at the time of the reactor trip. This case is sufficiently similar to the actual TMI-2 accident to suggest that flow through these valves could have been a substantial fraction of the outlet flow during portions of the actual actual.

Calculations have been made at Idaho National Engineering Laboratory to assess the impact of such flows through the vent valves on the overall heat removal from the core and the RCS. These calculations indicate that...[to be filled in later]

Figure 1-C-2

Core Barrel Vent Valve Flow
(Ref. 17)



I-C-1.7 Lack of Hot Leg Injection Capability

There exists a capability in some pressurized water reactors (i.e., those designed by Westinghouse) to inject emergency core cooling water directly into the reactor coolant system hot legs in addition to the cold legs. The TMI-2 plant, like all B&W plants, does not have such a capability. In situations such as at TMI-2 where uncovery of the top of the core occurs, a capability to pump water into the hot legs and directly onto the top of the core may be of significant benefit. Water flowing into the core in this manner provides a greater heat removal capability than a comparable amount entering into the core bottom, so that the fuel temperatures in the uncovered portion would tend to be lower.

The absolute degree to which the capability of hot leg injection would have enhanced the recovering and cooldown of the core is not readily apparent. However, it is clear that such a capability could have provided some additional cooling of the upper regions of the core. Therefore, while this "design deficiency" probably did not have a significant effect on the accident progression, the capability for hot leg injection could have provided a beneficial option for the operators in their efforts to control the accident.

Findings and Conclusions

The lack of hot leg injection capability in B&W plants probably did not significantly affect the course of the TMI-2 accident, but might have been a valuable option for use by the operating crew.

I-C-1.8 Decay Heat Removal System Not Designed for Operating Pressures

The decay heat removal system in pressurized water reactors are designed for use during a normal plant shutdown, rather than during accident situations. It is designed to be used after the plant has been cooled down and depressurized by other systems (e.g., the emergency feedwater system) to relatively low temperatures and pressures. After this is accomplished, the DHR system is initiated to provide the long term cooling of the reactor core.

At about seven hours into the accident, an attempt was made to depressurize the reactor coolant system to pressures at which the DHR system could have been used. It was believed by the operating crew that the use of the DHR pumps, which have a much higher pumping capacity than the makeup pumps, would have a better effect on the temperatures being seen in the reactor coolant system. However, the pressures in the RCS did not drop sufficiently low to use these pumps.

It has been suggested that the relatively low design pressure of the DHR system is a plant deficiency which was detrimental to the recovery from this accident. In one sense this is correct, in that the low design pressure did not permit use of the DHR pumps at the time period discussed above. In another sense, the low design pressure of the DHR system is not a real deficiency. For accidents such as that at TMI-2 where reactor coolant system pressures remain high, a cooling system with the capability to operate at high pressures is designed and installed, this being the high pressure injection (HPI) system. The DHR system thus may be considered as a backup system to the HPI system.

In the TMI-2 accident the high pressure injection system was automatically actuated and began to operate as designed a number of times. Subsequent crew actions reducing the flow from the HPI system greatly compromised the capability of the system and were the direct cause of the damage to the core. The apparent need for the DHR system is thus predicated on the prior compromising of the high pressure injection system.

A decay heat removal system designed for operating pressures thus may be thought of as additional equipment redundant to the high pressure injection system. This additional redundancy of equipment has the potential for improving the reliability of the high pressure cooling function. However, it seems likely that operator actions to compromise one system, such as was the case for the HPI system of TMI-2, would also likely compromise any additional equipment. It is therefore not readily apparent that the lack of a decay heat removal system designed for operating pressures is a significant deficiency contributing to the accident at TMI-2.

Findings and Conclusions

A decay heat removal system designed for operating pressures would in essence be additional equipment redundant to the high pressure injection system. It is not clearly evident that the presence of such a system would have significantly altered the course of the TMI-2 accident.

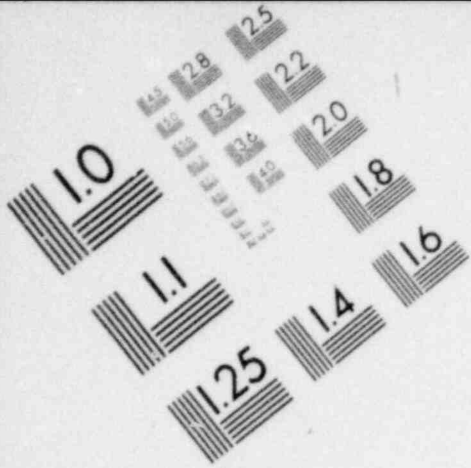
I-C-1.9 Adequacy of Debris Protection for the Reactor Building Sump

During the time of the accident, the gradual depletion of the primary water supply for the ECCS (the borated water storage tank) raised the possibility that ECC recirculation from the reactor building sump would be necessary. In the consideration of this, two concerns relating to the desirability of using the sump water arose. The first concern related to the possibility that debris might have entered the sump which could then be drawn into the ECC equipment and cause damage. The second concern related to radioactive contamination of the sump water. Since this water would have been drawn out into the ECC equipment in the auxiliary building, additional contamination of that building was of concern. This concern is discussed separately in Section I-C-1.2 above.

The reactor building sump design was considered in the licensing of TMI-2 to be part of the engineered safety features systems and as such was discussed in Chapter 6 of the final safety analysis report (Ref. 1). Section 6.2.2.2.1.11 of the FSAR specifically addresses sump debris elimination; the section indicates that in effect, the sump is completely enclosed in screens which minimize the likelihood of debris entry into the sump. Thus, for the conditions experienced during the TMI-2 accident, it appears that debris blockage of the reactor building sump was not a significant concern.

Findings and Conclusions

The reactor building sump design appears to have been adequate to protect vital equipment from debris damage in the event of sump water use in the recirculation mode of cooling.



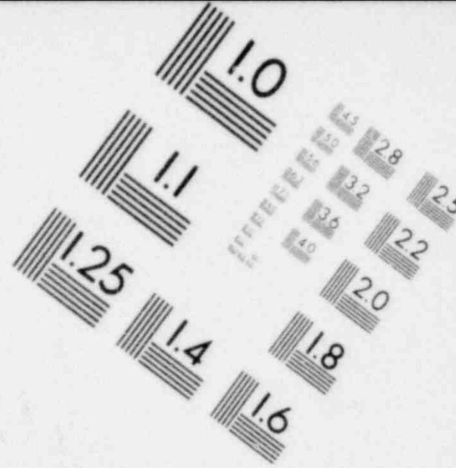


IMAGE EVALUATION
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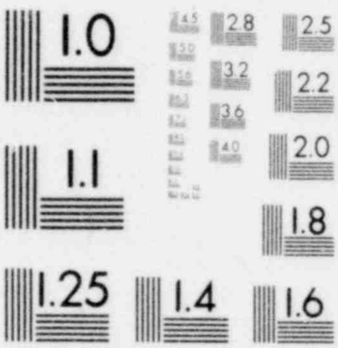

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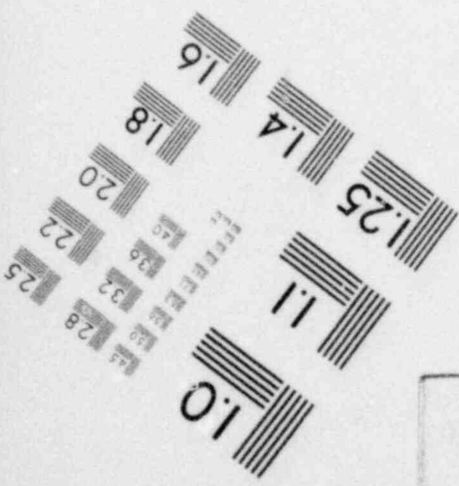
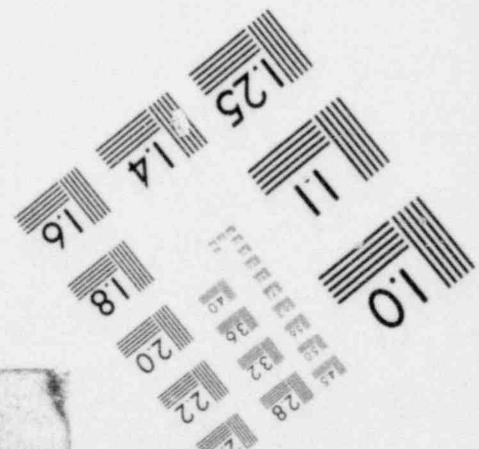
MICROCOPY RESOLUTION TEST CHART





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3. Final Safety Analysis Report -- San Onofre Nuclear Generating Station, Units 2, and 3.

4. Final Safety Analysis Report -- St. Lucie Plant

5. Final Safety Analysis Report -- Surry Power Station, Units 1 and 2.

6. Final Safety Analysis Report -- Sequoyah Nuclear Plant

7. Staff Report on the Generic Assessment of Feedwater Transients in B&W PWRs, NUREG-0560, May 1979.

8. Minutes of B&W/NRC meeting on losses of pressurizer level, February 14, 1979.

9. Report on Small Break Accidents for Westinghouse NSSS Systems, WCAP-9600, June 1979.

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11. Operating Units Status Report, NUREG-0020, Vol. 3, No. 9, September 1979.

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14. Letter from T.M. Anderson (Westinghouse) to Denwood Ross, NRC, May 1, 1979.

15. NRC TMI SIG deposition of staff of Radiological Assessment Branch, October 1979, pp. .

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17. (RELAP calculations).

18. (Natural circulation tests)

11/2/79
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I.C.1.X Emergency Feedwater block valves (EF-V12A and B) closed

Surveillance performed on the Emergency Feedwater System on March 26, 1979 may have resulted in the inadvertent closure of the block valves.

The surveillance procedure allows the simultaneous closure of the block valves when testing emergency feedwater pump operability. Such closure was required because of the known deficiency in the emergency feedwater level control valves (EF-V11A&B) to prevent leakage to the steam generators whenever the pumps were tested.

Surveillance procedures should not have permitted the simultaneous closure of both block valves since individual lines can be blocked and pumps tested when pump header valves EF-V5A and B are manipulated during test, to maitain at least one train of protection, at all times.

I.C.1.Y Leaks in reactor coolant system

It has been reported that the pressurizer relief valves were apparently leaking into the reactor coolant drain tank (RCDT) at approximately 6 gpm. This continuous leakage caused the boron concentration to continuously increase in the pressurizer and the relief valve exhaust to continuously indicate approximately 180-200 degrees F (the normal is 130°F).

Approximately 2600 gallons of water were transferred each shift (8 hours) from the RCDT to the makeup tank (MUT) via the RCBT prior to the shift on which the accident occurred. During the first 4 1/2 hours of the shift on which the accident occurred, 1800 gallons ~~were~~ was transferred which indicates that the transfer had increased to approximately 3600 gallons per shift on March 28, 1979 suggesting a substantial increase in the leak rate.

In violation of Technical Specification requirement the licensee was operating the facility during the March 22-28, 1979 period with an unidentified leak rate in excess of 1.0 gpm.

I.C.1.Z Leaks in makeup and letdown system

Since plant startup, there had been leaks detected in the waste gas system, and plant documents indicated that some efforts had been made to determine the source of the leaks. Some of the identified problems apparently were not corrected prior to the accident and may have caused releases to be larger that they normally would have been. Makeup tank vent valve had been suspected of leaking prior to the accident.

I.C.2 Command and Control Deficiencies

2.1 Reactor Coolant Pressure Control

Reactor coolant pressure is automatically controlled by pressurizer (a) electric heaters, (b) spray valve, and (c) power operated relief valve. This pressure control system is not classified as a system important to safety (Ref. 1) and therefore, the failure of the power operated relief valve (PORV), to reseat was not considered to cause unacceptable consequences in a transient mitigation sequence (Ref. 1, 2). Failures in the electric heaters further limited the ability of this control system to maintain system pressure above saturation temperature, at times that the operators judged necessary to increase system pressure. Pressurizer spray also became unavailable for pressure control when the forced reactor coolant circulation was interrupted by the isolation of the reactor coolant pumps.

Electric power supply for the pressure control system is provided by the offsite power source, whose interruption would have made the pressure control system unavailable indefinitely and the PORV block valve (located on the pressurizer side ahead of the PORV) unable to close at the operator's command. A higher availability electric power supply could be achieved if the onsite power sources (diesels) were made of sufficient capacity with appropriate power distribution interconnections to supply the required electric power (Ref. 7).

Additionally, since the PORV is regularly challenged from overpressure transients and has a long history of failures in the open position automatic closure of the block valve from coincidence signal (primary pressure and PORV exhaust temperature) could limit the need for operator interventions.

The failure on the part of B&W, Met Ed and NRC to acknowledge the safety significance of the reactor coolant pressure control system contributed to the lack of focus on the failure mode and effects analyses of this control system. Individual expert reviewing staff recognized that control systems may be important to safety of the plant but it remained NRC policy to exclude such systems from plant review (Ref. 3,4,5,21).

2.2 Lack of anticipatory reactor trip on turbine trip

An inadvertent main turbine isolation (trip) creates a mismatch between heat generation in the primary system and heat removal by the secondary system. This mismatch results in increase in temperature and pressure in the primary system leading to the opening of the power operated relief valve (PORV), in an attempt to retain primary system pressure below the high pressure reactor trip.

An anticipatory reactor trip following main turbine isolation requires a turbine steam stop valve closure or main generator breaker open signal to the reactor protection system for a near simultaneous reactor trip. The anticipatory reactor trip prevents, in most instances, the actuation of the PORV and negates control rod runback. The control rod runback is a feature of the integrated control system (ICS).

The ICS (by control rod runback and steam generator level control) and the pressure control system (pressurizer heaters, spray and the PORV) provide control of the primary and secondary systems, for transients that involve loss of load, turbine trip or feedwater loss. The above control systems attempt to maintain the primary system pressure above HPI and low pressure reactor trip and below high pressure reactor trip, in order to r[illegible]ain high unit availability. However, at TMI-2 with the opening of the PORV the pressure dropped below saturation, causing boiling in the primary, when the PORV failed to reseat to stop the loss of inventory. Additionally, the pressurizer level control further contributed to the loss of inventory by the automatic and manual throttling of makeup and HPI and from increases in letdown flow from the primary (Ref. 5, ~~33~~ 22, 31).

The NRC staff has not conducted reviews to assess the significance of anticipatory trips which were voluntarily introduced earlier, in Westinghouse & General Electric designs. It was a management decision to place the control systems and anticipatory trips out of the scope of staff review (Ref. 5,6).

An anticipatory reactor trip for all B&W plants appears desirable, at this time, until the adequacy of the primary pressure control system (pressurizer size, PORV, etc.) and the secondary heat removal system (steam generator size, ICS, etc.) to maintain the plant in a safe mode is ascertained (See Section I.C.1 of this report).

The introduction, however, of unnecessary anticipatory reactor trips can cause plant availability to suffer, increase the undesirable frequency of reactor trips and increase the likelihood of unacceptable consequences from anticipated transients without scram (Ref. 8). Therefore, a thorough evaluation of the controls involved and the implementation of adequate design features should improve system response and possibly minimize the need for anticipatory reactor trips.

2.3 Reactor Building Isolation

Reactor building isolation was actuated by safety features actuation signal (SFAS), on high reactor building pressure (4 psig), reached approximately four hours following the start of the accident.

For a considerable time prior to isolation radiation was released to the auxiliary building by reactor building sump discharge, reactor coolant letdown, reactor coolant drain tank vent, and reactor coolant pump seal injection return. Following isolation the plant operators manually defeated the isolation signal from the reactor coolant letdown and the reactor coolant pump seal injection in order to place both systems, letdown and seal injection, back in operation. This further contributed to radiation releases in the auxiliary building.

Reactor building isolation on high pressure is based on loss of coolant accident (LOCA) analyses that assume rapid increase in pressure in the reactor building prior to radiation releases resulting from the postulated fuel damage (Ref. 7).

The deficiency of the above design appears to be associated with the lack of (1) direct measurement of all important parameters (e.g., radiation), (2) sufficient LOCA analyses (small break) to determine accurately the values of all important parameters and (3) inadequate hardware and operating procedures that permit resetting of isolation signals and the reactivation of selected components and systems (Ref. 9).

2.4 ECCS (HPI) bypass

One of the most crucial contributors to the accident was the interruption of HPI flow and its subsequent throttling, as the accident progressed (Ref. 22, 29, 31). This interruption and throttling become possible only after reset (bypass) of the safety injection signal.

Emergency procedures for a number of abnormal conditions including small break LOCA mitigation made the requirements for immediate reset of HPI signal necessary because of the deficiency in the HPI pumps, DH pumps and reactor building spray pumps to withstand runout conditions when the pumping capacity exceeds their limiting design capability (Ref. 28). Additionally, the operators were specifically instructed early in the LOCA procedures to prevent the primary system from filing solid by interrupting coolant flow to the reactor. Such operator interventions are not in compliance with the NRC stated position that credit for operator action to mitigate the consequences of postulated accidents is only given if such actions can be taken 10 minutes or longer subsequent to initiation of the accident (Ref. 13).

An exception granted to TMI-2 in 1978 permitted manual reallignment of HPI lines to assure sufficient coolant injection to the core for small break LOCAs. To perform such reallignment manual control of injection valves is required following reset (bypass) of safety injection. Recognition of a small break LOCA by the operator is expected, in emergency procedures (Ref. 28), in less than 2 minutes for a successful mitigation by operator action. For the small break LOCA that actually occurred, however, recognition was not made until 2 hours and 20 minutes into the accident (see chronology of this report).

The NRC and B&W failed to act (on the repeated warnings from their own staff and the recommendations of the ACRS) to carry out their own regulatory responsibility to resolve the issue of reset, by operator action, before 10 minutes following safety injection initiation (Ref. 11, 13, 14). The survey conducted by IE (Ref. 32) erroneously reported that adequate procedures are in place in all operating reactors including TMI-2, to cover all necessary operator actions prior to and after SIS reset. The instruction in the procedure (Ref. 28) only requires reinitiation of RB isolation and cooling, <u>not</u> safety injection (See Section I.C.1 for additional discussion).

An additional deficiency that made safety injection reset necessary was the need to prevent inadvertent addition of sodium hydroxide to the core through the DHR system (Ref. 22). The DHR system at the injection phase, receives its water supply from the BWST as the containment spray does also. However, since the sodium hydroxide tank for containment spray discharges at the common suction line of containment spray and DHR systems, sodium hydroxide can enter the core requiring cleanup. Therefore, the operators were instructed to routinely defeat safety injection.

The vulnerability to loss of safety function following injection reset may be continuing to exist, to a varying degree, in most nuclear plants. The NRC staff erroneously testified before the Licensing Board (Ref. 23) for TMI-2 that the issue of safety injection reset was not applicable to this plant.

Deficiencies in operating plants continue to make operator intervention (early safety injection reset) necessary in order to prevent damage to safety components and systems. Some safety components and systems are not capable of performing at plant conditions (e.g., increased flow rates, pressure, temperature, etc.) expected (Ref. 24, 25) during an accident or transient mitigation sequence. A correct design consistent with the NRC's position, of no operator intervention for at least 10 minutes, would require for example that reactor coolant pumps, high pressure injection pumps, decay heat pumps, pressurizer relief valves, RB spray pumps, etc., be compable to withstand system conditions expected ~~from~~ during the course of the accident; without damage that can cause loss of safety function required to mitigate the consequences of the accident (Ref. 24, 25). The LOCA emergency procedures (Ref. 28) and other procedures, in effect at TMI-2, caution the operator and require manual actions within short ~~term~~ time following the accident to prevent damage to equipment.

2.5 Reactor building hydrogen concentrations control

In approximately 10 hours following the initial activation of the PORV hydrogen in the reactor building reached flamability concentraiton. The primary source of the hydrogen generation is attributed to the zirconium water reaction in the reactor core, when the core overheated as a result of its prolonged uncovery (See Section I.C.1 of this report).

The lack of an automatic hydrogen recombination system allowed the concentrated hydrogen to egnite, creating a pressure surge of about 31 psig in the reactor building. The building is designed to withstand pressures in excess of 60 psig (Ref. 1). Some reactor buildings however, can only withstand pressures of the order of 12 psig.

The regulatory criteria, applied to TMI-2, required provisions for hydrogen recombination systems to deal with slow (days) post-accident generation of hydrogen, following a LOCA, from (a) about 1 percent of clad metal-water reaction, (b) corrosion of materials inside the reactor building and (c) radiolytic decomposition of water (Ref. 1, 7). The primary source of hydrogen being that from corrosion of materials inside the reactor building and not the clad metal-water reaction which was the major source at TMI-2.

The provisions at TMI-2 called for post-accident installation and operation of external hydrogen recombiners. These recombiners would be hooked up at the 36 inch reactor building penetrations which were used for normal reactor building purging. However, such recombiners currently in use are not capable of preventing the rapid increase in pressure (31 psig) attributed to hydrogen egnition at TMI-2 (See Section I.C.1 of this report).

Reanalysis appears necessary to more accurately determine the principle sources of hydrogen generation, following an accident, for the implementation of a hydrogen recombination system that can successfully perform to control the hydrogen concentration in the containment building (Ref. 7).

2.6 High pressure injection control

Throughout the course of the accident high pressure injection pumps were either inadvertently tripping or were unable to start by automatic or manual commands (Ref. 22, 30).

Failure to maintain high pressure injection pumps operating was attributed to control component deficiencies and to undesirable operator actions. Control switches were placed in pull-to-lock off position, whenever the operator deemed necessary to take pumps out of service. Pull-to-lock is an off normal position prohibited during plant operation or during accident mitigation. At this position automatic command can not place equipment in service when required by system conditions (Ref. 1).

The inadvertent or deliberate placement of control switches in the pull-to-lock off position caused pumps to be inoperable when high pressure injection signals were reinitiated by abnormal system conditions later into the accident sequence.

The lack of automatic override features to remove the pumps from the off normal position, or to alarm that pumps are not alligned for safety injection is a deficiency that may have caused the operators to be confused regarding the operability of the pumps (Ref. 22).

Other unsuccessful attempts to automatically or manually start pumps appear to have been attributed to contact bounce of laching relays (Ref. 22).

2.7 Emergency Feedwater actuation and control

Loss of main feedwater, which initiated the accident, resulted in the actuation of the emergency feedwater system -- the emergency feedwater pumps were performing at full capacity within 40 seconds. However, since the discharge block valves were inadvertently closed water did not enter the steam generator until 8 minutes into the accident, when the block valves were manually opened. The steam generator automatically rose to a level of only 34 inches, as opposed to the required 32 feet analyzed for small break accidents (Ref. 26). Operator interviews have indicated that emergency level at TMI-2 is 21 feet (Ref. 27). It is uncertain, however, whether the course of the accident would have been altered even if the 21 feet level was automatic achieved since analysis does not appear to have been made by B&W for the 21 feet (Ref. 29).

By the time the block valves were opened the steam generators had boiled dry, the PORV had failed in the open position and high pressure injection actuation had been initiated. Hence, a small break LOCA had occurred and the emergency feedwater system if it had been properly designed should would have supplied water to the steam generators to raise the level to 32 feet; which is the required level for successful mitigation of small break LOCAs as postulated by B&W in their analysis. The emergency procedures (Ref. 28) did not include instructions for steam generator level requirements for the mitigation of small break LOCAs. Revised procedures (Ref. 29, 39) include level requirements.

Emergency feedwater level control is significant, according to B&W, for small break LOCAs. The analysis presented to the NRC by B&W in topical report BAN-10075A, Rev.1, was based on a 32 foot emergency feedwater level. This level, however, and its significance, to mitigation of accidents was not reported to the NRC and was also not included in the small break LOCA emergency procedures (Ref. 28) for TMI-2.

The 21 feet emergency water level in the steam generators, that the operators were aware of, might have been reached during the accident if high pressure injection actuation was coincident with loss of offsite power (reactor coolant pumps tripped) (Ref. 27). However, since offsite power was not lost at TMI-2 the ICS system controlled the steam generator level to a 34 inches. The ICS did not recognize the incident as a small break LOCA.

A design feature that controls steam generator level to 34 inches during feedwater transients appears to have been desirable in order to maintain pressurizer level indication, by limiting shrinkage in the primary coolant.

The need for dual level stepoint in the steam generator had become apparent in another B&W operating plant in the past. B&W did not inform its customers or NRC of the deficiency in the control system to recognize small break LOCAs with reactor coolant pumps running (Re. 26).

The deficiency in the system design to properly recognize the steam generator level requirement of 32 feet appears to have resulted in high pressurizer level indication, which the small break LOCA emergency procedures do not predict to occur. Emergency procedures for small break LOCA predict low pressurizer level. Hence, the operators did not apply the small break LOCA procedures and continued to throttle high pressure injection to prevent the primary system from going solid.

The reluctance on the part of B&W to properly identify the significance of the emergency feedwater system for mitigation of small break LOCAs has resulted in a system design classification of lesser stringency than that required for other systems important to safety. A properly designed system would probably have resulted, as a minimum, in diverse and redundant automatic actuation of pumps, discharge valve allignment (including the block valves), and high steam generator level control.

Operator interviews (Ref. 27) have revealed that the emergency level for the steam generators is 21 feet and could only be reached automatically by the ICS if all reactor coolant pumps were tripped. Therefore, since the pumps were not tripped the level remained at 34 inches.

2.2.8 Lack of automatic bypass on the demineralizer/polisher

The initial loss of main feedwater, prior to reactor trip, has been attributed to clogging of resin in the condensate polishers which resulted in the closure of the polisher outlet valves (Ref. 22). Bypass, electrically operated, valves around the polishers are manually controlled from the control room. Therefore, the initial transient would not have been prevented, since normal actuation of the bypass valves might not have been insufficient time to prevent reactor trip and the subsequent high pressure injection. Automatic actuation of the bypass valves with isolation of inlet or outlet valves at the polisher could have retained main feedwater and have prevented the PORV from exceeding its pressure setting (Ref. 27).

Efforts to open the bypass valve from the control room failed because the valve had previously been over-torqued in the closed position (Ref. 22, 30) making the motor operator unable to unseat the valve. The motor breaker was tripped by the torque limiting switches whose setting was exceeded. Surveillance requirements on valve operability, within expected settings, are not routinely applied by NRC staff on valves important to safety or nonimportant to safety. Hence, valve malfunctions from over-torque can be generally undetected. Periodic partial actuation of valves may not reveal all the torque requirements that can be applied on the motor operator during expected service conditions.

System designs should consider implementation of piping configurations that can permit periodic testing of valves to system conditions (e.g., differential pressure, temperature, etc.) expected during emergencies. Proper torque switch settings could be verified by comparison of the power/torque delivered to the valve assembly during test with the maximum setting of the torque switches for valve motor trip.

2.2.9 Instrument air system

The loss of the main feedwater pumps, that initiated the turbine trip followed by a reactor trip, has been attributed to the presence of water in the instrument air system which caused the condensate polisher air-operated outlet valves to close (Ref. 22, 30). Water at 100 psig in the condensate polisher entered the service air system, which is at 80 to 100 psig, through a failed open check valve.

Station service air used to free blockage in the resin transfer line is cross connected with the instrument air system. Inadequate capacity in the air system caused the licensee to cross-connect the service air to the instrument air as a normal mode of operation of the two systems. The mode of operation for air supply on March 28, 1979, was the cross-connected system.

For a total of about 11 hours until the feedwater transient operating staff were attempting to transfer resin from condensate polisher tank No. 7 to the resin regeneration tank. The inability to perform this transfer was attributed to a resin blockage in the transfer line (Ref. 22, 30).

It appears that as a result of the actions taken to clear the resin blockage in the transfer line the polisher outlet valves closed and condensate pump A tripped causing loss of condensate flow with an almost simultaneous trip of the main turbine.

The licensee had installed air dryers at various points in the instrument air system to prevent the accumulation of moisture. In particular, an air/water separator was installed in the condensate polisher instrument air line in series with two pressure regulators. This arrangement processed all air to the condensate valve controls and instruments located on the condensate polisher local control panel.

The licensee has performed tests on the condensate polisher instrument air system subsequent to March 28, 1979 and has indicated that upon isolation of instrument air from the condensate system, the condensate outlet valves for each polisher tank closed. Also, the tests indicated that introduction of water into the air system did not affect the polisher outlet valves in that the air-water separator functioned properly (Ref. 22).

2.10 Condenser hotwell control

2.10.1 Loss of Hotwell level control

Following the initial turbine trip and closure of the main steam isolation valves, steam release to the main condenser continued via the turbine bypass valves (Ref. 22). However, in the course of the accident the hotwell level control valve controller failed, in the low level setting and caused the hotwell to be flooded from the condensate storage tank. The failure of the level control valve controller caused the hotwell makeup valve to remain open allowing condensate storage tank water to discharge in the hotwell by gravity force, and therefore, steam release to the condenser was interrupted.

Subsequent to the flooding of the hotwell the operators attempted to reduce the level by discharging the hotwell to the condensate storage tank through a condensate pump (Ref. 22, 27). However, failure of the hotwell level reject valve did not permit the discharge until about 3 hours into the accident and after the reject valve was opened.

2.10.2 Loss of condenser vacuum

Following recovery of hotwell level, condenser vacuum started to decrease and eventually was lost (Ref. 22, 27). Condenser vacuum is also required in order to maintain the ability to release steam to the condenser. (Ref. 31).

Loss of vacuum resulted from loss of the auxiliary boiler that provides sealing steam for the interface between the turbine shell and the main shaft (Ref. 27).

2.11 Reactor coolant pump control

Throughout the course of the accident forced circulation of the primary coolant appeared necessary to assure decay heat removal from the primary; since natural circulation was inhibited, due to early voiding in the primary coolant system. Reactor coolant pump operability therefore, was necessary in order to maintain the required forced circulation (See section 1.9.X of this report).

During various phases of the mitigation sequence reactor coolant pumps were removed from service because conditions in the primary system exceeded those allowable for continued pump operation (Ref. 22, 27, 28). Hence, the required forced circulation was interrupted for extended periods of time (See chronology of this report).

At subsequent times operators were unsuccessful in their attempts to restart reactor coolant pumps because various permissives in the start circuit of the pump controls were not satisfied (Ref. 22). The ability to start the pumps, however, was regained when operators physically bypassed permissives and placed the pumps in operation.

A.C. electrical power supply to oil lift pumps, for the reactor coolant pumps, was lost when two motor control centers were inadvertently isolated making a reactor coolant pump permissive (oil lift pump running) not be satisfied in the coolant pump start circuit. Operators manually bypassed this permissive and started the reactor coolant pumps with oil lift provided by pumps powered from a D.C. power supply.

It appears that the deficiency existed in the reactor coolant pump start circuit, because the operator was required to manually satisfy (short relay contacts) permissives in order to start the pump. Since a redundant source of oil lift (D.C. powered pump) was available an automatic permissive in parallel with that from the failed A.C. powered lift pump, would not have required the operators to manually bypass permissives at the location of the electrical switchgear, for the reactor coolant pumps, in the auxiliary building.

It appears that lack of coordination in protective overcurrent relaying in the motor control centers may have resulted in their trip that removed from service the oil lift pumps, which provide the permissive for the reactor coolant pumps. The motor control centers were simultaneously tripped when the leakage closed cooling pumps in the containment were inadvertently tripped shortly after the high pressure explosion in the reactor building (See chronology of this report). An overcurrent condition that may have caused the pumps to trip simultaneously may have tripped the main feeder breakers to the motor control centers. A correct coordination would place main feeder breaker trip setpoints above individual branch circuit protection.

2.12 Diesel generator lockout

The first safety injection actuation signal initiation, in about two minutes into the accident, automatically started the emergency diesel generators. The generators provide an alternate onsite power supply to safety related equipment in the event of loss of the offsite power sources.

The diesel generators were turned off by the operators, as instructed by procedures (Ref. 22, 27), after it was established that offsite power was not lost and the diesels were running unloaded. The diesels at TMI-2 are not designed for prolonged operation unloaded, without damage to the exhaust system from excessive carbon deposits. Unloaded operation is only permitted for a few minutes. Therefore, following safety injection reset the diesels were turned off.

In order to prevent subsequent restarts of the diesel generators, following reinitiation of safety injection signal, the operators permanently defeated the automatic starting capability by isolation of the fuel at the fuel racks in the diesel rooms.

Isolation of the fuel to the diesels left the plant vulnerable to total loss of power supply for systems important to safety, in the event of loss of offsite power. The diesels could have been made available at a later time if it was recognized, in time, that the fuel was isolated and if it was reset. Operator interviews have revealed that the principle operating staff were not aware, at all times, of the fuel isolation (Ref. 27).

At a later time when the station electrical engineer arrived at the site he instructed the operating staff to reset the diesel fuel racks and isolate the diesels at the control room. Manual switches used to place diesels out service during maintenance were actuated at the control room and placed the diesels on manual control (Ref. 27).

Review of the emergency procedures (Ref. 28) have revealed that the instructions to the operator, regarding loss of offsite power during this accident, would not have resulted in the required engineered safety features actuation sequence. The instructions in the procedures call for manual reinitiation of the RB isolaton and cooling actuation. This actuation is not the safety injection actuation and therefore if offsite power was lost following safety injection reset, injection systems would not have operated. Fault appears to lie with lack of recognition that loss of offsite power can occur any time during an accident sequence and that RB isolation and cooling actuation is independent of safety injection actuation.

The deficiency in the diesel generators to run unloaded without damage resulted in a minimum redundancy in power supply, during a crucial period of the accident.

Diesel generators presently available commercially can run unloaded at sufficient length of time without damage from excessive carbon deposits.

2.13 Reactor building sump pump operation

There are two reactor building sump pumps, WLD-P2A and P2B each having a design pumping capacity of 140 gpm. The A pump started at 8 minutes 19 seconds into the event. The B pump started at 10 minutes 19 seconds. Both pumps were shut off by the operators at 38 minutes. The amount of water transferred totaled about 8000 gallons.

This sump water was likely a mixture of reactor coolant drain tank (RCDT) quench water and primary coolant vented through the RCDT relief valve until 15 minutes into the event.

The sump pump discharge is normally aligned to the miscellaneous waste holdup tank WDL-T-2. However, the tank level records do not show any level change in this tank. Therefore, the pumps may have been aligned to the alternative discharge point in the auxiliary building sump tank, which overflowed to the auxiliary building sump. (This tank had a blown rupture disk which was scheduled for later repair). The blown rupture disk allowed the water to spill from the tank to the sump causing the release in the auxiliary building (Ref. 22).

C.3 Deficiencies in Instrumentation

3.1 Lack of sufficient instrument indicating range

Some instrument readings, not previously anticipated, became important to proper operator actions for mitigation of the consequences of the accident. These readings, however, did not accurately display the true level of the measured variable because the level exceeded the indicated range of the recording instruments (Ref. 22).

Display instruments for incore thermocouples, used to measure temperature conditions in the core, have an indication range of about 700°F. Thermocouple temperatures during the accident exceeded 2000°F but were not indicated by the instrumentation available to the operator. External temporarily placed instruments (digital voltmeters) with sufficient range, recorded these higher temperatures. However, since such temperatures were not anticipated and provisions were not made for use or display, the operators did not place the proper significance to the higher temperature readings recorded. Operators have indicated in interviews that they were reluctant to attribute significance to the readings because the thermocouples were not assessed important to safety and were not designed to safety standards (Ref. 33, 34).

Reactor coolant temperatures also exceeded the indicated range of their display instruments during the accident. The indicated range for hot-leg temperatures is 0-600°F. Strip chart recordings have a range up to 800°F which were also exceeded.

3.2 Accuracy of pressurizer level instrumentation

Unexpected level indication in the pressurizer persisted as a result of swelling of the overheated primary coolant and perhaps the unavailability of emergency feedwater within a few seconds (See secton ~~1.9.X~~ of this report). The expected response in pressurizer level indication, following the TMI-2 accident, is a rapid decrease in level when the emergency feedwater system is immediately available. The lack, however, of the emergency feedwater system caused the primary coolant to expend leading to false high pressurizer level indication implying that the primary system was filing to a solid condition. This condition was unacceptable to the operators, who by approved written procedures (Ref. 28), intervened to interrupt HPI and normal makeup, and increase the letdown from the system.

Failure to correctly attribute safety significance to pressurizer level, even following a number of telling incidents (Ref. 35), allowed routine operational judgments to dictate reactor coolant system performance.

Level indication was provided by three physically independent level transmitters two of which failed during the accident mitigation sequence (two days later), causing the need for alternative indirect methods to be employed to assure continued level indication. This level indication remained important for the continued assessment of the primary coolant system pressure.

It appears that thermohydraulic analyses, from which most of the principal instrumentation and control for reactor protection is derived, lacks accuracy in predicting system variations. The unexpected level indication at TMI-2 and the incident at a foreign reactor (Ref. 36) seem to indicated the need for reassessment of some thermohydraulic models. In the case of the foreign reactor, of a different vendor, the automatic actuation of safety injection was derived from pressure and level instrumentation that requires simultaneous decrease of pressure and level in the pressurizer, with postulated accidents. However, as it has been demonstrated, for at least a certain range of postulated accidents, pressure and level do not decrease simultaneously.

3.3 RCS temperature instrumentation

In the course of the accident reactor coolant hot leg and cold-leg temperatures exceeded their measured maximum and minimum temperature ranges of 620°F and 520°F respectively. When the emperatures exceeded these values the computation for average temperature measurements remained constant for many hours (Ref. 22).

The average temperature computed remained at about 570°F, only 10°F lower than the normal operating level. This fact appears to have confused some operators who did not recognize that the average temperature reading of the instruments was in error.

Operator interviews revealed that there was confusion, for at least one control room operator, regarding the average temperature reading which possible misled him (Ref. 33, 34). The erroneous average temperature indication seems to have implied to the operator that decay heat was being successfully removed from the reactor by the secondary loop via the steam generators. The apparent fact that the average temperature, during a major part of the accident, was a few degrees lower than the operating values (582°F) appears to have convinced the operator that heat transfer was effective.

Decay heat removal by natural circulation via the steam generators, as attempted by the operators within 2 hours into the accident invokes a very complex procedure (Ref. 10) whose limits, precautions and prerequisites could not have been met by the conditions that existed at the time. The above procedure is invoked by the initial emergency procedure (Ref. 37) on loss of both main feedwater pumps.

3.4 Lack of flow recording for reactor water makeup

Throughout the accident flow indication of the makeup and high pressure injection was very important to the operators, particularly when these systems were placed on manual control. Makeup flow and high pressure injection were continuously throttled by the operators in order to control pressurizer level and the instantaneous flow indication was used for that control. This indication, however, was not recorded for later reference, as it became important for assessment of water inventory in the reactor.

The lack of flow recording for reactor coolant makeup, letdown and high pressure injection has hampered evaluation of reactor inventory assessment during and following the accident.

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